1000 MWe

L M F B R

Liquid Metal Fast Breeder Reactor

FOLLOW-ON STUDY

GENERAL ELECTRIC

TASK IV REPORT**Research and Development
Requirements for a 1000 MWe
Sodium-Cooled Fast Reactor**

Prepared for ARGONNE NATIONAL LABORATORY

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GEAP-5769
AEC RESEARCH AND
DEVELOPMENT REPORT
APRIL 1969

**RESEARCH AND DEVELOPMENT
REQUIREMENTS FOR A 1000 MWe
SODIUM-COOLED FAST REACTOR**

**TASK IV REPORT OF 1000 MWe
LMFBR FOLLOW-ON WORK**

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PERFORMED FOR ARGONNE NATIONAL LABORATORY
CONTRACT 31-109-38-1997

BREEDER REACTOR DEVELOPMENT OPERATION

GENERAL ELECTRIC

SUNNYVALE, CALIFORNIA

RESEARCH AND DEVELOPMENT REQUIREMENTS
FOR A
1000 MWe SODIUM-COOLED FAST REACTOR

TASK IV REPORT OF 1000 MWe LMFBR
FOLLOW-ON WORK

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Performed for Argonne National Laboratory
Contract No. 31-109-38-1997

April, 1969

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SECTION I

ELEMENT 1 - OVERALL

1.1 OBJECTIVES OF STUDY

This work was performed for Argonne National Laboratory as a part of a continuation of 1000 MWe Liquid Metal Fast Breeder Reactor (LMFBR) design studies previously performed for the AEC.^(1, 2) Major objectives of this follow-on work are to provide a relative comparison of the cost incentives for various design alternatives and to consider the cost incentives against the required development to achieve the state-of-technology implied. A further objective is to provide a reference design with sufficient detail to permit a better definition of absolute costs than was possible in previous studies. To implement these objectives, five design tasks were identified by Argonne National Laboratory:

- Task I

The development of one or more nuclear steam supply system concepts, with certain trade-off studies to aid in the definition of these concepts. The selection of a reference concept for further study.

- Task II

Preparation of system design descriptions defining the plant systems and components for the reference concept, and a cost estimate of the reference concept.

- Task III

Evaluation of trade-off and optimization studies in which numerous design parameters and economic ground-rules are varied to determine the effect on the economics of the reference concept.

- Task IV

Preparation of an R&D program to accomplish the state-of-technology implied by the previous three tasks and by Task V.

- Task V

Selected safety studies.

The results of the Task I effort have been described in the Task I Report: "Comparison of Two Sodium-Cooled 1000 MWe Fast Reactor Concepts," GEAP-5618, June 1968. The results of the Task II effort have been described in the Task II Report: "Conceptual Plant Design, System Description and Costs for a 1000 MWe Sodium Cooled Fast Reactor," GEAP-5678, December 1968. The result of the Task III and V work have been described in the Task III and V Report: "Methods, System Optimization and Safety Studies for a 1000 MWe Sodium-Cooled Fast Reactor," GEAP-5710, February 1969.

This document presents the results of the Task IV effort. Specifically, the new data, information and R&D programs required for the successful design, construction, and start-up of the Task II 1000 MWe LMFBR by the 1980 period are defined.

1.2 METHOD OF APPROACH

Information in this document pertaining to the required development work is provided as follows:

- Informational needs which are unique to the 1000 MWe LMFBR (as that LMFBR is defined in the Task II Report).

- Informational needs which are common to the Demonstration Plant and the 1000 MWe LMFBR. This includes a definition of the current state of technology, the informational base expected to be established by the Demonstration Plant, and the additional information required by the 1000 MWe LMFBR.
- The programmatic information for the 1000 MWe R&D (e. g., costs, manpower, schedule for non on-going programs).
- Identification of on-going programs which meet programmatic and informational needs of the 1000 MWe LMFBR.
- An estimate of the probability of success of R&D programs and a quantitative assessment of penalties for non-success, where applicable.

The format followed is that of the LMFBR Program Plan. The same identification as the Program Plan is used with the work divided into: Elements, Task Areas, and Tasks. Where possible, the same numbering system is used down to the task level. In some instances, tasks are reoriented somewhat from the Program Plan in order to be more responsive to the needs of this document. In other instances new tasks are added where work is needed which is not included in the Program Plan, or tasks are eliminated where they are not essential to the directed effort required for the development of the Task II LMFBR.

The development work is separated into the following elements which correspond to the same elements of the Program Plan:

- Element 1 - Overall
- Element 2 - Plant Design
- Element 3 - Components
- Element 4 - Instrumentation and Control
- Element 5 - Sodium Technology

- Element 6 - Core Design
- Element 7 - Fuels and Materials
- Element 9 - Physics
- Element 10 - Safety

Element 8, Fuel Recycle, which covers fabrication and reprocessing technology, is not included. Appendix III of the contract scope definition showed this work under the following studies:

- Task 2.1.6 A liquid-metal cooled shipping cask versus an air-cooled or other non-liquid-metal cooled shipping cask
- Task 2.1.7 On-site versus off-site fuel reprocessing and fabrication plant

Both of these studies were excluded from the General Electric contract.

The informational needs referred to above are related to the General Electric Company needs. That is, the program is developed to meet the technological needs of General Electric and not to fulfill the requirements of a national program or of some other contractor. Moreover, the cost estimates for the program and the schedule were prepared on the basis that General Electric will perform most of the work and will take advantage of available facilities at General Electric. Other assumptions would change the schedule and costs. All costs are given in 1968 dollars.

In general, the program is very much a directed program designed to meet the functional requirements of the Task II reference 1000 MWe LMFBR concept. Since many of the Task II design parameters are uncertain because of the present technological state, the program allows for alternative parametric choices, based upon the evolving work. The economic effect of altering

various design parameters from those used in Task II is quantitatively defined where possible. In this regard, the Task III report is a useful corollary to this document. In some instances it was not practical to list all of the economic penalties which might be incurred if particular design or development goals are not met. Instead, representative penalties are identified and the reader is encouraged to make use of the Task III Report which outlines numerous alternatives in more detail than is possible in this document.

The scope and specific nature of the program in this document are a strong function of the design philosophy adopted by General Electric. This philosophy can be stated as follows:

- a. The Demonstration Plant will be a "proof-of-principle" plant with functional goals compatible with a planned calendar 1976 start-up date. The major objective of the plant shall be the demonstration that a large, sodium-cooled, fast breeder, electrical generating complex can operate reliably and safely. Exact prototypicality of components or performance characteristics to those of larger commercial plants shall be of second order importance.
- b. The reference 1000 MWe LMFBR will be designed so that a first plant which has a start-up date at the end of calendar 1980 will be prototypical in both components and operating characteristics to a second 1000 MWe plant. The second plant will be commercially attractive and competitive with other forms of power expected to be extant in the 1980 period.

This particular philosophy was selected in order to maximize the probability of achieving a viable fast reactor economy in the

1980's. To this end, it is essential that the operating reliability of a large sodium-cooled fast reactor first be demonstrated on a utility network, and secondly, that the LMFBR performance characteristics be improved to achieve design goals established out of commercial considerations.

The second objective of achieving commercially desirable design goals plays a significant role in fixing the 1000 MWe development program. The Task II Report indicates that the expected value for the capital cost of the reference commercial plant is \$195/kWe, and this corresponds to a differential cost with water reactors of approximately \$37/kWe. These relatively high capital costs are not believed to be sensitive to the type of LMFBR system design (e.g., loop versus pool, hot-cell versus under-plug refueling, etc.), but are at least in part due to the nature of the numerous systems required for the LMFBR (e.g., secondary sodium loops, gas systems, etc.). Perhaps most importantly, the high capital cost differential between the LMFBR and water reactors is due to the relatively developed state of the water reactors compared to the LMFBR.

To offset the capital cost disadvantage of the LMFBR, a high availability factor and low fuel costs must be achieved for the operating 1000 MWe plant. This places a burden on the development program in four important elements:

- No. 3 - Components
- No. 4 - Instrumentation and Control
- No. 7 - Fuels and Materials
- No. 10 - Safety

In these elements the effect of Task II derived goals concerning plant availability factor (> 0.9) and fuel cycle costs (≤ 0.6 mill/kW-h) is to either increase the development compared to that which would be required for more modest design goals, or to

result in functionally duplicative development programs. For example, in the components element, it is essential that the development program demonstrate high component operating reliability at or near the operating conditions defined in the Task II Report. To obtain a high probability of success in meeting these goals, development work which functionally duplicates other on-going component development work is included on critical components, such as the steam generators. Such duplication was deemed to be warranted by the vital nature of these components to the overall success of the concept. It should be noted that although the development of the components meets similar functional goals to other on-going programs, the specific component design is not duplicative of the other programs. This enhances the probability of success of the LMFBR concept.

In an analogous manner, the Fuels and Materials element includes an ambitious oxide fuel development program designed to meet the functional requirements of the Task II core concept. Additionally, a carbide development program is included which serves the dual purpose of increasing the probability of meeting the economic goals of Task II and of potentially exceeding these goals.

The Instrumentation and Control and Safety elements have extensive development needs in order to assure that the low-neutron-leakage type core of the reference concept can meet certain minimum safety criteria. Failure to achieve this goal could result in economic penalties on the order of 1.0 mill/kW-h or greater. This would be sufficient to render the LMFBR non-competitive. Fortunately, the probability of demonstrating the ability to meet an acceptable safety criterion is high if the development program is carried out.

The 1000 MWe LMFBR development program is designed to take advantage of information obtained from anticipated Demon-

stration Plant development program work where the informational needs are common to the two programs. The informational base established by the Demonstration Plant program is identified in the individual 1000 MWe development tasks. Additionally, where possible the 1000 MWe program utilizes facilities which may be available from a Demonstration Plant Program. It should be cautioned however, that at the time of the preparation of this Task IV document, the development program for the Demonstration Plant was still under preparation and substantial changes could occur in these tentative plans.

1.3 SUMMARY OF PROGRAM

Table 1-1 shows a summary of the costs incurred in the different program elements and the total program costs. These costs are associated with new programs identified in this document, and not with on-going programs. On-going programs which are essential to the 1000 MWe development effort are identified in the individual task areas or tasks (e. g., Task Area 3-11, Liquid Metal Pumps).

Under the category "Overall" in Table 1-1, \$15,000,000 is shown as "First-of-a-kind" engineering and \$9,400,000 is shown as first-of-a-kind construction. These costs are also shown in the Task II Report (the construction costs are the difference in contingency between the "First" and the "Early Commercial" 1000 MWe LMFBR). These particular costs are early and imprecise estimates of more precise estimates which will be made later as a part of Task 2-1.11 Preliminary Plant Design (to be conducted during fiscal years 1974 and 1975 as part of the Plant Design Element). The first-of-a-kind engineering is intended to include all detail design of components, systems, plant, etc., which can be classified as being first-of-a-kind.

TABLE 1-1

NEW PROGRAM COSTS* FOR 1000 MWe LMFBR DEVELOPMENT

		Costs (1000 \$)	
<u>Element</u>		<u>Equipment and Facilities</u>	<u>Operating or Engineering</u>
1.	Overall		
	First-of-a-Kind Engineering	None	15,000
	First-of-a-Kind Construction	None	9,400
	PERT Program Planning	None	300
2.	Plant Design	400	5,800
3.	Components		
	3.6 Steam Generator	1,400	10,450
	3.7 Components Handling Equipment	2,200	1,470
	3.9 Reactor Vessels	550	2,250
	3.10 Control Rod Drives	300	2,400
	3.12 Intermediate Heat Exchangers	400	3,650
4.	Instrumentation and Control	2,350	5,600
5.	Sodium Technology	9,825	12,425
6.	Core Design**	1,850	6,570
7.	Fuels and Materials**	700	17,600
9.	Physics**	On-going Programs	3,855
10.	Safety	8,200	17,525
		<u>28,175</u>	<u>114,295</u>

* In terms of 1968 dollars

** Excludes irradiation unit charges, use charges for Pu, and withdrawal charges.

The "Equipment and Facilities" costs shown on Table 1-1 correspond to the needs for new equipment or modified equipment or facilities. The "Operating or Engineering" costs include expendable materials costs, contract labor, engineering-technician costs, computer charges, travel and living expenses, shop costs and other operating costs.

Table 1-2 shows the schedule for the 1000 MWe LMFBR development program. The assumed Fast Flux Test Facility (FFTF) and Demonstration Plant schedules are included under the "general" category, since important development tasks for the 1000 MWe LMFBR are scheduled around these other programs. For example, Breeder Test Facilities (BTF) are being built at General Electric and are planned to be used extensively for the Demonstration Plant component development program. The testing program of the 1000 MWe components is scheduled to take advantage of information from the Demonstration Plant Program, and of the facilities as they become available. Similarly, the Core Design and Fuels and Materials Elements of the 1000 MWe development effort are programmed to obtain in-pile testing of fuel pins and subassemblies at a high neutron flux level ($\sim 10^{16}$ n/cm²-sec). This will require the use of either the FFTF or the Demonstration Plant, and the program is, therefore, scheduled to take advantage of these facilities as they become available.

More detailed schedules and program milestones are shown in the individual elements. It should be observed at this point, however, that the timing of work conducted in specific elements is strongly dependent upon the scope and scheduling of work conducted in other elements. For example, it was noted previously that there are two functionally duplicative fuel development programs which are important to the overall success of the LMFBR. These two programs inter-relate with each other and with other elements in a highly complex manner. Their inter-relationships are summarized as follows:

- a. Initial fuel-swelling and steel swelling data are accumulated from the on-going PA-10 program under the Fuels and Materials Element during fiscal years 1969 and 1970.
- b. An updated oxide conceptual core design is performed under the Core Design Element in fiscal years 1971 and 1972 utilizing information from (a) and information from the Task II Report.
- c. A conceptual carbide core design is performed under the Core Design Element in fiscal years 1972 and 1973 utilizing similar steel swelling data and core design techniques as in (b).
- d. Oxide and fuel cladding development tests for functional conditions defined in (b) are performed under the Fuels and Materials Element during fiscal years 1973, 1974, and 1975.
- e. Carbide development for functional conditions defined in (c) is performed under the Fuels and Materials Element from fiscal years 1973 through 1982. Based upon the success of this effort in meeting the requirements of (c), a decision is made as to the desirability of designing a second or third core for the 1000 MWe LMFBR which is carbide. This would require additional design and development effort under the Core Design Element and the Safety Element which is not now included.
- f. A preliminary oxide core design is performed under the Core Design Element, based upon data from (d), and upon safety and control studies conducted under the Plant Design Element, during fiscal years 1974 and 1975. During this same period water flow tests of the core design defined in (b) and updated in (f) are performed.

TABLE 1-2

SCHEDULE FOR 1000 MWe LMFBR DEVELOPMENT

Calendar Year		1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	
Government Fiscal Year		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	
1.	General															
	1000 MWe Plant Detail Design															
	1000 MWe Plant Construction and Startup															
	Demonstration Plant (Assumed)															
	Detail Design															
	Construction and Startup															
	FFTF (Assumed)															
	Detail Design															
	Startup															
2.	Plant Design															
	Conceptual Studies															
	Preliminary Plant Design															
	Life Testing and Reliability Data Acquisition															
3.	Components															
	3.6 Steam Generators															
	Design															
	Test Prototypes															
	Fabricate and Deliver															
	3.7 Component Handling Equipment															
	Design															
	Proof Testing															
	Shipping and Installation															
	3.9 Reactor Vessels															
	Design and Analysis (excluding work under Task 2)															
	Tests															
	Fabricate and Install															

TABLE 1-2 (Continued)

Calendar Year		1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	
Government Fiscal Year		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	
3. 10	Control Rod Drives															
	Design															
	Tests															
	Fabricate and Ship															
3. 11	Pumps (Assumed)															
	Design															
	Test															
	Fabricate and Ship															
3. 12	Intermediate Heat Exchangers															
	Design															
	Test															
	Fabricate and Ship															
4.	Instrumentation and Control															
	Detector Development															
	Design Guidance															
	Design of Plant Instruments															
	Fabrication and Delivery															
5.	Sodium Technology															
	5-1, 7-11 Material Compatibility (Primary)															
	5-1, 7-11 Material Compatibility (Secondary)															
	5-4 Sodium Purification Technology															
	5-8 Plant Operation															
	5-11 Mass Transfer Engineering Mock-up															
	5-7 Cover Gas Atmosphere															

TABLE 1-2 (Continued)

Calendar Year		1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	
Government Fiscal Year		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	
6.	Core Design															
	Oxide Design															
	Carbide Design															
	Fuel and Core Water and Sodium Flow Tests															
	Fuel Subassembly In-Pile Tests															
	1000 MWe LMFBR Fuel Fabrication and Delivery															
7.	Fuels and Materials															
	Oxide Fuel and Cladding Development for 1000 MWe Conditions															
	Proof Testing of Oxide Pins															
	Carbide Development and Testing															
	Absorber Materials Testing															
	Bearings, Facing, Structural Materials Tests															
9.	Physics															
	Cross-Section Measurements and Data Evaluation															
	Demo-Plant Critical Experiments															
	1000 MWe Critical Experiments															
	Methods Development															
10.	Safety															
	Guidance and Special Studies															
	Code Development															
	Sodium Boiling															
	Fuel Failure															
	Post-Burst Phenomena															
	Containment Development															
	Transient Reactor Facilities															

- g. Proof tests of fuel pins are conducted in the FFTF and/or Demonstration Plant under the Fuels and Materials Element based upon the functional requirements of (f) from fiscal year 1975 through fiscal year 1981.
- h. In-pile tests of fuel subassembly mechanical and hydraulic features are conducted in the FFTF or Demonstration Plant under the Core Design Element, based upon the functional requirements of (f) and early test results of (g) from fiscal years 1977 to 1980.
- i. Final design and fabrication of the 1000 MWe core and fuel are performed under the Overall Element based upon early results from (g) and (h) and the functional requirements of (g) and data from the Plant Design Element from fiscal years 1976 to 1981.

As can be seen from this oversimplified listing, failure to accomplish a given task in a particular element can cause the entire program to slip with a consequent delay in the start-up of the 1000 MWe LMFBR. There are obviously other inter-relationships not included in the above list such as those with the Physics and Sodium Technology Elements. Similar lists could be prepared for numerous other components or technical disciplines in this program. It was beyond the scope of the present effort to show the complete inter-relationship in schedule and scope of the various elements. However, it should be recognized that the success of a directed development effort, such as this program, is strongly dependent upon a recognition of these inter-relationships, and upon taking appropriate action to alter the program as necessary to maintain a useful total effort as the work evolves.

Because of the problems of assuring the complete integration of the effort in this pro-

gram, a task is provided in the Overall Element (Task 1.1 PERT Program Planning) which includes the preparation of a PERT diagram for the 1000 MWe LMFBR development, design, and construction effort. This diagram will be continuously monitored and updated as the program proceeds. The objective of Task 1.1 is to obtain information from the PERT diagram to allow General Electric management and the AEC to determine critical paths in the development schedule, and thus assist in fixing priorities for performing the work. Input and planning for the PERT diagram should be provided principally by General Electric since alteration of the program to achieve selected intermediate goals can best be identified by the Design-Manufacturer of the 1000 MWe LMFBR complex.

Priorities for performing the work are discussed under the individual elements. As will be noted in that discussion, these priorities change with time. From an overall program point of view, the high priority work tends to fall into two categories:

- Work which is scheduled to commence very soon and could holdup other critical development effort if the schedule slipped, and
- Work which is in a technical area so vital to the total program that failure to accomplish it would jeopardize the 1000 MWe viability, and the current technology of the technical area must be extended significantly to reach the Task II reference conditions.

The following tasks are in the first category:

- Plant Design Element
 - Task 2-1.2 Pool Primary System Study
 - Task 2-1.4 Probabilistic Safety Study
 - Task 2-1.7 Dynamic Analysis and Control Study.

- Components Element
Task 3-7.2 Conceptual Design of
Components Handling Equipment.

In addition to this new work, there is of course continuing or on-going programs such as PA-10, and pump development which also fall in this first category of high priority work.

In the second category of high priority work are the following:

- Element 7 Fuels and Materials
- Element 4 Instrumentation and Controls
- Element 10 Safety
- Task Area 3.6 Steam Generators (in the Components Element)
- Task Area 3.9 Reactor Vessels (in the Components Element)

With regard to this second category of high priority work, it is noted that entire elements are included. The Fuels and Materials Element was selected because of its strong influence on economics. The Instrumentation and Control and Safety Elements were included because of their influence on determining the viability of the particular concept selected in Task II. The Steam Generators and Reactor Vessels Task Areas of the Components Element were chosen because of the present state-of-technology of these components and their importance to the overall plant. In any of this second category of high priority work, proper scheduling is also important. For example, under Element 10, Safety, a study of the possible modification of SEFOR to accomplish fuel failure testing is scheduled to commence in fiscal year 1970. The design and modification of SEFOR is tentatively scheduled for completion by fiscal

year 1975. Testing of short burnup fuel, or fuel previously irradiated in EBR-II to study failure mechanisms could commence in fiscal year 1975. Other fuel could be accumulating burnup in the FFTF or the Demonstration Plant until fiscal year 1977 or 1978 and then also tested in the SEFOR modified safety test facility. This would allow incorporation of the results into the final design of the 1000 MWe LMFBR and would materially assist in providing safety information for licensing bodies.

From the above examples, it is clear that an urgency already exists for accomplishing some of the required development for the 1000 MWe LMFBR. If a directed program with well defined priorities is not put into effect, a setback in schedule of from 5 to 10 years could easily occur. Indeed, some of the work is of such a nature (such as the Safety Element) that failure to launch a properly directed and timely program could force the adoption of economically damaging policies from which there is never a recovery. Carried to the extreme, such policies could jeopardize the economic competitiveness of the LMFBR.

The material which follows in individual elements, task areas and tasks describes in more detail the specific requirements of the program. It should be borne in mind that in the course of performing this development work, the results which are obtained could cause significant changes in the program from that which is presented in this document. Sufficient flexibility should be retained so that as events which are now unforeseen suggest alternative development routes, these alternative routes are quickly recognized and adopted.

SECTION II

ELEMENT 2 - PLANT DESIGN

The succeeding material on Plant Design R&D includes the identification of new design information needed in order for the testing and development program to proceed, and in order for the Title I and detailed design of the reference concept identified in the Task II and III work to commence. Only those tasks of the Plant Design Element from the Program Plan are included which are necessary to meet these goals. Tasks which can be deferred until the Title I or detailed design phase are not shown. Illustrative of deferred tasks is 2-1.5 Plant Design Study of Vented and Defective Fuel. Other tasks, such as 2-1.6 Refueling System Study have been included in other elements with a changed scope. (The refueling study is included as Task 3-7.2 of the Components Element.)

Much of the engineering for deferred tasks is "first-of-a-kind" engineering, and will be so identified during the Title I and detailed design phases. Some of the deferred engineering will result in the identification of development and testing programs. Where possible, these testing programs have been anticipated and included in other program elements. Later engineering work on the 1000 MWe reference design, or the Demonstration Plant development program may show that some of the testing programs included in this document will need to be modified.

As with the other program elements in this report, the priorities for performing the work will change with time. The most urgent near-term needs are for the performance of the Probabilistic Safety Analysis Study (Task 2-1.4), and the Pool Primary System Study (Task 2-1.2). Both of these studies will result in the identification of long-term development and testing programs which are difficult to project in advance of the studies. The Probabilistic

Safety Analysis Study is fundamental to the success of the reference concept and has long term implications concerning the viability of the LMFBR (as will be discussed). The Pool Primary System Study is important because of the complex structural design, system design, safety analysis and construction inter-relationships. This particular work on the pool primary system concept and the development and testing program which will result from it tend to be strongly a function of the specific arrangement details, and the design and safety criteria established by the individual reactor designer. For that reason, it would be difficult for someone other than the designer of the reference reactor to perform the definitive study which evaluates the alternative arrangement possibilities and defines the R&D.

Table 2-1 shows the milestones for the Plant Design Element, and Table 2-2 shows the program outline by task. Also shown in Table 2-1 are the schedule for the detail design, construction, and start up of the 1000 MWe LMFBR. The detail design is covered in the Overall Element.

2.1 TASK 2-1.2—POOL PRIMARY SYSTEM STUDY

2.1.1 Definition and Current Technology

This task is defined as the conceptual and preliminary design of the vessel complex which contains the primary pumps, intermediate heat exchangers, reactor, and refueling equipment. The primary sodium tank in the reference design is a 48-foot diameter, open-top stainless steel tank which contains approximately 3 million pounds of sodium. Its upper rim is reinforced and provides a support point

TABLE 2-1

PLANT DESIGN MILESTONES

	Government Fiscal Year												
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
Task 2-1.2 Pool Primary System Study		△1	△2	△3									
Task 2-1.4 Probabilistic Safety Analysis Study	△1	△2	△3										
Task 2-1.7 Dynamic Analysis and Control Study		△1	△2	△3	△4					△5			
Task 2-1.8 Containment Study			△1		△2								
Task 2-1.11 Preliminary Plant Design					△1		△2						
Detail Design of 1000 MWe Plant							△1				△2		△3
Construction and Startup										△1			△2

MILESTONESTask 2-1.2 Pool Primary System Study

- △1 Commence Pool Primary System Study.
- △2 Conceptual design of alternative pool arrangements completed.
- △3 Final design completed, R&D program defined.

Task 2-1.4 Probabilistic Safety Analysis Study

- △1 Commence Probabilistic Safety Analysis Study.
- △2 Preliminary survey and development of probabilistic methods completed.
- △3 Probabilistic safety analysis of 1000 MWe reference design completed; future R&D defined.

TABLE 2-1 (Continued)

Task 2-1. 7 Dynamic Analysis and Control Study

- △1 Commence development of computer program for study of control systems.
- △2 Apply computer program to different control arrangements.
- △3 Complete Dynamic Analysis and Control Study.
- △4 Initiate a program for life testing and reliability data acquisition for key subsystems and instruments.
- △5 Complete life testing and reliability data acquisition.

Task 2-1. 8 Containment Study

- △1 Commence Containment Study.
- △2 Complete Containment Study.

Task 2-1. 11 Preliminary Plant Design

- △1 Commence preliminary plant design of 1000 MWe LMFBR.
 - △2 Complete preliminary design and specifications for 1000 MWe plant.
-

Detail Design of 1000 MWe Plant

- △1 Commence detail design of plant.
- △2 Complete detail design of plant, commence engineering follow of work.
- △3 Complete engineering follow.

Construction of 1000 MWe Plant

- △1 Start construction.
- △2 Load fuel and startup.

TABLE 2-2

PLANT DESIGN PROGRAM OUTLINE

<u>Task</u>	<u>Title</u>	<u>Facility Requirements</u>	<u>Costs (1000 \$)</u>		<u>Manpower (Man Years)</u>	<u>Schedule Fiscal Year</u>
			<u>Equipment and Facilities</u>	<u>Operating or Engineering</u>		
2-1.2	Pool Primary System Study	None	None	1200	25	1970 to 1972
2-1.4.1	Probabalistic Safety Analysis Study	None	None	1000	24	1969 to 1972
2-1.4.2	Life Testing and Reliability Data Acquisition*	Lab, Shop and Loops	400	800	20	1972 to 1978
2-1.7	Dynamic Analysis and Control Study	None	None	800	16	1970 to 1972
2-1.8	Containment Study	None	None	500	13	1971 to 1973
2-1.11	Preliminary Plant Design	None	None	1500	30	1974 to 1975

* This program is scoped on the assumption that a relatively limited number of small tests will be run. If the approach is accepted and successful, this program might be expanded and extended to the Component, Instrumentation and Control, Core Design elements, etc.

for the weight of the tank. This weight reaction is taken on a ring of tension studs connecting to support points near the top of the shield deck. Horizontal alignment and seismic load are taken by guide members at the level of the center of the tank. These guide members are arranged to permit radial thermal expansion of the tank, but no net horizontal movement of its vertical centerline.

Outside of the primary sodium tanks is a carbon-steel secondary or containment tank which is insulated on the inside and cooled by nitrogen on the outside. This containment tank is supported from and sealed to the upper shield deck.

The reactor vessel, primary sodium pumps and intermediate heat exchangers of the primary system are supported from the upper shield deck. Since the primary system tank is an open-top vessel, the thermal expansion of the vessel relative to the fixed location of the components poses no problem. However, in order to prevent sodium condensation on cool component surfaces, and to prevent leakage of fission product gases, a system of static seals (O-rings, and multiple plates), and a continuous in-flow of clean argon is provided at the upper tank surface. This is described in detail in the Task II and Task III Reports.

Flow between the reactor vessel and the pumps and intermediate heat exchangers is effected by interconnecting pipes. These pipes expand from thermal effects, and this expansion is accommodated by a series of sliding seals which allow some sodium leakage (see Task II and Task III Reports).

The reactor vessel and some of the interconnecting pipes in the primary tanks contain sodium at 1150°F while the surrounding primary tank sodium is at 800°F. This 350°F temperature drop would impose excessive thermal stresses across the steel boundary. Therefore, the reference design includes an insulating argon gas blanket between two steel layers. At sliding

joints, some of the gas is allowed to leak out into the pool sodium.

Alternatives to the reference primary system arrangement discussed above are included in the Task III Report. The current state-of-art is covered in that discussion. From that discussion and work, the following problem areas are identified:

- Pipe expansion in primary vessel
- Thermal stress in structural boundaries between hot and cold sodium
- Cover gas control
- Vessel construction and support
- Vessel inspection
- Maldistribution of flow
- Thermal transients
- Containment of energy release
- Leakage of equipment seals
- Equipment maintenance

Although the reference design treats these problems, additional work should be performed to examine possible alternatives to the reference concept, and to perform more detailed analyses of the proposed solutions.

2.1.2 New Information Needed

Several alternatives to the reference design primary system are discussed in the Task III Report. These alternatives need to be explored in more detail to determine whether the optimal configuration has been obtained. For example, the sliding seals on the reference design cannot be maintained, except with great difficulty. Means for providing replaceable seals should be examined.

Also, the piping seals allow some leakage of argon into the pool sodium. Studies of the ability to separate this gas from the sodium by means of baffles, low velocity regions, etc. should be performed. Alternatives to the gas insulating barrier should be considered in more detail, such as multiple steel barriers with stagnant layers of sodium. In short, each of the problem areas mentioned in subsection 2.1.1 should be reviewed to determine whether more attractive conceptual means could be found for solving them.

One of the problem areas mentioned above received very little attention during the 1000 MWe LMFBF Follow-On Study. This was equipment maintenance. Although plans were described for removing pumps and intermediate heat exchangers into a cask for maintenance, and the refueling cell is provided for removal and minor maintenance of core components, the details of how this would be done were not studied. Since the viability of the reactor system will depend upon these maintenance procedures, they need to be developed, and their influence on the primary system arrangement needs to be better understood.

Two of the problem areas mentioned in subsection 2.1.1 require new information from other tasks in the Plant Design Element. These are: "thermal transients," and "containment of energy release." The thermal transient problem is related to the need for a set of design criteria for the primary system components. Since no system transient studies were performed on the 1000 MWe Follow-On work, the rate of change of temperature as a function of load change or system malfunction is unknown. Some of this information can be obtained from Task 2-1.4 which will consider transients as they affect the system safety. More detailed information can be obtained from Task 2-1.7 which will evaluate various control systems and modes. The need for information concerned with containment of energy release is related to a definition of loads imposed upon the primary system components or tank by

accident conditions. This information should be supplied from Task 2-1.4, the Safety Analysis Study. Ultimately, a more detailed evaluation of primary system containment adequacy should be performed as a part of Task 2-1.8.

Following the conceptual design effort to define alternative solutions to the problem areas, a more detailed design of the primary system tank and components is needed. This detailed design should include a determination of the methods used to fabricate the tank and assemble the primary system components. Specifically, this more detailed design should include the following:

- Analysis of stress distribution in vessel and supports which result from:
 - Weight of tank and contents
 - Horizontal and vertical seismic loads
 - Temperature distribution: steady state and transient
- Determination of best method of support in horizontal and vertical directions
- Development of construction methods including the construction of the outer tank, concrete shielding and steel liner
- Development of criteria and methods for tank inspection after construction and after operation
- Definition of fuel handling system in sufficient detail to permit tank and component mounting design to proceed
- Re-estimate of design and construction costs

Finally, there is a need for more specific R&D testing definition in order that the pool primary system development for large reactor systems may proceed. Although some of this testing and analysis may be performed on the Demonstration Plant, work conducted to date

on the 1000 MWe LMFBR, and compared with conceptual design work for the Demonstration Plant indicated that in going from a 350 MWe facility based upon current technology to a 1000 MWe facility based upon 1975-1980 technology, many of the features of the smaller design may be different from those of the larger design. For example, the Demonstration Plant may use a center-located reactor in the primary system vessel, while the 1000 MWe design uses an off-center reactor arrangement. The center design simplifies structural analysis, coolant flow patterns in the vessel, and reduces the containment problem during large energy releases. On the other hand, the off-center design reduces the primary tank size and minimizes the refueling cell interference with equipment arrangement in the vessel. The off-center arrangement will undoubtedly require flow mock-ups of scaled vessel arrangements. Other features such as the gas blanket-leakage system of the larger design will need testing. The testing itself will be conducted under Task area 3-9 of the Component Element, but a better definition of what should be tested must await the design studies of Task 2-1.2.

2.1.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of Task 2-1.2 follow:

- To determine alternative conceptual design solutions to the problem areas identified during the 1000 MWe LMFBR Follow-On Work (and listed under subsection 2.1.1.).
- To perform a more detailed design of the primary system arrangement and of the vessel design in conjunction with a vessel fabricator.
- To identify construction processes and steps by which the vessels might be fabricated and the components installed. To determine engineering and construction costs.

- To identify a specific development and testing program to be carried out under Program Element 3, "Components."

The probability of achieving the objectives of this task are high. However, the probability of achieving a primary system arrangement which includes the specific features of the reference concept is estimated to be medium or low. Fortunately, other features are possible, and it is the objective of this task to explore them. For example, further analysis (and later testing) may demonstrate that the non-replaceable sliding leaky seals are not reliable for long term operation. Alternatives to these seals include bellows expansion joints, replaceable seal joints, braided stainless steel pipe, or smaller more numerous pipes with expansion loops. Further analyses, design studies and later testing of some of these alternatives should make the probability for achieving a viable solution to the piping thermal expansion problem high. Similar situations exist for the other problem areas. Therefore, the achievement of a high probability for developing the pool primary system concept is dependent upon an adequate exploration (in Task 2-1.2) of a alternative solutions to the problems previously mentioned, and upon properly defined (in Task 2-1.2) and carried out tests (in Program Element 3) for demonstrating the characteristics of proposed components.

If the development steps and program for the pool primary system are not carried out then it would be necessary to base the design upon current technology. This would require a more conservative approach and would increase costs. For example, if the development of a high temperature insulating barrier between the hot and cold sodium in the primary system (gas insulator in the reference design) were deleted, it might be necessary to reduce the temperature difference between these two regions in order to reduce the magnitude of the thermal stresses. The Task III Report indicates that these stresses would approach 60,000 psi for the 340°F

design temperature difference. EBR-II has acceptable thermal stresses in their pool design arrangement with a 200°F ΔT through an insulating barrier of multiple layers of steel and static sodium. Assuming that this represents the current state-of-the-art and that it would be necessary to reduce the temperature difference in the reference LMFBR from 350°F to 200°F to correspond to present technology, a cost penalty can be determined. Extrapolation from work conducted during Task III shows that this penalty might be between 0.1 and 0.3 mill/kW-h. Similar penalties can be assigned for a center mounted reactor vessel, a stainless steel primary tank head (instead of the open top arrangement), and elimination of special piping joints. These several changes would increase the diameter and construction problems of the large primary sodium tank, and would increase the complexity of the refueling system. No detailed cost trade-offs were conducted during Task III to determine these particular penalties. However, analogous studies on the arrangement cost differences associated with varying the number of loops demonstrated cost differences of from 10 to 12 million dollars (> 0.1 mill/kW-h) for changes in the primary and secondary system arrangements. Assuming that the primary system cost variances for the causes noted above are of the same magnitude or greater, the total penalties for a design based solely on present technology might be from 0.2 to 0.5 mill/kW-h.

2.2 TASK 2-1.4—PROBABILISTIC SAFETY ANALYSIS STUDY

2.2.1 Definition and Current Technology

This task is defined as the study of probabilistic methods to ascertain the utility of these methods for determining the safety characteristics of a 1000 MWe LMFBR. Application of the methods will be made on the Task II reference concept.

To demonstrate the present state-of-art with respect to safety, a brief review of the Task I, III and V safety efforts is presented below.

In the course of carrying out studies on the Task I effort two contrasting approaches to LMFBR safety were evaluated. The consequences of these two approaches, which lead to two different reactor designs, were considered in terms of economic effects, safety effects, and development requirements. From these some conclusions were drawn about the type of safety approach that we believe should be pursued for the LMFBR, and about the corresponding general characteristics of the reactor. These conclusions, if implemented, would have significant consequences for certain aspects of the LMFBR development program.

The first approach to LMFBR safety is essentially that assumed in previous conceptual designs of LMFBR plants, and actually applied in the design of Fermi, EBR-II, and SEFOR. In the Task I Report, it was defined as a "conservative" approach, not in the sense that it is more cautious, but simple because it follows past procedures. A distinguishing feature of this approach is that it assumed that a core disassembly accident (usually caused by meltdown, or possibly also by sodium voiding in some core designs) cannot be prevented, and containment features are provided to accommodate it. In the conservative approach, the use of probabilistic considerations is usually avoided, if possible. Where they cannot be avoided, they appear implicitly in the form of judgments which are made generally without any formal analysis of probabilities.

It is clear, qualitatively, that the magnitude of the energy release in a core disassembly is dependent on the core design. In particular, the dependence on Doppler coefficient has been estimated.⁽³⁾ A dependence on sodium void reactivity is expected because of the possible

contribution of sodium voiding to reactivity insertion rates, either independently or in conjunction with reactivity changes due to fuel movement. Therefore, modifications of core design to increase the Doppler coefficient and to decrease the sodium void reactivity should, in principle, reduce the magnitude and/or probability of core disassembly accidents. Previous conceptual designs have made significant compromises in core design in order to achieve this result and, in addition, have made provision for containing energy releases up to 1000 MW-sec of available work.

Although these provisions certainly would seem to increase safety, there was no way of determining the degree of improvement or of assessing whether the improvement was worth the cost. It should also be realized that given sufficiently pessimistic assumptions no absolute assurance exists that the core designs proposed are not capable of calculated energy releases exceeding the capability of the proposed containments. Consequently the degree to which the core design is compromised, and the design basis accident for the containment, are arbitrary to a considerable extent. With this in mind, the following criteria were assumed to define a "conservative" core design:

- Complete voiding of the core and blankets shall result in a zero or negative reactivity effect.
- The Doppler $T \frac{dk}{dT}$ with sodium voided from the core and blanket shall be -0.004 or greater (in magnitude).
- Containment shall be furnished for a core disassembly accident. In the absence of detailed studies, 2000 MW-sec of available work was assumed to be representative of the amount of energy to be contained.

The first of these criteria will avert large sodium reactivity effects as long as voiding

occurs at different times in different parts of the core. This should be the case in many circumstances that might be postulated. The second and third criteria are based on the aforementioned calculations of the effect of the Doppler coefficient on energy release.

The second approach to LMFBR safety is to make extensive and explicit use of probability analysis to optimize the plant design under the constraint of achieving a defined, acceptable level of safety. It appears likely that one could assure by design that severe core disassembly accidents have such low probability that containment need not be provided for them. It then follows that there is no point in making compromises in core design to reduce energy releases. The core can be optimized for nuclear performance with design attention concentrated on the measures necessary to reduce the consequences of the more probable minor accidents and to provide an acceptable degree of assurance that neither gross sodium voiding nor core melting will occur. This approach will require not only use of the techniques of probability analysis, but also higher reliability requirements on components and systems, and a more elaborate instrumentation system than does the conservative approach. It also implies a need for substantial change in currently accepted procedures and attitudes. The assumed result of this second approach is termed the "advanced" design.

The conclusions that were drawn from the Task I study are summarized as follows:

- The fuel cost incentive for a low-leakage core design is large, being at least 0.2 mill/kW-h, and possibly 1 mill/kW-h or higher. The economics of a conservative core are much more vulnerable to uncertainties in nuclear data and unit fabrication costs and to possible revisions in void reactivity criterion.

- In the more probable accidents the advanced and conservative cores are equally safe.
- The only safety drawback of the advanced design is its large sodium void reactivity effect, which becomes important only if gross sodium voiding occurs. It is expected that systematic probability analyses will eventually show that accidents involving gross sodium voiding can be made to have very low probability, so that severe compromises in the design to reduce the void reactivity effect are not merited (given achievement of reasonable development goals).
- A strong incentive therefore exists to support the choice of an advanced core design. It should be possible to make the safety of the advanced design comparable or better than that of the conservative design for equivalent economics.
- The development and application of probabilistic methods for the safety analysis of fast reactors should be undertaken in the near future. For the advanced design, the need for this is obvious. Even for a conservative design, however, it is impossible to make a well-founded choice of core safety design criteria or of a containment design without the use of a probabilistic approach.

The Task III and Task V work further confirmed these conclusions. In effect, this latter work again showed that safety evaluations must be made more quantitative (that is, safety must be "measured" in some sense) and this can only be done by application of probability theory. The probabilistic approach recognizes that all accidents which are not contrary to the laws of nature have a finite probability of occurring and are therefore "credible." Application of sound engineering principles should result in the more serious of these accidents having an acceptably low probability

of occurrence. For all accidents, the probability of occurrence may be lessened, or the consequences of the accident decreased, by addition of suitable safety instrumentation or by system design changes.

To implement a quantitative, probabilistic safety approach, the following must be accomplished:

- Establishment of a risk criterion, in terms of an overall damage* probability which will be considered acceptable for a plant.
- Evaluation of the risks associated with all accidents that contribute significantly to the overall plant risk.
- Variation of the plant design and reiteration in the above step until the overall risk criterion is satisfied in an economically optimum design.
- For those key subsystems or instruments which have a strong effect on the analysis, reliability data acquisition and life testing should begin where possible.

It is sometimes assumed that application of a quantitative, or probabilistic, approach to safety must await the availability of good data on equipment failure rates. This is not the case, since the techniques of probability analysis and decision analysis are actually the only logical methods of dealing with uncertainties, either in failure rate data or in the knowledge of physical processes in accidents. It should also be noted that, in fact, failure rates are never perfectly known, and the use of a single number, rather than a probability density distribution, to characterize a failure rate, is a simplification which is only valid if the uncertainty in the failure rate being considered is

* Damage is meant in a broad sense to include intangibles that may not be susceptible to economic evaluation.

insignificant in its effect on the analysis. Only analytical experience and sensitivity analyses can determine when this condition is satisfied.

It must, therefore, be recognized at the beginning that the probability of any degree of damage occurring with any particular frequency is itself uncertain, and the overall risk from a plant is also uncertain. The degree of uncertainty will depend directly on the uncertainties in the input data, such as the equipment failure rates; both these inputs and the answers will be in the form of probability density distributions, and a lack of data on a particular failure rate will reflect itself in a broad distribution with a poorly defined peak, as indicated in Figure 2-1.

Even with no data, it is seldom true that one knows nothing about the failure rate. For example, if Figure 2-1 shows the failure rate for a pump, the "no data" distribution is certainly defensible. If the failure rate is greater than one per year, the plant will hardly operate, and the pump would be quickly modified or replaced; on the other hand, there would probably

be general agreement that a failure rate of less than once every hundred years has negligible probability.

Since the risk level of the plant is always uncertain, one must assess the probability that the risk exceeds the criterion. (In effect, probabilities of probabilities are involved.) The probability that the risk exceeds the criterion must be kept below the number that may also be set as a criterion. The use of the "no data" distribution would reflect itself in a greater uncertainty in the risk and in a higher probability that the criteria were exceeded. More expense would result in order to achieve an acceptable design, and this expense could be weighed against the cost associated with the R&D necessary to obtain increased data.

A major conclusion, therefore, which results from a consideration of alternative safety analysis methods is that because of the lack of quantitative data upon which to base the design of certain safety features in an LMFBR, the probabilistic approach is the only method

FIGURE 2-1. ILLUSTRATIVE PROBABILITY DISTRIBUTION WITH EXTENSIVE DATA AND WITH VERY LITTLE DATA

which can be utilized to determine the most urgent R&D needs. The R&D needs can be determined by sensitivity studies which examine the ability of a total system to meet a given safety criterion with different assumptions on individual component reliability. If the system is relatively insensitive to varying assumptions on individual component reliability or failure probability, then the need for additional data is low; if the sensitivity is high, then additional R&D is needed. Other methods of safety approach which assume equal probability of failure for all accidents, or which depend upon qualitative intuitive judgments, or arbitrary accident definitions are implicitly open-ended in terms of research and development since there is no logical way of fixing priorities.

2.2.2 New Information Needed

New information is needed from the Safety Program Element (No. 10), as well as from Task 2-1.4. Following is a list of items needed for a thoroughgoing application of the quantitative approach to reactor safety:

- a. Methodology for analyzing system reliability and the probabilities for undesired events.
- b. Criteria to establish whether a given probability for a particular accident is sufficiently low to be acceptable, and whether the total accident risk for a plant is acceptable.
- c. Adequate components and systems, as needed to meet the requirements deduced for:
 - . Reliability
 - . Response characteristics
 - . Possible preclusion of some accidents through design.

- d. Sufficient data on reliability to support the calculations of accident probabilities.
- e. Sufficient knowledge of the physical processes involved in accidents to support the calculations of accident probabilities and resulting damage.

Item a, the methodology, is essentially available as standard practice in the aerospace industry, although additional code development will be necessary to adapt these methods to reactor safety. Item c, adequate components and systems, will require development particularly to achieve the desired reliability levels. An immediate need is to perform further analysis that will provide the basis for preliminary reliability goals for components, subsystems, and systems. This is necessary to determine in each case where additional development is needed and where the desired reliabilities should be achieved through redundancy in already available components. Item d is the data needed to show that the reliability goals are indeed satisfied. Again, work is needed to define exactly what data is needed. Some of the information should be generated through testing, but system data under operating conditions will be available only after prototype operation. It will, therefore, not be possible to thoroughly exploit the probabilistic approach to safety for early plants.

Item e applies particularly to acquiring additional knowledge of fuel failure modes, possible sodium vapor pressures, resistance of the core lattice to damage, and the possible damage corresponding to different energy releases during core disassembly. Item b is the most difficult but the most necessary, of these requirements. The concept of an acceptable level of risk is now accepted by regulatory bodies, industry, and the public. It must, however, be quantitatively defined in terms that allow its application to a reactor design.

It is obvious that the non-technical aspects of this will make it a difficult thing to achieve. What may not be as obvious is that even from a technical standpoint the definition of a level of risk in such a way that it can be compared to a computed characteristic of a plant design is a problem requiring careful study.

2.2.3 Task Objectives and Assessment of Effect of Non-Achievement

Plant safety analyses on the Task II 1000 MWe LMFBR design will be performed from the point of view of establishing the probability that given levels of damage will be exceeded for various accidents. Concurrently, the problems involved in defining the acceptable risk level for an individual plant will be studied, and alternative possibilities will be evaluated. Procedures for combining computed damage probabilities for different accidents and comparing them to an overall criterion will be studied, as will the problem of allowing for chains of events not explicitly considered in analyses. To whatever extent possible, uncertainties in the computed probabilities will be considered. The objectives of the work for this task are as follows:

- To demonstrate the potential utility of the probabilistic approach to safety, even in the face of uncertainties in failure rate data and in the understanding of physical events during an accident.
- To make recommendations of preferred methods, criteria, and approaches, if possible.
- To make recommendations for required further work, if necessary.
- To identify development needs uniquely associated with implementing the probabilistic approach, such as needs for new

or improved computer codes and analytical methods, and needs for reliability data.

- To place the other areas of safety development identified in the program plan in the framework of a probabilistic approach, and to identify corresponding high priority work.
- Based on the results of the above, to initiate a program for acquiring reliability data and life testing of key subsystems and instruments which have a strong effect on the analysis, and which can be economically tested. Such items as control drives, safety instrumentation, isolation valves, control switches, relays, and circuits are examples of the type of subsystems and instruments which might be tested.

The probability of meeting the foregoing objectives is high. The probability that the quantitative or probabilistic approach to reactor safety can be effectively utilized to assist in licensing fast reactors is also estimated to be high providing that the steps indicated above and in Element 6, Safety, are taken.

The effect of non-achievement of the objectives of this task and/or non-achievement of the work related to establishing probabilistic methods for dealing with reactor safety would be to continue to rely upon procedures now extant in the industry. These procedures utilize "engineering judgment" to identify possible accidents, design criteria and other arbitrary data and definitions for achieving some unknown degree of safety. Because of the arbitrary nature of these procedures, they could result in the requirement for design compromises in core performance, as well as other "passive" safeguard features in the plant. The Task I work demonstrated that this approach

could result in economic penalties in excess of 1 mill/kW-h. This magnitude of penalty appears to be sufficient to make the LMFBR non-competitive with other forms of power expected to be available in the 1980's.

2.3 TASK 2-1.7—DYNAMIC ANALYSIS AND CONTROL STUDY

2.3.1 Definition and Current Technology

This task is defined as the conceptual design of a plant control system for the Task II 1000 MWe LMFBR, the development of methods for evaluating the control system, and the evaluation of alternative control arrangements.

Current technology for control of liquid metal cooled power plants is represented by the Enrico Fermi, EBR-II and Hallam Power Plants. The control arrangement described in the Task II Report advances the technology somewhat from these earlier plants. The reference concept described in the Task II Report has a plant control system which utilizes no large sodium control valves, and exercises temperature control by sodium pump speed control, steam bypass of the reheater, and attemperation of the superheated steam with feedwater. Task III work demonstrated that temperature control in the sodium and steam-water circuits was feasible by this method, and could be improved upon with increased cost by the addition of feedwater preheating. A primary goal in this work was to maintain the reactor inlet and outlet sodium temperatures approximately constant throughout a control range of 20% to 100% load. Other goals included the ability to change load at the following rates:

<u>Range</u>	<u>Load Change Rate</u>
100% to 50%	10%/minute
50% to 20%	5%/minute
20% to 0%	Depends on reason for such action

The control system which was conceptually designed to meet these goals included provision

for automatic control over the range of 20% to 100% load.

The methods used during Tasks II and III to determine the feasibility of controlling by the proposed scheme involved steady state analysis of partial load operation. No transient studies were conducted. Moreover, operation from 0 to 20% load was not evaluated. No consideration was given to alternative control schemes in which sodium flow control valves are used. No analysis was made of the number of temperature cycles, or rate of change of temperature change at different points in the system; hence the ability of components to withstand these cycles was not determined.

Although specific goals were established in Task II and III to minimize the thermal stress problems introduced by load changes in the plant, it is not clear that these goals are the most appropriate ones. For example, it was assumed that it would be desirable to hold constant core outlet temperature with changing load in order to minimize thermal stress of components. From the point of view of core structural components, such as fuel bundle channels, this is probably true. However, from the point of view of the fuel cladding, it is not necessarily true. As the load changes, the temperature of the fuel also changes. If the coolant inlet and outlet temperatures are maintained approximately constant (by flow control), then the cladding temperature also remains approximately constant (modified slightly by the ΔT across the cladding as heat flux changes). Since the cladding temperature does not change, but the fuel temperature does, thermal expansion and contraction is imposed upon the fuel as the load changes, but not the cladding. If the fuel is near the end-of-life, it may have been expanded from irradiation effects so that the cladding is in the strained condition at full power. By changing load, then, the thermal expansion and contraction of the fuel imposes a cyclic strain on the cladding, which may lead to fuel failure. If, on the

other hand, the core outlet coolant temperatures were allowed to vary so that as the fuel temperature dropped, the cladding temperature also dropped, the cyclic strains on the cladding could be reduced. Unfortunately, all fuel pins in the core do not behave similarly, so it would not be possible to remove all cyclic strain from all fuel cladding. Moreover, if the coolant outlet temperature varied with load, cyclic thermal stresses would be imposed on other structural components.

In summary, the present state-of-the-art is such that for a high performance plant like the reference LMFBFR, the inter-relationships between various possible control modes and the structural integrity of plant and core components is not well defined. These inter-relationships are important in order to define the type of control system which should be developed, and to determine design requirements for the various components.

2.3.2 New Information Needed

The new information needed for this task includes both methods development and analytical work. Computer programs should be developed which allow the transient characteristics of the total system to be determined for different control arrangements, different control modes and different system arrangements. These programs may be digital, analog or hybrid.

Following development of the necessary computer programs, they should be applied to the reference design to determine its transient characteristics. Information needed from this study includes:

- Rate of change of temperature as a function of load at all points in the system where a boundary occurs prior to an interchange of energy.
- Stability of the system under transient conditions.

- Relationship between pump speed control accuracy or valve position accuracy, and system temperature response.
- Identification of possible component problem areas resulting from system transients.
- Determination of the effect of rate of change of load, and of alternate control ranges on component performance or reliability.
- The above items for alternative control modes and arrangements from the reference arrangement.

2.3.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task follow:

- To prepare a computer program which will allow the simulation of the reference control arrangement and alternatives thereto as discussed in subsection 2.3.2.
- To apply the developed program to the reference and alternative control arrangements and thereby determine their characteristics.
- To select an attractive control arrangement from the above and utilize it to identify troublesome thermal stress areas in components; to evaluate these troublesome stress areas and design to accommodate them or modify the control arrangement to reduce the causative factors.
- To define specific R&D needs for the control system and/or R&D needs for components subjected to transient thermal stresses.

The probability of achieving the objectives of this task is high. The probability of designing an acceptable control arrangement is also

high. This control arrangement may deviate from the reference concept, however, at some added expense. For example, the Task III work demonstrated that if it were necessary to include additional feedwater heating in order to limit the rate of change of temperature in certain components, approximately 1 million dollars would be added to the plant cost (~ 0.02 mill/kW-h). Larger penalties might accrue from limitations on fuel or component performance which result from the inability to reduce cyclic thermal stresses. For example, it might be necessary to limit the rate or the range of load changes, or alternatively to reduce fuel burnup for a given control performance requirement. If it were necessary to reduce fuel burnup from 100,000 MWd/Te to 80,000 MWd/Te, the energy costs would be increased by 0.06 to 0.1 mill/kW-h.

2.4 TASK 2-1.8—CONTAINMENT STUDY

2.4.1 Definition and Current Technology

This task is defined as the study of energy release-containment requirement relationships. The reference design described in the Task II Report does not include a large steel dome as part of the containment system. This does not mean that containment is not included on the reference LMFBR; numerous containment barriers are provided. These barriers to the uncontrolled release of radioactivity include the following:

- Fuel matrix and cladding
- Sodium
- Primary system tank and piping
- Inert gas blankets and seals
- Secondary tank
- Refueling cell

- Concrete building substructure and steel liner
- Metal panel building superstructure
- Building ventilation and filtering system
- Stack

In Task 2-1.4 of this document a discussion is given of the basic safety philosophy which evolved from the Task I, II, III and V work. Essentially, this philosophy is based upon a quantitative or probabilistic approach to safety. All safety features, whether "passive" or "active" are treated quantitatively. If a passive feature, such as a containment barrier, shows a lower probability for exceeding a given safety criterion under all accident conditions than an active safeguard feature, and they both have equivalent cost, then the passive feature is included; if not, then the active feature is provided. If one feature has lower cost than the other, and they both meet the specified safety criterion, then the lower cost feature is used.

In the Task II and III work, it was assumed that a complete quantitative analysis of the safety aspects of the reference design would show that the steel dome type of containment was unnecessary, and that the containment barriers noted above together with instrumentation and other active safeguards systems would meet an acceptable safety criterion. It was recognized, however, that much work would have to be performed to demonstrate this point (as discussed in Task 2-1.4). Very little work was performed during Tasks II, III and V to quantitatively determine the adequacy of the barriers noted previously to absorb energy release and/or contain radioactivity for abnormal conditions. It was assumed that these barriers could be designed to meet the requirements imposed upon them by a complete analysis of the system; this needs to be verified.

Much of the safety work which is needed for a basic understanding of the system will be performed under Task 2-1.4. Another area which will contribute to a better understanding of the containment adequacy is Task 2-1.7, the Dynamic Analysis and Control Study. This particular task will provide information on thermal transient conditions at various points in the primary system. This will allow an analysis of the transient thermal stresses in key structural components, such as the reactor and primary system vessels. Task 2-1.2 will generate more detailed information on the primary system structure. This detailed information will allow a more complete review of the ability of the primary system to act as a containment barrier.

2.4.2 New Information Needed

Information is needed from Tasks 2-1.2, 2-1.4, and 2-1.7, as discussed in these tasks (and as discussed above) before work in Task 2-1.8 can commence. These earlier tasks will provide basic information which will permit an analysis of the structural adequacy of the various component barriers under different conditions. In addition to these tasks, information is needed from Element 6, Core Design. Task 6-1.1 provides for the updating of the reference 1000 MWe core design based upon additional test data on irradiated materials. Since the core structure is also one of the containment barriers, this updated information is needed to allow proper analysis. Task 6-1.1 and 2-1.8 can proceed simultaneously if conducted by the same designer.

In addition to the background information needed from other tasks, the following information or tools are needed:

- Means for evaluating work-energy relationships on specific containment barriers.

- Cost sensitivity of the core structure to the rate and magnitude of temperature changes.
- The adequacy of the core structure and primary system to absorb some level of energy release.
- Sensitivity of the primary system to alternative arrangements for given accident conditions.

2.4.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task are as follows:

- a. Collect information from Tasks 2-1.2, 2-1.4, 2-1.7 and 6-1.1 pertinent to the design of the containment barriers described in subsection 2.4.1.
- b. From Task 2-1.4, select certain high and low probability accidents and establish a tentative containment design criterion.
- c. Determine alternative conceptual design methods of accommodating energy release suggested by (b).
- d. Determine the applicability of the Argonne code on energy release-containment work relationships to the requirements of (b) and (c). If not applicable, define needs for additional work.
- e. Determine sensitivity of alternative structural layouts to accidents defined in (b).
- f. Identify needed testing or development program.

The probability of achieving all of the objectives is estimated to be medium. Most

of the objectives have a high probability of success but the fifth objective, related to determining the sensitivity of alternative structural layouts is estimated to be low for two reasons: (1) the Argonne energy release-containment work relationships code may not be applicable to the specific reference design, and (2) any work-energy model must be related to a specific testing program of scaled structures in order to build confidence in the model. Failure to achieve the objectives listed above would not cause any significant programmatic penalties providing that objective number (f) is met and implemented. Failure to achieve this last goal could seriously affect Safety, Element 10, since it seems likely that the development program will involve modeling and testing of the reference containment barriers for the purpose of developing empirical relationships in an analytical model which simulates absorption of energy release. Such modeling is important to the quantitative understanding of various phenomenon related to safety. As pointed out in Task 2-1.4, failure to implement

the quantitative approach to reactor safety could lead to significant compromises in core performance (in order to meet arbitrary safety goals) with economic penalties of 1 mill/kW-h or more.

2.5 TASK 2-1.11—PRELIMINARY PLANT DESIGN

2.5.1 Definition, Current Technology, and New Information Needed

This task is defined as the preliminary design of a 1000 MWe LMFBR. Current technology is represented by the Task II conceptual design of the 1000 MWe LMFBR.

The new information needed is a complete plant design in sufficient detail to permit the solicitation of component and construction quotations for the fabrication of equipment and the construction of the plant. This preliminary design should include updated information from other elements and tasks including:

<u>Element</u>		<u>Task or Sub-Task</u>
Element 2, Plant Design	2-1.2	Pool Primary System Study
	2-1.4	Probabilistic Safety Analysis Study
	2-1.7	Dynamic Analysis and Control Study
	2-1.8	Containment Study
Element 6, Core Design	6-1.1.1	Updated Oxide Conceptual Design
	6-1.1.3	Oxide Core Preliminary Design
Element 3, Components	3-6.2	Steam-Generator Scope Design
	3-7.2	Conceptual Design of Components Handling Equipment
	3-9.1	Vessel Design Technology
	3-9.4	Vessel Fabrication Technology
	3-10.2	Preliminary Design of Control Rod Drives
	3-11	Liquid Metal Pumps
	3-12.2	IHX Scope Design

Task 6-1.1.3 Oxide Core Preliminary Design should be done simultaneously and as an integrated part of Task 2-1.11.

The schedule for this task (Table 2-1) is planned so as to allow accumulation of information from the other tasks, and to allow completion of this updated preliminary design of the complete plant with integration of equipment and component information in time for the detail design of the plant to commence. Prior to detail design, the preliminary design task should generate updated cost estimates in order to allow preparation of proposal information.

2.5.2 Task Objectives and Assessment of Effect of Non-Achievement

The objective of this task is to perform a preliminary design of the complete 1000 MWe LMFBR in sufficient detail to allow preparation of a proposal for the detail design and construction of the plant.

The probability of achieving the objective is high. Failure to achieve the objective would result in the inability to properly assess the risks associated with the construction of the plant. This would make the cost ceiling of the plant limitless and the schedule for construction could not be fixed.

SECTION III

ELEMENT 3 - COMPONENTS

The succeeding material on the components R&D defines the new information needed for the development of components similar to those described in the Task II and Task III reports. The program is nearly complete in providing for conceptual and preliminary design, proof testing, and development of fabrication technology. Detail design of the final plant components is not included (except for schedule requirements) since this is provided in Element 1, Overall.

This particular element has a strong interrelationship with the Demonstration Plant development program. In some cases, components or information are assumed to be developed on the Demonstration Plant program which will adequately meet the needs of both the Demonstration Plant and the 1000 MWe LMFBR. In most instances, additional information needs or component development are required to extend the technological base beyond that of the Demonstration Plant. Both situations are covered in the following material, and the informational base established by the Demonstration Plant program is described.

Because of the interrelationship of this Element with the Demonstration Plant program and with other elements such as the Plant Design, Sodium Technology, and Fuels and Materials, the schedule has been planned in some detail to be compatible with these various efforts. For example, a majority of the test work is assumed to be carried out in facilities at General Electric. The development effort for this element requires the use of a large water-flow loop, a sodium-water reaction test rig, and a 2 to 3 MWt steam-generator test rig. It is anticipated that similar facilities may be used in carrying out the Demonstration Plant development

program. If such equipment and facilities are provided for the Demonstration Plant program, then it may be possible to use them, with suitable modifications, for the 1000 MWe development. The schedule is planned to be compatible with this possibility.

As with other elements in this report, the priorities for performing the work will change with time. Because of the large body of new information needed, and because of the long development, design, and fabrication times required, the task areas with the highest present priority are Task Area 3-6, Steam Generator and Task Area 3-9, Reactor Vessel and Related Components. Task Area 3-9 includes as a major informational need work performed under Task 2-1.2, Pool Primary System Study. This work could start immediately. Except for materials test data procured from the Sodium Technology Element, work on Task Area 3-6 is not scheduled to commence until the middle of fiscal year 1971 in order to take advantage of information which may be developed on the Demonstration Plant program. Another high priority effort in this Element is that of Task Area 3-11, Liquid Metal Pumps.

Other work which could commence at an early date but with a lower priority than Task Areas 3-6, 3-9, and 3-11 is Task 3-7.2, Conceptual Design of Components Handling Equipment. This work could commence in fiscal year 1970 and would be of benefit to effort on Task 2-1.2, Pool Primary System Study in the Plant Design Element.

Table 3-1 shows the milestone schedule for work performed in this Element. The programmatic needs are shown in Table 3-2. More detailed schedules and discussion are included in the individual tasks.

TABLE 3-1
COMPONENTS MILESTONE SCHEDULE

	Government Fiscal Year										
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Task 3-2.2 Development of Standards				1	2		3		4		
Task Area 3-6 Steam Generators		1 2			3	4				5	
Task Area 3-7 Components Handling Equipment	1		2			3		4	5		6
Task Area 3-8 Coolant Transport	1		2		3	4					
Task Area 3-9 Reactor Vessels	1		2 3		4				5		
Task Area 3-10 Control Rod Drives				1	2	3	4	5			6
Task Area 3-11 Liquid Metal Pumps	1		2		3						
Task Area 3-12 Intermediate Heat Exchangers		1	2	3	4	5	6				

Milestones

Task 3-2.2 Development of Standards

- 1 Complete review of material from Tasks 2-1.2 and 2-1.4 for incorporation in Standards.
- 2 Distribute preliminary version of vessel code for industry comment.
- 3 Publish updated preliminary version of vessel code based on industry comments.
- 4 Publish final version of vessel code.

Task Area 3-6 Steam Generators

- 1 Initiate preliminary scope design.
- 2 Initiate fabrication technology development.
- 3 Commence model tests.
- 4 Complete final scope design and fabrication technology development, commence detail design.
- 5 Complete model tests of steam generator.

Task Area 3-7 Components Handling Equipment

- 1 Commence conceptual design.
- 2 Fabricate critical portions of component handling equipment; commence tests.
- 3 Complete bench tests of critical portions, commence detail design and fabrication.
- 4 Commence proof testing of full-scale equipment.
- 5 Modify test equipment or fabricate new equipment.
- 6 Ship equipment to plant.

TABLE 3-1 (Continued)Task Area 3-8 Coolant Transport and Storage Systems

- ① Initiate fabrication of prototype valves for 1000 MWe LMFBR service.
- ② Commence testing of prototype valves; evaluate pipe expansion methods from Task 2-1.2.
- ③ Write preliminary test report and specifications for valves.
- ④ Complete development of piping expansion devices.

Task Area 3-9 Reactor Vessels and Related Components

- ① Start Task 2-1.2, Pool Primary System Study.
- ② Complete Task 2-1.2; commence development of new analytical design methods; commence fabrication technology development.
- ③ Commence design and fabrication of mock-up sections of vessels.
- ④ Commence flow tests and component stress-strain tests.
- ⑤ Complete components tests, fabrication technology development, and preliminary specification.

Task Area 3-10 Reactivity Control Rod Drives

- ① Commence preliminary design of modified Demonstration Plant drives and design of Task II drives.
- ② Commence fabrication of critical portions of drives.
- ③ Commence testing of critical portions of drives.
- ④ Complete design of full-scale prototype drives; commence fabrication.
- ⑤ Commence testing of prototype drives.
- ⑥ Complete testing of prototype drives.

Task Area 3-11 Liquid Metal Pumps

- ① Detail design of LMFBR prototype pump completed.
- ② Prototype pump fabrication completed; seal and bearing development completed.
- ③ Prototype pump testing completed.

Task Area 3-12 Intermediate Heat Exchangers

- ① Commence preliminary scope design.
- ② Commence fabrication of prototypes and critical parts.
- ③ Commence insulation development.
- ④ Complete fabrication of prototypes; commence testing.
- ⑤ Insulation development completed.
- ⑥ Prototype model tests completed.

TABLE 3-2

COMPONENTS PROGRAM OUTLINE

Task	Title	Facility Requirements	Cost (\$1000)		Manpower (Man Years)	Schedule Fiscal Year
			Equipment and Facilities	Operating or Engineering		
Task Area 3-6, Steam Generator.						
3-6.2	Design					
	1. Preliminary Scope	None	None	250	7	1971 to 1972
	2. Prototypic Model Designs	None	None	600	16	1972 to 1973
	3. Final Scope and Specifications	None	None	350	10	1975
3-6.3	Fabrication Technology					
	1. Tube Joint Fabrication	Lab-Shop	None	400	9	1972 to 1973
	2. Prototype Model Fabrication					
	a. 2 to 3 MWt	Lab-Shop	None	900	18	1974 to 1975
	b. 15 to 30 MWt	Lab-Shop	None	1,500	28	1973 to 1974
3-6.4	Materials and Small Components Test					
	1. Materials Tests	In Sodium Technology Element				1971 to 1974
	2. Fluid Flow and Heat Transfer	Large water flow loop and steam generator test rig	300*	1,000	8	1972 to 1975
	3. Na/H ₂ O Reactions	Na/H ₂ O leak rig	200*	1,000	8	1975 to 1978
3-6.5	Model Tests					
	1. 2 to 3 MWt Model Test	Steam Generator Test Rig	500*	1,500	12	1975 to 1979
	2. 15 to 30 MWt Model Test	SCTI (LMEC)	400*	3,000	20	1976 to 1980
Task Area 3-7, Components Handling Equipment						
3-7.2	Conceptual Design					
	1. Fuel Handling	None	None	100	3	1970 to 1971
	2. Fuel Shipping	None	None	100	3	1971 to 1972
	3. Component Handling	None	None	70	2	1972
	4. Maintenance Equipment	None	None	200	5	1972 to 1973

* Allowance for modification of equipment or fabrication of new equipment.

TABLE 3-2 (Continued)

Task	Title	Facility Requirements	Cost (\$1000)		Manpower (Man Years)	Schedule Fiscal Year
			Equipment and Facilities	Operating or Engineering		
3-7.3	Bearings, Seals and Mechanisms	LMEC	On-going program at LMEC			1970 to 1975
3-7.5	Development and Proof Testing					
	1. Fuel Handling and Shipping	SEFOR hot-cell mock-up or equivalent	800*	300	6	1977 to 1978
	2. Component Handling	SEFOR mock-up or equivalent	1000	300	6	1977 to 1978
	3. Maintenance Equipment	SEFOR hot-cell mock-up or equivalent	400	400	8	1978 to 1979
Task Area 3-9, Reactor Vessels and Related Components						
3-9.1 and 3-9.2	Design Technology					
	1. Review of Existing Analytical Methods	None	None	100	2	1973
	2. Generate New Analytical Methods	None	None	400	11	1973 to 1977
	3. Design and Fabricate Mock-ups and Perform Tests	Water loop, sodium loop, and strain equipment	400**	900	15	1973 to 1977
3-9.4	Fabrication Technology					
	1. Define Construction Techniques	Fabricator shop	None	400	6	1973 to 1976
	2. Develop Quality Control Methods	Lab-Shop	50**	150	3	1973 to 1975
	3. Generate Preliminary Specifications for Field Fabrication	None	None	100	2	1966 to 1977
	4. Develop Inspection Methods	Lab-Shop	100**	200	4	1975 to 1976
Task Area 3-10, Reactivity Control Rod Drives						
3-10.2	Preliminary and Detail Design					
	1. Adaptation of Demonstration Plant Drives	None	None	200	6	1973 to 1974
	2. 1000 MWe Task II Reference	None	None	300	8	1973 to 1974
	3. Full Size Prototypes	None	None	500	12	1974 to 1975

* Allowance for possible modification of SEFOR Hot Cell Mock-Up or construction of new facility

** Allowance for purchase of new equipment or modification of possible existing equipment in Facilities.

TABLE 3-2 (Continued)

Task	Title	Facility Requirements	Cost (\$1000)		Manpower (Man Years)	Schedule Fiscal Year
			Equipment and Facilities	Operating or Engineering		
3-10.4	Fabrication of Prototypes					
	1. Modified Demonstration Plant Drives	Shop	None	200	2	1976
	2. Task II Reference	Shop	None	500	4	1976 to 1977
3-10.5	Proof Testing of Prototypes					
	1. Modified Demonstration Plant Drives	Control rod test stand and support equipment	300*	300	3	1977 to 1978
	2. Task II Reference			400	4	1977 to 1979
Task Area 3-12, Intermediate Heat Exchangers						
3-12.2	Design					
	1. Preliminary Scope	None	None	250	5	1971 to 1972
	2. Prototype Designs	None	None	350	9	1972 to 1973
	3. Final Scope and Specifications	None	None	350	10	1975 to 1976
3-12.3	Fabrication Technology					
	1. Critical Parts	Lab Shop	None	300	6	1972 to 1974
	2. Prototypes					
	a. 2 to 3 MWt Prototype	Lab Shop	None	600	14	1973 to 1974
	b. Flow-Test Model	Lab Shop	None	400	10	1972 to 1974
3-12.4	Model Tests					
	1. Water Flow Test	Water flow loop		600	7	1975 to 1976
	2. 2 to 3 MWt Model	Sodium loop	400*	800	9	1975 to 1976
3-12.5	Materials, Insulation, and Maintenance					
	1. Materials Tests	In Sodium Technology Element				1970 to 1974
	2. Insulation Fabrication and Test	Lab Shop	None	200	4	1973 to 1975
	3. Maintenance Tests	In Task Area 3-7				1975 to 1976

* Allowance for equipment modification or construction of new equipment.

3.1 TASK 3-2.2—DEVELOPMENT OF STANDARDS

The objective of this task, as defined in Task Area 3-2 of the Components Element of the Program Plan, is: "to prepare RDT standard for use in the LMFBR component development program." LMEC and ORNL have been designated as having the principal responsibility in this area and there are on-going programs at ORNL and LMEC to carry out the objective of this task.

The task is mentioned here because there is a particular need to develop codes and standards for LMFBR primary system vessels and reactor vessels which provide for minimum standards of safety and which meet the unique requirements of the LMFBR. For example, current pressure vessel design is guided by the requirements of ASME Section 3 of the Boiler and Pressure Vessel Code for Class A Vessels. This code is oriented towards the low temperature, high pressure, high thermal inertia water reactor systems. It is sensitive to the needs of these systems in such areas as nil-ductility failure, large section design, etc. Requirements include hydrostatic testing of finished vessels and energy-impact failure tests of portions of the material used in fabrication. In some instances, direct application of this code to the LMFBR vessels could result in less safe vessels than if no code requirements were followed. For this reason, a new code is required (as differentiated from a revision of the existing code) which recognizes the unique characteristics of the LMFBR system. To be most useful, the code should be based upon design work performed under Tasks 2-1.2, Pool Primary System Study and 2-1.4, Probabilistic Safety Analysis Study and performed as a part of the Plant Design Element. These efforts are scheduled to be completed at the end of the fiscal year 1972 (presuming a start-date at beginning of fiscal year 1970 for Tasks 2-1.2 and end of fiscal year 1969 for Task 2-1.4). A copy

of the code should be available for use during the detail design phase of the 1000 MWe LMFFR vessels (scheduled to commence beginning of fiscal year 1977). This leaves a 4-year time period for development of the code, review by industry and professional societies, and publishing of an early version of it.

Since this task is already a part of an on-going program, no additional program is suggested. The implementation of the work as outlined above is recommended, however. In particular, the incorporation of ideas from Tasks 2-1.2 and 2-1.4 is important, as is a thorough review of the suggested code by industry and professional societies.

3.2 TASK AREA 3-6—STEAM GENERATORS

This task area is defined as the design, development, and fabrication technology necessary to put into operation steam generators with operating conditions similar to those conceptually described in the 1000 MWe LMFBR Task II report. These operating conditions include the use of a once-through reheat steam cycle with an operating pressure of approximately 2600 psia and steam temperatures from the reheater and superheater of 950°F. The reference conceptual design shown in the Task II report for the steam generator indicates that the units are of U-tube construction and utilize austenitic stainless steel in the superheater and reheater. But before deciding on the precise type of tubing configuration used and the kind of materials selected, further conceptual design work and materials testing needs to be done. Also, feedback must come from the ESADA development program, which is currently being carried out at General Electric to develop steam generators for the Demonstration Plant.

Although the Demonstration Plant and its supporting development programs may utilize a considerably different steam cycle

and steam generator arrangement than that ultimately selected for the 1000 MWe plant, it is planned that the Demonstration Plant development program for the steam-generator effort will complement that of the 1000 MWe design. For example, the Demonstration Plant steam generators may utilize Croloy for the superheater materials. Development work is being carried out on the ESADA program concerned with the stabilization of carbon in Croloy at the temperatures expected in the Demonstration Plant. The 1000 MWe design will require higher operating temperatures and will, therefore, extend the technology beyond that developed for the Demonstration Plant. However, information obtained in the development program for the Demonstration Plant should permit a better basis for determining the potential usefulness of Croloy at these higher temperatures. Work is included in the following material which extends the investigative area for Croloy and other materials and will enhance the ability to select proper materials for the intended service conditions. Other areas where a technology overlap may occur between the Demonstration Plant and 1000 MWe plants include: fabrication technology, fluid flow and heat transfer, and sodium-water reactions. Programs are provided in the following sections which extend the technology from the Demonstration Plant base to that required for the 1000 MWe design.

Table 3-3 shows a schedule of the various tasks to be performed in this task area. These particular tasks are designed to complement similar work which may be performed for the Demonstration Plant and to build upon the technological base formed by that work. Utilization of the facilities is timed so that as various loops, which may be available from the Demonstration Plant work, are phased out of that program, they can be used for the 1000 MWe steam-generator development program. Some modifications to the loops may be necessary as is shown on the schedule.

Alternatively, if the loops are not available from the Demonstration Plant program because of scheduling or other reasons, new loops would be required. This could increase the cost of facilities and equipment shown in Table 3-2 by approximately one million dollars since the costs shown are based upon modification of existing facilities and equipment.

As noted on the schedule, the detailed design, fabrication, and delivery of the plant steam generators are included. Delivery is planned near the end of calendar year 1977 (early fiscal year 1979), and would allow for a 16-month installation and check-out schedule prior to startup of the plant. Because of the long testing program required for the steam generators, this is one of the critical components for the 1000 MWe plant in terms of the long lead time required between development and delivery.

The current technology for this task area is described in nearly complete detail in Task Area 3-6 of the Components Element of the Program Plan. At the time when the majority of the 1000 MWe steam generator work is expected to begin, the technology should have improved considerably from that indicated in the Program Plan. For example, as discussed above, it is expected that understanding of materials under high temperature conditions, such as Croloy and stabilized Croloy, will have improved to the point where application of these materials with greater confidence will be possible. Additionally, the fabrication technology for particular configurations required for the Demonstration Plant may have progressed to the point where large components can be fabricated with the expectation of a reliable service life. Model tests may have been performed to demonstrate this reliability on full-scale sections of the Demonstration Plant steam generators. Finally, the fluid flow, heat transfer, and sodium chemistry problems under Demonstration Plant conditions may be well understood.

TABLE 3-3

SCHEDULE FOR TASK AREA 3-6—STEAM GENERATOR

Task	Government Fiscal Year									
	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
3-6.2 Design										
Preliminary Scope										
Model Designs										
Final Specification										
3-6.3 Fabrication Technology and Quality Control										
Tube Joint Fabrication										
Model Fabrication										
2 to 3 MWt										
15 to 30 MWt										
Quality Control Methods										
3-6.4 Materials and Small Components Tests										
Material Tests										
Tube Supports; Flow and Vibration Tests										
Na/H ₂ O Reactions										
3-6.5 Model Tests										
2 to 3 MWe Model Tests										
15 to 30 MWe Model Tests (S.C.T.I. or Equivalent)										
Demonstration Reactor Steam Generator Tests and Surveillance*										
Facility Construction or Modifications										
Steam Generator Test Rig										
Water Loop										
Na/H ₂ O Rig										
Plant Steam Generators										
Place Order and Detail Design										
Fabricate										
Deliver and Install										
Commission										

* May be carried out as part of the Demonstration Plant program.

Operation

This understanding could include a knowledge of the system characteristics during a postulated tube failure such that detection of the early phases of this failure are possible and accommodation of the results of the reaction products can be made.

3.2.1 Task 3-6.2 Design

3.2.1.1 Definition, Functional Requirements, and Current Technology

This task is defined as the preliminary scope design of the 1000 MWe LMFBR steam generators, the design of prototype test units, and the final scope design and specification of the plant steam generators. The preliminary scope design will include conceptual design studies to select the reference steam-generator type and materials; it will also define the plant operating conditions. The prototype model designs will be based upon the preliminary scope designs and will include multiple-segment tube sections of the reference steam generator. Detailed design of these prototype sections will be carried out so that fabrication will be possible under Task 3-6.3. It is anticipated that a 2 or 3 MW thermal prototype will be designed for basic heat transfer studies, and a 15 to 30 MW thermal unit will be designed for possible use in the SCTI. The final scope design and specifications will be prepared for the purpose of soliciting bids for the procurement of the 1000 MWe steam generators.

The functional requirements for these various designs will include the following information:

- Full-load sodium temperature and steam temperature and pressure conditions,
- Required load factor,
- Required rate of change of load and anticipated system conditions as a function of this load change,

- Frequency and severity of system-imposed transient extremes (e.g., loss of load), and
- Frequency of scheduled refueling and down-time for other plant components.

Data for these functional requirements will be obtained from different sources as the program progresses. During the preliminary scope design work, the other program elements will not have produced sufficient definitive information to permit sole dependence upon them. Therefore, the information will be generated from the current 1000 MWe LMFBR Follow-On Study and possibly from the Demonstration Plant steam-generator development work with some assistance from early Plant Design Element work. Some modification to these functional requirements will be necessary in the prototype design work and in the final design scope and specifications. These latter design efforts will obtain updated information on functional requirements from work performed under the Plant Design Element. In particular, Task 2-1.4, Probabilistic Safety Analysis Study, and Task 2-1.7, Dynamic Analysis and Control Study, will provide necessary information.

The current technology for this task has been discussed at the beginning of subsection 3.2.

3.2.1.2 New Information Needed

New information needed for this task includes the comparison of alternate design concepts which will most properly satisfy the system conditions described in the 1000 MWe LMFBR Task II report and will be compatible with knowledge which may be obtained on the Demonstration Plant development program. Based upon this comparison of the advantages and disadvantages of various design approaches, information is needed to permit the specific testing program to proceed for development of the selected steam generator. Finally,

information is needed from the testing program to allow a final scope design and specifications to be written so the detailed design and fabrication of the steam generator may be done.

3.2.1.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task follow:

- Prepare a preliminary scope design based upon a consideration of alternative design possibilities. Functional requirements for this preliminary scope design shall be based upon the 1000 MWe LMFBR Task II report, upon work which may be accomplished for the Demonstration Plant, and upon early information obtained from the Plant Design Element.
- Prepare prototype designs of sections of the reference design. One of these sections shall be designed for a thermal rating of 2 or 3 MW and the other shall be a multiple tube segment designed for a thermal rating of from 15 to 30 MW.
- Prepare a final scope design and specification for the 1000 MWe steam generator suitable for soliciting bids for procurement of the plant steam generators.

The probability of achieving the objectives of this task are high. It is possible, however, that the functional requirements defined early during Task 3-6.2 may have to be altered to less stringent conditions based upon information obtained during the testing program. If it were necessary to reduce the steam temperature from the 950°F design goal shown in the Task II report to 850°F, the economic penalty is estimated to be in excess of 0.05 mill/kW-h.

3.2.2 Task 3-6.3 Fabrication Technology

3.2.2.1 Definition, Functional Requirements, and Current Technology

This task is defined as the development of the technology necessary to fabricate critical

parts and prototype sections of the reference steam generator selected in Task 3-6.2.

This task requires that the fabrication techniques developed meet certain quality control standards so reliable manufacture of the components can be assured. The specific quality control standards will be developed as a part of the task.

The current technology for this task has been discussed in general terms under the task area. At the time when the work for this task is begun, it is anticipated that manufacturing capability may have been developed to the extent required to manufacture reliable steam-generator components for the Demonstration Plant. Many of the techniques developed during the Demonstration Plant program would then be applicable to the 1000 MWe program. However, it is anticipated that the once-through design of the 1000 MWe reference will require modification of components. Moreover, the specific tube arrangement and attachment to the tube sheets may be different than in the Demonstration Plant. Therefore, applied development work to generate the proper procedures for fabrication of these units will be needed.

3.2.2.2 New Information Needed

Since the technology required for the 1000 MWe design will be of an applied nature and will be quite specific to the selected concept, it is anticipated that the major requirements for new information will be for the development of techniques for fabricating these specific components; therefore, it is anticipated that two major efforts will be needed to supply the desired information. One effort will consist of the development of techniques for producing reliable tube-to-tube sheet joints for the reference design. Information will be needed on the jigs, fixtures, and welding techniques necessary to assure that repeatability of quality welds is

possible for the specific design selected. Modifications to the design may be necessary in order to achieve the desired quality under repetitive conditions. Testing of the completed joints by thermally cycling and stressing them in and out of sodium will be required to develop information concerning the reliability of the joints.

In addition to the technique for fabricating tube joints, information is needed on methods for fabricating complete components. This information can best be obtained by the fabrication of prototypical segments. One segment will be designed for use in accumulating basic heat-transfer information and will have only limited usefulness in generating fabrication technology. The other will be a larger section and will include portions of the shell and nozzles and other features which may prove troublesome in manufacture. This latter section will be capable of transferring from 15 to 30 MWt of heat.

In addition to information concerning the fabrication of the component parts of the steam generator, information is needed concerning quality control techniques which will assure the fabrication of these components to a desired minimum standard of quality.

3.2.2.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task follow:

- Develop techniques for the fabrication of tube joints for the reference steam-generator design. Alter the design, if necessary, as these techniques are developed.
- Develop techniques for the fabrication of other critical portions of the steam generator. These techniques will be developed by fabricating mock-up sections of the full scale steam generators,

including a multi-tube segment prototype model. One prototype segment will have the capability of transferring 2 to 3 MWt of heat and the other will have the capability of transferring 15 to 30 MWt of heat. Both units will use full-size tubes.

- Develop quality control procedures which will assure the achievement of a standard of quality required by the final specifications for the steam generator.

The probability of achieving the objectives of this task are high. Failure to meet the objectives might require that a completely different type of steam generator be developed, such as the double-walled tube units used in EBR I and II, SRE and HNPF. Such units would be considerably more costly than the single-walled reference of the type shown in the Task II report. No comparison of double-walled units with single walled units was made during the 1000 MWe Follow-On Study, so precise estimates of cost penalties are not given. It is entirely conceivable, however, that the cost of the exchangers could be doubled for given system conditions if double-walled units were used. Studies conducted during Task III showed that if steam-generator costs were doubled, the cost of energy would increase by 0.26 mill/kW-h. Alternative to the development of a new type of steam generator, failure to achieve the objectives of this task might force dependence on a less reliable steam generator which would cause plant outages. Unscheduled outages could cost the utility from 1 to 2 mills/kW-h during the period of the outage.

3.2.3 Task 3-6.4 Materials and Small Components Tests

3.2.3.1 Definition, Functional Requirements, and Current Technology

This task is defined as the development of technology by specialized tests of materials

and components to form a base of information concerning materials compatibility, fluid flow, heat transfer, and sodium-water reactions. There are three functional requirements for this task: the selected materials must be capable of meeting the defined criteria established in Task 3-6.2, the fluid flow and heat transfer characteristics of the heat exchangers must be satisfactory for operations under the conditions and over the load range specified in Task 3-6.2, and a leak in a tube must not result in catastrophic destruction of the heat exchanger.

The current technology of this task has been discussed in general terms under the task area. As with Tasks 3-6.2 and 3-6.3, each of the test areas described under this task (and illustrated in Table 3-3) will build upon technology which may be developed for the Demonstration Plant. However, because of the more severe conditions for the 1000 MWe design and because of the specific design requirements of the particular heat-exchanger configuration selected, additional development is required. For example, the Demonstration Plant may utilize a recirculating evaporator. In a recirculating evaporator, the problem of significant changes in the mode of heat transfer, as occurs in a once-through boiler, is not encountered. In the once-through boiler, during the boiling process, the fluid changes from nucleate-boiling heat transfer to film boiling. This change occurs at the departure from nucleate-boiling (DNB) point. At either side of the DNB point, the heat transfer coefficients are significantly different, and this results in temperature changes in the tubing material. The wall temperature may fluctuate at the DNB point even at steady-load conditions. Moreover, deposits tend to accumulate at this point. As load is changed in the steam generator, the DNB point changes location, which causes additional cyclic temperature changes. These phenomena need to be studied further to determine the long-term fatigue life to these temperature transients and to determine possible

deterioration of performance or corrosion effects from the deposits.

3.2.3.2 New Information Needed

It is possible that the steam generator for the Demonstration Plant will use Croloy or stabilized Croloy materials. The suitability of these materials at higher temperatures than used in the Demonstration Plant needs to be examined as well as the alternative of 300-series stainless steels. In the case of stainless steel, additional information needs to be obtained concerning stress corrosion at the temperatures and environmental conditions expected in the 1000 MWe steam generators. Interaction of tube supports on tubes must be investigated for material compatibility, fretting, friction forces, and degree of support. Work on materials development will be carried out under the sodium technology element.

As discussed under current technology, new information is needed concerning the heat transfer mechanisms for once-through steam generators. This information can be obtained from single-tube test sections which are representative of the tubes of the reference steam generator. Basic data for those heat transfer regions which are identified as having potential for improved thermal performance are needed. Areas of potential problems should also be studied. These areas include both the nucleate boiling region and the film boiling regions as well as the DNB point between the two. On the shell side of the heat exchanger, non-uniform flow distribution can reduce the effectiveness of the steam generator at locations where the velocity is low. To overcome this problem, it is customary to resort to baffles with partial crossflow. Unfortunately such baffles can introduce tube vibrations and fretting wear at the support points. At part-load operation, even with the baffles, flow instability is possible because of the reduced mass flow rates,

which cause unequal flow distribution. This is also true in the tubes at very low flow rates. Information is needed concerning the flow characteristics and stability of the fluids as a function of load for the specific steam-generator configurations being developed. It was mentioned earlier that tube vibration might be a problem by reason of improper baffle design. Another source of tubing vibration is the high velocity flow of a two-phase fluid inside the tubes. Of major importance to the solution of the tube vibration problem is the design of a tube support system which can supply the required damping without producing wear problems by fretting corrossions. Information is needed on the characteristics of the specific component being developed.

Of importance to the design of the steam generator is the ability to detect small leaks before major damage occurs. If the damage does proceed to the point where major tube failures result, then it is equally important to be able to vent the reaction products without destruction of the steam generator or supporting equipment. To this end, information is needed for the specific configuration under development on the threshold of detectability for a sodium-water reaction. Instruments with the potential for detection of sodium-water reactions need to be tested with small leaks at different locations in mock-ups of the steam generators to determine the sensitivity of the instruments to the system design. Major leaks need to be studied for the design configuration to determine whether relief devices function properly in order to prevent destruction of the components.

3.2.3.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task follow:

- Perform materials tests under the Sodium Technology Element designed to supply information concerning suitable materials which are compatible with the sodium and steam-water conditions defined in the Task II report of the 1000 MWe Follow-On Study. Determine interaction of tube supports and tubes to determine their compatibility in sodium.
- Perform fluid flow and heat transfer tests to meet the informational needs of subsection 3.2.3.2 described above and in accordance with the functional requirements defined in Task 3-6.2.
- Perform sodium-water reaction tests which meet the informational needs of subsection 3.2.3.2 described above and in accordance with the functional requirements defined in Task 3-6.2.

The probability of achieving the objectives of this task is medium. It is not certain at this time that all of the functional requirements described in the Task II report for the reference steam generator can be met. If, for example, the high temperature system requirements result in a reduced lifetime of the unit, it might be necessary to reduce the design temperature shown in the Task II report. If it were necessary to lower the sodium temperature by 100°F, the energy costs would increase by approximately 0.1 mill/kW-h.

3.2.4 Task 3-6.5 Model Tests

3.2.4.1 Definition, Functional Requirements, and Current Technology

This task is defined as the testing of mock-up portions of the reference steam generators to determine the suitability of the selected design for the performance and functional requirements defined in Task 3-6.2. These functional requirements will include the complete range of operating characteristics for the steam generator.

The current technology was described in general terms under the task area discussion. At the time that these model tests are to be performed it is anticipated that a technological base for performing the tests may have been formed on the Demonstration Plant program and under Tasks 3-6.2, 3-6.3, and 3-6.4, which will permit the performance of specific tests directed towards proving the reliability of the selected concept.

3.2.4.2 New Information Needed

New information is needed on multiple tube segments to verify the steady-state thermal performance of the units. Information is also needed to determine the long-term endurance capability of the unit to withstand multiple thermal cycles. The duration of these tests should be on the order of 10,000 hours and the number of cycles should be approximately twice the number expected to occur during the lifetime of the units installed in the 1000 MWe plant.

In addition to the normal operating thermal performance tests, information is needed on the ability of the components to withstand severe thermal shocks. These thermal shocks can result from rapid changes in load or from reactor scram. Small, high-frequency shocks from the DNB region can also be important to the design of the steam generator. This information should be accumulated in a series of tests.

Since it is anticipated that the Demonstration Plant steam generators will be operational before the final design of the 1000 MWe steam generators are completed, information from the operation of these steam generators could be useful in this design effort. Information is needed on the operating characteristics of the steam generators and on the comparison of these characteristics against the design performance predictions. Instrumentation may be included

on one of the steam generators in the Demonstration Plant which would permit accumulation of this information. This is particularly true for specific components which may closely duplicate the components used in the 1000 MWe design, such as the superheater units.

3.2.4.3 Task Objectives and Assessment of Effect of Non-Achievement

There are three objectives to this task.

- Perform proof-performance tests to meet the informational needs of subsection 3.2.4.2 in a prototype model of the reference steam generator which is capable of transferring approximately 2 to 3 MW thermal of heat.
- Perform additional tests on a large multiple-tube prototype model of the steam generator with the capability of transferring from 15 to 30 MW thermal of heat.
- Perform tests on the Demonstration Plant steam generator and record the performance characteristics of this steam generator so the data may be used to assist in the final design of the 1000 MWe steam generator. This latter objective is included in this task because of its importance to the program. Actual implementation and funding of the work has not been included in the 1000 MWe development program.

The probability of achieving the objectives of this task are high. Failure to meet the objectives of this task would cause steam generators to be designed without prototypical test information. This could result in reduction of the reliability of the final units. No studies on the 1000 MWe Follow-On Program were made to determine to what extent the lack of such tests might affect the reliability

of the completed units; thus, no quantitative estimates of the energy cost penalties can be made. They could be quite high, however, since the plant could be out of service for long periods of time as a result of inoperative steam generators. The cost of unscheduled outages has been discussed under Task 3-6.3.

3.3 TASK AREA 3-7—COMPONENTS HANDLING EQUIPMENT

This task area is defined as the design, development, and fabrication technology necessary to put into operation fuel handling and shipping equipment, component handling equipment, and maintenance equipment suitable for conditions similar to those conceptually described in the 1000 MWe LMFBR Task II report or for conditions which will be more fully developed in the Plant Design Element and in Task 3-7.2 of this task area.

The Task II report conceptually described a fuel handling system which is based upon a hot cell arrangement. A shield plug with penetrating control rods is lifted within the hot cell after disconnecting control rods so that fuel transfer may occur. Used fuel is transported under sodium to temporary positions around the periphery of the core into transportable storage thimbles. These thimbles are subsequently raised from the reactor, using a natural convection cooling device, to a separate fuel storage pit in the hot cell. Fuel is stored for decay in this storage pit under sodium. An inert atmosphere of argon is maintained in the refueling cell and over the reactor sodium.

Subsequent to a decay period of approximately 1 month, the fuel is removed from the storage pit in its thimble and placed in a liquid hydrocarbon with a different specific gravity than sodium. In this hydrocarbon pool the fuel is removed from the thimble and placed in a shipping container which also is under the hydrocarbon. The container is

ultimately closed and removed for shipment. The outside of the container consists of a finned surface for natural or forced convection cooling by air during shipment.

Other components requiring removal for maintenance include control rods, instrument probes, and reflector elements. These particular components are removed within the hot cell by mechanisms provided for this purpose. Components outside of the hot cell which must be removed for maintenance include intermediate heat exchangers and the primary pumps. These latter components are removed into a container which is sealed to the floor and filled with an inert gas prior to removal of the components. Following the removal, a temporary valve located over the opening where the component was removed is closed, and the transportable container with the component inside is transferred to the maintenance cell. All of the equipment described above is radioactive from both contamination and activation.

Although a conceptual method for performing the component movements as described above was defined in the Task II and III work, relatively little consideration was given to alternatives in sufficient detail to justify a selection of any of these alternatives. Moreover, the reference design selected was not in sufficient depth to justify the conclusion that the state-of-technology had improved significantly from that of SEFOR or EBR-II (this state-of-technology is discussed in some detail in Task Area 3-7 of the Components Element of the Program Plan). For example, although a fuel shipping procedure is discussed above which utilizes a liquid hydrocarbon, the state-of-technology in this area is quite limited. However, it is expected that by the time development and proof testing on the 1000 MWe components handling equipment begins, the technology will have improved considerably by reason of Demonstration Plant development work. This Demonstration Plant

work may include fuel shipment preparation development and fuel cell housekeeping development such as sodium vapor cleanup, contamination removal, and liquid sodium removal. Additionally, the ability to service component handling equipment, such as fuel transfer mechanisms, may be developed during the Demonstration Plant program. The fuel transfer scheme on the Demonstration Plant may be similar to that of the 1000 MWe LMFBR design, and therefore, the development work on this equipment could be useful. However, extension of much of this technology will be necessary for the following reasons:

- Fuel assemblies will have a higher decay heat removal requirement on the 1000 MWe LMFBR.
- The fuel of the 1000 MWe LMFBR will be vented, but the Demonstration Plant first core may not.
- The specific components requiring movement or maintenance will be larger and sometimes of different design than on the Demonstration Plant.

It is possible that some of the facilities provided for the Demonstration Plant component handling development work will be available for the 1000 MWe program. These facilities might consist of a modification of the SEFOR mock-up refueling cell or they may be special facilities constructed which will allow testing of components in an inert gas atmosphere. An allowance is included in the program for possible modification or construction of these facilities.

Table 3-4 shows a schedule of the tasks identified for this task area. Also shown on the schedule are the facility modification work and the detailed design, procurement, shipping and installation schedule for the components which will first be proof-tested, then modified or refabricated and shipped to the plant for

installation. As with other components in the Components Element, funding for the detailed engineering and procurement is not included in this task area but is shown in Element 1, Overall.

3.3.1 Task 3-7.2 Conceptual Design

3.3.1.1 Definition and Functional Requirement

This task is defined as the conceptual design of the following 1000 MWe LMFBR component handling equipment:

- Fuel handling
- Fuel shipping
- Other radioactive component handling
- Radioactive component and equipment maintenance.

The functional requirements for this task are that reference methods be devised for the handling functions described in the Task II and III reports.

3.3.1.2 New Information Needed

New information is needed concerning each of the four component handling equipment items listed in subsection 3.3.1.1. For fuel handling, a comparison should be made of the reference fuel handling concept in the Task II report with other possible schemes for performing the same functions. Analysis of the ability of the proposed scheme to remove 50 kW of heat during the transfer process should be carried out in more detail than was done on the 1000 MWe program. A review of the bearings, seals, and mechanisms development program in progress at the LMEC should be made to ascertain the usefulness of particular devices for incorporation into alternative concepts. From these considerations

TABLE 3-4

SCHEDULE FOR TASK AREA 3-7—COMPONENTS HANDLING EQUIPMENT

Task	Government Fiscal Year										
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
3-7.2 Conceptual Design											
Fuel Handling											
Fuel Shipping											
Component Handling											
Maintenance Equipment											
3-7.3 Bearings, Seals, and Mechanisms*											
3-7.5 Development and Proof Testing											
Critical Portions of Components Handling Equipment											
Full Scale Equipment Tests											
Facility Modification or Construction											
Design											
Modification											
Plant Components											
Detail Engineering** and Procurement or Fabrication											
Fabrication and Testing (Under 3-7.5)											
Modification or New Fabrication											
Shipping and Installation											

*Continuing programs at LMEC

**Carried out in Overall Element

of alternatives and increased detail on the reference concept, a final conceptual design reference should be selected.

The fuel shipping preparation procedure described above should be used as the basis for arrangement studies. A review of the compatibility of fluids for washing and shipping the fuel should be made. Similar work may be carried out on the Demonstration Plant program but this work will be for fuel

having much lower decay heat removal requirements than the 1000 MWe fuel sub-assemblies. The conceptual design of the fuel shipping cask should be completed and a brief review made of the hazards associated with transportation of the cask.

Conceptual design information is needed on specific arrangement possibilities and on estimated costs of equipment for handling various radioactive components both within the hot cell and external to the hot cell.

For the internal heat exchanger (IHX) and primary pumps, the conceptual arrangement of hoists, valves, inert gas purging equipment and transporting mechanisms should be worked out for the casks which will move these components to the maintenance cell.

The 1000 MWe Follow-On Study did not include an evaluation of specific means for maintaining various items of equipment. Typical of equipment which will require maintenance and replacement are the following:

- Fuel shuffling machine for moving fuel within the core,
- Fuel transfer machine for transferring fuel to the storage cell,
- Fuel transfer machine for removing fuel to the shipping area,
- Control drives and rods,
- Instrument probes,
- Shield plug lifting mechanism,
- Intermediate heat exchangers, and
- Primary pumps.

A complete conceptual design study is needed to determine possible methods for maintaining all of this equipment under varying conditions of radioactive contamination and activation. Some information in this regard may be generated as part of the Demonstration Plant program. In particular, a multi-purpose machine with the capability of performing numerous small non-specialized maintenance tasks within the hot cell may be developed for the Demonstration Plant. However, there is need for evaluating the usefulness of such machines as well as conceptually studying others which meet the requirements for the specific components used in the 1000 MWe design.

Two major component handling equipment items which will require design and development testing for the 1000 MWe components are the primary pump and IHX removal casks. The conceptual design effort may demonstrate that a dual purpose cask can provide for maintenance removal of both the pump and the IHX.

Table 3-4 shows a schedule that is planned so the subtasks described above coincide with work being performed under Task 2-1.2 (Pool Primary System Study of the Plant Design Element). To be most useful, Task 3-7.2 should be carried out as a part of the effort conducted under Task 2-1.2. Those changes in the pool concept which result in fuel handling changes would be reflected in alterations to the work being performed under Task 3-7.2 and vice versa.

3.3.1.3 Task Objectives and Assessment of Effect of Non-Achievement

The objective of this task is to perform the conceptual design necessary for the selection of final reference concepts for the following components:

- Fuel handling mechanisms,
- Fuel shipping equipment and facilities,
- Other radioactive component handling equipment, and
- Radioactive component maintenance equipment and facilities.

The probability of achieving the objectives of this task are estimated to be high. Lack of achievement of the objectives within the scheduled time period would result in a delay of the 1000 MWe LMFBR program. No assessment of the economic penalty associated with such a delay was made during the Task III effort, and therefore, a quantitative determination of this penalty is not possible.

3.3.2 Task 3-7.3 Bearings; Seals and Mechanisms

This task is shown in Table 3-4 and is a part of continuing programs at PNL and LMEC. Other work on bearings and seals may be carried out as part of the Demonstration Plant Development Program. The current technology is discussed in Task 3-7.3 of the Components Element of the Program Plan. Also discussed in the same place is the need for new information. It is included here because of its importance to other work carried out in this task area. It is assumed that the work carried out by PNL and LMEC, and possibly the Demonstration Plant, will meet the needs of the 1000 MWe LMFBR. The probability of achieving suitable bearings, seals, and mechanisms for the components required in Task Area 3-7 is estimated to be high. Failure to achieve these objectives would require that additional tests of specific bearings, seals, and mechanisms be conducted with a possible delay in schedule.

3.3.3 Task 3-7.5 Development and Proof Testing

3.3.3.1 Definition and Functional Requirements

This task is defined as the development, fabrication, proof testing, and modification or refabrication of the components defined in Task 3-7.2. Some of these components will be used directly in the 1000 MWe plant, unless the testing program indicates that necessary alterations are of such a nature as to require a new component. Detailed design of the components will not be carried out under this task but will be performed as a part of the prototype engineering required for the 1000 MWe LMFBR plant, and this is included in the Overall Element. In some cases, such as the IHX transfer cask and appurtenances, the complete cask will not be fabricated for testing, but rather selected mechanisms used

on the cask and mock-up sections of the IHX. These mechanisms will be tested, and if successful they will be used in the final cask assembly.

The functional requirements for the development and proof testing will be defined under this task from material generated under Task 3-7.2, and possibly from information developed on the Demonstration Plant components handling development program. These functional requirements will include a consideration of the effect of economic penalties resulting from plant downtime during handling or maintenance of components.

3.3.3.2 New Information Needed

New information is needed concerning the ability of the specific components handling equipment and maintenance equipment to perform their appointed tasks as defined in the functional requirements. Tests need to be carried out under conditions which simulate the environment expected for this equipment. Information is needed on the ability of the components to function repetitively and under different load conditions. Based upon these tests, information is needed to allow minor modifications which would improve the performance of the equipment and these minor modifications need to be incorporated in the equipment. Alternatively, new equipment needs to be manufactured which meets the functional requirements and the design criteria established during the tests.

3.3.3.3 Task Objectives and Assessment of Effect on Non-Achievement

The objectives of this task follow.

- Provide new facilities or modified facilities from the Demonstration Plant Program which are suitable for proof testing of components identified under Task 3-7.2.

- Perform development and proof testing of the components handling and maintenance equipment conceptually defined in Task 3-7.2 and of the detail design under prototype engineering included in the Overall Element.
- Modify components handling and maintenance equipment as determined to be necessary from the performance tests, or provide new equipment if modification is not practical.
- Ship the modified components to the plant for installation.

The probability of achieving the objectives of this task is assessed to be high. Lack of success would require additional development and testing with a consequent delay in the startup of the 1000 MWe LMFBR plant. A quantitative estimate of this delay is not possible since it was not studied as a part of the Task II or Task III effort.

3.4 TASK AREA 3-8—COOLANT TRANSPORT AND STORAGE SYSTEMS

The objective of this task area as defined in the Components Element of the Program Plan is to achieve "the technology to permit design and fabrication of coolant transport and storage systems for commercial LMFBR plants." Two tasks are shown in the Program Plan in this task area which have particular importance to the 1000 MWe LMFBR as defined in the Task II Report. These tasks are Task 3-8.5, Development of Valves and Task 3-8.8, Evaluation of Piping Expansion Devices. Both are continuing programs at LMEC (Task 3-8.5) and ORNL (Task 3-8.8). In addition, work related to Task 3-8.8 is in progress at LMEC and Materials Technology Incorporated.*

*Work in progress on sliding seals at high temperatures. (MTI, 968 Albany Shaker Road, Latham, New York)

The Task II 1000 MWe LMFBR reference concept utilizes no valves in the primary or secondary sodium systems except for a check valve in the primary pump outlet. The check valve is of the Enrico Fermi type of vertical check. Because of the absence of valves, the mechanical design of these major systems is simplified. However, compromises are necessary in the plant control system because of the inability to throttle sodium flow. It is not clear that the improvements in mechanical design justify the increased complexity of the control system. Furthermore, numerous smaller valves are needed in the auxiliary sodium systems. Thus, the continuing program for development of these valves is fully justified. Contracts have been let by LMEC to Aerojet General, Rockwell Manufacturing, and Crane Company for initiating work which is expected to result in the fabrication of large throttling, stop, and check valves. It is planned that these valves will be tested in the LMEC starting in 1973. In addition to this work, contracts have also been let for the development of valves for the Sodium Pump Test Facility (SPTF) at the LMEC and for the Fast Flux Test Facility (FFTF). Development of valves by 1974 or 1975 would be in sufficient time to be incorporated in the 1000 MWe LMFBR final design. A more aggressive schedule would be necessary for the Demonstration Plant.

The Task II reference concept also uses a sliding seal arrangement to accommodate pipe expansion. This particular method of handling thermal expansion may not be the best method; it is scheduled to be studied further under Task 2-1.2, Pool Primary System Study, of the Plant Design Element. Work is currently in progress at LMEC on frictional wear of sliding materials in sodium and on high-temperature sliding seals in sodium at MTI. ORNL is also working on this task. It is also anticipated that specific mechanical joint work may be performed as a part of the Demonstration Plant effort. It is suggested

that all of this work be continued and be directed towards solutions which are developed as a part of Task 2-1.2 (scheduled for completion at the end of fiscal year 1972, presuming start of work at the beginning of fiscal year 1970) or as a part of specific Demonstration Plant requirements.

Both the valve development work and the mechanical joint development efforts should be directed to the specific requirements of particular LMFBR designs. These requirements will be defined in more detail as a part of the following Tasks:

- 2-1.2, Pool Primary System Study,
- 2-1.4, Probabalistic Safety Analysis Study,
- 2-1.7, Dynamic Analysis and Control Study, and
- 2-1.11, Preliminary Plant Design.

Additionally, functional requirements for valves and mechanical joints may be generated for the Demonstration Plant and the FFTF. The continuing development effort should be redirected as necessary to be responsive to these various and specific needs. If further study during Task 2-1.2 indicates that these needs will not be met with continuing programs, then additional development work will be suggested.

3.5 TASK AREA 3-9—REACTOR VESSELS AND RELATED COMPONENTS

This task area is defined as encompassing the technology necessary to design, develop, and fabricate the primary sodium vessel, secondary containment vessel (if required), reactor vessel and ancillary support structure as conceptually defined and described in the Task II and Task III reports of the 1000 MWe Follow-Up Study. The functional requirements for this work

will be described in the individual tasks in this task area. A major requirement in this task area in terms of further design work on the reference pool primary system is satisfied by Task 2-1.2, Pool Primary System Study, of the Plant Design Element. Task 2-1.2 will provide input to Task Area 3-9 in the following important areas:

- Conceptual design alternates to the present primary system arrangement,
- Preliminary design of a specific primary system arrangement and vessel design in conjunction with a vessel fabricator,
- Identification of construction processes and steps, and
- Identification of a specific development and testing program for implementation under Task Area 3-9.

The state of current technology is described in Task Area 3-9 of the Components Program Element from the Program Plan, and in Task 2-1.2 of this document. However, these descriptions do not make reference to the problems of establishing proper coolant distribution in the primary tank or of the problems of cover-gas control.

In the reference design described in the Task II and Task III reports, the primary tank serves as a plenum between the exits of the three intermediate heat exchangers and the entrances to the three primary sodium pumps. The velocities are in general very low and pressure drops small, but an understanding of the flow distribution is important to assure that the assumption of isothermal tank conditions is valid in transients. In particular, under low-flow operation, the buoyant forces in the sodium tend to predominate over the pump pressure forces. In this situation, the temperature in the sodium may stratify with hot sodium at the top and cold

sodium at the bottom. In a low-flow transient where the secondary sodium temperature drops, a stream of cold sodium may slowly flow from the lower ends of the intermediate heat exchangers and form a cold layer at the bottom of the tank. A considerable time will elapse before this cold stratified sodium reaches the inlets of the primary sodium pumps. A further complication results from the temperature changes which may be imposed upon the primary vessel, reactor vessel, intermediate heat exchangers, and primary pumps during load transients. For the very large systems such as the 1000 MWe reference, there are no precedents as to the type of transients to be expected during load changes within the reference control concept.

Another area of concern is the flow condition in regions which are subject to argon gas bubbles, such as might leak past seals in the insulation space around the flow ducts. The flow distribution must not entrain these gas bubbles into the pump inlets. Furthermore, the reference concept has a dynamic gas upper closure at the vessel top. In this concept, no solid structure serves as a cover over the primary vessel. Instead, a series of plates form loose seals around all penetrations, and argon gas is continually bled through these seals to prevent sodium vapor from condensing on cold surfaces.

The coolant distribution in the vessel complex, gas bubble entrainment in the sodium, and the dynamic gas seals all present problems in design which do not lend themselves to purely analytical solutions. Instead, experimental mock-ups to develop empirical solutions based upon model theory are desirable. Additionally, experimental mock-ups are useful in studying stresses and strains in complex mechanical shapes which do not lend themselves to a purely theoretical approach. To this end, the tasks included in this task area will propose model testing where necessary.

These model tests will be similar to those performed for SEFOR for the study of the flow patterns and the vibration potential of various vessel components. For determining flow patterns, it is anticipated that a reduced scale model of the vessel and its components will be used with water as the simulated coolant. For vibration analysis, it is anticipated that a full scale section of the reactor vessel and core components will be needed and water flow tests will be conducted. This latter work will be conducted as a part of the Core Design Element under Tasks 6-2.2 and 6-2.3. Finally, tests of proposed insulation methods to reduce the thermal stresses across stainless-steel barriers separating regions of hot and cold sodium are needed. These later tests will mock-up complex sections such as the sliding seals and gas-blanket insulation of the Task II reference design. The Demonstration Plant Development Program may also perform insulation tests, and this information will be of value to the 1000 MWe LMFBR, but new tests will still be needed to verify the adequacy of the specific arrangement and design conditions of the LMFBR.

Equipment and facilities for the tests performed as part of this task area include the water and sodium loops which may be available from the Demonstration Reactor work. If this equipment is available, some modification may be necessary for the specific requirements of the 1000 MWe program, and an allowance is included in the program for that modification or for the fabrication of new equipment.

The schedule for this work is shown in Table 3-5 and is compatible with Task 2-1.2, Pool Primary System Study, of the Plant Design Element and with the possible availability of facilities from the Demonstration Plant Development Program. It should be noted that although the Demonstration Plant

TABLE 3-5

SCHEDULE FOR TASK AREA 3-9—REACTOR VESSEL AND RELATED COMPONENTS

Task	Government Fiscal Year										
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
2-1.2 Pool Primary System Study*											
3-9.1 and 3-9.2 Design Technology											
Review of Existing Analytical Methods											
Generate New Analytical Methods											
Mock-ups											
Design											
Fabrication											
Flow Test											
Component Stress-Strain Tests											
3-9.4 Fabrication Technology											
Define Construction Techniques											
Quality Control Methods											
Preliminary Fabrication Specification											
Develop Inspection Methods											
Facilities Equipment Modification or Fabrication											
Design											
Procurement and Installation											
Plant Vessels											
Detail Design											
Fabricate and Install											
Shop Work											
Field											

*Performed under Element 2, Plant Design

work may assist in advancing the technology, particularly in the area of vessel fabrication, the informational needs of the 1000 MWe program are quite extensive. For example, as discussed in Task 2-1.2, the Demonstration Plant primary system vessel, component arrangement, and other design features may be considerably different than those of the 1000 MWe design. These differences could

include the method of providing the top closure, the reactor arrangement in the vessel, the flow arrangement and pressure drop requirement, the IHX shell and tube coolant flow arrangement, thermal expansion accommodation method, and component sizes. This will result in differences in construction techniques for the 1000 MWe LMFBR, and most certainly will require detailed study of the flow patterns

for the particular design selected. Moreover, the Demonstration Plant will make use of design techniques which are sufficiently crude that sacrificial steel will be required in certain areas to allow for design uncertainties. In order to design for the higher temperature and more stringent conditions of the 1000 MWe vessel which will result in a commercially attractive design, these methods will need improvement. The tasks which follow are intended to provide the necessary technology upgrading for the 1000 MWe LMFBR.

3.5.1 Tasks 3-9.1 and 3-9.2—Development of Design Technology for Vessels

3.5.1.1 Definition, Functional Requirements, and Current Technology

This task is defined as the generation of analytical and empirical tools for the performance of detailed design of the primary sodium and reactor vessels and appurtenances. The functional requirements are that the tools must be capable of imparting information for the analysis of the structure components for stresses and strains resulting from dynamic and static loadings and from temperature differentials and transients. These tools should also enable the user to predict thermal transients from models of coolant distribution as a function of position in the component and as a function of time during a transient. Additionally, they should permit the prediction of cover and insulating gas behavior as a function of the seal design, based upon model tests. For proper analysis of the reactor vessel and component parts, the design tools should be capable of providing analytical information up to the following goals:

- Vessel lifetime: 35 years,
- Maximum temperature: 1200°F,
- Neutron fluence in structural members: 10^{21} n/cm² (En > 1 MeV),
- Instantaneous temperature drop in non-loading bearing members: 350°F, and
- Ramp temperatures in load bearing members: 200°F in 30 seconds.

In the reference design, described in the Task II report, care was taken to separate load bearing components from the most severe radiation damage and temperature extremes. For example, the entire reactor vessel is maintained at the reactor inlet sodium temperature of 800°F, and the fluence in a 35-year life is considerably below the goal noted above. This is accomplished by utilizing shields (both in the core structure and the reactor vessel) which absorb neutrons, and shield the reactor vessel from high temperatures. The reference design uses a gas blanket between the thermal shield in the vessel and the reactor vessel. For this arrangement, the non-load bearing thermal shield is maintained at 800°F on its lower portion and 1100°F over its upper portion. During scram, the upper sodium temperature drops from 1100°F to 800°F in seconds, and the shield is thus subjected to a severe thermal shock. Although its fluence (10^{20} n/cm² at En > 1 MeV) is also considerably below the design goal noted above, tools are needed to analyze the stresses imposed in the structure as a function of the fluence to determine the number of cycles before failure. Moreover, by reason of the temperature gradient in the shield during normal operation, creep will affect the shape of the structure and the stresses imposed during scram. The reactor vessel itself is exposed to a maximum fluence of 10^{19} n/cm² during its lifetime, and at the upper portion experiences a temperature gradient of about 700°F. This gradient is established over several feet of length. Questions of neutron embrittlement and creep strength need to be analytically resolved for this structure. In the event that the gas insulating barrier is inadvertently lost during operation, the vessel

could be subjected to a severe thermal stress, and the practical consequences of this eventuality need to be analyzed.

The above comments are intended as a brief review of some of the problems of the present conceptual design and to indicate areas for which design tools are needed. Other comments on the current state of technology are made in Section 3.2 of the Components Element of the LMFBR Program Plan. The state-of-art noted in Section 3.2 does not refer to the requirements for model analysis and testing to predict coolant flow behavior, gas seal flow, and gas entrainment and removal in coolant.

This was briefly discussed above under Task Area 3-9. Since a primary objective of Task 2-1.2 in the Plant Design Element is to define alternative conceptual and preliminary design methods of dealing with these problems, as well as the structural analysis of the vessel complex, a more detailed state-of-art description, and a more specific definition of analytical and testing requirements will be supplied by that work.

3.5.1.2 New Information Needed

There is a need for analytical methods for evaluating structures and components under a variety of stress and loading conditions as described above. These methods should be based upon long-term creep data at elevated temperature conditions obtained from the Fuels and Materials Element and from the Sodium Technology Element. Techniques for analysis of structures subjected to cyclic strain or cyclic stress conditions should also be developed based upon test data supplied from Fuels and Materials and Sodium Technology. In general, information is needed for a more precise and reliable analysis of reactor vessels and structures as described in subtask 3-9.1.2 of the Components Element of the Program Plan.

Additional information needed includes flow models for analytically predicting temperature and gas bubble distribution in the primary system as a function of alternate operating characteristics and design conditions. Also needed are methods for predicting gas distribution and flow rates through various seals. In these latter cases dealing with coolant and gas distribution and flow, experimental mock-ups of the designs selected in Task 2-1.2 of the Plant Design Element are needed. Model theory can be used in conjunction with these mock-ups to provide the necessary design tools. Experimental mock-ups are also needed of certain complex shapes identified within Task 2-1.2, which do not lend themselves to a purely analytical approach. These mock-ups are necessary to obtain information on strains (and stresses) and failure data under various simulated operating conditions.

3.5.1.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task follow.

- Review existing test data and analytical tools available for analysis of structures of the type described by the reference 1000 MWe design. Based upon this review, recommend additional testing to extend the data base.
- Utilize the recommendations for R&D from Task 2-1.2 (Pool Primary System Study) of the Plant Design Element; generate specific analytical tools which will meet the requirements of the defined systems in Task 2-1.2, based on test data from the Fuels and Materials and Sodium Technology.
- Where purely analytical tools are not practical, perform experimental mock-ups of specific components and design conditions to verify the adequacy of the

design performed in Task 2-1.2. Where necessary, recommend changes to the design. The experimental mock-ups are expected to include critical sections of the vessel (such as nozzles) for thermal or loading tests, or both, and a reduced scale model flow test of the primary vessel and components arrangement. They will also include tests of proposed insulation methods.

The probability of success in meeting these objectives is rated as high. However, as discussed in subsection 2.1.3, if the development steps suggested in Task 2-1.2 and above are not carried out, it would be necessary to base the design upon current technology. As noted in subsection 2.1.3, this could result in a cost penalty of from 0.2 to 0.5 mill/kW-h.

3.5.2 Task 3-9.4 Development of Fabrication Technology

3.5.2.1 Definition, Functional Requirements and Current Technology

This task is defined as the development of processes for field fabrication and quality control of the primary system tank and reactor vessel and ancillary equipment. The functional requirements are that these processes must be compatible with the design requirements developed under Task 2-1.2 of the Plant Design Element and with the development of codes and standards established under Task 3-2.2 of the Components Element.

The current technology of field fabrication of vessels and related components and of quality control procedures is described in Task 3-9.4 of the Components Element of the Programs Plan. This work area was not studied as a part of the 1000 MWe LMFBR Follow-On Study, but is included because of the importance of the Task.

3.5.2.2 New Information Needed

A major objective of Task 2-1.2 is to define the processes and steps for construction of the reference vessels and components described by work performed on that task. Based upon this definition, there is a need to develop techniques for carrying out these processes and steps. There is also a need for establishing quality control procedures and inspection techniques to assure that the construction processes meet certain minimum standards. Further, these quality control and inspection techniques need to be developed so that continuous monitoring of critical components can be made during the operational life of the components.

3.5.2.3 Task Objectives and Assessment of the Effect of Non-Achievement

The objectives of this task follow.

- First, define primary system tank and reactor vessel (and ancillary component) construction techniques based upon the construction steps generated in Task 2-1.2 of the Plant Design Element. These techniques are to be developed with use of mock-up sections of critical components where necessary.
- Second, develop quality control methods for assuring a minimum standard of quality following construction of the tank, vessel and components.
- Third, generate a sample field construction specification for the tank, vessel and components described in Task 2-1.2.
- Fourth, define continuous or regular but intermittent monitoring methods for assuring reliability of the tank and components during operation.

The probability for successfully achieving the first three objectives is high. The probability for achieving the fourth objective is estimated to be medium, since continuous or intermittent methods for non-destructive examination of an operating primary tank may not be feasible. Lack of achievement of all of these objectives could lead to in-service failure of portions of the tank. The reference design is arranged so this would not constitute a significant hazard to the general public, but it could result in unscheduled down-time for repair of the affected member. For short periods of time (2 to 4 weeks) utilities can sometimes purchase power from adjacent utilities to make up lost load. The energy cost for this replacement energy may vary from 1/2 to 2 mills/kW-h, depending upon the circumstances. If replacement power is not available or if the down-time is for a more extended period, the costs can be much higher.

3.6 TASK AREA 3-10—REACTIVITY CONTROL ROD DRIVE

This task area is defined as the design, development, proof testing, and fabrication of reactivity control rod drives for a 1000 MWe LMFBR. These drives include both the shim control rod drives and the scram control rod drives.

As a part of the Demonstration Plant Development Program, reactivity control rod drives will be developed. These Demonstration Plant control drives are different from the type of drives indicated in the Task II report for the 1000 MWe LMFBR. The principal differences are that the Demonstration Plant drives may use sliding seals or bellows seals in much the same manner as is discussed in Task Area 3-10 of the Components Element of the Program Plan. The 1000 MWe drives, as described in the Task II report, utilize a new principle which eliminates the need for sliding seals or bellows seals. A discussion of these 1000 MWe drives is included later.

Table 3-6 shows a schedule of the development program for the 1000 MWe reactivity control rod drives. It is noted on this schedule that work is included for adaptation of the Demonstration Plant drives to the requirements of the 1000 MWe plant. It is not certain at this point that this effort will be entirely successful. Moreover, the Task II reference control drives represent an improved type of design compared with possible Demonstration Plant drives and are more fitted for the requirements of the 1000 MWe plant. Therefore, two programs are conducted in parallel so there is an assurance that an acceptable drive will be available for the completed plant.

The schedule is arranged so the 1000 MWe development program is compatible with the timing assumed for the development of Demonstration Plant drives. It is possible that equipment and facilities used for the development of Demonstration Plant drives can also be used for the 1000 MWe drive development. The 1000 MWe program will require a control drive test stand and various loops and sodium component equipment. An allowance is included in the program for this task area for modification of possible Demonstration Plant Development Program facilities and equipment or for construction of new equipment.

Information is needed from several other program elements in order to complete the requirements of this task area. These elements include the following: Physics, Fuels and Materials, Sodium Technology, Plant Design, and Core Design. The Physics Element will provide information on the reactivity worth of various control rod materials and shapes. Information on the compatibility of these materials and shapes to a radiation and sodium environment will be developed in the Fuels and Materials Element. The gas buffer system for the reactor cover gas will be developed on the Sodium Technology Element. This gas buffer system makes possible the elimination of the seals in the

TABLE 3-6

SCHEDULE FOR TASK AREA 3-10—REACTIVITY CONTROL ROD DRIVES

Task	Government Fiscal Year										
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
3-10.2 Preliminary and Detail Design											
Adaptation of Demonstration Plant Drives				┌───┐							
1000 MWe Task II Reference				┌───┐							
Full Scale Prototypes					┌───┐						
3-10.4 Fabrication of Prototypes and Critical Portions											
Modified Demonstration Plant Drives							┌───┐				
Critical Portions of 1000 MWe Drives					┌───┐						
1000 MWe Task II Reference							┌───┐				
3-10.5 Proof Testing of Prototypes											
Critical Portions of 1000 MWe Drives					┌───┐						
Modified Demonstration Plant Drives							┌───┐	┌───┐			
1000 MWe Task II Follow-On Reference								┌───┐	┌───┐		
Facilities Modification											
Design						┌───┐					
Modification							┌───┐				
Plant Components											
Detail Engineering								┌───┐	┌───┐	┌───┐	
Fabrication, Testing, and Shipping									┌───┐	┌───┐	┌───┐

Task II reference control rod arrangement. Finally, the Plant Design and the Core Design Elements will provide information on updated core and primary system arrangements within which the control rods will be located. The schedule shown in Table 3-6 is intended to be compatible with information developed on the Pool Primary System Study (Task 2-1.2) of the Plant Design Element so updated arrangement information can be used to set some of the functional requirements for the control rod drives.

A discussion of the current technology for control rod drives is included in the Components Element (Task Area 3-10) of the Program Plan. In order that the Task II reference design improvements in this technology may be better understood, a brief discussion of this reference for both the shim and scram rod drives follows:

The Task II reference shim rod drive operates with a linear motion of 34 inches driven by a rack and pinion. Motion is

very slow in both the upward and downward direction. The entire mechanism, including the motor, reduction gear, and rack and pinion, are operated in a cooled reactor cover gas ambient, thus eliminating the requirement of a rotating or sliding seals. Static seals are used in the installation of the mechanism, and a 180-degree rotary seal is used for the unlatching mechanism. The ambient atmosphere is provided by cooling and removing sodium vapor from a small stream of reactor cover gas. This gas passes downward along the control rod drive shaft to return to the reactor cover gas plenum. Downward flow velocity in this annulus would be selected to be sufficient to prevent upward thermal convection currents. Pressure of the reactor cover gas is about 5 psi greater than that of the refueling cell, thus causing any seal leakage to be in the direction of (radioactive) reactor cover gas leaking into the refueling cell.

Electrical connections are made through a hermetically sealed connector in the mechanism housing. This includes leads for the electric motor drive, as well as for position indication instrumentation (such as a selsyn and limit switches).

The design approach suggested in the reference design offers the possibility of a mechanism which will not be subject to leakage of highly radioactive reactor cover gas into the refueling cell and which will not use potentially troublesome sliding seals or long-stroke bellows. Such sliding seals would be normal state-of-the-art if it were not for the problems presented by lubrication, which would tend to coat the sliding shaft with a thin oil film that could be attacked by sodium vapor. Even small gas leakage would be quite troublesome since it would tend to release radioactivity into the refueling cell. This requirement makes it particularly difficult to develop a dry sliding seal.

Another incentive for the reference approach to the shim rod drive is that the overall height of the mechanism can be limited to only a slightly greater than one stroke length. If a sliding shaft seal is used, at least two stroke lengths are required above the seal. With a bellows seal the problem is similar, except that one must also accommodate the length of the bellows. This height of mechanism above the shield plug is important from the standpoint of the overhead space required in the refueling cell. This height affects directly the height of the recessed space above the raised shield plug. It also affects the height of the rails for the overhead crane-manipulator and therefore the height of the refueling cell ceiling, since the reference arrangement has just sufficient head room for withdrawal of a control drive mechanism. The consequence of adding 3 feet to the height of the control rod drive is to raise the ceiling of the refueling cell by 3 feet and the ceiling above the shield plug by 6 feet, and the height of the reactor building by 6 feet (so the overhead crane will clear the top of the refueling cell). Also, it is necessary to increase the reach of the overhead manipulators and the shield plug hoist by 3 feet.

The Task II reference scram-rod drive operates over a stroke of 34 inches with actuation by argon gas and a vacuum. During reactor operation the drive rests in the uppermost end of its stroke with its piston held firmly against a face seal by vacuum. When operation is initiated, a high pressure of argon is applied to the upper end of the piston to cause it to move rapidly downward. Deceleration is accomplished by restricting the exhaust of argon below the piston through one or more relief ports and by a final cushion. Recocking the scram rod is accomplished by connecting the upper end of the cylinder to a high-capacity vacuum system until it is firmly seated against the top face seal. This

would be done with one scram rod at a time, whereas all scram rods would be held in the cocked position by a single low-capacity vacuum system.

As with the shim control drive, disconnection of the control rod is accomplished by rotation of the drive shaft 90 degrees to disconnect and, after lifting to the top of the stroke, an additional 90-degree rotation to assure that it is disconnected. Sealing around the drive shaft is only sufficient to prevent excessive bypass when the rod is decelerating at the lower part of its stroke.

The incentives for the development of this type of rod are similar to those of the shim rod drive; namely, the possibility of limiting the height requirement to only slightly greater than one stroke length and the elimination of critical sliding seals and their leakage.

3.6.1 Task 3-10.2 Preliminary and Detail Design

3.6.1.1 Definition and Functional Requirements

This task is defined as the preliminary design necessary to adapt Demonstration Plant reactivity control-rod drives to the requirements of the 1000 MWe LMFBR, adapt the preliminary design of the Task II conceptual reference for the shim and scram rod drives, and adapt the detail design of full-scale prototypes for both the Demonstration Plant type of drives and the Task II reference concepts.

The functional requirements for the design of the control rods will be established by reference to the Task II report, to the updated primary system arrangement study conducted under Task 2-1.2, and by reference to the Demonstration Plant design for the reactivity control drives.

3.6.1.2 New Information Needed

New information is needed concerning the Demonstration Plant drives to determine their compatibility with the reference concept selected for the primary system of the 1000 MWe LMFBR. The 1000 MWe plant will use vented fuel with a very large primary system vessel which has stringent requirements on the minimization of leakage from the upper region of this vessel to the refueling cell or to the reactor room. The ability needs to be determined of the Demonstration Plant control-rod drives to meet these requirements, and also to meet the space requirements, the rate of reactivity change requirements, and the control rod weight requirements of the 1000 MWe LMFBR. Some modification to these drives will undoubtedly be necessary. Means for accomplishing this modification so as to have the least impact on the remainder of the system should be determined.

New information is needed to secure a more detailed information on the reference design of Task II. Additional information is needed from Task 2-1.4, Probabilistic Safety Analysis Study, to define more precisely the functional requirements for the rate of change of reactivity under different routine and emergency conditions. This information in turn needs to be related to the mechanical design of the mechanisms needed to produce this movement. This should include an evaluation of the stresses that the scram and shim rods will be subjected to during acceleration and deceleration movement as well as during shim rod control. More detailed information also needs to be developed concerning the proposed method of sweeping argon gas through the various complex mechanisms in the shim and control rods at a rate sufficient to prevent sodium vapor condensation in these mechanisms.

Following the preliminary design of the modified Demonstration Plant and of the

Task II reference drives, design information is needed in order for full-scale prototypes to be fabricated. This design information should be sufficient for the fabrication of single units. It will not include the final design and specifications for the plant control rods.

3.6.1.3 Task Objectives and Assessments of Effect of Non-Achievement

The objectives of this task follow.

- Prepare preliminary designs of the Demonstration Plant type shim and scram rod drives suitable for use on the 1000 MWe LMFBR.
- Determine what modification of these rod drives will be necessary to adapt them to the 1000 MWe LMFBR conditions.
- Prepare a preliminary design of the conceptual reference scram and shim rod drives contained in the Task II report.
- Prepare detailed designs of both the Demonstration Plant type and the Task II concept of shim and scram rods in sufficient detail to permit fabrication of prototypes of these rods for testing.

The probability of achieving the first two of these objectives is estimated to be medium, and the probability of achieving the last two objectives is estimated to be high. Until the Demonstration Plant program is farther along, it is not possible to determine whether complete success can be achieved in modifying the drives for the use of the 1000 MWe LMFBR plant. Lack of achievement of this objective would not seriously hamper the program since development of the Task II reference concept drives would proceed. However, early success of the first two objectives would allow a reassessment of the necessity for

proceeding with the Task II reference concepts. Considerable savings in the development program would be realized if it were possible to eliminate the development of the more advanced type of drives. The magnitude of this savings can be ascertained by examining development program costs for these items (at the beginning of this Element).

3.6.2 Task 3-10.4 Fabrication of Prototypes

3.6.2.1 Definition and Functional Requirements

This task is defined as the fabrication of prototypes of modified Demonstration Plant drives suitable for testing under 1000 MWe conditions and the fabrication of prototypes of the Task II type of drives defined under Task 3-10.2.

The functional requirements for this task are that the prototypes will be fabricated to demonstrate the functional requirements for the plant conditions as defined under Task 3-10.2 and to be compatible with the available testing facilities.

3.6.2.2 New Information Needed

Since the primary objective of this task is the fabrication of prototypes to be tested under Task 3-10.5, a large body of information will not be developed in Task 3-10.4. However, some information will be developed concerning quality control procedures. This information will later be applied to the specifications for the finished plant components. Means are presently available for the fabrication of the mechanisms and parts required for the drives, so there will not be the requirement for the development of a unique fabrication technology.

3.6.2.3 Task Objectives and Assessment of Effect of Non-Achievement

The objective of this task is to fabricate prototype units suitable for testing in the Breeder Test Facilities of General Electric.

These prototype units will include the Demonstration Plant type of control rod drives and the Task II reference type of drives. The detailed design for these prototypes will be accomplished under Task 3-10.2. Critical portions of the Task II reference drives will also be fabricated for bench testing.

The probability of achieving the objective of this task is estimated to be high. Failure to achieve this objective would result in the lack of test data upon which to base the final design. This would be a serious deficiency and could result in faulty performance of the control rod drives with consequent plant outages or compromises to the plant safety. No assessment of the economic penalty is possible with regard to possible plant safety compromises. Unscheduled plant outages result in a deficiency of power which must be made up by purchased power or by operation of less efficient plants. Purchased power or makeup power can result in cost penalties of from 0.5 to 2.0 mills/kW-h during the period that the plant is inoperative.

3.6.3 Task 3-10.5 Proof Testing of Prototypes

3.6.3.1 Definition and Functional Requirements

This task is defined as the proof testing of modified Demonstration Plant shim and control rod drives and proof testing of prototypes of the Task II reference type of shim and control rod drives. Additionally, critical portions of the Task II drives will be bench tested. The functional requirements for this task will be established by reference to information generated under Task 3-10.2. These functional requirements will include a definition of the number and type of loading cycles to which the prototypes will be subjected. These loading cycles will be sufficient to simulate or exceed conditions expected in the reactor environment. Certain

of the tests will include a dry argon atmosphere environment with a heated sodium pool to simulate conditions expected in the reactor.

3.6.3.2 New Information Needed

New information is needed on the ability of the modified Demonstration Plant control rod drives to satisfactorily operate under 1000 MWe conditions. In particular, information is needed concerning the ability of the rods to minimize leakage of cover gas outside of the reactor environment under repetitive cycling of the rods and at high temperature conditions. Information is also needed concerning the ability of these drives to meet the normal rate of reactivity change and the rapid acceleration and deceleration requirements under the loading conditions expected for the 1000 MWe plant.

Similar information to that required for the modified Demonstration Plant drives is also needed for the Task II reference type of drives. Additionally, information is needed concerning the control of the buffer gas which sweeps through the mechanisms of the drives. Since the Task II reference concept will not have had the advantage of a previous proof testing program as with the Demonstration Plant drives, a more extensive testing program is necessary for these drives. This testing program should extend the life cycle of the prototype units to that approaching or exceeding the life cycle expected during operation of the plant.

Based upon the testing program, information is needed concerning possible modifications to improve the performance of the drives to be fabricated for the plant. This information will be incorporated in the specifications for these drives.

3.6.3.3 Task Objectives and Assessment of Effect of Non-Achievement

The objective of this task is to proof-test full-scale prototype models of the two

types of scram- and shim-rod drives defined in Task 3-10.2 and to bench-test critical portions of the Task II drives, and by such bench- and proof-testing, to obtain information concerning the adequacy of these designs. A further objective is to obtain additional information which will permit incorporation of desirable modifications into the final design and specifications of the plant scram- and shim-rod drives. Provision is made for final testing all drives prior to shipment to the plant. The probability of achieving the objectives of this task is high. The probability of achieving a scram- and shim-rod drive which is capable of operating reliably under conditions required for the 1000 MWe LMFBR plant is also estimated to be high since two separate drive types are included in this development program. If at a given point in the development program it appears that one drive type should be dropped, some savings in development costs could be realized. Failure to achieve the objectives of this task would result in penalties similar to those discussed under Task 3-10.4.

3.7 TASK AREA 3-11—LIQUID METAL PUMPS

The objective of this task area as defined in the Components Element of the Program Plan is to achieve "...the tech-

nology base to permit the design and fabrication of liquid metal pumps for commercial LMFBR plants." Work is currently in progress on the development of these pumps. Contracts have been let to Westinghouse and Byron Jackson to develop large sodium pumps and the LMEC is providing test facilities for this development work. The work includes the development of seals and bearings as well as full-scale prototype testing of the operating pumps under simulated reactor conditions.

The current state-of-technology and the informational needs for this task area are described in the Program Plan. Also indicated in the Program Plan is the schedule for development and testing of large prototypic pumps. The reference conditions selected for these prototypic pumps together with the characteristics of the primary and secondary pumps from the Task II report are shown in Table 3-7. As can be seen in the table, the characteristics of the pumps are sufficiently close that development of the reference prototype should be sufficient to allow normal design extrapolation to meet the Task II conditions. The schedule indicates that prototypic units will have completed their testing program in the LMEC at the end of fiscal year 1974. This will be in a sufficient time to be incorporated in the final design of the 1000 MWe LMFBR.

TABLE 3-7
SODIUM PUMP CHARACTERISTICS

	Development Pump Reference Conditions	Task II Report	
		Primary Pump	Secondary Pump
Flow (gpm)	60,000	63,800	49,000
Head (ft)	350	240	243
Net Positive Suction Head (ft)	40	51	130
Flow Control Range (%)	?	20 to 100	20 to 100
Operating Temperature (°F)	?	810	650

It should be noted that the development of seals and bearings being carried out on the on-going program is of particular importance to the Task II reference design. The reason is that the pump support base and drive are mounted on the shield deck, and the pump barrel and shaft penetrate directly into the sodium pool of the primary system. Failure of a bearing in either the pump or the driver must not produce a mechanism which could destroy the integrity of the seal that prevents the release of cover gas from the primary system (or prevents the entrance of oxygen into that cover gas). Moreover, means should be provided for repairing a damaged seal without removing the pump from its mounting place. Additional work on development of bearings and seals may be performed for the Demonstration Plant.

Work will be carried out in the Plant Design Element which is important to the pump development and design. This work is Task 2-1.7, Dynamic Analysis and Control Study; it is scheduled to be completed in fiscal year 1972 (presuming commencement at the beginning of fiscal year 1970). This work will permit a better definition of the functional requirements of the pumps in terms of transient temperatures to which pumps will be subjected and in terms of the necessary pump, speed-control accuracy and range. Unfortunately this work will be too late for incorporation in the detail design of the prototypic pumps, which is underway now, but it should be helpful in the testing program and it will most certainly be useful for the final design requirements of the 1000 MWe LMFBR pumps.

Since Task Area 3-11 is already part of a continuing program which includes pump ratings and characteristics close to Task II 1000 MWe conditions, and since the Demonstration Plant and the Fast Flux Test Facility (FFTF) are expected to provide additional applied pump development work, no additional

program is suggested at this time. It is obvious that this is a vital part of the over-all LMFBR development program. Failure to achieve the objective of the task area would be sufficiently serious that the LMFBR program would be jeopardized. Failure to meet certain intermediate goals such as a design efficiency goal or a limitation on the controllable range of the pump could cause economic penalties which, although serious, would not retard the over-all program. A 1 percent change in the efficiency of the primary and secondary pumps would be equivalent to an additional capital requirement of \$310,000.* No quantitative assessment has been made of penalties associated with control limitations. If specific work carried out for the Demonstration Plant, the FFTF, and on-going programs, later prove to be insufficient for the 1000 MWe program, additional work will be suggested.

3.8 TASK AREA 3-12—INTERMEDIATE HEAT EXCHANGERS

This task area is defined as the design, development, and fabrication technology necessary to put into operation intermediate heat exchangers (IHXs) with operating conditions similar to those described in the 1000 MWe LMFBR Task II Report. This task area is closely related to Task Area 3-6, Steam Generators; the technology is similar and many of the same development test facilities will be used. Moreover, in an analogous manner to the steam-generator development program, this task area will build upon a base established by the Demonstration Plant Development Program for intermediate heat exchangers. Indeed because of the similarity of the 1000 MWe and Demonstration Plant intermediate heat exchangers, the technology will be even more applicable than in the case of the steam-generator development work.

*Assuming that incremental plant output is worth \$85/kWe and a fuel cost of 3.44¢/MBtu or a capitalized differential fuel cost worth of \$1860/Btu-kW-h.

The reference-design intermediate heat exchangers indicated in the Task II report are vertical shell and tube units with the support for the heat exchangers provided at the upper concrete deck. Piping to and from the exchangers enters the units from above in a concentric arrangement and penetrates the shield deck. A concrete shielding plug and external shielding is provided around this piping to protect against gamma radiation. A unique feature of the reference design is that the primary sodium is located on the tube side and the secondary sodium is located on the shell side. The reason for this selection was to assure low pressure drop and predictable flow characteristics on the primary fluid side. It is anticipated that the Demonstration Plant IHX may have the fluids in a reverse order with the secondary sodium on the tube side. Nevertheless, the technology developed for the Demonstration Plant Heat Exchangers will be very similar to that of the 1000 MWe LMFBR. Facilities needed for this task area are similar to those required for the steam generator. The majority of the work is to be carried out at General Electric with some subcontract work in supplier shops. The equipment and facilities required may be similar to possible facilities and equipment used for the Demonstration Plant program. An allowance is provided for modification of equipment in the facilities or for fabrication of new equipment. There is a possible need for either the LCTL or SCTI (at the LMEC) for one part of the work in this task area, if availability of possible Demonstration Plant program loops becomes restrictive during the testing program. It is currently planned that the program will be carried out without requiring the LMEC facilities.

A discussion of the current technology for this task area is included in Task Area 3-12, Components Element, of the Program Plan. It is anticipated that by the time the

majority of work under this task area is in progress, the technological basis will have improved substantially by reason of the Demonstration Plant IHX development. A schedule for work under Task Area 3-12 is shown in Table 3-8. This schedule is compatible with possible Demonstration Plant development work with facilities programmed for use as they become available from assumed Demonstration Plant work.

3.8.1 Task 3-12.2 Design

3.8.1.1 Definition, Functional Requirements, and Current Technology

This task is defined as the work necessary to produce preliminary scope designs, prototype designs, final scope designs, and specifications for the 1000 MWe LMFBR intermediate heat exchangers (IHX). The functional requirements for this work will be derived in a similar manner to those of the steam-generator task area. For the preliminary scope design the functional requirements will be generated from three sources if possible: (1) the 1000 MWe LMFBR Follow-On Study, (2) preliminary work on Tasks 2-1.2, 2-1.4, and 2-1.7 of the Plant Design Element, and possibly (3) the Demonstration Plant Program. Functional requirements for the final scope and specifications will be generated principally from the information obtained from the aforementioned tasks of the Plant Design Element and from the results of the testing work carried out during Task Area 3-12. The functional requirements will include the following information:

- System characteristics under steady-state full-load conditions,
- Rate of change of temperature and flow during normal load changes,
- Rate of change of temperature and flow under rapid transients such as during plant scram, and

TABLE 3-8

SCHEDULE FOR TASK AREA 3-12—INTERMEDIATE HEAT EXCHANGERS

Task	Government Fiscal Year										
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
3-12.2 Design											
• Preliminary Scope											
• Prototype Designs											
• Final Scope and Space											
3-12.3 Fabrication Technology											
• Critical Parts											
• Prototypes											
• 2 to 3 MWt Prototype											
• Flow Test Model											
3-12.4 Model Tests											
• Water-Flow Test											
• 2 to 3 MWt Model											
• Demonstration Plant Intermediate Heat Exchanger*											
3-12.5 Materials, Insulation, and Maintenance											
• Materials Tests**											
• Insulation Fabrication and Tests											
• Maintenance Tests***											
Plant Intermediate Heat Exchangers											
• Detail Design											
• Fabricate and Deliver											

* May be carried out under the Demonstration Plant Development Program

** Carried out in the Sodium Technology Element

*** Performed under Task Area 3-7

- Conditions expected during two-loop operation with one loop out of service.

The current technology for this task is covered in the discussion under the task area.

3.8.1.2 New Information Needed

New information is needed concerning alternative arrangements to the reference IHX design described in the Task II report. A comparison of the proposed Demonstration Plant IHX arrangement with other possible conceptual arrangements should be made. Based upon this comparison, a final refer-

ence should be selected. Other design information is needed to permit the component fabrication and testing program to proceed. Finally, information is needed from the testing programs to allow preparation of functional requirements, final scope design, and specifications so that the detailed design and fabrication of the plant intermediate heat exchangers may progress.

3.8.1.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task are as follows:

- Prepare a preliminary scope design of a reference IHX based upon a consideration of alternative design possibilities. Functional requirements for this preliminary scope design shall be based upon the 1000 MWe LMFBR Task II report, upon possible work performed for the Demonstration Plant, and upon early information obtained from the Plant Design Element.
- Prepare the designs necessary to fabricate particular prototypic sections required for tests performed in Task 3-12.4.
- Prepare the final functional requirements, scope design, and specifications necessary for the solicitation of quotations on the plant intermediate heat exchangers.

The probability of achieving the objectives of this task is high. Tests carried out during the program, however, may show that the performance goals noted in the Task II report should be more modest. For example, the measured pressure drop in the heat exchanger may be higher than the calculated values shown in Task II. This could result in the requirement for an increased pressure rating in the cover gas over the reactor vessel, and this, in turn, could complicate the design of the shield deck over the vessel. The control of the cover gas regions would also be more difficult. No evaluation of this possibility was made during the Task III work so no quantitative assessment of the economic penalties is possible.

3.8.2 Task 3-12.3 Fabrication Technology

3.8.2.1 Definition, Functional Requirements and Current Technology

This task is defined as the development of the technology necessary to fabricate the

intermediate heat exchangers for the 1000 MWe LMFBR. This will be accomplished by fabrication of critical parts which are portions of the intermediate heat exchanger. These parts will include full-scale portions of the IHX, a two or three MWt multiple tube segment of the exchanger, and a scale model portion suitable for performing flow tests.

The full scale portions will be subjected to strain cycling and temperature transient tests to confirm the adequacy of the design. The two or three MWt prototypic section will be utilized in a thermal and flow test of this segment of the exchanger, and the scale model portion of the exchanger will be used for flow pattern determination and tube vibration experiments.

The functional requirements of this task are that the fabrication and manufacturing technology must be developed to the extent necessary to meet certain minimum quality control standards to assure the reliability of the final product. These quality control standards will be established as a part of the task and will draw upon standards which may be developed during the fabrication and development of the Demonstration Plant IHXs.

The current technology for this task was described under the task area discussion. As with the other tasks in this task area, it is expected that at the commencement of the fabrication technology work, the state-of-technology will be improved from the current state because of work which may be performed on the Demonstration Plant IHX's. In particular, tube-to-tube sheet joints may have been fabricated, and work on thermal sleeves, nozzles, special expansion bellows, and shield plug assembly, could be well along as a part of the manufacturing development program for the Demonstration Plant. Much of this work would be directly applicable to the fabrication technology development work required by the 1000 MWe program.

3.8.2.2 New Information Needed

New information, beyond that presently available and in addition to the information from a possible Demonstration Plant program, is required concerning the fabrication of the 1000 MWe intermediate heat exchangers. This informational need results from two principal causes:

- The more severe service requirements of the 1000 MWe IHX's compared with the Demonstration Plant, and
- The specific differences in the design of the 1000 MWe units compared with the Demonstration Plant IHX's.

Thus, although much of the work from the Demonstration Plant can be used, additional directed information is required to obtain knowledge concerning reliable methods of fabricating these particular components. It is believed that quality control procedures which may be established for the Demonstration Plant IHX's will also be adequate for the 1000 MWe designs. Some of these procedures will be applied on a trial basis in the fabrication technology task to determine their suitability for final application on the plant IHX's.

3.8.2.3 Task Objectives and Assessment of Effect on Non-Achievement

The objective of this task is to obtain fabrication and manufacturing experience suitable to implement the manufacture of the plant IHX's by the fabrication of selected portions of the IHX's. These portions will include the following:

- Full-scale portions of critical parts of the IHX,
- A 2 to 3 MWt multiple tube full-scale prototypic section of the exchanger, and

- A scale model of a portion or all of the shell side of the intermediate heat exchanger.

The probability of achieving the objectives of this task is estimated to be high. Failure to achieve the objectives could result in less reliability in the finished components. This could affect the plant availability factor by reason of unscheduled outages for maintenance. During the period that a 1000 MWe LMFBR is out of service for maintenance, the utility must purchase or generate replacement energy. This replacement energy can cost from 0.5 to 2.0 mills/kW-h during the period of the outage presuming that replacement energy is available. If it is not available, the costs can be much higher.

3.8.3 Task 3-12.4 Model Tests

3.8.3.1 Definition, Functional Requirements, and Current Technology

This task is defined as the testing of mock-up portions of the reference intermediate heat exchangers to determine the suitability of the selected design for the performance and functional requirements defined in Task 3-12.2. As information is developed from these tests, modifications will be incorporated where necessary into the final scope design. Three major test areas are defined:

- A water flow and gas blanket test on a scale model of the shell side of the heat exchanger,
- A thermal performance test on a 2 to 3 MWt section of the IHX, and
- Full-scale tests on the Demonstration Plant intermediate heat exchangers during operation of the plant (possibly).

The functional requirements for these tests will be established from the preliminary

scope design effort and from the Demonstration Plant performance specifications. These functional requirements will include definition of the range of flow and temperature extremes as well as the rate of change of these parameters through which the components will be tested.

The current technology for this task was discussed in general terms under the task-area description. Additional information beyond the present state-of-the-art may be available from the Demonstration Plant program since analogous model tests from that program may be completed at commencement of the work on this task. It is possible, for example, that shell-side flow tests for the Demonstration Plant intermediate heat exchangers will yield information concerning the reliability of predictive techniques to determine flow patterns, tube vibration, and controllability of liquid level in the exchanger. Presuming that such tests are performed and that the predictive techniques are shown to be reliable, the need for an extensive parametric investigation in Task 3-12.4 will be reduced.

3.8.3.2 New Information Needed

New information needed on this task beyond that supplied from a possible Demonstration Plant Development Program includes investigation of the flow pattern on the outside of the tubes for the particular design arrangement of the reference 1000 MWe exchangers. The effectiveness of the selected baffle arrangement to adequately distribute the coolant flow and to prevent tubing vibration must be determined. The possibility of reverse flow patterns or stagnant areas occurring as the flow through the exchanger is reduced must be established.

The Task II reference design 1000 MWe intermediate heat exchangers utilize a separate cover-gas pressure than that over the reactor sodium. Under conditions when the

intermediate heat exchanger is removed from service (e.g., one loop taken out of operation), the reference concept achieves shut-down by increasing the gas pressure so that the level of the primary sodium drops below a flow baffle and prevents further fluid flow. The control of sodium-flow rate on the shell side of the exchanger by varying the gas pressure needs to be studied to assure that an adequate range of control can be established without entrapment of gas bubbles in the sodium.

To assure that a development and testing program such as that proposed under Task 3-12 can yield a reliable operating unit and, also, to benefit in the final design stage from knowledge gained in the operation of a full size heat exchanger, it is desirable that information be obtained concerning the operating characteristics of the Demonstration Plant IHX's. Thus, Task 3-12.4 shows tests of Demonstration Plant IHX's during operation of the plant. This work is not funded on the 1000 MWe program on the premise that it will be included as part of the Demonstration Plant program. It is shown under Task 3-12.4 because of its importance to the 1000 MWe program.

3.8.3.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task follow.

- Perform water flow tests on a scale model portion of the IHX in order to determine the flow characteristics of the exchanger. These tests will include a study of the flow control characteristics with varying cover-gas pressure. Gas entrainment effects will be studied.
- Perform a thermal performance test of a multiple tube segment of the intermediate heat exchanger.

- If possible, obtain information on the operating characteristics of the Demonstration Plant intermediate heat exchangers during operation of that plant so the information might be used in the final design phase of the 1000 MWe IHX's.

The probability of achieving these objectives is estimated to be high; however, as discussed under Task 3-12.2, it is possible that the tests will demonstrate the need for certain reduced performance characteristics compared with those shown in the 1000 MWe LMFBF Task II report. The effect of this possible reduced performance is also discussed under Task 3-12.2.

3.8.4 Task 3-12.5 Materials and Insulation

3.8.4.1 Definition, Functional Requirements, and Current Technology

This task is defined as the accumulation of experimental data on materials suitable for the conditions defined in the Task II report and in Task 3-12.2. It also includes test information concerning certain proposed methods of insulating structural parts of the IHX. The insulation methods and functional requirements for these methods will be defined in Task 3-12.2.

Concerning the current technology of materials, it is discussed in some detail in Element 5, Sodium Technology. There currently exists a considerable body of data relative to the mass transfer characteristics of austenitic stainless steel in sodium. However, there is very limited information concerning the material depletion and deposits which might occur over a 35-year lifetime in the specific IHX configuration selected as a reference. There is also very little information concerning the heat transfer and frictional characteristics of these deposits, or of practical means for removing them.

Various methods have been used to insulate materials in sodium. The principal

problem involves the reduction of the temperature drop across structural barriers separating hot sodium from cold sodium. In the Task II reference heat exchanger, for example, portions of the exchanger are exposed to temperature differences of 300°F or greater. These temperature differences would produce excessive thermal stress in a single stainless-steel barrier. The method for reducing these temperature differences in SEFOR and EBR-II represent current technology. In these reactors, multiple steel plates with close fitting gaps, filled with stagnant sodium were used across such barriers (as on the reactor vessel). This worked well for the temperature differences involved, which were on the order of 200°F. The Demonstration Plant Program may study methods for providing insulation on the intermediate heat exchanger. Again, however, the temperature differences are less than with the Task II reference design. On the reference design it was proposed that a gas insulating barrier be provided between two stainless-steel members.

3.8.4.2 New Information Needed

New information is needed concerning the materials' properties and compatibility with sodium in the intermediate heat exchangers, under conditions defined in the Task II report. Since the IHX operates at a higher temperature than the steam generator, these materials will probably be austenitic stainless steel. Work may be performed on the Demonstration Plant Program to investigate suitable materials for the Demonstration Plant IHX's. New information is needed which extends the technology beyond that of the less severe conditions required for the Demonstration Plant.

One study which was carried out during the 1000 MWe Task III study involved the identification of problems associated with mass transport of materials within the primary sodium system. It became evident

from data obtained from the Mass Transfer Program that under pessimistic conditions, substantial quantities of materials could deposit on the colder surfaces of the IHX compared with the core. These deposits could be sufficient to materially reduce the performance capability of the intermediate heat exchanger. Indeed under very pessimistically assumed conditions, the required heat-transfer surface could increase by a factor of two compared with more realistic assumptions. This represents a major uncertainty in the design of the IHX and should be resolved by specific tests oriented to the reference conditions and configuration of the selected exchanger. In addition to an understanding of the possible deposits on the exchanger, data should be obtained on methods for removing these deposits during the heat exchanger lifetime. For example, by thermally cycling the exchanger, it may be possible to loosen the deposits. Alternatively, the exchanger could be removed periodically and treated chemically, and information is needed on possible cleaning methods. The more attractive alternative, of course, is to design the exchanger in such a way that the deposits do not seriously reduce the performance characteristics during its lifetime.

Because the intermediate heat exchangers operate in a radioactively contaminated environment, and must periodically be maintained, information is needed on methods for maintaining them. This information should include basic data on the behavior of fission products in sodium and additionally should include specific maintenance procedures with mock-up sections, possibly contaminated with tracer isotopes. The informational need concerning the behavior of fission products in sodium will be met in the Sodium Technology Element. The informational need concerning the mock-up maintenance tests will be met by work performed under Task Area 3-7. They are mentioned here and shown in the schedule because of their importance to Task Area 3-12.

Information may be developed on the Demonstration Plant IHX's concerning the insulation problem and much of this information would be directly applicable to the 1000 MWe IHX's. However, because of the more severe service conditions in the 1000 MWe exchangers, additional information will be needed for the more severe conditions of the 1000 MWe LMFBR. Mock-up sections of the exchangers will provide the necessary configuration to perform tests designed to supply the needed information.

3.8.4.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives for this task follow:

- Obtain data on materials suitable for use in an IHX with conditions as defined in the Task II report and in accordance with the informational needs of subsection 3.8.4.2. This test data on materials is to be obtained from the Sodium Technology Element.
- Experimentally study methods for insulating structural members against excessive temperature drop across these members and to extend the technology for such insulation beyond that of the Demonstration Plant design.
- Perform mock-up maintenance tests and establish effective maintenance procedures under radioactively contaminated conditions; this work will be performed under Task Area 3-7.

The probability of achieving the objectives of this task without some modification to the performance goals of the Task II report is estimated to be medium. Although it is not anticipated, for example, that the severe buildup of materials on the tubes will occur as pessimistically assumed in the Task III Report, some deterioration is possible. Under cost penalty could be from 3 to 4 million dollars. Similarly, the insulation performance should be sufficient for the 300 °F and greater temperature drops indicated in the Task II report. Again, however, some deterioration could occur. The penalty for this possible drop in ΔT was not estimated.

SECTION IV

ELEMENT 4 - INSTRUMENTATION AND CONTROL

The General Electric LMFBR design is predicated on the assumption that reliable instrumentation will be developed to monitor the performance characteristics of the core. The fundamental purpose of the core instrumentation is to decrease the probability that there can be developments within the core area that would compromise reactor safety without the effects of these developments being observed. Developments that might affect the safety of the reactor are:

- Any phenomenon that reduces the normal total coolant flow through a fuel channel,
- Any phenomenon that reduces the local coolant flow within a fuel channel, and
- Any phenomenon that increases the reactor power level above the normal operating range.

The special emphasis for the core instrumentation is to detect malfunctions originating within one fuel channel before this local disturbance might propagate through the core to the extent that reactor safety would be compromised. The LMFBR design is expected to be feasible on the basis of bulk flow and outlet temperature measurements on each fuel channel. Local sodium temperature, and acoustic measurements to detect sodium boiling may also prove to be desirable.

Additional instrumentation systems are described that

- Monitor neutron flux,
- Provide an estimate of reactor subcriticality,

- Provide diagnostic information on mechanical components based on acoustic principles, and
- Determine the location of a water-sodium leak in the steam generator.

An extensive discussion is contained in the Task III Report concerning the functional requirements of in-core instrumentation as related to the detection or prevention of propagation of local accidents. It is pointed out in that discussion that numerous probes may be of value in detecting and preventing such propagation, but two detectors in particular appear to be useful for this purpose. These detectors or probes are coolant flow detectors on individual channels and thermocouple temperature measurements for bulk coolant outlet temperature on each channel. Other instruments such as acoustic detectors, thermal noise analysis, and delayed neutron sensors can also prove useful, and may ultimately be more important than bulk sodium flow and temperature, but the present state-of-art would suggest that these two devices are especially important for the 1000 MWe LMFBR.

The precise nature of the flow and temperature probes are not uniquely defined in the Task III Report. Venturi type flowmeters are shown, and thermocouples are described which extend to the upper region above each fuel channel, but the instrument characteristics and details are imprecisely described. This lack of attention to detail is because present knowledge is inadequate to justify the selection of unique instruments. For example, flow could be sensed on each channel by a venturi type device, an electromagnetic flowmeter, a flow noise sensor, or other means.

Each has advantages and disadvantages which have not been adequately explored in existing testing programs. Similarly, there are various types of thermocouples and methods for diagnosing temperature readings. On the ESADA program at General Electric, for example, work is being performed on thermocouple arrangements which evaluate characteristic temperature patterns produced by a flowing coolant in a channel. This work is being performed for the Demonstration Plant. It is expected that both flow and temperature probes will be developed for the FFTF. At this time, however, it is not possible to predict the precise characteristics or even the type of instruments which will be most successful. Furthermore, it is not clear whether the instruments developed for the FFTF or the Demonstration Plant will meet the specific needs of the 1000 MWe LMFBR.

In an analogous manner to the individual channel instruments, other safety and diagnostic instruments are included in this program for which work will be performed on the FFTF and the Demonstration Plant. For example, in-core neutron sensors may be useful to detect power shifts with burnup and regions of peak power at enrichment zone changes in the core. This type of instrumentation development may be pursued on the Demonstration Plant. However, the neutron sensors may be of the non-regenerative type and accordingly be less suitable than regenerative sensors for the 1000 MWe LMFBR. Moreover, the temperature regime of the Demonstration Plant will be less than that of the 1000 MWe LMFBR. Thus an extension of technology is required.

Because of the problem of identifying the most attractive instruments for use in the 1000 MWe LMFBR, and because of the importance of this work to the 1000 MWe concept (see discussion under Safety in Task III Report), development is included in this program for a variety of instrument types. Some of this work may ultimately prove to be dupli-

cative of the FFTF or Demonstration Plant Development Programs. If at a later point in the program, this duplication becomes evident, specific instrument development will be redirected or discontinued. It is assumed that certain basic instrumentation systems, e.g., those for determining sodium level, failed fuel-element location, high-temperature strain gages, cables and connectors, sodium pressure measurement, and hydrogen evolution from a steam generator water to sodium leak, will be developed successfully during the Demonstration Plant or FFTF programs. Accordingly, these instrument systems are not specifically discussed in this section. Evaluation of these systems to meet the potentially more demanding requirements of the LMFBR is discussed in Task 4-7.15.

Another area discussed in the Task III Report which extends the technology involves the application of multiple instrument probes in a system complex which evaluates the signals received and determines the corrective action, if any, which should be taken. The logic which must be incorporated into this system should be such as to reject spurious signals generated by faulty instrument probes or systems, and accept valid signals. The reliability and redundancy necessary to accomplish this task with specific devices yet to be developed needs to be determined. The guidance for this type of development work is included as part of Task 4-7.13 in this program.

The General Electric LMFBR design has a conceptual plant control system which utilizes no large sodium valves, and exercises temperature control by sodium pump speed control, steam bypass of the reheater and attemperation of the superheated steam with feedwater. Steady state analyses of partial load operation were given in the Task III Report as the basis for determining the feasibility of the conceptual plant control system. Transient analyses of the performance of the reference plant are required to establish the

actual plant control system and implications on the plant design. Because of the advanced performance characteristics of the LMFBR it is essential that there be a good understanding of plant dynamics and that suitable control hardware be developed by industry to meet the specifications that will be determined by the control system studies.

Table 4-1 shows the instrumentation and control development milestones and Table 4-2 describes the program outline which reflect the above considerations. Each of these tasks is discussed in more detail in the remainder of this section.

It should be noted that the milestone schedule is planned to be compatible with information provided from other elements and tasks of this program. These elements or tasks include:

	<u>Task</u>	<u>Element</u>
2-1.4	Probabilistic Safety Analysis Study	Plant Design
2-1.7	Dynamic Analysis and Control Study	Plant Design
6-1.1	Reference Design	Core Design
3-6.4	Steam-Generator Materials and Small Components Tests	Components

The informational needs satisfied by these various tasks are described in detail in the task descriptions. Briefly, the Probabilistic Safety Study will permit a more quantitative assessment of the ability of different types of instrument devices to meet a given safety criteria. The Dynamic Analysis and Control Study satisfies all of the requirements of Task 4-4.1. Information from the Reference Design work will allow an updated evaluation of geometric considerations for instrument probes; and an updated version of the environmental conditions to which the in-core instruments will be exposed. It will also

provide further guidance on the need for in-core neutron detectors (Task 4-2.1). The steam-generator tests will provide facilities for carrying out the steam-generator leak location tests.

In the discussion of each of the tasks which follows, economic penalties are evaluated where possible to assess the possible non-achievement of certain design or development goals. One economic penalty which was studied on the Task I effort is associated with this entire element. This penalty involves the possible non-achievement of any in-core instrumentation which would be utilized to relegate a core voiding or melt-down incident to a sufficiently low probability as to disregard it in the plant design effort. In the event that this were the case (e.g., in-core instrumentation to detect and prevent fuel failure propagation is not developed) then it would be necessary to resort to compromise in core design to enhance "inherent" characteristics (e.g., void reactivity and Doppler effect). Studies in Task I showed that a compromise core design to achieve "inherent" safety could result in economic penalties in excess of 1.0 mill/kW-h. This would be sufficient to render the LMFBR non-competitive with other forms of power expected to be available in the 1980's. For this reason, all of the in-core instrumentation development described in this element which is utilized to diagnose and prevent propagation of fuel failure is of the highest priority.

4.1 TASK 4-2.1—CIRCUITS FOR NEUTRON SENSORS

4.1.1 Definition, Functional Requirements and Current Technology

This Task is defined as the effort to develop, test, and qualify circuits for use with the neutron detectors of Task 4-7.1/7.2.

Functionally required are circuits for both ionization and self-powered type detectors,

TABLE 4-1
INSTRUMENTATION AND CONTROL MILESTONES

Task	Title	Government Fiscal Year										
		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	
4-2.1	Circuits for Neutron Sensors			1	2	3						
4-3.1	Systems for Subcriticality Measurement						1		2	3	4	
4-3.4	In-Core Sodium Boiling Detection			1	2		3	4	5		6	
4-3.6	Steam Generator Leak Location				1		2			3	4	
4-3.8	Diagnostic Systems for Plant Components		1	2	3					4	5	6
4-4.1	Control System Studies					1						
4-4.2	Computer Control System Applications					1						
4-7.1/ 4-7.2	In- and Out-of-Core Neutron Sensors				1	2	3				4	
4-7.6	Sodium Flow Measurement in Large Pipes		1	2	3	4						
4-7.12	Fuel Channel Temperature and Flow Measurement Systems							1	2		3	
4-7.13	Design Guidance for Instrumentation Development			1		2		3			4	
4-7.14	Advanced In-Core Instrumentation										1	
4-7.15	Homologous Process Instrumentation Testing			1		2	3			4		

Task 4-2.1 Circuits for Neutron Sensors

- 1 Reference neutron detector systems selected
 2 Complete prototype circuit
 3 Complete testing of final design

TABLE 4-1 (Continued)

Task 4-3.1 Systems for Subcriticality Measurement

- 1 Select reference subcriticality determination technique
- 2 Install hardware in a reactor for proof test
- 3 Complete evaluation of test and recommend design modifications, if necessary
- 4 Complete hardware for LMFBR application

Task 4-3.4 In-Core Sodium Boiling Detection

- 1 Complete preliminary evaluation of current technology regarding sodium vapor generation and collapse characteristics
- 2 Select vendor to develop transducer
- 3 Complete detection system hardware for out-of-core tests
- 4 Complete out-of-core testing program
- 5 Complete hardware for in-core testing
- 6 Complete in-core testing and LMFBR hardware design

Task 4-3.6 Steam Generator Leak Location

- 1 Complete investigation of water-sodium interactions
- 2 Complete steam-generator test-rig data evaluations
- 3 Complete power plant test program
- 4 Complete design of LMFBR leak location system

Task 4-3.8 Diagnostic Systems for Plant Components

- 1 Complete identification of LMFBR components for which these diagnostic techniques are preferred
- 2 Complete preliminary theoretical analysis for selected components malfunction characteristics based on predetermined malfunction modes
- 3 Acquire initial test data on LMFBR components
- 4 Complete most test data acquisition programs
- 5 Complete theoretical analysis related to extrapolation to LMFBR conditions
- 6 Specify LMFBR diagnostic systems

Task 4-4.1 Control System Studies

- 1 Complete control system studies and specify the LMFBR control system

Task 4-4.2 Computer Control System Applications

- 1 Complete plant applications involving control system theoretical considerations

TABLE 4-1 (Continued)

Task 4-7.1/4-7.2 In- and Out-of-Core Neutron Sensors

- ① Complete analysis of breeding coatings
- ② Complete testing of detectors at EBR-II
- ③ Complete initial testing of detectors at EBR-II
- ④ Complete tests of prototype detectors at FFTF or Demonstration Plant

Task 4-7.6 Sodium Flow Measurement in Large Pipes

- ① Design large pipe flowmeter concepts, with priority given to electromagnetic type
- ② Fabricate flowmeters and initiate concept testing program
- ③ Evaluate calibration techniques
- ④ Design LMFBR prototype flowmeters

Task 4-7.12 Fuel Channel Temperature and Flow Measurement Systems

- ① Complete mockup test program
- ② Complete prototype probe hardware
- ③ Complete probe test program and specification for LMFBR probe

Task 4-7.13 Design Guidance for Instrumentation Development

- ① Complete first round functional requirements and specifications for instrumentation systems under development
- ② Update requirements
- ③ Update requirements
- ④ Complete development design guidance

Task 4-7.14 Advanced In-Core Instrumentation

- ① Complete advanced instrumentation evaluations

Task 4-7.15 Homologous Process Instrumentation Testing

- ① Initiate review of on-going programs and relate to LMFBR requirements
- ② Initiate hardware procurement for testing
- ③ Initiate testing program
- ④ Complete test program

TABLE 4-2

INSTRUMENTATION AND CONTROL PROGRAM OUTLINE

<u>Task</u>	<u>Title</u>	<u>Facility Requirements</u>	<u>Costs (\$1000)</u>		<u>Manpower (Man Years)</u>	<u>Schedule Fiscal Year</u>
			<u>Equipment and Facilities*</u>	<u>Operating or Engineering</u>		
4-2.1	Circuits for Neutron Sensors	None	150	250	8	71-73
4-3.1	Systems for Subcriticality Measurement	EBR-II, FFTF	150	350	11	74-77
4-3.4	In-Core Sodium Boiling Detection	FFTF	150	300	10	71-77
4-3.6	Steam Generator Leak Location	BTF rig **	150	400	12	70-77
4-3.8	Diagnostic Systems for Plant Components	Demonstration Plant	200	500	16	70-78
4-4.1	Control System Studies	None	None	refer to 2-1.7	refer to 2-1.7	70-72
4-4.2	Computer Control System Applications	None	None	refer to 2-1.7	refer to 2-1.7	70-72
4-7.1	In- and Out-of-Core Neutron Sensors	EBR-II, FFTF or Demonstration Plant	400	1000	21	71-77
4-7.2						
4-7.6	Sodium Flow Measurement in Large Pipes	Secondary Sodium Loop	100	200	7	71-74
4-7.12	Fuel Channel Temperature and Flow Measurement Systems	EBR-II, FFTF or Demonstration Plant	600	1400	39	74-78
4-7.13	Design Guidance for Instrumentation Development	None	None	450	12	70-78
4-7.14	Advanced In-Core Instrumentation	BTF	200	400	4	75-78
4-7.15	Homologous Process Instrumentation Testing	Secondary Sodium Loop and Laboratory Loop Facility	250	350	11	70-76

* Costs shown are for equipment or equipment modifications

** Testing of these instruments will be carried out in facilities provided for steam-generator development under Task Area 3-6.

operating in the counting, period or D. C. modes, for representative fast flux and gamma levels in- and out-of-core, preferably covering at least 10 decades in one range with digital output. The circuit packaging shall be standardized and a failure mode analysis satisfactorily completed.

Current circuit technology exists for both types of detectors, all three operating modes, the flux values and 10 decades in one range.

4.1.2 New Information Needed

The following information is needed:

- Basic decision information from Core Design and Safety Elements on the need for in-core neutron detectors,
- Basic decision information from plant design tasks as to display format and packaging concepts, and
- Design guidance information from Task 4-7.13.

4.1.3 Task Objectives and Effect of Non-Achievement

The task objectives follow:

- Perform conceptual design and analytical performance analysis of circuits for both ionization and self-powered detectors and recommendation which should be used to achieve the overall LMFBR objective. With current technology so close to the final objective, the probability of success is high. Non-achievement will result in double work effort in circuit designs for both types. No economic penalties to the plant would result unless failure to achieve either neutron sensor circuit resulted. This could necessitate designing for increased peaking factors in the

core. If the overall peaking factor were increased by 5%, the economic penalty would be approximately 0.2 mill/kW-h.

- Perform design, prototype construction, and prototype testing of a circuit for the selected detector type having a 10-decade range, adequate gamma discrimination and a digital readout of the specified format. Probability of success is very high.
- Complete the final design, packaging, testing, and safety analysis. Probability of success, with one recycle, is very high. Final testing should be done with the sensors of Task 4-7.1/7.2.

4.2 TASK 4-3.1—SYSTEMS FOR SUBCRITICALITY MEASUREMENT

4.2.1 Definition, Functional Requirements and Current Technology

This task is defined as encompassing the design, development and testing of a measurement system which may be used to estimate the degree of reactor subcriticality. The functional requirements of the measurement system are:

- Compatibility with normal refueling operations,
- No significant decrease in plant availability, and
- Provision of a sufficiently accurate and reliable estimate of subcriticality that it is a meaningful backup to normal operational and procedural controls aimed at preventing a refueling reactivity accident.

The current technology is generally as described in Task 4-3.1 in the Instrumentation and Control Element. General Electric evaluated the two-neutron detector, neutron-noise

cross-power spectral-density function as a vehicle for estimating the subcriticality of the FFTF during the course of the FFTF Backup Design work. The results were encouraging that this technique would be suitable for the LMFBR (Final FFTF Backup Design Summary Report).⁽⁴⁾ This technique does not depend on knowledge of the neutron source level, or require the movement of control rods or changes in the core power level to estimate reactivity.

4.2.2 New Information Needed

This task provides the supporting theoretical model developments described in Task 9-5.1.5 in the Program Plan Physics Element, and integration with Task 4-7.2 regarding sensors for the detection of neutrons out-of-core, core design, and plant refueling procedures. Circuits will be developed as necessary as part of this task. Development design guidance will be given by Task 4-7.13.

4.2.3 Program Objectives and Assessment of Effect of Non-Achievement

The objectives of this Task are as follows:

- Evaluate techniques for estimating subcriticality, with emphasis on the two-detector cross-correlation method,
- Design, fabrication and in-core testing of a monitoring system which is to include suitably shielded neutron detectors, data acquisition and data reduction means,
- Perform analytical support in the area of physics models and interpretation of the effect of the data acquisition and data reduction procedures, and
- Demonstrate that subcriticality can be estimated reliably and with sufficient accuracy to meet the functional requirements.

A medium to high probability of success is expected in meeting these objectives. The critical path item is a large, efficient, suitably shielded detector that will operate in the intense gamma field in or at the periphery of the core. This work is an element of the effort for eliminating the refueling accident from consideration. No estimate was made in the Task II and III work concerning economic penalties associated with a refueling accident so a quantitative assessment of non-achievement is not possible. A reliable subcriticality measurement system that does not depend on control rod movement may increase the time the reactor is available for power production. No estimate of this reduction in refueling time was made in the Task II and III work. Each hour the LMFBR is not on the line represents a penalty of approximately \$2000.

4.3 TASK 4-3.4—IN-CORE SODIUM BOILING DETECTION

4.3.1 Definition, Functional Requirements and Current Technology

This task is defined as encompassing the design, development and testing of a measurement system to detect sodium boiling and sodium vapor collapse within the reactor core utilizing acoustic techniques. The functional requirements of this measurement system can not be defined precisely at this time because of limited knowledge regarding sodium vapor generation characteristics under plant conditions. The general requirements for this system are:

- Operation in the vicinity of the fuel channel outlet (preferably as an in-core transducer to detect the acoustic boiling signals or the less desirable approach of a waveguide) and
- Time constant, reliability and sensitivity consistent with specifications developed under Task 4-7.13 to ensure that the system will be effective in detecting

local failures that result in local boiling before the failure has propagated extensively throughout the core.

The current technology is generally as described in Task 4-3.4 in the Instrumentation and Control Element. Discussions with vendors have indicated that the severe environmental conditions at the core outlet presently exceed the capabilities of commercially available acoustic transducers. The waveguide approach is within current technology but represents a serious complication of the core instrumentation system with respect to practical design and maintenance.

4.3.2 New Information Needed

This task requires integration with the related Safety Element studies (Task 2-1.4) regarding sodium superheating and the effect of local blockages within a fuel channel on local flow restriction. The detector and lead-out system must be compatible with the core design and maintenance procedures. All sodium boiling experimental work required to develop and proof test the sodium boiling detection system will be done as part of this Task. General Electric will provide guidance to a vendor who will develop the acoustic transducer to our specifications. Circuits will be developed as necessary for this Task. Development design guidance will come from Task 4-7.13.

4.3.3 Program Objectives and Assessment of Effect of Non-Achievement

The objectives of this Task follow:

- Design, fabricate and proof test a sodium boiling detection system based on acoustic principles; the preferred approach is an acoustic transducer located near the core outlet, an alternate is to use a mechanical waveguide to a less severe environment.
- Develop a commercial vendor source for the acoustic transducer.
- Perform boiling tests to the extent necessary in support of the first objective.

The probability of achieving the objectives is high. The degree of difficulty associated with the development of an acoustic transducer to operate near the core is higher than for the waveguide. However, the incentive is considerably greater to develop the transducer because of the simplifications in mechanical design and layout that result. At this time it can not be determined that a sodium boiling detector will be required instrumentation on the LMFBR in order to meet the safety criteria on which the design is based. It is more likely that future analysis and experiments will show that there is considerable inherent margin against rapid failure propagation within a fuel channel. However, because of the importance placed on core instrumentation as a critical part of the plant safety system, it seems reasonable that development of these sensors be carried along as a backup to the temperature and flow detectors discussed in Task 4-7.12.

4.4 TASK 4-3.6—STEAM GENERATOR LEAK LOCATION

4.4.1 Definition, Functional Requirements and Current Technology

This Task is defined as encompassing the design, development, and testing of a measurement system which may be used to locate a water-sodium leakage interaction in the LMFBR steam generator. The functional requirements of the reference measurement system which is based on acoustic techniques are:

- Will locate the approximate radial and transverse position of a leak within the steam generator,

- Will locate a leak before there is serious damage to tubes adjacent to the leaking tube, and
- Will have high reliability of operation and sufficient signal-to-noise resolution that spurious indications of a leak are eliminated for all practical purposes.

The current technology is in a relatively primitive state. The DFR had its first indication of a significant leak in one of 24 reactor vessel inlet nozzles on May 9, 1967. An extensive experimental program using acoustic techniques was required to finally locate the leak on November 27, 1967. The location of a leak implies that the leak has been detected. The measurement system would also complement detection systems developed for the Demonstration Plant based on more direct methods, such as detection of hydrogen. Tests will be required to establish the sensitivity of the acoustic techniques for detecting and locating small leaks.

4.4.2 New Information Needed

This Task requires integration with the steam generator development in Task 3-6.4 of the Components Element and the materials evaluation of tube wastage in the presence of leaks of the Sodium Technology Element. Circuits will be developed as needed for this Task. This Task is expected to build on a relatively insensitive Demonstration Plant acoustic location scheme. Development design guidance will be given by Task 4-7.13.

Experimental data are required on the frequency composition of typical water-sodium interactions in a steam generator. Analysis of the effect of the steam-generator geometry and material composition on the transmission and reflection characteristics of the acoustic signal created by the water-sodium interaction is required to correlate test data and allow extrapolation to LMFBR steam generator conditions.

4.4.3 Program Objectives and Assessment of Effect of Non-Achievement

The objectives of this Task are:

- Execute an experimental program that establishes the basic characteristics of the water-sodium interaction to guide selection of the requirements for the measurement system.
- Perform analysis to support the selection of the reference measuring system.
- Demonstrate by test where practical, and appropriate extrapolation of analysis that the measurement system based on acoustic principles developed under this Task will meet the functional requirements of the LMFBR steam generator.

A high probability of success is expected in meeting these objectives. Because of the expense of an LMFBR steam generator, it is not practical to have a spare generator available in the LMFBR to use when one unit is down because of a leak. On the other hand the LMFBR must exhibit a high availability consistent with other sources of power. Steam-generator repair can be completed in an efficient manner if the approximate location of the leak can be determined. The DFR experience where 6 months were required to locate an inlet nozzle leak would be intolerable. The proposed system would be capable of locating the leak region before it is necessary to reduce plant power to effect the repair. The more precisely the leak can be located the faster the generator can be returned to service.

Failure to develop an instrument which can detect a leak in the steam generator at an early stage could result in excessive downtime for repair. If a loop were removed from service, the power generating capability would be reduced. In the extreme case where multiple

loops are out of service and the plant is shut down, purchased or generated replacement power can cost from 0.5 to 2.0 mills/kW-h during the outage.

4.5 TASK 4-3.8—DIAGNOSTIC SYSTEMS FOR PLANT COMPONENTS

4.5.1 Definition, Functional Requirements and Current Technology

This Task is defined as encompassing the development of measurement systems and means of interpretation to diagnose the performance characteristics of select plant components throughout their life to ascertain that plant safety is not degraded due to deterioration in the action of these components. At the present time these systems are based on the fact that the component structures hear sound and vibrations due to the movement of internal parts. Analysis of the changes from the norm in these patterns provides a diagnosis of conditions within the structure. The functional requirements of the measurement systems are:

- They do not interfere with the routine operation or complicate the design of the components and
- They perform under remote conditions with automatic data acquisition, processing, interpretation, and display.

The current basic technology related to sensors, data acquisition and reduction is relatively advanced. There have been few applications to reactor plant components, however, because the basic technology was developed in non-nuclear related fields, such as aerospace and defense. An initial application to a General Electric control rod drive has shown promise. It is to be expected that sensors, and data reduction techniques will become more sophisticated and carry more information as experience with reactor components is acquired.

4.5.2 New Information Needed

This Task should be integrated with component development programs for the LMFBR so that data for both normal and abnormal operation can be acquired prior to installation of the components in the plant. The pre-installation data will increase the precision with which the plant component performance can be diagnosed. The basic technology is in place today. As experience is gained new techniques specially suited for the components will be developed. Development design guidance will be provided by Task 4-7. 13.

4.5.3 Program Objectives and Assessment of Effect of Non-Achievement

The objectives of this task follow.

- Identify the plant components in the LMFBR for which these diagnostic techniques are appropriate, for example, control rod drives, pumps, and valves.
- Determine the most suitable measurement system elements and apply them to components during their development and testing.
- Perform analysis of the component to predict malfunction characteristics, and compare with test data for select malfunction modes, thereby minimizing the amount of test data required to perform the diagnostic function accurately.
- Specify LMFBR plant diagnostic systems.

There is a high probability of success in meeting these objectives. These diagnostic systems hold the potential for increasing the probability that the plant safety characteristics will be maintained throughout the plant lifetime, because of improved preventive maintenance on plant components. These systems hold the

potential for increasing the plant availability because of maintenance procedures based on specific intelligence about these components. These systems represent an added value to an LMFBR on which it is difficult to place a dollar value because the failure characteristics of LMFBR components cannot be determined at this time. Increase in the plant availability by one percentage point would decrease plant energy costs by approximately 0.04 mill/kW-h.

4.6 TASK 4-4.1—CONTROL SYSTEM STUDIES

The practical intent of this Task is to assure that dynamic plant response and control models will be developed and solved on modern computers in order to determine the required control characteristics of the LMFBR. These elements of the Task have been incorporated as an integral part of Task 2-1.7 of the Plant Design Element. This combination is a necessity because plant design considerations completely outweigh purely control considerations in importance. It is our opinion that the advanced control methods development described in the Program Plans should be deferred until a clear justification is established based on the studies done as an integral part of Task 2-1.7. It is our assumption that industry can provide control systems to meet the design specifications that are developed in Task 2-1.7 and accordingly no LMFBR funding is required.

4.7 TASK 4-4.2—COMPUTER CONTROL SYSTEM APPLICATIONS

This Task will be accomplished by industry as required as part of normal plant engineering upgrading after commercial fast breeder nuclear power has been demonstrated. This Task requires no specific LMFBR funding. However, consideration to important practical aspects of this Task will be given during the plant transient simulations and control studies of Task 2-1.7 of the Program Plan Element.

4.8 TASK 4-7.1/4-7.2—SENSORS FOR DETECTION OF NEUTRONS IN AND OUT OF CORE

4.8.1 Definition, Functional Requirements, Current Technology

This Task is defined as the effort to develop, test and qualify a statistical quantity of neutron detectors covering the startup and power range requirements.

This Task is shown as being a dual one under the assumption that both in-core and out-of-core detectors will be used in the LMFBR.

Functionally required are fast neutron detectors, either ion-chamber or self-powered type to meet the in-core and out-of-core environmental conditions and to meet the life and reliability requirements.

Current technology is described in the Program Plan.

4.8.2 New Information Needed

The following information is needed:

- Ability of self-powered detectors to meet the environmental and sensitivity requirements,
- Ability of all types to accommodate sodium immersion,
- Life of self-regenerative (breeding) coatings in the fast flux spectrum, and
- Basic decision after appropriate evaluations by core design and safety tasks as to the proper mix of in-core and out-of-core detectors that should be utilized in the LMFBR.

4.8.3 Task Objectives and Effect of Non-Achievement

The task objectives are as follows:

- Supply and testing of a statistically important quantity of both in-core and out-of-core detectors for testing at an AEC facility. Probability of success is very high for out-of-core and for in-core sensors after one engineering recycle of the sensor design.
- Prototype design, construction and initial testing of self-powered detectors for the environmental conditions. Probability of success is high for out-of-core and medium for in-core sensors. Non-achievement would leave ion chambers as the alternative.
- Analytical analysis of breeding coatings in the flux spectrum. Probability of success is high. Non-achievement would require more frequent detector replacement and calibration.
- Final testing at AEC facility of type selected in full environment, including sodium immersion and failure mode analysis. Probability of success is very high with one engineering recycle.

Failure to succeed in developing in-core detectors would necessitate dependence on out-of-core units or bulk channel outlet flow and temperature systems. This could result in a deterioration of control or in-core power shapes and would necessitate increased peaking factor allowances. A 5% increase in the overall peaking factor would result in a power cost penalty of approximately 0.2 mill/kW-h.

4.9 TASK 4-7.6—SODIUM FLOW MEASUREMENT IN LARGE PIPES

4.9.1 Definition, Functional Requirements and Current Technology

This Task is defined as encompassing the design, fabrication and testing of a measurement system that will give reliable and accurate indication of sodium flow in pipes as large as 60 inches inside diameter. The functional requirements of the system are:

- Capability of long-term operation under LMFBR temperature and radiation conditions,
- Time constant consistent with requirements of safety systems, and
- Sufficient accuracy in individual applications to meet normal plant requirements over as wide a range as practical.

The current technology is generally as described in Task 4-7.6 in the Instrumentation and Control Element.

4.9.2 New Information Needed

There is a need for basic development of flowmeters suitable for larger size piping. Electromagnetic flowmeter designs that have been proven on intermediate size piping (10 to 14 inches inside diameter) may be impractical when scaled to larger systems.

4.9.3 Program Objectives and Assessment of Non-Achievement

The objectives of this Task are as follows:

- Design and development of an electromagnetic flowmeter for large piping;

exploration of velocity head measurement techniques and designs,

- Fabrication and concept testing of electro-magnetic and velocity-head flowmeters,
- Evaluation of calibration techniques, and
- Design prototype flowmeters for testing under Subtask 4-7.15, Homologous Process Instrumentation Testing.

Non-achievement of this Task would result in increased uncertainty in plant sodium flow characteristics, especially for below-rated flow conditions. Because the extent of this uncertainty is not known, the capital cost and plant availability cost can not be satisfactorily estimated at this time.

4.10 TASK 4-7.12—FUEL CHANNEL TEMPERATURE AND FLOW MEASUREMENT SYSTEMS

4.10.1 Definition, Functional Requirements and Current Technology

This Task is designed as encompassing the design, development and testing of measurement systems which may be used to determine the fuel channel outlet temperature and flow rate. The functional requirements of the system are:

- Compatibility with normal refueling and maintenance procedures,
- High accuracy, reliability and lifetime,
- Accommodation of expected environmental and operational conditions,
- Outlet sensor probe dimensions consistent with fuel channel outlet space available,
- Time constant less than one second, and

- Channel outlet measurement probe not an integral part of fuel channel.

The current technology is relatively highly developed regarding temperature measurement by means of thermocouples. The current technology of in-core flowmeters is in an embryonic stage today as described in the Program Plan. There are a number of potentially suitable flowmeter concepts. It is anticipated that both the redundant temperature sensors, and the flowmeter at the channel outlet will be contained within the same probe assembly for compactness, however alternate approaches would not be excluded if shown to be necessary. Channel inlet flow-measuring concepts such as the Venturi flowmeter described in the Task II report represent new approaches to in-core measurement that have potential for successful development.

4.10.2 New Information Needed

This Task receives design guidance from Task 4-7.13 of the Control and Instrumentation Element. Because of the strong dependence of the conceptual LMFBR design on the assumption that this in-core instrumentation system will be available, this Task has an interface with many areas such as core and plant design and safety. It is anticipated that bulk outlet fuel channel coolant temperature thermocouples and possibly a flowmeter will be developed for the Demonstration plants. The FFTF plans to develop an in-core outlet flowmeter. It is not clear at this time what degree of success will be achieved on the FFTF flowmeter. However, because of the importance of this in-core instrumentation to the LMFBR, it will be assumed that additional development will be required beyond what is accomplished for these earlier plants. Inlet flow-monitoring schemes must be evaluated with special attention to their effect on core design, internals designs, and maintenance. Task 4-7.14 provides for the possibility of more advanced instrumentation concepts to supplement this Task.

4.10.3 Program Objectives and Assessment of Effect of Non-Achievement

The objectives to this Task follow:

- Based on the current technology at the start to this Task, design, develop, and proof test a channel outlet temperature and flow probe and, if further evaluation continues to show promise, an inlet flow system, and
- Perform analytical, testing and mock-up support as required to accomplish the first objective.

A high probability of success is expected in meeting these objectives with regard to the outlet temperature and flow probe. The inlet flow monitor has a low probability of meeting all design objectives. The incentives for achieving the advanced performance characteristics have been delineated throughout this report. Instrumentation systems, of which this Task is a prime example, are required to support the selection of the conceptual LMFBR design. An estimate of the cost savings that result from the selection of this LMFBR design, rather than one which does not depend on instrumentation to increase the overall plant safety, is 1.0 mill/kW-h.

4.11 TASK 4-7.13—DESIGN GUIDANCE FOR INSTRUMENTATION DEVELOPMENT

4.11.1 Definition, Functional Requirements and Current Technology

This Task is defined as encompassing the engineering analyses and evaluations required to support the selection of functional requirements and design specifications for in-core instrumentation systems. The functional requirement of this Task is to ensure that the basis for the development of instrumentation systems is well founded. This Task focuses the development of the instrumentation systems to meet the plant design

and safety considerations of the LMFBR. The current technology for this guidance activity is adequate to accomplish the desired results.

4.11.2 New Information Needed

This Task must be flexible to ensure that the proper instrumentation systems are developed in a timely manner for the LMFBR. This Task requires information relevant to providing this guidance from a number of interfacing disciplines such as the Core and Plant Design, and Safety Elements.

4.11.3 Program Objectives and Assessment of Effect of Non-Achievement

The objectives of this Task are as follows:

- Evaluate the needs and objectives of the instrumentation systems on a continuing basis,
- Update instrumentation design specifications to meet LMFBR needs, and
- Ensure that the proper instrumentation systems to meet the LMFBR needs are developed in a timely and efficient manner.

A high probability of success is expected in meeting the objectives of this Task. This Task is difficult to assign a cost incentive. However, experience has amply illustrated the need for a continuing reassessment and focusing of development goals to meet the needs for which the development was originally intended.

4.12 TASK 4-7.14—ADVANCED IN-CORE INSTRUMENTATION

4.12.1 Definition, Functional Requirements and Current Technology

This Task is defined as encompassing the evaluation and development of advanced

instrumentation systems not otherwise discussed. New and promising schemes presently unknown or at a very low state of development should be considered during the next 10 years. For example, it may be practical to detect acoustic emissions associated with cladding failures. Functional requirements can not be given. However, it is known that more sophisticated in-core instrumentation or alternate ways of detecting core performance characteristics might be shown to be desirable. The current technology on all advanced concepts is necessarily low at this time.

4.12.2 New Information Needed

No meaningful definition of the new information needed can be given at this time. However, provision for this information must be included in this plan.

4.12.3 Program Objectives and Assessment of Effect of Non-Achievement

The objectives of this Task are to provide the LMFBR with better or less expensive in-core instrumentation systems.

This work has a low priority at this time, but may become of real value in light of new concepts and techniques that may arise in the future. This Task should have the flexibility to be initiated sooner than shown in Table 4-1, if new information warrants development of new concepts.

4.13 TASK 4-7.15—HOMOLOGOUS PROCESS INSTRUMENTATION TESTING

4.13.1 Definition, Functional Requirements and Current Technology

This Task is defined as encompassing the procurement, installation and testing of process instrumentation for:

- Large pipe sodium-flow measurement device,

- Sodium pressure measurement and transmittance,
- Sodium-level measurement and transmittance,
- High-temperature strain gages, and
- High-temperature instrumentation connectors for in-sodium service.

The functional requirements of this Task are that these instrumentation systems and components will be tested under conditions sufficiently realistic and representative of the LMFBR that the items can be considered standard for plant installation.

The current technology on these items is generally as listed in the Instrumentation and Control Element.

4.13.2 New Information Needed

It is assumed that the technology will be developed in support of the FFTF and Demonstration Plant programs in each of the areas described above. This Task extends this technological base to the more demanding LMFBR requirements. Improvements will be made for these items. Evidence of satisfactory performance under representative LMFBR conditions will be obtained by this test program.

4.13.3 Program Objectives and Assessment of Effect of Non-Achievement

Objectives are to:

- Procure process instrumentation hardware,
- Test under representative LMFBR conditions, and
- Demonstrate that hardware will perform satisfactorily under simulated plant service conditions.

A high probability of success in meeting these objectives is estimated. Non-achievement may delay plant startup, decrease plant availability, and decrease plant reliability.

SECTION V

ELEMENT 5 - SODIUM TECHNOLOGY

This section defines the sodium-technology information needed for the development of the 1000 MWe reactor described in the Task II and III reports. In several areas, development programs will be described which are not directly discussed in sections of the Tasks II and III reports, because these areas were not specifically covered or were only marginally covered in the study. It is felt, however, that a particular need exists for this work.

This element is very dependent on the continuation of the PA-15 programs, on the work carried out in the Fuels and Materials Element, and on the success of the Demonstration Plant research and development program. The program is in general designed to extend the technologies beyond that required for the Demonstration Plant. Advances are needed in order to make the 1000 MWe plant economically competitive with other sources of electrical energy. The advances required in sodium technology, materials development, and plant operations come about primarily because of higher-system temperatures, a decrease in design margins, and a distinct need for more efficient operation.

Although it is not specifically covered in Sodium Technology of the LMFBR Program Plan, Volume 5, all materials development for the steam generator, for the intermediate heat exchanger, and for that material development which does not involve neutron irradiation will be considered in this element.

Priorities for performing the work could change with time, depending on the progress made in the various technologies; however, the areas which appear to be most critical and require the longest lead time would be Task Area 5-1, Materials Compatibility and Materials

Development. A high priority effort is also needed in Task Areas of Sodium Purification Technology.

Task Area 5-11, Engineering Mock-up, and Task Area 5-8, Plant Operation, are given lower priority because they depend on the results of the high priority effort or because they do not require as much lead time.

Table 5-1 has the milestone schedule for work performed in this Element. The programmatic needs are shown in Table 5-2. More detailed discussions are included in the individual tasks.

5.1 TASK AREA 5-1—PLANT SODIUM SYSTEM STUDIES

5.1.1 Definition, Functional Requirements, and Current Technology

Recent data obtained in flowing sodium loops revealed potentially serious problems which are likely to affect the performance of major components in the 1000 MWe LMFBR. These are: (a) metallic and particulate deposition on heat transfer surfaces of the intermediate heat exchanger (IHX), (b) radioactive system contamination from species, e.g., Mn, transported under thermal gradients in the coolant stream. These problems are likely to increase in intensity and extent as the system temperatures approach 1150°F, which is the projected IHX inlet temperature for the 1000 MWe plant.

It has been estimated that degradation of heat transfer in the IHX from these deposits may require as much as 100% increase in heat transfer areas. The same degradation may occur to a lesser extent in certain areas of the

TABLE 5-1

SODIUM TECHNOLOGYMILESTONE SCHEDULESPlant Sodium System Studies

		Government Fiscal Year										
		1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Task Area 5-1 Radioactive (Primary) Sodium Systems Study	①		③	④	⑥	⑦		⑧				
	②			⑤								
Task Area 5-1 Non-Radioactive (Secondary) Sodium System Studies	①		③	④	⑦	⑧		⑨				
	②			⑤								
				⑥								

MilestonesTask Area 5-1 Radioactive (Primary) Sodium System Studies

- ① Initiate loop designs.
- ② Complete loop designs.
- ③ Initiate sodium loop fabrication.
- ④ Complete loop fabrication.
- ⑤ Initiate primary chemistry studies and deposition rate measurements.
- ⑥ Initiate a study of radioactive transport dynamics and abatement.
- ⑦ Complete primary studies and deposition rate measurements.
- ⑧ Complete radioactive transport work.

Task Area 5-1 Non-Radioactive (Secondary) Sodium System Studies

- ① Initiate (phase I and phase II) secondary sodium loop design. (See Element 3, Components)
- ② Complete (phase I and phase II) secondary sodium loop design.
- ③ Initiate phase I secondary sodium loop fabrication.
- ④ Complete phase I secondary sodium loop fabrication.
- ⑤ Initiate study of deposition in-plant secondary system loop.
- ⑥ Initiate study of on-line sensors and monitors in secondary system loop.
- ⑦ Complete phase II secondary loop design and fabrication.
- ⑧ Initiate (1000 MWe condition) deposition studies, materials evaluations, and on-line sensor studies in phase II secondary sodium loop.
- ⑨ Complete phase II secondary sodium loop studies.

TABLE 5-1 (Continued)

Materials Evaluation

		Government Fiscal Year										
		1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Task Area 5-1	Sodium Service Evaluation	①	②		③	④	⑤					
Task Area 5-1	Steam/Water Service Evaluations	① ②			③	④	⑤					

MilestonesTask Area 5-1 Sodium Service Evaluations

- ① Initiate laboratory studies of carburization resistant austenitic alloys.
- ② Initiate evaluation of carburization resistant, austenitic steels decarburization resistant ferritic, and bimetallic steels in phase I's secondary sodium loop.
- ③ Complete laboratory studies of carburization resistant austenitic and decarburization on resistant ferritic and bimetallic steels.
- ④ Initiate induction of carburization resistant and decarburization resistant steels in phase II's (1000 MWe conditions) secondary sodium loop.
- ⑤ Complete phase I and II secondary sodium loop alloy evaluations.

Task Area 5-1 Steam/Water Service Evaluations

- ① Select candidate austenitic ferritic and bimetallic alloys for steam/water evaluation.
- ② Initiate laboratory steam/water corrosion evaluations of candidate austenitic ferritic bimetallic alloys.
- ③ Complete laboratory steam/water corrosion evaluations.
- ④ Initiate evaluation of candidate austenitic, ferritic, and bimetallic alloys in phase II (1000 MWe conditions) secondary sodium loop.
- ⑤ Complete (1000 MWe conditions) secondary sodium loop steam/water evaluations of candidate alloys.

TABLE 5-1 (Continued)

Sodium Purification Technology

		Government Fiscal Year										
		1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Task Area 5-0	Purity Design Basis	①		②		③						
Task 5-2.3	Cold Trap Design Criteria	①	②									
Task Area 5-4	On-Line Sensors (O, H, C, N)		①			②						
Task 5-4.4	On-Line Monitors	①			②							
Task Area 5-5	Broadband Qualitative Chemical Analysis Technique				①	②						
Task Area 5-5	Backup Laboratory Methods		①							②		
Task Area 5-10	Proof Test of Integrated Purification Systems		①	②	④	⑤	⑦				⑧	⑨
					③		⑥					

MilestonesTask Area 5-0 Purity Design Basis

- ① Initiate study of sodium purity design basis.
- ② Complete study.

Task 5-2.3 Cold Trap Design Information

- ① Initiate development of cold trap design information.
- ② Complete test facility.
- ③ Complete test to establish design information.

Task Area 5-4 On-Line Sensors (O, H, C, N)

- ① Develop and evaluate on-line sensors.
- ② Complete evaluation.

Task 5-4.4 On-Line Monitors (Fission Products, Gas, Fuel)

- ① Initiate development of on-line monitor.
- ② Complete development.

Task Area 5-5 Broadband Qualitative Chemical Analysis Technique

- ① Evaluate broadband qualitative analysis instrument.
- ② Complete evaluation.

Task Area 5-5 Backup Laboratory Methods

- ① Initiate development of backup laboratory methods.
- ② Complete development.

TABLE 5-1 (Continued)

Task Area 5-10 Proof Test of Integrated Purification Systems

- ① Initiate design of test facility.
- ② Initiate construction of test facility.
- ③ Initiate developmental test phase.
- ④ Complete design.
- ⑤ Complete construction.
- ⑥ Complete developmental test phase.
- ⑦ Initiate operational test phase.
- ⑧ Complete operational test phase in engineering mock of loop.
- ⑨ Complete special test phase.

		Plant Operation										
		Government Fiscal Year										
		1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Task 5-8.3 Plugs and Deposits		①		④		⑥						
				②		⑦						
						③			⑤			
Task Area 5-8 Repair of Contaminated Sodium Systems					①			④				
				②			⑤					
				③				⑥				
		Milestones										

Task 5-8.3 Plugs and Deposits

- ① Initiate program aimed at developing methods of detecting plugs and deposits in operating systems.
- ② Initiate program to determine the composition of plugs and deposits in operating systems.
- ③ Develop operating procedures for removal of plugs and deposits from operating systems.
- ④ Test method for detecting plugs or deposits in operating systems.
- ⑤ Complete program on plugs and deposit removal.
- ⑥ Complete program for plug and deposit detection.
- ⑦ Complete program to determine composition of plugs and deposits.

Task Area 5-8 Repair of Contaminated Sodium Systems

- ① Develop methods of repairing large contaminated systems.
- ② Develop methods for dealing with large sodium spills.
- ③ Develop methods for disposal of bulk radioactive sodium.
- ④ Recommend procedures for repairing large contaminated systems.
- ⑤ Recommend procedures for dealing with a large sodium spill in sodium.
- ⑥ Recommend procedures for disposal of radioactive sodium.

TABLE 5-1 (Continued)

Mass Transfer Engineering Mock-Up

	Government Fiscal Year										
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Task Area 5-11 Mass Transfer Engineering Mock-up	①	②	③	④	⑤				⑥		

MilestonesTask Area 5-11 Mass Transfer Engineering Mock-Up

- ① Initiate design of test facility.
- ② Begin construction of mock-up facility.
- ③ Complete design of test facility.
- ④ Complete construction of loop facility.
- ⑤ Start 30,000-hour test.
- ⑥ Complete 30,000-hour test.

Cover Gas Atmosphere

	Government Fiscal Year											
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
Task 5-7.5 Purification Methods for Recycle Cover Gas			①	②			③	④				

MilestonesTask 5-7.5 Purification Methods for Recycle Cover Gas

- ① Define functional requirements for cover gas atmosphere from Task 2-1.2.
- ② Complete comparison of helium and argon cover gas.
- ③ Initiate tests of cover gas system based on preliminary design work conducted under Task 2-1.11.
- ④ Complete tests of cover gas system.

TABLE 5-2

SODIUM TECHNOLOGY PROGRAM OUTLINEPlant Sodium System Studies

<u>Task</u>	<u>Title</u>	<u>Facilities Required</u>	<u>Cost (\$1000)</u>		<u>Manpower (Man Years)</u>	<u>Schedule Fiscal Year</u>
			<u>Equipment and Facilities</u>	<u>Operating and Engineering</u>		
5-1	Radioactive (Primary) Sodium System Studies					
	Sampling loop design fabrication	Modify sodium loops*	200	200	5	1970 to 1971
	Primary chemistry and deposition studies	Laboratory facilities and sampling loops	100	700	20	1972 to 1976
	Radioactive transport studies	Laboratory facilities and sampling loops	100	200	5	1973 to 1976
5-1	Non-Radioactive (Secondary) Sodium System Studies					
	Phase I secondary sodium loop design and fabrication	Secondary sodium loops**	200	100	2	1970 to 1972
	Secondary chemistry and deposition studies	Laboratory facilities and experimental loop	100	600	17	1972 to 1976
	Phase II, secondary sodium loop design and fabrication	Laboratory facilities and experimental loop	1,500	700	20	1970 to 1973
	On-line sensor and monitor studies	Laboratory facilities and experimental loop	200	200	5	1972 to 1974

* Sampling loops on the 350 MWe Demonstration Plant may be used.

** A loop in the Secondary Sodium System of the 350 MWe Demonstration Plant may be used.

TABLE 5-2 (Continued)

Task	Title	Facilities Required	Cost (\$1000)		Manpower (Man Years)	Schedule Fiscal Year
			Equipment and Facilities	Operating and Engineering		
5-1	Sodium Service Evaluation					
	Laboratory carburization and decarburization studies	Laboratory loop facilities	100	400	11	1970 to 1973
	Phase I loop carburization and decarburization studies	Secondary sodium loops	100	300	9	1972 to 1975
	Phase II loop carburization and decarburization studies	Secondary sodium loops	200	200	5	1973 to 1975
5-1	Steam/Water Service Evaluation					
	Laboratory steam/water corrosion studies	Laboratory facilities	100	200	5	1970 to 1973
	Phase II loop carburization and decarburization studies	Secondary sodium loops	500	300	9	1973 to 1975
Sodium Purification Technology						
5-0	Purity Design Basis	None	None	70	3	1970 to 1971
5-2	Cold Trap Design Information	Double loop facility (Below)	None	1200	28	1970 to 1974
5-4	On-Line Sensors Development and Evaluation (O, H, C, N)	See large proof test loop below	None	400	6	1971 to 1976
5-4.4	On-Line Monitors Development (Fission Products, Gas, Fuel)	Bench Test Equipment	300	600	8	1970 to 1972
5-5	Broadband Qualitative Analysis Evaluation	See large proof test loop below	None	300	12	1973 to 1974
5-5	Backup Laboratory Methods	Chemistry Laboratory area	200	300	5	1971 to 1977
5-10	Proof Test of Integrated Purification Systems	Engineering Mockup-primary Engineering Mockup-secondary				
	Test Facility		3,000	None	22	1971 to 1973
	Developmental Test Phase	See above	None	370	13	1974 to 1975
	Operational Test Phase	See above	None	1500	45	1975 to 1978

TABLE 5-2 (Continued)

Plant Operation						
Task	Title	Facilities Required	Cost (\$1000)		Manpower (Man Years)	Schedule Fiscal Year
			Equipment and Facilities	Operating and Engineering		
5-8.3	Plugs and Deposits in Operating Systems					
	Detection (5-8.3)	Modify existing facility (77-78)	250	250	9.5	1971 to 1974
	Composition (5-8.3.2)	Modify existing facility (77-78)	250	250	9.5	1973 to 1974
	Removal (5-8.3.3)	Modify existing facility (76-77)	225	285	10.5	1975 to 1977
5-8	Repair of Contaminated Sodium Systems					
	Repair procedures	Remote equipment Fabrication facilities Auxiliary sodium system	800	700	25.0	1974 to 1976
	Radioactive sodium spill	Lined trenches, Disposal and Safety Systems Building	500	450	17.0	1973 to 1975
	Radioactive sodium disposal (5-8.4)	Existing and remote equipment	700	450	17.0	1973 to 1976
<u>Cover Gas Atmosphere</u>						
5-7.5	Purification					
	<u>Methods for Recycle Cover Gas</u>					
	Define functional requirements	None	None	20	1	1972
	Compare cover gas systems	None	None	80	2	1973
	Perform cover gas tests	Laboratory shop	200	500	15	1976 to 1977

steam generator as the temperature of operation increases beyond 1000°F. Thermal shocking or thermal dissolution (i. e., by heating the Na above deposition temperature) to remove deposits may damage tube welds in the case of the former and may be too slow in the case of the latter.

Corrosion and subsequent mass transfer of highly radioactive material may create serious problems from the standpoint of maintenance and repairs during the reactor operations. Present mass transfer studies of corrosion and deposition have indicated that there is preferential dissolution of some elements from the stainless steel in flowing sodium. In a fast flux, several isotopes of Cr, Fe, and Mn, with relatively long half lives, will be formed. High density depositions of these active elements in sections where routine maintenance is required can be of acute concern. Ways should be sought to monitor the deposition pattern, and promote the deposition of critical isotopes in non-critical areas.

It is required to study the deposition of solids and radioactive species in primary circuit components (which presently are planned to be constructed from 300-series stainless steels) under environmental conditions which are described initially in the Task II report and which will be revised in fiscal year 1972.

It is necessary to develop in-line carbon detectors, carbon sources, and sinks to minimize the steam-generator decarburization and IHX carburization. It is required to study the control of carbon movement (through the use of inhibitors) in sodium systems for use in the 1000 MWe plant.

5.1.2 New Information Needed

The new information needed to overcome the problem areas identified above follows.

- In depth knowledge of flowing sodium system chemistry including carbon movements and carbon detection in sodium (batch chemistry and on-line detection methods), under conditions that simulate temperatures, flow, and hydraulic conditions typical of the 1000 MWe plant.
- Verification of adequate performance of newly developed analytical and monitoring techniques in large experimental loops. See Sub-Tasks 5-1.2.2, 5-1.2.3, 5-5.7.1, 5-6.1.1, 5-6.1.3.
- Measurements of the effect of primary and secondary circuit deposits and the effects of secondary circuit carburization upon the heat transfer characteristics of tubes including methods to minimize or eliminate deposit formation. See Sub-Tasks 5-1.2.1, 5-1.2.6, 5-5, 7.1.
- Understanding of the character, movement, and quantitative magnitude of radioactive isotopes associated with deposition products, including methods for removing radioactive isotopes at convenient locations in the primary coolant circuit. See Sub-Tasks 5-1.1.5, Task 5-3.

5.1.3 Discussion

The studies must include sodium chemistry, movement of carbon under the influence of temperature gradients, on-line monitoring methods to detect progress of deposition or carburization. In order to obtain the higher temperatures envisioned for the 1000 MWe reactor, the direction of development work must proceed along the lines of focusing attention to methods of preventing the particulates from depositing on heat-transfer surfaces.

Mechanical separation methods such as hydro-clones, magnetic collectors, micro-metallic filters, or centrifuges should be studied for use prior to sodium entry into the IHX and steam generator. Other methods of separating these particulates from the sodium prior to entrance to the IHX and steam generator should also be investigated. The corrosion process itself, which is the source of the particulates, might be decreased by the addition of inhibitors or the use of more corrosion resistant alloy grades.

Movement of radioactive species must be studied by means of tracer techniques. The objective will be to define the character of deposits, study the system conditions which lead to accumulation of troublesome species, and find ways of removing them from the coolant stream at convenient locations.

To make the information meaningful and easier to extrapolate to reactor systems, the following approaches are suggested.

- Fabrication and operation of a special secondary plant loop to duplicate and/or simulate plant chemistry and geometry (for example, equivalent carbon steel to stainless steel surface ratios). Operate the loop and study secondary system chemistry as a function of temperature of operation, flow rates, and chemical additions.
- An alternative approach is that a special loop be attached to one of the Demonstration Plants. The loop will be a valved off section which will be operated along with the plant, using coolant from the secondary circuit. Thus, the chemistry conditions of an actual plant will be used for the studies proposed. Heaters may have to be used to achieve 1000 MWe plant thermal conditions. For studies that require slight departures from the available plant chemistry, separate provisions must be

made to have the chemical species added for study. The sodium can either be returned to the plant main stream (after appropriate cleanup) or discarded after use.

5.1.4 Task Objectives and Assessment of Non-Achievement

The following objectives are also discussed above.

- Study flowing sodium chemistry to ascertain the exact conditions as to how the deposits form, what properties they have, and how they can be controlled.
- Study the transport of carbon in flowing sodium and the rates at which carburization of IHX takes place as a function of temperature, time, and hydraulic conditions.
- Study transport, distribution, and quantitative deposition of radioactive species.

The probability of achieving the objectives of these tasks are high. However, if the objectives of the tasks are not achieved, the functional requirements will have to be made less stringent, based on the information obtained during the program. If the core outlet temperature has to be reduced from 1150°F to 1050°F, the economic penalty is estimated to be in excess of 0.10 mill/kW-h. If the IHX has to be replaced, the cost would be \$5.8 million for the IHX plus downtime costs.

5.1.5 Facilities

At present, chemistry of flowing sodium systems is being studied by GE, PNL, and others. Most systems are of an exploratory nature, small in size and of limited capacity. It is assumed that these facilities will provide the basic understanding needed to focus the experiments proposed for the 1000 MWe plants.

Primary sodium loop facilities and a large experimental secondary sodium loop are needed. It is possible that such loops could be attached to the 350 MWe Demonstration Plant to accomplish the proposed work. The cost of the equipment and facilities shown in Table 5-2 are based on this premise. Alternatively, at increased cost, independent loops could be provided, or other facilities might be modified such as those at LMEC or PNL.

5.1.6 Task 5-1 Materials Evaluation

5.1.6.1 Definition, Functional Requirements, and Current Technology

The demonstration plant will probably feature intermediate heat exchangers and secondary steam generators constructed from materials which have been developed through 1970. For example, the intermediate heat exchangers may be built from an austenitic stainless steel (304 or 316), thus, providing a monometallic primary loop. A variety of materials are being considered for the steam generators. These include Type-321 stainless steel (France), 2-1/4 Cr - 1 Mo - 1 Nb (Britain and Germany), and 2-1/4 Cr - 1 Mo (United States). The choices of materials for the 1000 MWe plant, IHX's, steam generators, and reheaters will similarly be based on the relative merits of existing ferritics and austenitics.

Stainless steels of the 300 series are ideal both from strength and fabrication standpoints, but are susceptible to aqueous stress corrosion cracking and carburization from both primary and secondary sodium coolants. Non-stabilized ferritics (e.g., 2-1/4 Cr 1 Mo steel) have lower high temperature strength than austenitics and suffer from decarburization and a 30% loss in strength when exposed to sodium. Stabilized grades of 2-1/4 Cr 1 Mo are now being studied for possible steam-generator application, but these materials have not been accepted by the ASME code. In

addition, practical experience with stabilized alloys for large structures and in large quantities is lacking.

A limited amount of effort has been expended in developing bimetallic materials in which the ferritic materials (which are resistant to aqueous stress corrosion) are exposed to steam, water, or both. The austenitics (which are stronger and more resistant to decarburization and to sodium water reactions) are exposed to sodium. At present, insufficient information on welding technology and mechanical properties after long thermal treatments (i.e., interface carbide formation and separation) limit the usefulness of the bimetallic materials.

In summary, there exist two major material problems which must be solved for the 1000 MWe plant. These follow.

- Carburization of the intermediate heat exchanger
- Decarburization of steam generator (particularly the superheater) components.

Carburization of the IHX tube walls will cause severe loss of ductility and, thus, reduce the ability of the unit to absorb operating thermal shocks. Because of plant licensing limitations, reduction of the IHX capability to sustain normal thermal shock may mean plant derating or replacement of the units.

Decarburization of the steam-generator components will require excessive section thickness in order to maintain adequate creep strength at operating temperatures. A solution to the steam-generator decarburization problem will reduce (by 1/2) the carburization problem of the IHX.

The following material requirements must be met to have satisfactory options in

the choice of materials for the 1000 MWe plant.

- Austenitic materials resistant to carburization must be evaluated and developed for reference 1000 MWe IHX conditions.
- Ferritic materials resistant to decarburization must be evaluated from 1000 MWe evaporator conditions. Stabilized ferritics which have greater high temperature strength and greater resistance to general steam corrosion must be developed and inducted for the referenced superheater and reheater conditions.
- Austenitic materials which have shown promising resistance to aqueous stress corrosion must be evaluated for potential steam generator and reheater applications.
- Bimetallic materials which by design are resistant to stress corrosion and decarburization must be evaluated for reference steam generator and superheater conditions.

5.1.6.2 New Information Needed

Present day materials technology, which can be applied directly to steam generators for FCR's, dictates the need for selecting alloys (a) that have and retain adequate strength at temperatures in sodium, (b) that are resistant to stress corrosion cracking (steam side), (c) that can be fabricated by familiar techniques, and (d) that can be coded.

The work required which will permit improved materials performance as sodium temperature increases to 1070°F follows.

- Evaluate the carburization resistance of austenitic stainless steels by choice of proper alloy chemistry (i. e., high-silicon content).

- Evaluate the decarburization resistance of ferritic alloys such as 2-1/4 Cr 1 Mo by stabilizing the carbon with suitable alloy additions (e. g., N⁶).
- Improve the high temperature mechanical properties of ferritic alloys for superheater and reheater applications.
- Evaluate austenitic alloys which are most familiar to industry, and for which there exists a high probability for resistance to stress corrosion attack.
- Evaluate the decarburization and stress corrosion resistance of bimetallic ferritic austenitic alloys.

5.1.6.3 Discussion

The relationship to present programs follows.

- Ferritic Steels: Work on stabilized chromium steels has been going on in Great Britain, France, Germany, Sweden, as well as in this country, especially in General Electric, Atomics International, Foster-Wheeler, and B&W. Facilities to study decarburization rates (or resistance to decarburization), long term mechanical properties, and fabricability are needed to develop a practical technological base which will enable industry to use these materials for high temperature steam-generator components.
- Austenitic Steels: At the present time, there is little systematic effort in this country to develop materials for sodium steam generators. There is some effort in industry to develop austenitic stainless steels which are resistant to stress-corrosion attack. This effort is largely supported by the metals industry, such as INCO, U.S. Steel, Crucible Steel, as

well as theoretical work in the Universities. Preliminary data show that appropriate alloying modifications of the 300-series stainless steels are possible to enhance the resistance of these alloys to stress corrosion. It is proposed that this program be based upon these findings and that this program will not attempt to undertake development of stress corrosion information. In short, apply what is known today so as to reduce to practice the methods suggested by basic research.

At present, there are no systematic studies in progress to investigate the carburization-resistant austenitic stainless steels. Such steels do exist and have been widely used in carburizing fuel applications. The various alloys which have previously been developed for carburization resistance should be evaluated in carburizing sodium environments.

- Bimetallic Alloys: Extensive systematic evaluation of bimetallic (ferritic-austenitic) alloys for breeder reactor applications have not been made. These materials should be evaluated with respect to weld ability and resistance to interfacial embrittlement.

5.1.6.4 Task Objectives and Assessment of Non-Achievement

The following specific information will be needed to develop materials for the steam generator according to design needs and desired performance.

- Develop a basic understanding of the metallurgy involved in using stabilized ferritics, low-carbon ferritics, austenitic stainless steels, or bimetallic alloys for 1000 MWe application. This information is needed to understand processing methods for alloy fabrication,

weld ability and stability in sodium and steam environments (corrosion).

- Determine mechanical and chemical properties needed to design the components stress to rupture, creep, and tensile properties at temperatures of 1000°F to 1200°F, and decarburization resistance and stress corrosion resistance, Na or steam at 1000°F to 1200°F.
- Obtain sufficient information on improved materials to permit the issuance of ASME coding.

The probability of achieving the objectives of these tasks is high. If, however, the objectives are not achieved, the functional requirements will have to be made less stringent based on the information obtained during the program. A reduction of 100°F in the maximum temperature of the secondary system would result in an economic penalty of about 0.10 mill/kW-h. If the reheaters and superheaters had to be replaced, the cost would be over \$7.3 million plus downtime costs.

5.1.6.5 Facilities

Approximately 1 million dollars in new facilities and equipment will be needed for the above tasks. Major loop facilities could be attached to the secondary coolant of the 350 MWe demonstration plant and would be used for 1000 MWe new-condition evaluations of materials. Modest increases in laboratory facilities will be required.

5.2 TASK AREAS 5-0, 5-2, 5-3, 5-4, 5-5, 5-10-SODIUM PURIFICATION TECHNOLOGY

5.2.1 Definition, Functional Requirements, and Current Technology

The succeeding material is related to the development of purification systems for the LMFB primary and secondary sodium

cooling systems. Two systems are involved: one for the primary circuit and one for the secondary. They both must perform the same function, that is maintaining the desired system purity; however, they operate under different conditions and must remove different types of impurities. The technical area considered covers four task areas from the LMFBR Program Plan: Task Area 5-2, Sodium Purification; Task Area 5-3, Fission Products in Sodium; Task Area 5-4, On-Line Monitoring Methods; and Task Area 5-5, Laboratory Methods.

The present state of sodium purification technology is rudimentary and is, in general, limited to relatively small-size systems. The problem is twofold: first, it is necessary to devise means of continuously monitoring the level of all impurities in the sodium system, and secondly, it is necessary to devise means of removing them down to a tolerable concentration. Promising development efforts are underway on four types of monitors: the general impurity monitor, oxygen activity meter, hydrogen activity meter, and the carbon activity meter. Except for the analysis of one of the expended Fermi cold traps, there has been little effort reported on cold trap development. With the exception of the PFR, there has been no development effort concerning the unique problems and requirements related to a cold trapped based purificator system designed to adequately provide the high purity rigs such as that planned for the 1000 MWe.

The current technology is also summarized in the LMFBR Program Plan, Volume 5, Page 5-2. Briefly SIR, EBR-II, SRE, and Fermi, and Hallam in the U.S. have demonstrated plant size operation in the range from 500°F to 900°F steam temperature with a maximum sodium temperature of 1000°F. Present development programs are aimed at developing mass transfer, carbon transfer, and material compatibility for various parts of the systems at high temperatures up to

1300°F sodium, and 1050°F to 1100°F steam temperatures.

It is expected that several demonstration-size sodium-cooled fast breeder plants will be built for operation between 1970 and 1980. It is expected that these will operate with reactor outlet sodium temperatures of approximately 1000°F to 1050°F and peak steam temperatures in the range from 900°F to 950°F. These plants will be in operation and will have accumulated several years of operation prior to the start of the 1000 MWe design.

5.2.2 New Information Needed

Development of on-line general impurity meters, hydrogen activity, oxygen activity, and carbon activity meters must be pursued to produce successful industrial type designs. In addition to these developments, the following additional work is required.

- The development is needed of design criteria for cold traps with respect to different impurities expected in the primary and secondary systems.
- A design basis is needed for the primary system impurity control of fission products released from failed fuel. (What elements are released to the system? What form do they take and how can they be removed?)
- Develop on-line monitors for fission gases and for fuel particles in the sodium.
- Develop a broad band qualitative analysis technique to identify all elements in the sodium system.
- Develop laboratory methods to provide calibration backup for the on-line monitoring devices.

- Develop designs for and proof testing of two integrated purification systems capable of removing suspended and dissolved impurities from the primary and secondary systems.
- Investigate additives to the sodium which have the effect of suppressing corrosion and mass transfer or, alternately, which make control of the products easier.

Other information needed is a knowledge of how material transfer will affect operation of the components with the higher reactor outlet temperature of 1150°F and higher loop differential temperatures. This information is best obtained by use of an engineering mock-up of portions of the system. The major areas of interest are deposition on the primary side of the IHX, carburization of the IHX, and decarburization of the steam generator tubing. The system in which the tests are carried out should be representative of the operating LMFBR systems.

5.2.3 Discussion

The successful use of sodium as a reactor coolant will depend to a large extent on the plant's ability to maintain the sodium in a state of purity for which corrosion of containing materials is minimal and is at plugging temperatures well below loop operating temperature. The Program Plan presented by the LMFBR Program Office, (Tasks 5-2, 5-3, 5-4, and 5-5) is excellent insofar as it goes, but it does not provide for integration of the various tasks into working systems and thus may fall short of achieving its stated objective. Experience in operating loops at GE in San Jose has shown that impurity plugging could interfere with proper operation of a pressure measuring device operating at 1000°F even though the loop sodium was cold trapped at 300°F and showed an appropriate plugging indicator reading. This experience shows that it is possible for impurities to hide out in a large system and to show

locally large deviations from overall average conditions.

The accumulation of experience on cold trap operation indicates that current designs, associated equipment, and operating modes lead to relatively short-lived (inefficient) cold traps and low plant availability. Little experience or design data is available to optimize designs for cold traps to meet the expected needs for the 1000 MW plant.

5.2.4 Task Objectives and Assessment of Non-Achievement

The objectives of this task are to (a) obtain design criteria for cold traps consisting of trap efficiencies for control of sodium impurities such as sodium oxides, carbonates, hydroxide and hydride as a function of design parameters, (b) develop optimum designs and proof test an integrated system for control of impurities expected in the primary and secondary system of the 1000 MW plant, (c) measure the effect of material deposition on the heat transfer performance of simulated portions of the IHX, and (d) measure the amount and effect of carbon transfer on the IHX tubes.

The incentive for doing this work out-of-reactor is that it can be done much more expeditiously because of the absence of reactor shielding and containment which always results in very difficult working conditions. In the system for the primary loop, the absence of high radiation levels in the test environment will allow much more freedom of experimentation.

Before the sodium-system design can be finalized, the purity design basis must be established.

It has been assumed that by 1976 on-line sensors (O, H, C, N) and broad-band qualitative analysis techniques will have been developed and will only have to be evaluated in the proof tests.

The objective of the on-line monitor task is to develop an on-line method for continuously measuring the overall impurities (i.e., fission products, gas and fuel) in an operating sodium system.

Analytical laboratory methods and procedures must be developed for LMFBR plant surveillance.

Performance of the material deposition and carbon-transfer tests is expected to primarily influence the IHX design. Its function will be to give some warning of material problems in the IHX which would require removal of this unit for cleaning or retubing. The advanced knowledge of the units degradation with time will allow adequate design margins or cleanup procedures to avoid lengthy unplanned outages.

There is an excellent probability of accomplishing the planned work with the funds allocated. The proposed schedule is very tight and extraordinary effort will be required to accomplish the tasks as scheduled. It is estimated that over the 30-year life of the plant, about 3 years of operating time could be saved by timely solution of sodium purity and contamination problems independent of the reactor installation. It is expected that some progress will be made in developing purification systems for the demonstration plants; however, the problem of scaling this work up to the proper size for a 1000 MWe plant still remains. Failure to accomplish the materials deposition and carbon transfer effects on the IHX could result in severe deterioration in performance of these units, to the extent that they would need to be replaced if such performance deterioration had not been planned. Replacement of the IHX's could cost 6 million dollars in direct costs.

5.2.5 Facilities

The primary purification system differs from the secondary in a number of respects,

but the most obvious is that all the primary cooling systems share a common primary tank and share sodium which mixes in its passage through the reactor. The secondary systems are in general separate closed loops.

To do the required development, it will be necessary to build a model system connected to a tank which simulates the primary tank, though this tank need not be to scale. The primary cleanup system would consist of pumps, full-size cold trap and associated purification equipment, flow meter, heat exchangers, and a tank through which impurity additions could be made.

Tests would be performed in this system to determine the disposition and subsequent removal of introduced impurities which should range from normal dirt and material found in newly constructed loops to deliberate additions of oxygen, nitrogen, argon, oil, moisture, as might be introduced through the cover gas, and simulated fission products, as would be released from failed fuel.

In the secondary system, the options exist of making one common system to service all of the secondary loops or to drive each loop with its separate cleanup system. In either case, it is expected that two cold-trap systems corresponding to two levels of cleanup will be required: one to the secondary system to handle relatively large quantities of impurity at high concentration levels following a sodium-water reaction and the second at the primary system, a continuous impurity accumulation and removal operation to work impurities down to an allowable operating level. The choice in this area is primarily a design problem; however, its solution requires knowledge of the performance of how specific designs of large flow units operate. It is expected that the design of equipment will have progressed through efforts for the 350 MWe Demonstration Plants, but that it will be necessary to proof test models of the large flow units envisaged

for quick recovery from the effects of a sodium-water reaction.

The facility required for this development consists of a purification system sized to handle about 1% of the flow of one loop through the polishing system and 5% through the quick recovery system. Specific pieces of equipment required are cold traps, high flow particle separator, flow instrumentation, impurity instrumentation, recording and control equipment, and an impurity injection station. This equipment is to be installed as a part of the engineering mock-up loop.

Tests would be performed in the secondary purification system to determine the efficiency and operating limits of the equipment and to provide information by which the capability could be increased.

At the present state of development, it is not possible to specify the exact size of the test facility required; this can only be done when the reference core and IHX designs are set.

5.3 TASK AREA 5-7-COVER GAS ATMOSPHERE

This task area is defined as the development of techniques for using helium or argon as the cover gas for the 1000 MWe LMFBR. The Task II report identified a particular arrangement of cover gas for the LMFBR. This particular arrangement involves the use of argon in a buffer-seal method of circulation, which minimizes the possibility of gas escaping through penetrations in the upper plug over the reactor. Relatively, pure argon is continuously bled through seals around the penetrations and leaks downward into the upper reactor region. Between the upper plug and the sodium level, a series of loose-fitting steel plates are included, which require that the incoming gas flow through passages adjacent to penetrations. This down-flowing gas prevents sodium vapor

from condensing on cooler surfaces at the under surface of the plug.

Argon gas is continuously removed from the cover-gas region to a series of decay tanks. Following a suitable decay period, these gases are released to the atmosphere through the stack. The amount of gas released corresponds to the amount leaking past the buffer seals, and makeup argon is provided to replace that discharged through the stack.

Additional gas handling is required to provide for the control of sodium level in the different regions of the reactor. These different regions include the reactor cover-gas region, the intermediate heat exchanger region, the sodium pump region, and the primary sodium tank cover-gas region. As sodium flow rate changes in the different parts of the primary system, the pressure drop through the system changes and results in pressure fluctuations over the different regions noted above. Control is provided by utilizing control valves, compressors, and pumps with regenerative heat exchangers for alternately cooling and heating the argon as it is removed from and returned to the system. During the cooling and heating process, sodium vapor is extracted from the coolant stream.

During refueling the argon is purged from the cover-gas region over the reactor so that fission products may be discharged to the radioactive gas handling system and clean argon replaced prior to opening the refueling plug to the hot cell.

The foregoing discussion is amplified in the Task II and Task III reports. The system described represents an extension from current technology in that continuous removal and return of portions of a radioactively contaminated cover gas are carried out during operation of the reactor. The current state of technology is discussed in Task Area 5-7 of the LMFBR Program Plan.

The functional requirements of this task area are that a cover gas system be provided, which is compatible with the following system requirements:

- Use of vented fuel
- The pool primary system arrangement concept
- Refueling in an open hot cell
- Safety regulations contained in 10CFR20 and 10CFR100

Although an argon cover gas system is described in the Task II report, the Task III report makes reference to the possibility of a helium cover-gas system. The functional requirements would be similar to those noted above for the argon system. The advantage of a helium system, as pointed out in the Task II report, might be that long-lived radioactive isotopes, such as Krypton 85, can be more readily cold-trapped from the helium system than from argon. This would permit complete or almost complete removal of radioactive isotopes without the necessity of discharging to the atmosphere.

5.3.1 Task 5-7.4 Analytical Methods for Cover Gas and Maintenance Atmosphere

As defined in the Program Plan, the objectives of this task follow.

- Continuous on-line monitoring methods for cover gas and maintenance atmosphere.
- Sampling and laboratory methods for sodium technology R & D programs.
- Standardized analytical procedures for LMFBR plant surveillance.

This work is presumed to be an on-going program as part of the FFTF effort. It is anticipated that the techniques for developing

monitoring systems as performed for the FFTF will be suitable for the LMFBR.

5.3.2 Task 5-7.5 Purification Methods for Re-Cycle Cover Gas

5.3.2.1 New Information Needed

A principal informational need of this task is a better definition of the functional requirements for the large-pool primary-system concept so that more definitive development work can be performed on cover-gas systems. Task 2-1.2 in the Plant Design Element will provide this definition since one objective of that task is to study alternative primary system arrangements and the necessary cover-gas control for these arrangements. Following this definition of the primary system in Task 2-1.2, information is needed concerning the relative merits of helium versus argon for the functional requirements defined. Based upon the information developed in such a comparison, new information is needed concerning the practicality of suggested methods for handling radioactive contaminants and minimizing the spread of these contaminants. If it is decided for example that the Task II reference arrangement is an appropriate arrangement, then it would be desirable to fabricate models which simulate the Task II vessel arrangement for the determination of gas dynamics and thermal characteristics in a replica of the actual reactor system. This information could be used to determine the adequacy of the proposed baffles and seals and to determine the effectiveness of the down-flowing argon in preventing sodium-vapor condensation on the upper portions of the reactor vessel. Other tests could be performed which would permit evaluation of the ability to obtain a desired low-level fission product concentration in the cover-gas region by continuous cleanup of a bypass stream or by periodic purging.

5.3.2.2 Task Objective and Assessment of the Effect of Non-Achievement

The objectives of this task follow.

- Utilize information from Task 2-1.2 to define the functional requirements for a cover gas system based upon an updated and more thorough evaluation of the large-pool concept than that carried out in Task II.
- Perform a comparison of argon and helium cover gas systems for the primary system functional requirements identified in the above objective.
- Perform tests as necessary for firming up design parameters which will permit the use of one of the cover-gas systems selected from the above objective.

The probability of achieving the objectives of this task are high. Since two cover-gas systems are selected for investigation, and since both have advantages and disadvantages, high probability exists that one of these cover-gas systems will satisfy enough of the functional requirements identified during the work that a viable system will be defined. Development work carried out in pursuit of this task area, however, may demonstrate that changes in the arrangement of the primary system vessel are required, or that increased development is necessary in the handling of fission products in order for the sodium reactor to use vented fuel. Alternatively it may be shown that it is necessary to operate with sealed fuel pins instead of the vented fuel selected as a reference during Task II. The use of sealed pins instead of vented fuel could result in a penalty of 0.1 mill/kW-h.

5.4. TASK AREA 5-8—PLANT OPERATION

5.4.1. Definition, Functional Requirements and Current Technology

The current reference LMFBR design of the plant sodium system and its operation

are based upon the premise that the maintenance practices are established and scalable to larger systems. The limited nature of the conceptual design effort and the limited experimental data upon which to base the design and procedures resulted in numerous areas of uncertainty. Some problems could result from the following.

- Lack of experience in handling large quantities of radioactive sodium in large pipes and equipment, particularly under radioactively contaminated conditions.
- Failure to consider development of non-routine procedures (preventive maintenance).
- Lack of experience in sodium system scale-up.
- Inability to detect and remove deposits from sodium systems.
- A lack of development of sound design and preventive maintenance that uses new technological diagnostic tools.
- A lack of development of sodium-related technology; for example, making remote internal pipe welds on surfaces contaminated with sodium.

The Demonstration Plant and its supporting development programs will develop technology which will compliment the base for operating procedures of the 1000 MWe design. Development work is being carried out on the ESADA program particularly in the area of steam generators where instrumentation is planned to be included on the steam generator.

It is required that operating procedures be developed which will make possible the efficient surveillance, repair, and maintenance of the plant without seriously jeopardizing economics.

5.4.2 New Information Needed

One basic need is to specify the 1000 MWe sodium system design and operation methods to correspond to data developed in other program elements. Specifically, some of the needed information follows.

- How to operate very large sodium systems typical of the reference 1000 MWe sodium systems.
- How to best utilize preventive maintenance and early warning diagnostics.
- Methods of detecting, identifying, and removing plugs and deposits from sodium systems.
- Determine response to abnormal operating conditions.
- Determine safety system adequacy.
- Procedures for repair and maintenance of contaminated sodium systems.
- Development of techniques for remote welding.
- What the effect is of spilling large quantities of hot sodium on equipment.
- Procedures for handling and disposal of radioactive sodium.

5.4.3 Discussion

The task areas are concerned with the general problems associated with the use of sodium in a large scale operating reactor system and the integration of systems and techniques which have sodium technology requirements. The size of the primary and secondary sodium systems require an extension of the sodium technology. Also, the presence of high level fission product

activity in the primary sodium introduces an unknown into the performance of auxiliary systems that will be used in plant operations. Otherwise routine operations must be done remotely. Sufficient data is not available to permit adequate design. For example, the current reference LMFBR design of the cold trap is based upon oversizing the cold trap by greater than a factor of six. No sound bases are available for both design and operational procedures. The removal of a contaminated cold trap for example, and its replacement with a new vessel, is of particular importance. It may require considerable dexterity with remote manipulation techniques and devices. It is important that all sodium system maintenance be performed with a minimum of plant down time. Good diagnostic, preventive maintenance will minimize all operations.

There has been sufficient operating experience with liquid-metal cooled reactors to recognize the recurring problems the reactor operator must face in a properly performing reactor plant. These problems are generally solved on an individual basis as they have arisen. Less than optimum procedures for solving these problems have resulted in delays in plant operation. Similar practices with the 1000 MWe plant could cause large economic losses.

Also, a problem is the detection, identification, and removal of plugs or adherent deposits in large piping in both the primary and secondary systems. Plug detecting and locating is viewed from the preventive maintenance standpoint rather than after-the-fact maintenance. The plug detection task interfaces closely with the development of on-line diagnostic procedures to monitor and detect incipient failure. Probabilities of complete piping failures are low if small cracks or leakage can be detected in time. Instrumentations development similar to those discussed in Task Area 4-2 is required for sodium-loop diagnostics since these techniques are not presently in use.

In the large primary systems, the sources of precipitated material are mass transfer, precipitated contaminants, system contaminants entrapped during maintenance procedures, detached "frosting" compounds, and accidental infusion of bulk fluids or particles. Such plugs normally occur in large static line or tees that are used only intermittently and in lines in which temperature gradients may exist.

The methods which will be developed for removal of plugs or deposits follow.

- Internal mechanical "descaling hammers"
- Dissolution and trapping
- Chemical reaction (additives) and trapping
- "Open" system removal by boring and scrapping chemical etchants.

5.4.4 Task Objectives and Assessment of Non-Achievement

The objective of the tasks which are discussed above follow.

- Develop methods of detecting, identifying, and removing plugs and deposits in operating systems.
- Develop procedures for repair and maintenance of contaminated sodium systems (remote welding, etc.).
- Develop procedures for handling a sodium spill accident.
- Develop procedures for handling and disposal of radioactive sodium.

The probability of accomplishing the tasks is high. However, failure to meet the objectives could result in long delays in plant operation and a resulting economic penalty of over \$0.5 million per week of downtime.

5.4.5 Facilities

Much of the equipment required now exists or will be available at General Electric when the tasks are scheduled to begin.

Remote equipment, instrumentation, mock-ups, lined trenches, and a safety system building will be required.

SECTION VI

ELEMENT 6 - CORE DESIGN

The succeeding material on the Core Design R&D includes new information necessary for the development of a high performance LMFBR core as discussed in the Task II and Task III reports. In some cases, because of a medium or low probability of success (instead of a high probability), the development program includes effort to develop alternatives to the reference design identified in the Task II report. In particular, effort is included to examine and develop alternatives to the non-free-standing fuel channels designated as the reference in the Task II design.

Priorities for performing the work are not shown on the program outline, since these priorities will change with time. At the present time, however, the most urgent need is for additional data on steel swelling and allowable strain at long burnup; this data will be supplied from the Fuels and Materials Element. The available data are not sufficient to permit with any confidence the conceptual design of a core structure with low steel content.

The second highest priority is for an updated core design using better data (and possibly for alternative materials which may show better characteristics under irradiation than austenitic stainless steel).

As with other elements in this document, the Core Design Element will draw upon technology and utilize development facilities provided for the Demonstration Plant. The schedule is designed to be compatible with work performed in other elements and with the Demonstration Plant development program plans. In general, the current state-of-technology is defined under the individual tasks. However, comment should be made on how the Demonstration Plant relates to this current technology.

As noted above, one of the most urgent tasks in this element is the accumulation of additional data on steel swelling. In this regard, it is expected that work carried out for the Demonstration Plant may improve on the current technology both with regard to a knowledge of swelling mechanisms, rates, etc., and with regard to design methods for accommodating this swelling in the core. However, because the Demonstration Plant is being designed principally as a "proof-of-principle" plant, and not as a commercial reactor, the design goals are more modest than in the Task II 1000 MWe LMFBR. Thus the temperatures are lower than in the 1000 MWe core, and there is less incentive to reduce the structural material in the core and to extend fuel burnup. These latter factors could result in a core structure for the Demonstration Plant which includes the "free-standing" type of channel with sufficient clearance between channels at critical locations to allow for swelling. This type of design would result in substantial economic penalties for the 1000 MWe LMFBR. Therefore, knowledge will have to be obtained for the 1000 MWe LMFBR which defines steel swelling and other characteristics at higher temperatures and burnups than for the Demonstration Plant, and different core design techniques will have to be developed to permit low structure fractions to be used.

Table 6-1 shows the core design milestones for the tasks in this element, and Table 6-2 shows the core design program outline. Each one of the tasks is discussed in more detail in succeeding material. Table 6-1 also shows the schedule assumed for availability of a fuel testing facility such as the FFTF or the Demonstration Plant, and the schedule for performing the final design and fabrication of fuel for the 1000 MWe

TABLE 6-1
CORE DESIGN MILESTONES

	Government Fiscal Year											
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
Task 6-1.1 Reference Design												
Updated Oxide Conceptual Design		△1	△2									
Carbide Core Conceptual Design			△3	△4								
Oxide Core Preliminary Design				△5		△6						
Task 6-1.3 Core Structure												
Fuel Subassembly Prototype		△1		△2	△15 △3	△16						
Core Structure Mock-Up			△4		△5	△6						
Fuel Subassembly Performance				△7		△8			△9	△10	△11	
Critical Component Tests						△12	△13		△14			
FFTF or Demonstration Plant Availability						△1		△2				
1000 MWe LMFBR Fuel									△1		△2 △3	

MILESTONES

Task 6-1.1 Reference Design

- △1 Updated oxide core conceptual design commenced.
- △2 Updated oxide core conceptual design completed.
- △3 Carbide core conceptual design commenced.
- △4 Carbide core conceptual design completed.
- △5 Oxide core preliminary design commenced.
- △6 Oxide core preliminary design completed.

Task 6-1.3 Core Structure

- △1 Fuel subassembly prototype designs commenced.
- △2 Water flow tests of fuel subassembly prototypes commenced.
- △3 Water flow tests of fuel subassembly prototypes completed.
- △4 Core structure mock-up design commenced.

TABLE 6-1 (Continued)

Task 6-1.3 Core Structure (Continued)

- 5 Water flow and mechanical assembly tests of core structure mock-up commenced.
- 6 Water flow and mechanical assembly tests of core structure mock-up completed.
- 7 Commence design of 3 or 4 instrumented fuel subassemblies.
- 8 Instrumented fuel subassemblies fabricated and ready for in-pile testing.
- 9 Removal of first thermal test of instrumented subassembly from FFTF or Demonstration Plant.
- 10 Removal of burnup test of instrumented subassembly from FFTF or Demonstration Plant.
- 11 Removal of long burnup test of instrumented subassembly from FFTF or Demonstration Plant.
- 12 Design and fabrication of critical core components commenced.
- 13 Testing of critical core components commenced.
- 14 Testing of critical core components completed.
- 15 Sodium flow tests commenced.
- 16 Sodium flow tests completed.

FFTF or Demonstration Plant

- 1 Earliest assumed availability of FFTF for fuel testing.
- 2 Latest assumed availability of FFTF or earliest assumed availability of Demonstration Plant.

1000 MWe LMFBR Fuel

- 1 Detail design of 1000 MWe LMFBR core and fuel commences (carried out in Overall Element).
- 2 Fuel shipped to 1000 MWe LMFBR for first core load.
- 3 Load fuel, start to power for 1000 MWe LMFBR.

TABLE 6-2

CORE DESIGN PROGRAM OUTLINE

<u>Task</u>	<u>Title</u>	<u>Facility Requirements</u>	<u>Equipment and Facilities</u>	<u>Operating or Engineering</u>	<u>Manpower (Man Years)</u>	<u>Schedule Fiscal Year</u>
6-1.1	Reference Design					
	Updated Oxide Conceptual Design	None	None	400	8	1971, 1972
	Carbide Core Conceptual Design	None	None	400	8	1972, 1973
	Oxide Core Preliminary Design	None	None	800	12	1974, 1975
6-1.3	Core Structure Fabrication, Development, Testing and Analysis					
	6-1.3.1 Fuel Subassembly Prototype					
	Design and Fabrication	Fuel Fabrication Facility	50*	400	7	1972, 1973
	Water Flow Tests	Water Loop	500*	400	6	1973, 1974
	Sodium Flow Tests	Sodium Loop	500*	500	4	1974, 1975
	Analysis	None	None	100	2	1974, 1975
	6-1.3.2 Core Structure Mock-Up					
	Design and Fabrication	Shop	None	700	8	1972, 1973
	Assembly Tests	BTF	None	200	3	1974
	Water Flow Tests	Water Loop, BTF	500*	200	3	1975
	6-1.3.3 Fuel Subassembly Performance					
	Design	None	None	300	7	1974
	Fabrication	Fuel Fabrication Facility	100*	1000**	9	1975
	In-Pile Test of Subassemblies	FFTF or Demonstration Plant	None	400	7	1977 to 1980
	Hot-Lab Examination	Hot-Cell	100*	400	7	1978 to 1980
	6-1.3.4 Critical Component Stress Strain Tests					
	Channels and Subassembly	Lab Shop	50*	120	1	1975 to 1978
	Grid Plates	Lab Shop	50*	150	2	1975 to 1978
	Analysis	None	None	100	2	1978

* Allowance for special equipment required in facilities.

** Fabrication only, does not include fuel use or inventory charges which are assumed to be waived by the AEC.

LMFBR. This final design is not included in this element, but is a part of the Overall Element.

6.1 TASK 6.1.1—REFERENCE DESIGN

6.1.1 Definition, Functional Requirements, and Current Technology

This task is defined as the conceptual and preliminary design of the 1000 MWe Task II reference to update the design based upon improved materials irradiation data, and the conceptual design of an advanced carbide core.

The functional requirements for the reference designs will be defined as a part of this task, and will be based upon the Task II reference and upon updated materials data produced as a part of the Fuels and Materials Element.

The current state-of-technology for this task has been discussed briefly in the introduction to this element. With regard to the Task II reference concept, the core and core structure are based upon an oxide-fueled reactor with non-free-standing type of fuel channels. As a part of the conceptual design work performed for Task II (and Task III), a number of problem areas were identified which illustrate specific areas of deficiency in the current technology as represented by the Task II core design. The deficiencies result from the lack of basic materials data and from the limited nature of the Task II work. The deficiencies follow.

- Lack of high-burnup irradiated fuels and structural materials data upon which to base fuel pin and core structure designs.
- Inadequate definition of coolant distribution within a fuel channel.

- No attempt to determine effect of cyclic load changes on fuel and core structure.
- Long-term potential of alternate fuels (e. g., carbide) was not adequately explored.
- Optimum control rod design, configurations, and replacement was not determined.

It is intended that these deficiencies in the current state-of-technology will be corrected by three major steps. The first step will involve an updating of the Task II reference design to be carried out in fiscal years 1971 and 1972 (Milestones 1 and 2 of Task 6-1.1). Improved materials data will be obtained from the on-going Project Agreement 10 program described in the Fuels and Materials Element.

The second step in correcting the deficiencies noted above will include the conceptual design of an advanced carbide core to be carried out in fiscal years 1973 and 1974 (Milestones 3 and 4 of Task 6-1.1).

The third step will involve the preliminary design of the updated reference concept. This preliminary design will be based upon 1000 MWe oxide fuel development performed for the LMFBR and defined in the Fuels and Materials Element, as well as work performed in Tasks 2-1.2, 2-1.4, and 2-1.7 of the Plant Design Element, and work performed in Tasks 6-1.3, 6-2.2, 6-2.3, 6-2.4 and 6-2.5 of this element. The schedule for this last step calls for performing the work in fiscal years 1974 and 1975 (Milestones 5 and 6 of Task 6-1.1).

6.1.2 New Information Needed

The basic need of this task is for a series of evolutionary core designs which

draw upon data developed in other elements as defined in subsection 6.1.1. The first conceptual design step (Milestones 1 and 2) has an informational need for alternative methods of accommodating high steel swelling rates (up to 15% $\Delta V/\Delta V$ for 100,000 MWd/Te core average burnup) and to relate these steel swelling rates to core design parameters. Additionally, information is needed concerning alternative methods of achieving low structure fractions in the core which are compatible with the postulated swelling rates. Based upon the updated core design, information is needed to define specific tests to be performed in the oxide fuel development work of the Fuels and Materials Element and tests to be performed in Tasks 6-1.3, 6-2.2, 6-2.3 and 6-3.2.

The second evolutionary step involving the carbide design includes an informational need for the potential economic gain to be realized in a high performance carbide core with steel swelling characteristics and core structure design similar to that of the updated oxide reference design. If a significant improvement in fuel cycle economics over the oxide is found to exist then there is a need to define directed research and development which will lead to the realization of this potential.

The third and final evolutionary step prior to the detail design of the 1000 MWe core (discussed in the Overall Element of this document) is a preliminary design which is again updated and uses information from the oxide fuel development work in the Fuels and Materials Element and information from tests performed in the other tasks of this element (Core Design). The principal informational need of this preliminary design work is to define all of the major core design parameters in sufficient detail so that final design, specification, and fabrication can proceed on a

schedule compatible with a calendar 1980 start-up date.

6.1.3 Task Objectives and Assessment of Effect of Non-Achievement

The objective of this Task is to perform a series of evolutionary core designs based upon the increased state of knowledge as test data from this element and other program elements is obtained and to iterate with these testing programs so as to provide a strongly directed development effort. Tentatively (unless the iterative procedure suggests otherwise), the evolutionary core designs shall include:

- An updated conceptual oxide core design (from the present Task II reference) which provides for increased steel swelling and low structure fraction.
- An advanced carbide fueled core conceptual design with similar structural features to the above.
- A preliminary design oxide core in sufficient detail to permit final design, specifications, and fabrication to proceed.

The probability of achieving the objectives of this task is estimated to be high. However, work on the design may indicate that the Task II reference core structure is inadequate. This is discussed under Task 6-1.3.

6.2 TASK 6-1.3—CORE STRUCTURE FABRICATION DEVELOPMENT, TESTING, AND ANALYSIS

6.2.1 Definition, Functional Requirements, and Current Technology

This task is defined as the fabrication, development, and testing of specific

core structure components or mock-ups and the supporting analysis to demonstrate the suitability of these components to meet the design goals and criteria identified in Task 6-1.1. The functional requirements for the development and testing program will be defined in Task 6-1.1.

The schedule for this work allows for iteration of information with work performed in Task 6-1.1. Facilities in which the work will be carried out will include the BTF, possibly the LMEC, and the Demonstration Plant or FFTF.

The current technology for the core design testing is similar to that described for the reference design task. However, it may be useful to discuss some particular areas of difficulty, as represented by current technology, which will be explored by this task.

A basic uncertainty caused by limited predictive analytical tools involves thermal-hydraulic data on non-uniform geometric shapes. Flow patterns in fuel subassemblies at corners, around fuel-pin spacers, around subassembly handling hardware, etc., are difficult to predict by purely analytical means. This is also true on a more macroscopic scale around fuel subassembly channels, through control rod bundles, in flow plenum distribution chambers, etc. Lack of ability to properly predict these flow patterns and the resulting thermal effects can result in excessive pressure drop or temperature, distortion of core components, or vibration of components with possible fretting corrosion or fatigue failure.

Other uncertainties are caused by mechanical assembly and disassembly problems of core components. These uncertainties relate to the accumulation of tolerances in manufactured or assembled components.

Finally, uncertainties are associated with the irradiated characteristics of specific components. For example, the grid plate of the Task II design is exposed to a fluence of 10^{22} n/cm² at an energy above 1 MeV at the end of its design life. If it later develops that this fluence is excessive it would be necessary to include additional steel shielding in the fuel subassemblies, at some expense.

With regard to fabrication technology, it is expected that the Demonstration Plant will provide some information on the manufacture of components, such as fuel subassemblies. There is a need, however, for defining the techniques for fabrication of the specific core components for the 1000 MWe design, particularly the fuel subassemblies. It is expected that quality control procedures developed for the Demonstration Plant will be adequate for the 1000 MWe needs. The component fabrication technology will be developed in the fabrication of the test prototypes or mock-ups.

6.2.2 New Information Needed

New information is needed on techniques for fabricating specific core components, as discussed above, on the ability of these components to be properly assembled and disassembled in the core, and on the performance of these components in accordance with the functional requirements defined in Task 6-1.1.

The information is expected to be obtained in four separate testing programs as defined in subtasks 6-1.3.1, 6-1.3.2, 6-1.3.3 and 6-1.3.4 on the core design program outline. The specific information needed and expected to be obtained from these separate subtasks is discussed below.

6.2.2.1 Subtask 6-1.3.1 Fuel Subassembly Prototype

This subtask will include fabrication of several full-scale prototype models of the fuel subassembly. Information will be generated on the ability to fabricate fuel pins and complete subassemblies based on designs from Task 6-1.1. Some of the pins will include the complete fabrication steps expected in the final design but uranium oxide may be used to simulate plutonium-oxide. Techniques developed during development of fuel fabrication procedures for the Demonstration Plant will be applied where possible. Completed subassemblies will be fabricated with varying types of fuel pin spacers (wire wrap versus grid) in order to obtain test information on flow and vibrational characteristics. Some of the simulated fuel pins may be resistance heaters suitable for obtaining information on the heat transfer characteristics of different spacer devices. The subassemblies will be fabricated complete with channel walls and handling hardware.

Subsequent to the fabrication of the subassemblies, water and sodium flow tests will be performed. Information necessary from these tests includes:

- Flow-pressure drop characteristics for different spacer types.
- Flow mixing characteristics for different spacer types.
- Angular flow characteristics (in order to determine the effects of flow obstruction and the possibility of controlling the flow distribution across a single fuel bundle).
- Heat transfer effects of the spacers for different spacer types.
- Susceptibility to plugging for different spacer types.

- Consequences of failure of a spacer.
- Effect of spacers on fuel pin distortion.
- Tendency toward flow-induced vibration.
- Effects of deviations from nominal geometry.
- Variations in flow between the center of the bundle and the edges.
- Distortion of fuel channel walls under pressure loading.
- Leakage past fuel subassembly hydraulic holddown seals.
- Result of failure of hydraulic holddown seal.
- Orifice performance in subassembly.

During the flow testing work, methods will be developed which permit empirical analyses to determine the performance of the fuel subassemblies.

While the water flow tests are sufficient to define the hydraulic characteristics of the fuel subassemblies operating in sodium, they will not simulate the thermal characteristics. Sodium flow testing is required. This is the only way the effects of such parameters as the ratio of thermal-to-momentum diffusivity (E_h/E_m) can be determined. The results of the sodium flow tests will be used to confirm and/or correct the analyses developed during the water tests.

6.2.2.2 Subtask 6-1.3.2 Core Structure Mock-Up

A full scale fractional portion of the core will be mocked-up for determining the adequacy of assembly and disassembly procedures and for determining the effect of

cumulative tolerances on core assembly and disassembly. The reference core clamping method will also be tested for adequacy in this mock-up. The fuel subassemblies for this core mock-up will be furnished from work in Subtask 6-1.3.1.

Subsequent to fabrication and assembly tests, water flow tests will be conducted to obtain the following information.

- Flow patterns in the core structure as a function of flow.
- Coolant distribution at low flow rates and with unequal temperature distribution.
- Component vibrational tendencies.
- Cover gas entrapment in coolant as a function of coolant flow rate and cover gas pressure.
- Adequacy of hydraulic holddown devices and mechanical back-up.
- Vibration of instrument probes as a function of flow rate.

6.2.2.3 Subtask 6-1.3.3 Fuel Subassembly Performance Tests

Three or four prototype fuel subassemblies will be designed and fabricated for irradiation testing in the FFTF or in the Demonstration Plant. These subassemblies will include fuel pin and spacer designs identical to that expected in the 1000 MWe LMFBR except for minor modifications which may be necessary to allow insertion in the test facility. The subassemblies will be tested to obtain the following information.

- Early thermal-hydraulic performance data for incorporation in final design specifications.

- Irradiation performance data.

The precise design of these prototype subassemblies will be a function of the available test facilities. It may not be possible within the allotted space, for example, to include all of the pins within the subassembly. In this event a few rows of fuel pins may be eliminated. Additionally, the flux of the test facility will be different from the 1000 MWe LMFBR. The design peak total flux and capacity factors for the 1000 MWe LMFBR, FFTF, and Demonstration Plant follow.

	Peak Total Flux	Capacity Factors
1000 MWe LMFBR	1.14×10^{16}	0.8
FFTF	0.95×10^{16}	0.75
Demonstration Plant	0.86×10^{16}	0.8

From the above it can be seen that irradiation will be accumulated in either the FFTF or the Demonstration Plant at a rate approximately 30% less than that of the LMFBR. This corresponds to a peak burnup in the FFTF or Demonstration Plant of approximately 55,000 MWd/Te-yr (assuming no increase in enrichment from that required for these facilities). Thus, the peak burnup of 145,000 MWd/Te which occurs in the 1000 MWe LMFBR will occur in either the FFTF or the Demonstration Plant in approximately 2-1/2 years. At the end of 2 years, 110,000 MWd/Te will have been accumulated.

As noted above, two different types of information are required from these prototype subassemblies. The thermal-hydraulic performance data will require different prototype design features than the irradiation data. In order to simulate the thermal performance of the peak pins in the 1000 MWe LMFBR, slightly increased enrichment will probably be necessary. Since the Fuels and Materials

Element has a continuing program for obtaining thermal performance data of individual pins. The principal informational need from this subassembly test will be to evaluate the performance of full length (or nearly full length) fuel pins in an array which mock-up the total fuel subassembly. Coolant distribution, local peaking factors, rod bowing, and possible fuel redistribution are the type of information which is needed from the thermal-hydraulic performance tests. Much of the necessary data can be accumulated in a relatively short period of time (6 months).

Assuming that either the FFTF or the Demonstration Plant will be available before the end of calendar year 1976 (mid fiscal year 1977), the thermal test subassembly could be inserted and irradiated for a 6-month schedule, removed and evaluated in the hot-cell over a 3- to 6-month period. This would still allow all of the data to be incorporated in the final design of the 1000 MWe fuel which commences in fiscal year 1978.

The irradiation performance tests require that the total fluence for the 1000 MWe LMFBR be simulated in both the fuel and the subassembly structural materials such as cladding and fuel pin spacers. As with the thermal performance tests, the Fuels and Materials Element has a continuing program for irradiation testing of individual pins. Therefore information needed from the subassembly irradiation performance tests involves the evaluation of the interacting effects of the various fuels and structural materials to determine the adequacy of the design. For example, information is needed on the ability of the multiple pin array to withstand internal swelling of fuel and external swelling of cladding and structural materials without failure of individual pins or undue warpage. Additionally, information is needed on the long-term behavior of fuel in full-length (or nearly full-length) fuel pins.

For the irradiation tests, it would be possible to accelerate fuel burnup by increased enrichment (as with the thermal tests). However, the cladding and structural materials burnup effects cannot be accelerated beyond that imposed by the test facility peak flux. Therefore, one of the subassemblies may be irradiated commencing at mid fiscal year 1977 for a period of 1.8 years to achieve a burnup of 100,000 MWd/Te. This schedule would allow removal and examination in time for incorporating the data and other performance information (including thermal data extending over a period of longer burnup than for the higher thermally rated thermal test subassembly) in the final design. Finally, a second subassembly will be irradiated to the full burnup of 145,000 MWd/Te experienced by the peak subassembly in the 1000 MWe LMFBR. This subassembly would be removed prior to completion of the fabrication of the 1000 MWe LMFBR fuel, and would be in sufficient time to incorporate minor changes (with a possible slight delay in the schedule).

The design for all of the prototype subassemblies will be performed as a part of this sub-task and will be based upon information obtained from the preliminary core design scheduled to be performed during fiscal years 1974 and 1975 under Task 6-1.1.

It should be noted that earlier availability of the FFTF than the assumed fiscal year 1977 with a capacity factor as indicated in the table will enhance the ability to meet the scheduled start-up of the 1000 MWe during calendar year 1980. Delay of the FFTF and the Demonstration Plant, or lack of achievement of the indicated capacity factor could delay the 1000 MWe LMFBR because of a lack of necessary test information upon which to base the final fuel design.

6.2.2.4 Subtask 6-1.3.4 Critical Component Stress-Strain Tests

There is a need for test information on the ability of certain core structural members, such as the channels and the grid plates to withstand cyclic and thermal stresses. The principal need is for information concerning the ability of these components to withstand the stresses and strains under irradiated conditions. Since it is impractical to irradiate large members, such as the grid plates, unirradiated members will be stressed and strained in testing jigs designed to meet or exceed the conditions expected in the reactor. The behavior of the unirradiated members will be compared with predicted behavior. Test data of materials properties of irradiated specimens will also be accumulated from the Fuels and Materials Element and data from both sources will be used to predict the ultimate behavior of the irradiated components.

6.2.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task follow:

- Develop fabrication technology and determine flow and vibrational characteristics of individual subassemblies by fabricating full scale prototypes and testing these prototypes in water and sodium flow tests designed to meet the informational need of Sub-Task 6-1.3.1.
- Determine the adequacy of the core structure to meet assembly and disassembly requirements, to have an acceptable coolant flow distribution over the expected range of flow rates, and to meet functional requirements defined in Task 6-1.1 and defined in the Safety Element with regard to minimizing gas entrapment in coolant passing through the core. This objec-

tive is to be satisfied by a mock-up and water flow test of a full-scale fractional portion of the core.

- Determine the adequacy of fuel subassemblies to meet thermal-hydraulic and irradiation requirements defined in Task 6-1.1 by fabricating and testing these subassemblies in the FFTF or the Demonstration Plant in accordance with the schedule and informational needs identified in Sub-Task 6-1.3.3.
- Determine the ability of key core structure components, such as the grid plate or fuel channels, to meet anticipated cyclic or steady state loads. This would be done by testing models of these components in the unirradiated state and combining the unirradiated component test data with irradiation materials test data from the Fuels and Materials Element to analytically predict the performance of the component in the reactor.

The probability of meeting the objectives of this task is estimated to be medium. It is not clear at this point, for example, that all of the functional requirements established in Task 6-1.1 will be met by the proposed testing program and within the allotted schedule. Failure to meet all of the objectives could result in either a delay in the 1000 MWe LMFBR start-up schedule or a degradation of performance. Performance degradation could occur in a number of areas: typical economic penalties follow:

	Penalty, Mills/kW-h
Free-standing channels instead of non-free-standing	0.1
Burnup reduction from 100,000 to 80,000 MWd/Te	0.05 to 0.1
Reduced reactor outlet temperature (1000°F versus 1150°F)	>0.1

6.3 TASK 6-2.4 AND 6-2.5—FUELS AND MATERIALS AND SAFETY COORDINATION

The Fuels and Materials and Safety Program Elements produce information which is vital to the Core Design Element. There is a need to coordinate, interpret, and guide the other Program Elements in accordance with the needs of the Core Design Element. Current technology concerning this coordinating and guiding task is delineated under Tasks 6-2.4 and 6-2.5 of the Program Plan. The guidance and interpretive functions should be performed by a continuing engineering analysis of recent data and its implication on core design. This guidance function should provide analytical methods for performing the design work

specific to the reference concepts being carried out in the Plant Design Element and in Task 6-1.1 of this element. For example, these methods should include updated fuel design models which are compatible with the test data and are suitable for use in performing specific design of fuel pins which meet the functional requirements defined in Task 6-1.1.

The fuels and materials and safety coordination tasks are part of an on-going program so the specific program descriptions are not included in the Core Design Program outline or milestones. The on-going program is the FCR Development Program (PA-10) being performed at General Electric.

SECTION VII

ELEMENT 7 - FUELS AND MATERIALS

The required fuels and materials development described herein is the work necessary to realize the reference design, or its equivalent in performance, as described in the Task II Report. The operational date for this 1000 MWe plant is fiscal year 1981, which implies that final design will begin in fiscal year 1978, at the latest. Fuel fabrication must probably begin in early fiscal 1979, and development work for the oxide fuel must therefore be largely completed before that time. Without considering in detail the required schedule for each of the various materials areas, it has generally been assumed for scheduling purposes that all the materials development necessary for the initial operation with the reference oxide fuel performance must be completed by the end of fiscal 1978.

A carbide fuel development effort is also projected, anticipating that the incentives indicated in the brief Task III study will be verified and that carbide fuel may well replace the oxide at some point in the life of this plant. The carbide development proceeds well beyond 1978. It should be recognized that provision of plans and funds for the carbide development also allows for the possibility that additional oxide development may be needed, if the anticipated first core performance is not achieved initially, and if carbide development is not proceeding as rapidly as planned. In effect, a continuing fuel development program to achieve better fuel performance is a certainty, though the specific work contained therein may not be exactly as described here.

The materials development encompasses not only fuel and fuel cladding, but also structural materials, both in- and out-of-core; control materials; and ma-

terials for miscellaneous applications such as bearings, springs, seals, etc. The specific area of sodium, and its interaction with structural materials, is, however, covered under Sodium Technology. Extensive requirements exist for information from the Sodium Technology development effort as input to the Fuels and Materials Section. The information would normally be in the form of the effects of the sodium environment on the material physical properties or on the physical dimensions (corrosion-erosion). Since measurements of out-of-sodium properties are in effect merely a point of comparison and reference for the in-sodium property measurements, all development of structural materials where high temperature sodium is the only novel element (essentially, Task 7-12.4) has been placed in the Sodium Technology Section. In some instances, however, integrated tests are necessary to determine the simultaneous effect of the total environment (temperature, neutron fluence, presence of sodium, flow velocity). Such tests are included under Fuels and Materials, since the planning and interpretation of the tests must be an integral part of the overall materials development effort. Materials development related specifically to the steam generator, such as studies of tube wastage due to leaks and sodium-water reaction products, is included in the Components Section.

The existence and successful operation of a ~300 MWe Demonstration Plant is assumed, but it is assumed for current purposes that the Demonstration Plant has minimal performance goals, at least in its early operation, and much of its development effort may be directed toward specific problems which are unique and not extrap-

relative to a much larger plant. In short, a pessimistic view is taken of the amount of development required to make the step from the Demonstration Plant to the 1000 MWe reference design. The Demonstration Plant or the FFTF is presumed to be available for fuel testing in fiscal 1975.

The highest priority within the Fuels and Materials area is obviously the development of the fuel pin to give the performance projected for the reference design, with sufficient reliability so that warranties can be given, and with sufficient tolerance for operation in abnormal or partially failed conditions so that plant availability is acceptable. The most critical areas currently involved in achieving this goal are the following:

- Properties of the cladding under conditions of high neutron fluence, high temperature, and low strain rates, including creep, stress-rupture, and cyclic fatigue,
- Fuel meat swelling as a function of temperature, burnup, density, and restraint,
- Behavior of defective or failed fuel, particularly tendencies toward propagation of damage, and
- Cladding swelling as a function of fluence and temperature.

The second highest priority is the in-core structural materials, in particular the hexagonal assembly can material. This is discussed under Core Design. The material problems here are similar to those of the cladding except that maximum temperatures are somewhat lower, and the stress-strain conditions as a function of time more predictable, since the swelling behavior of the fuel meat is not a factor. It is probable that successful development of the cladding

material will also provide the information necessary to design the core structure.

Priorities for the remaining materials development work are difficult to set at this time. In most cases, designs could be made which would avoid the need for significant additional development, at some additional cost. These incremental costs have not been estimated, but are thought to be considerably less, taken individually, than the cost incentives associated with the fuel pin and core structure materials development needs.

Table 7-1 shows milestones for the fuels and materials development, and Table 7-2 summarizes the program costs and facility requirements by Task. An attempt has been made to follow the organization of the LMFBR Program Plan, but that framework was not always appropriate for categorizing the work required. Therefore the emphasis or content of task areas has been altered where necessary and some new tasks added. In addition it was felt desirable to make carbide development more explicitly defined by assigning it separate task numbers.

Inasmuch as the program is directed specifically toward the General Electric 1000 MWe LMFBR concept, it is anticipated that a major fraction of the work shown would be performed by General Electric. The timing of the 1000 MWe LMFBR development has therefore been established to be consistent with overall G.E. plans and commitments. The fuels and materials development work for the 1000 MWe LMFBR has (with the exception of one Task) been shown as beginning in fiscal 1973 for two reasons. First, Project Agreement 10 should provide a base of information and experience that will allow much more exact definition of incremental development requirements than is now possible. This in turn should result in better efficiency and lower development cost,

TABLE 7-1

FUELS AND MATERIALS MILESTONES

	Government Fiscal Year												
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982
Task Area 7-5 Fundamental Behavior, Oxide			1			2							
Task Area 7-6 Evaluation of Alternate Designs, Oxide			3			4		5					
Task Areas 7-7, 7-8, 7-9 Cladding Development			6, 7			8		9					
Task 7-10.1 Proof-Testing, Oxide, Steady-State					10	11	12		13			14	
Task 7-10.2 Proof-Testing, Oxide, Off-Normal Conditions			15			16				17			
Task Area 7-15 Accommodation of Fission Gas	18	19	20			21							
Task Area 7-16 Fundamental Behavior, Carbide			22					23					
Task Area 7-17 Evaluation of Alternate Designs, Carbide			24					25					
Task Area 7-18 Proof-Testing, Carbide							26	27	28				29
Task 7-11.3 Absorber Materials			30	31				32	33				
Task Area 7-12 Structural Materials								34					
Task 7-13.3, 7-13.4 Bearing, Facing, and Seal Materials								35					

TABLE 7-1 (Continued)

- △1 Begin studies and model development for fundamental behavior
- △2 Complete development of models for 1000 MWe LMFBR design
- △3 Begin fabrication and testing using single pins—small groups—alternate designs
- △4 Choose designs for proof-testing
- △5 Complete testing and evaluation of alternate fuel concepts
- △6 Begin alloy modification work and testing for 1000 MWe LMFBR
- △7 Begin alloy development for carbide fuels
- △8 Make reference choice of alloys for LMFBR first core
- △9 Complete testing and evaluation of alloys for oxide fuel
- △10 Begin fabrication of oxide bundles for proof-testing
- △11 Begin proof-testing oxide bundles in FFTF or Demonstration Plant
- △12 Complete fabrication of oxide bundles for testing
- △13 Basic fuel design fixed for 1000 MWe LMFBR first core fabrication
- △14 Proof-testing of oxide completed
- △15 Begin pre-irradiation for off-normal proof-testing
- △16 Begin proof-testing of off-normal conditions

TABLE 7-1 (Continued)

- 17 Proof-testing of off-normal conditions completed
- 18 Reinsert stored vented fuel test capsules for irradiation to goal burnups
- 19 Complete irradiation of diving bell tests to 100,000 MWd/Te peak burnup; begin study of alternate vent designs and out-of-pile tests
- 20 Begin in-pile tests of vent concept(s)
- 21 Complete in-pile tests; choose 1000 MWe LMFBR concept
- 22 Begin studies and model development for fundamental behavior, carbide
- 23 Complete development of models for carbide behavior
- 24 Begin fabrication and testing of alternate carbide designs
- 25 Complete testing of carbide alternates and choice of designs for proof-testing
- 26 Begin fabrication of carbide bundles for proof-testing
- 27 Begin proof-testing of carbide bundles
- 28 Complete fabrication of carbide bundles for proof-testing
- 29 Complete testing of carbide bundles; begin design of carbide core
- 30 Begin out-of-pile materials compatibility tests
- 31 Begin irradiation tests of control materials in prototypical conditions

TABLE 7-1 (Continued)

- 32 Complete testing of control materials
- 33 Complete examinations; choose reference concept for detail design
- 34 Recommend materials and properties for non-cladding structural components
- 35 Materials properties established for service in sodium of bearings, facings, and seals

TABLE 7-2

FUELS AND MATERIALS PROGRAM OUTLINE

<u>Task or Task Area</u>	<u>Title</u>	<u>Facility Requirements</u>	<u>Costs (1000\$)</u>		<u>Manpower (Man-Years)</u>	<u>Schedule (Fiscal Year)</u>
			<u>Equipment and Facilities</u>	<u>Operating or Engineering</u>		
7-5	Fundamental Behavior, Oxide	Additional Pu Lab equipment	200	550	20	1973 to 1975
7-6	Evaluation of Alternate Designs, Oxide	Fast flux testing at high sodium temperature with defected and vented fuel; single pins and 7-19 pin assemblies		2380	26	1973 to 1977
7-7, 7-8, 7-9	Cladding Development	In-pile creep testing capability, fast flux, high Na temperature		1660	30	1973 to 1977
7-10.1	Steady-State Proof Testing, Oxide	Interim fuel inspection capability, FFTF or Demonstration Plant		3500	15	1973 to 1981
7-10.2	Off-Normal Proof Testing, Oxide	Fast flux high temperature sodium, transient test facility; single pins; initial steady-state conditions		3000	15	1973 to 1979
7-15	Accommodation of Fission Gas			330	8	1971 to 1975
7-16, 7-17, 7-18	Carbide Fuel Development	Fabrication and examination facilities in inert atmosphere. Irradiation facilities as for oxide	500	5200	30	1973 to 1982
7-11.3	Absorber Materials	In-pile loops, fast flux, high temperature sodium		530	12	1973 to 1978
7-12	Structural Materials	Fast flux irradiation, low and high temperatures		330	7	1973 to 1978
7-13, 7-14	Bearing, Facing and Seal Materials	Assumed to be carried out largely under on-going programs		120	4	thru 1977

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than would be the case if this program were initiated earlier. Secondly, a major expansion of company development commitments before this time does not appear desirable, since it would tend to dilute efforts toward the current primary goal, which is a Demonstration Plant. On the other hand, the period of time then available before irrevocable decisions must be made for the 1000 MWe LMFBR is only about 5 years. This is thought to be sufficient, but does not permit much schedule slippage.

7.1 TASK AREAS 7-5, 7-6, 7-7, 7-8, 7-9, 7-10 AND 7-15—OXIDE FUEL DEVELOPMENT

7.1.1 Definition, Functional Requirements, and Current Technology

This group of Task Areas is defined to cover the development necessary to allow the design and fabrication of oxide fuel that will achieve the performance specified for the reference 1000 MWe LMFBR design. Furthermore, sufficient information must be developed by approximately 1975 so that the economic risks due to possibly falling short of design performance are tolerable.

The functional requirements of the fuel pin are described in detail in the Task II Report. The most important of them are repeated below.

TABLE 7-3
FUEL PIN FUNCTIONAL
REQUIREMENTS SUMMARY

Linear Power, kW/ft	
Normal peak location	16.1
With overpower	17.9
Core average	9.6
Average Specific Power, kW/kg U + Pu	
(requires fuel pin diameter of ~0.25 inch)	157

Cladding thickness/diameter (around 10 mil cladding)	~0.04
Burnup, MWd/Te U + Pu	
Central pin, average over core length	~100,000
Central pin, axial center	~120,000
Core periphery, maximum local burnup	~150,000*
Core average	~100,000
Residence times, years	
Core center	2
Core periphery	3
Cladding Temperature, ° F	
Peak, full power	1300
Coolant Temperature, ° F	
Peak outlet at full power	1275
Cladding Fluence, neutrons/cm ²	
Peak location, >0.825 MeV	7.4×10^{22}
Peak location, total fluence	56.4×10^{22}
At the estimated design limiting location, >1 MeV	$\sim 6 \times 10^{22}$
Reliability	Must be high, but requirements not yet determined.
Tolerance for abnormal conditions	Must be capable of operating with common types of failures without tending to propagate damage.
Transient conditions	Design requirements for transient conditions in normal operation, incidents such as load rejection and scrams, and accidents not yet determined.

*Occurs in low power pin; not assumed to be the limiting location for design.

It is expected that the numbers and types of transients to which the fuel pin will be subjected will be defined by the Dynamic Analysis and Control Study, Task 2-1.7 under the Plant Design Element. Also, changes in the functional requirements may result from Task 6-1.1 of the Core Design Element, which involves updating of the reference design. Functional requirements for fuel pin reliability may be better defined as information becomes available from a variety of sources, but are strongly dependent on the overall design of the primary system and the behavior of fission products within it. Even after operating experience is obtained it may be impractical to determine the actual number of defective fuel pins within a core, and reliability requirements will therefore not be definable except relative to previous fuel performance.

The status of current technology from the point of view of the designer is well illustrated by the discussion of fuel pin optimization in the Task III report. Critical items which remain highly uncertain for the conditions of interest are the following:

- Fuel radial swelling rate, as a function of the following variables; burnup, smeared density, deployment of void, degree of cladding restraint, length of fuel column (axial restraint), operating temperature profiles, and stoichiometry.
- Cladding ductility, as a function of neutron fluence, neutron spectrum, irradiation temperature, test temperature, and strain rate.
- Cladding swelling rate, as a function of neutron fluence, neutron spectrum, and irradiation temperature.
- Fuel-cladding interaction at long burnups and high temperatures, and

fuel-sodium-cladding interaction in defected fuel.

From a testing standpoint, Table 7-4 summarizes the situation approximately as of September 1968.

TABLE 7-4
OXIDE FUEL TEST STATUS

<u>Parameter</u>	<u>EBR-II Testing (9/1/68)</u>
Cladding Thickness, mils	
- primarily	15
- a few at	10
Peak Cladding Temperatures, °F	1100
Power at Peak Position, kW/ft	17.5
Burnup (Peak), MWd/Te	75,000
Fast Flux Exposure of Cladding, $\times 10^{22}$ n/cm ² , >1 MeV	1.2
Gas Containment	Non-vented
Fuel Density, Void Deployment Smeared	Largely 87 to 94%

From this tabulation, it is seen that major areas in which performance must be extrapolated beyond current test results are cladding thickness, cladding temperature, fuel burnup, fast flux cladding exposure, and accommodation of fission gas. A further consideration which is not obvious from the above tabulation is that tests which combine representative cladding conditions with representative fuel burnup have not been performed and cannot be achieved before operation of FFTF or the Demonstration Plant. The flux levels in available facilities are too low, so that if test pins are overenriched to achieve typical linear power ratings, the resulting cladding fluence is too low. As a result, there is no test experience that combines the effects of fuel swelling on cladding strain with the appropriate corresponding cladding fluences and cladding mechanical properties.

Underlying the situation in the current technology as it applies to the fuel pin as a

whole are corresponding problems and needs in the current technology of many individual areas, such as fuel meat gas release, fuel-cladding interactions, fundamental cladding behavior under irradiation, and alloy development. These are discussed in the Program Plan, and to some extent in the material following on the individual Tasks and Task Areas.

7.1.2 Task Area 7-5 Fundamental Behavior, Oxide

7.1.2.1 Definition, Functional Requirements, and Current Technology

This Task Area is defined as encompassing basic studies of fuel and cladding behavior under the conditions experienced in the 1000 MWe LMFBR. The purpose of this task is to systematically relate the macroscopic behavior of the fuel and cladding to the microscopic structures and composition, and to the effects of irradiation, stress, and other variables on that structure and composition. The in-pile testing for this work will come largely from Task Area 7-6, and from previous testing. Required out-of-pile experimental work, additional testing of available irradiated specimens, analysis of the data on a microscopic scale, and development of theories and models to explain the observed behavior will be done under Task Area 7-5. It is intended that Task Areas 7-5 and 7-6 be very closely coordinated, and have concurrent schedules. The planning and priorities in Task Area 7-5 will be determined by the specific needs for theoretical understanding of macroscopic behavior in various areas. These needs will be determined most immediately by Task Area 7-6. The value of Task Area 7-5 is in allowing prediction of the effects of different variables on fuel and cladding properties. Although these predictions must still be verified by irradiation testing of fuel pins, the number of such tests can be minimized, and extrapolation of the re-

sults to different conditions can be made with more confidence. In addition, the theoretical models of material behavior will serve as guides to the most promising fuel designs and cladding alloys. In sum, the functional requirements of the work in Task Area 7-5 are to provide information which will minimize the cost and time required in the remainder of the oxide fuel development.

An extensive discussion of the current technology may be found in the LMFBR Program Plan under Task Area 7-5. It may be summarized here by saying that although explanations have been proposed for the observed behavior of fuel and cladding materials, they do not as yet provide a completely satisfactory basis for prediction and extrapolation. In addition, data currently available is not adequate in many instances to allow verification of predictions. This situation will certainly improve greatly by fiscal year 1973, due to ongoing programs such as Project Agreement 10. Nevertheless, it is improbable that the needs in this area will have been completely satisfied for the 1000 MWe LMFBR.

7.1.2.2 New Information Needed

The new information required may be summarized as follows:

- a. A theoretical explanation of observed volumetric fuel swelling rates, and how they are affected by burnup, external restraint, operating temperatures, stoichiometry, and Pu/U ratio.
- b. An explanation of observed fuel plasticity as a function of burnup and temperature.
- c. An understanding of the mechanisms of radiation embrittlement and swelling in cladding alloys.

- d. Improved understanding and data on the effects of reactor operating conditions on irradiated cladding material. These conditions involve the areas of low-cycle fatigue and long-term creep-rupture at derated temperatures with complex biaxial stress distributions.
- e. Understanding of the possible chemical reactions between fuel and cladding or fission products and cladding at interface temperatures above 1300° F and residence times of 2 to 3 years.
- f. Understanding of the possible chemical reactions between high-burnup fuel cladding, and sodium in defected fuel, and of the subsequent physical effects (e.g., possible swelling of resultant compounds).

7.1.2.3 Task Area Objectives and Effect of Non-Achievement

This Task Area divides itself naturally into three tasks, as in the Program Plan:

- Task 7-5.1 Fuel Behavior Mechanisms
objectives a and b above
- Task 7-5.2 Cladding Behavior Mechanisms
objectives c and d above
- Task 7-5.3 Chemical Interactions
objectives e and f above

Complete success in achieving the objectives of Tasks 7-5.1 and 7-5.2 is not to be expected, particularly when all the significant variables are considered. However, it is expected that there is a high probability that the basic causes and mechanisms of fuel swelling, cladding swelling,

and cladding loss of ductility will be better identified and that the principal features of observed behavior will be explained. Furthermore, at least qualitative understanding of the effects of the significant variables will be achieved with a high probability. Limited predictive ability is to be expected. In Task 7-5.3, a high probability of success exists.

If the degree of success is low, the penalties will be significant, but are very difficult to evaluate. In general they will appear as a need for more fuel testing in Task Area 7-6, which will require more funds and a longer schedule. More of the testing will turn out to be non-productive. Alternately, higher risks may be accepted in building the 1000 MWe plant. The most critical Task is probably 7-5.2, since non-achievement here will severely cripple the development effort which may be required on new or modified cladding alloys. If such development is needed, it is probably on the critical path of the 1000 MWe LMFBR schedule.

7.1.3 Task Area 7-6 Evaluation of Alternate Designs, Oxide

7.1.3.1 Definition, Functional Requirements, and Current Technology

This Task Area is defined as studies of the fuel pin as an integral system, with the objectives of evaluating alternate possible designs and choosing one or more for statistical proof testing. Included is extensive irradiation testing of pins, with a number of design variables. The performance of these pins will be screened to identify the most promising designs. Supplemental out-of-pile tests and measurements will be performed where appropriate. The data on macroscopic fuel and cladding behavior and properties will be supplied to Task Area 7-5, and test specimens themselves will be made available to Task Area 7-5 for further measurements that may be required for

study of microscopic behavior. Integration of Task Areas 7-5 and 7-6 will be carried out by Task Area 7-6, and coordination with the Core Design Element will be accomplished largely in this Task Area. The cladding development being carried out in Task Areas 7-7, 7-8, and 7-9 would reflect itself in the alloys chosen for the fuel pin tests in Task Area 7-6, and results of these tests would therefore be utilized in the cladding development tasks. Similarly, the development of vent mechanisms under Task Area 7-15 will determine the vent designs used in tests in Task Area 7-6, and the test results will be available to aid in the vent development.

Formulation of models to explain the macroscopic behavior of the fuel pin will also be carried out, utilizing the theoretical understanding gained in Task Area 7-5.

The functional requirement of this Task Area is to identify one or more fuel pin design concepts that will achieve approximately the fuel performance described in Table 7-3. The identification would be made in terms of the recommended cladding alloy, cladding thickness, fuel smeared density, void deployment, necessary measures to limit fuel-cladding interaction to acceptable rates, necessary measures to avoid propagation of damage in defective fuel, and recommended fission gas vent design. The fuel pin concept(s) identified would then be subjected to statistical proof testing in Task Area 7-10.

The state of the technology at the time this Task begins cannot be accurately forecast. It has been assumed that current ongoing programs will provide adequate information on questions of thermal conductivity, thermophysical properties, stoichiometry, and migration behavior. Adequate does not necessarily imply very complete or extremely accurate information, but is taken here to mean that the available data and understanding, together with evaluations of

the significance of these variables to reactor safety or economy, will lead to the conclusion that further efforts in these areas should be minimal. One item which may, however, require further work is the effect of smeared density and void deployment on linear power.

It appears at this time that the critical variables in these tests will be cladding thickness, fuel smeared density, cladding temperature, and burnup. Also of importance may be the deployment of void (e.g., what is the most advantageous combination of pellet density, fuel-cladding gap, and possibly fabricated central void to achieve a given smeared density in pellet fuel) and the use of barrier layers on the inside of the cladding to restrict attack by fission products. Cladding alloy will also be very important, and is discussed under Task Areas 7-7, 7-8, and 7-9.

7.1.3.2 New Information Needed

While it is not certain the cladding thinner than 15 mils can satisfactorily restrain fuel at high burnup, particularly in view of the higher operating temperatures discussed in the next section, testing to ascertain such behavior must be performed. Cladding at both 10- and 12-mil thickness, along with the conventional 15-mil thickness, will be tested. The 12- and 15-mil thick cladding is to serve as a back-up effort in the event that 10-mil tubing is unsuitable.

The cladding temperature of 1300° F in the large LMFBR is about 200° F above conventional EBR-II testing conditions and is also above the probable temperatures of a Demonstration Plant. Operating at such high temperatures raises serious questions on element performance. Some of these concerns are listed below.

- Reduced fuel restraint will occur with the higher cladding temperature, so that fuel swelling may increase.

Cladding swelling, however, may be lower at these higher temperatures.

- Lessened cladding ductility would be expected.
- Potential fission product reaction with the cladding could significantly reduce its "effective thickness," and thereby reduce its restraint effect further.
- Potentially increased fuel-cladding surface reactions could take place.

Fuel burnups to 100,000 MWd/Te have not yet been achieved in a fast flux and a significant increase in the fuel pin swelling rate appears to develop between 50,000 and 75,000 MWd/Te. The attainment of at least 100,000 MWd/Te exposure in a fast flux irradiated pin is not expected until fiscal year 1970.

Analysis of fuel elements at burnups to about 75,000 MWd/Te has shown for 15-mil thick tubing at about 1100° F that fuel density does not appear to have a pronounced effect on fuel swelling provided the initial fuel-cladding gap exceeds 2.5 to 3 mils. However, for burnups in excess of 100,000 MWd/Te one may expect that the larger amount of void volume present in low density fuel would more readily accommodate fuel swelling. Most fuel elements in test include fuel of >85% T.D. On this basis the testing of lower density fuel in the 75 to 90% range will be necessary to establish optimum density for these high burnups.

Any test program proposed at this time must be viewed as purely speculative. However, a not unreasonable program might involve four 37-pin assemblies, as an order of magnitude. The schedule would require that these tests be performed in EBR-II.

Essentially pins of a given stoichiometry would operate at cladding temperatures of 1225 and 1300° F to burnups of 120,000 MWd/Te. Cladding of 10, 12 and 15 mils would be utilized with various methods of accommodating fission gas. Barrier layers on advance cladding materials might be utilized in about half of the pins. Pins would operate at 16.5 kW/ft, with cladding temperatures varied by controlled orificing within the bundle. These pins would be unencapsulated for interim examinations to obtain incremental, non-destructive information on their behavior. Some pins operating at peak temperatures would be removed for destructive examination at 40,000 MWd/Te to observe the effectiveness of the vent and the extent of fuel and fission product interactions with the cladding at elevated temperatures. A similar total examination and limited removal would be scheduled at 75,000 MWd/Te. These examinations would permit the early removal of pins showing unsatisfactory performance and permit their replacement with pins of alternate parameters. Back-up pins would therefore be required as replacements.

In addition to the steady-state tests of fuel in good condition, there must be tests of damage propagation tendencies. These require bundles of perhaps 19 pins or more, with purposely defected fuel at one or more locations. The defects would be varied in severity to attempt to identify the approximate type or magnitude of defect which would tend to propagate. An in-reactor loop is required, preferably in a fast flux.

Analytical work and model development for macroscopic fuel pin behavior will be carried out. Included here will be the achievement of a systematic description of fuel radial and axial swelling as a function of cladding restraint, operating

temperature, burnup, smeared density, and void deployment. Also required is a model of the fuel configuration (gap size, central void, etc.) as a function of time, and burnup, so that the allowable linear power can be predicted.

7.1.3.3 Task Area Objectives and Effect of Non-Achievement

As previously indicated, the basic objective of this Task Area is to identify a fuel pin design concept or concepts which appear to offer good assurance of satisfying the performance requirements of Table 7-3, and which can be used in subsequent proof-testing. It is this Task Area which largely determines the success of the overall oxide fuel development program. This basic objective can be separated into individual objectives corresponding to the various design goals of the fuel pin performance. Those individual objectives relating specifically to the cladding performance are most appropriately part of Task Areas 7-7, 7-8, and 7-9, and those relating specifically to fission gas accommodation are covered under Task Area 7-15. The objectives listed below are those more uniquely associated with this Task:

- a. Integration of the oxide fuel development efforts (prior to proof testing) to achieve a functioning fuel pin concept,
- b. A fuel pin concept which can reach about 120,000 MWd/Te U + Pu in fuel with high linear powers, and about 150,000 MWd/Te in low power fuel,
- c. A fuel pin concept which can operate at about 160 kW/kg U + Pu,
- d. A fuel pin concept with cladding thickness-to-diameter ratios of about 0.04,
- e. A fuel pin concept with reasonable assurance that the expected types of defects do not tend to propagate damage to adjacent fuel pins, and
- f. Models for macroscopic fuel pin behavior that can be used to optimize designs and as guides for possible evolutionary improvement.

The probability of success of objectives a and c is high. The probability of success of objective d is high because it is largely dependent on development of a good vent design and on successful operation of vented fuel, and it is felt that both of these have high probability of success. The probability of success for objective b, in conjunction with the other objectives, is largely dependent on the cladding development tasks, and may possibly be no better than 50 to 70%. As the burnup goal is reduced, the probability of success rises; if a core average burnup of 50,000 to 60,000 MWd/Te U + Pu were found to be acceptable, the probability of success would be very high, perhaps 90%. The assumed cladding temperature naturally affects these estimates; probability of success would increase if peak cladding temperatures were reduced below the 1300° F chosen for the reference design.

The penalty associated with non-achievement of the burnup goals depends on whether compensating adjustments can be made in other fuel and plant design parameters. For example, if a lower burnup is necessary but is associated with a higher allowable cladding and coolant temperature, then a reduction of 10,000 MWd/Te in core average burnup might cost only 0.01 to 0.02 mill/kW-h in power cost, based on the Task III studies using optimistic fabrication and reprocessing costs. Using the alternate, more conservative unit costs, the penalty becomes

0.02 to 0.4 mill/kW-h. The penalty for further reductions in burnup becomes greater for each successive 10,000 MWd/Te. These are minimum penalties, and it is highly improbable that they would be achieved. A simple uncompensated reduction of burnup by 10,000 MWd/Te U + Pu would cost 0.04 mill/kW-h using the reference fuel cost basis even though the unit costs of the reference fuel costs are closer to the optimistic basis than to the conservative basis. Halving the burnup to 50,000 MWd/Te might cost up to 0.38 mill/kW-h. These penalties are more realistic than those based on a compensating gain in coolant temperature, but may still be underestimates if fabrication and reprocessing unit costs prove to be higher than estimated.

The incentives for thin cladding can be estimated as about 0.05 mill/kW-h for a decrease in thickness from 15 to 10 mils, at 10\$/gram of plutonium. This incentive is not dependent on the fabrication and reprocessing cost assumptions.

The desired specific power is very heavily dependent on fabrication costs. Assuming that reductions in specific power would normally be accompanied by increases in fuel pin diameter (that is, linear power not affected) there is less than 0.01 mill/kW-h penalty incurred by reducing specific power from 160 kW/kg U + Pu to 135 kW/kg, using the optimistic fabrication costs. If, however, it were necessary to reduce linear power by a similar percentage, the penalty would be about 0.07 mill/kW-h (at 10 \$/gram Pu). It is thought that such linear power reductions are unlikely.

The non-achievement of objective e, resistance to damage propagation due to common types of defects, would be extremely serious and make the feasibility of the 1000 MWe LMFBR doubtful. However, it is felt that the probability of success is high.

It is expected that objective f will have a fair measure of success, sufficient at least to allow the development program to proceed efficiently.

7.1.4 Task Areas 7-7, 7-8, and 7-9 Cladding Development

7.1.4.1 Definition, Functional Requirements, and Current Technology

These Task Areas have been grouped together and will be discussed as a whole here because they cover a single work area which has well-defined objectives and must be well integrated. The work covered is the development and improvement of cladding, leading first to the selection of a reference cladding for the oxide fuel which is to be proof tested under Task Area 7-10, and subsequently to the selection of a reference cladding for carbide, where proof-testing is to begin in 1977. Included within the work scope of these Task Areas are efforts to modify alloy compositions or to develop new alloys with improved properties; irradiation of cladding material samples; testing of cladding material to determine relevant mechanical properties; studies of fabrication variables and of defects to determine their significance to cladding properties; development of barriers against fuel and fission product attack; development of means of fabrication for such barriers, which may mean in effect a composite two-layer cladding; obtaining relevant data and material samples from testing carried out in Task Area 7-6; and coordination of fundamental structural materials work being carried out in Task Area 7-5. Task Areas 7-7, 7-8, and 7-9 taken together will form the focus of the cladding development effort, and must constitute a very unified, directed effort.

The functional requirements of the cladding for the oxide fuel can be extracted from Table 7-3; they are the following:

Peak temperature	1300° F
Peak fluence, >0.825 MeV	7.4×10^{22} n/cm ²
Peak fluence, total	56.4×10^{22} n/cm ²
Residence time	2 to 3 years
Fuel fission product content	representation of up to 150,000 MWd/Te U + Pu
Fuel-cladding bond	gas

It is desired that these conditions be withstood by a cladding thickness of about 10 mils. It is expected that a critical requirement will be for sufficient ductility after irradiation to accommodate the strain due to fuel swelling, plus any non-uniform strains caused by spacers. In addition, swelling of the cladding itself must be held within limits that make mechanical design of the fuel assembly feasible and do not unduly penalize core composition by requiring excessive space for expansion.

The functional requirements for carbide cladding are similar in many respects to those of oxide, but have not been established as definitively. In addition, there are increased problems of fuel-cladding chemical interaction, particularly with sodium-bonded carbide. A general discussion of carbide fuel, including cladding considerations, is given under the carbide fuel Task Areas.

The current technology of cladding development is discussed in the LMFBR Program Plan. The critical problems currently appear to be the loss of ductility due to irradiation, particularly at high temperatures, and swelling due to the creation of porosity under irradiation. A further problem may be fission product attack. This reaction has been observed in test pins, B3B, B8A, B4D and B5B,

operated at cladding temperatures approaching 1300° F in the U.S. AEC PA-10 Program.^(5, 6, 7) Potential barrier materials would be vapor deposited Cr, ZrO₂ or Al₂O₃, or a compound deposited from a solution in which the sample is immersed. Development, evaluation and testing of new cladding alloys may be required if gross cladding-fuel reactions are still observed.

Of the five alloys presently being considered for FCR cladding, Type-304 and 316 are receiving the most attention. Task-L of PA-10 will concentrate on these alloys also since at the present time no clear technical advantage can be established for choosing either alloy. Both have performed satisfactorily in fuel irradiation experiments in DFR, EBR-II, and Rapsodie. Radiation damage experiments to date show both alloys will become embrittled at elevated temperatures after extensive neutron fluence, $> 10^{22}$ total n/cm², in the fast flux facilities presently available. The post-irradiation ductilities appear to be a strong function of strain rate, Figure 7-1, and test temperature, Figure 7-2. Both alloys are strengthened after fast neutron radiation. The degree of strengthening depends on the irradiation temperature and fluence, Figure 7-3. Both cold worked and annealed structures are under investigation. Recent results suggest that the advantages of higher initial strength due to cold work are lost after extensive high temperature irradiation.

Swelling of both alloys has been observed after elevated temperature irradiation, $> 700^\circ$ F (371° C). Swelling rates are believed to be dependent on the alloy composition and structure, irradiation temperature, fluence, flux, amount of cold work, and the applied stress. Recent results suggest that stabilized grades of stainless steels, Types-321 and 347, exhibit less swelling

FIGURE 7-1. STRAIN RATE DEPENDENCE ON DUCTILITY FOR STAINLESS STEELS

FIGURE 7-2. TEMPERATURE DEPENDENCE ON DUCTILITY FOR STAINLESS STEEL

FIGURE 7-3. FLUENCE DEPENDENCE ON STRENGTH FOR STAINLESS STEEL

than the unstabilized grades. It is presently thought that the voids observed in all alloys irradiated at high temperatures are responsible for the radiation-induced swelling. Excellent agreement has been found between swelling as determined by actual immersion density measurements and that determined by void observations.

Estimates of the structural material expansion to be expected in the 1000 MWe core suggest that it may be serious enough to cause severe difficulties in mechanical design and significant penalties in economics, although this has not been evaluated with any precision. The estimates of the amount of swelling range upward from 15% $\Delta V/V$ for the goal burnups. It should be understood, however, that these approximations represent extensive extrapolations of existing data. If these forecast expansions are incompatible with core design requirements, it may be necessary to consider some

mode of in-core annealing of the cladding voids, such as raising the sodium temperature to about 1600° F, the temperature at which Cawthorne and Fulton caused cluster migration and elimination during a 1-hour treatment,⁽⁸⁾ or to investigate different cladding alloys.

Additional ferrous alloys may include FV-548*, the United Kingdom PFR backup alloy to 316, and high nickel alloy PE-16** which also is undergoing irradiation in Dounreay. Vanadium-base alloys are being investigated by ANL and Westinghouse, and are possible alternate materials. Preliminary results of tests of a V-20 w/o Ti alloy by ANL showed a considerably smaller concentration of

* Compositions 18.14% Cr - 10% Ni - 1.3% Mo - 0.85% Nb - 1.08% Mn - 0.07% C

** 16.45% Cr - 43.6% Ni - 3.5% Mo - 1.3% Ti - 1.2% Al - 0.1% Mn - 0.06% C

voids than in 304 stainless steel irradiated under the same conditions of 1030° F temperature and 3×10^{22} n/cm² total exposure, ⁽⁹⁾ although loss of ductility was considerable.

Besides the V-20 w/o Ti alloy, the following related ones are also in test:

1. V-10 w/o Ti,
2. V-9 w/o Cr - 3 w/o Fe - 1.3 w/o Zr - 0.05 w/o C,
3. V-8 w/o Cr - 10 w/o Ta - 1.3 w/o Zr - Bal. Fe, and
4. V-6 w/o Fe - 5 w/o Nb - 1.3 w/o Zr - 0.05 C.

It may be noted, however, that acceptable corrosion resistance of the vanadium alloys in sodium appears to depend on the maintenance of extremely low oxygen content, less than 10 ppm, and this may place severe requirements on plant operating procedures or cause significant capital cost penalties due to changes in system design.

7.1.4.2 New Information Needed

Planned experiments within Task L of PA-10 include evaluation of cladding embrittlement and swelling for EBR-II exposures of:

Irradiation Temperature	700° F to 1300° F
Total fluences	10^{20} to 10^{23} n/cm ²
Total fluxes	1.5×10^{15} n/cm ² -sec.

These flux and fluence values are approximately seven times less than that proposed for the 1,000 MWe core where the peak total flux is estimated to be 1.1×10^{16} n/cm²-sec. Therefore, the effects of total accumulated radiation damage will be determined only for fluences $\leq 16\%$ of that proposed for the 1,000 MWe core. The consequences of higher fluxes and fluences must be determined in later development programs.

Only limited in-pile data will become available from the PA-10 program. These data will become available primarily from EBR-II developmental fuel rod experiments. Residual ductility of cladding from highly irradiated fuel rods, 120,000 MWd/Te, will be determined but for cladding temperatures significantly below the 1300° F peak cladding for the 1000 MWe core.

Further work directed toward the 1,000 MWe plant design should focus on a broad based post-irradiation test program whereby the fuel cladding would be characterized at specific fluence levels. Both physical and mechanical properties of the cladding should be determined to allow ductility, strength and cladding swelling to be evaluated as a function of burnup and irradiation temperature and provide the data necessary for projecting end-of-life behavior. In addition, in-pile work is needed to measure the strength and ductility under dynamic conditions similar to that being experienced by the 1,000 MWe fuel cladding. The in-pile creep data presently available are confusing, regarding strength of cladding in a high-flux, high-temperature environment. Available in-pile ductility data when compared to post-irradiation data show similar results for similar test conditions. However, since only a limited amount of in-pile data will become available before 1972, none of which will have been performed in a flux of 10^{16} n/cm² sec, further in-pile ductility data should be generated in this program at the higher flux levels.

At the high burnups projected for the 1000 MWe LMFBR, the performance limits of existing alloys may be exceeded. As knowledge continues to be gathered on the specific effects of impurity or tramp elements and the interaction between major alloying constituents, it will be advisable to implement this knowledge in developing an alloy "tailored" for performance in a

particular reactor environment. A basic assumption in the schedule and program for the oxide cladding development is that it will be possible to meet the development goals with alloys that are basically similar to the 300 series stainless steels or to other alloys which are currently available and on which some work is being done. Thus, the cladding development program contemplated involves possible alloy modification or close control of impurities, but not, for oxide, the development of a new class of alloy.

The problem of fuel and fission product attack on the cladding will be better defined when this program begins in fiscal 1973, due to additional data from on-going programs. Promising barriers will be tested possibly both in- and out-of-pile.

New information requirements for carbide fuel cladding are discussed under the general introductory discussion for the carbide fuel Task Areas.

7.1.4.3 Task Area Objectives and Effect of Non-Achievement

The objective relevant to oxide fuel is to develop a cladding which has the following properties:

- a. An allowable uniform strain (for steady-state design purposes) of at least 1.5% at 1300° F, with a fluence of about 7.5×10^{22} n/cm² greater than 1 MeV, and at the very low strain rates characteristic of fuel swelling.
- b. A volumetric swelling of no more than 15% at the peak nvt, assuming the swelling is isotropic. This is an unusually tentative objective, since no analysis was done of the mechanical design and volumetric

composition penalties attached to cladding swelling.

- c. Compatibility with fuel and fission products generated by burnups of 120,000 to 150,000 MWd/Te U + Pu, over periods of 2 to 3 years, at temperatures of 1300 to 1350° F (inside cladding surface).

The probability of success for the first objective is thought to be high. The sensitivity of costs to the allowable strain rate appears to be significant. Decreasing the allowable strain by 25% of the 1.5% goal would cost about 0.02 mill/kW-h, based on optimistic cost assumptions. Further decreases would be expected to increase the penalty more than proportionately.

Two further considerations are important in connection with objective a. First, the 1.5% strain requirement is for use in design. Since some margin is presumably required, this could mean a requirement for perhaps 3% in actual cladding capability. Furthermore, the fuel pin tolerance for transients is probably dependent on retaining some ductility in the cladding. Some residual ductility is therefore required over and above the steady-state design value. This residual ductility must be available at the high strain rates characteristic of the time scales involved in accidents. Secondly, the strain due to fuel swelling is not experienced after peak fluence is reached, but accumulates over a long period of time, most of which is spent at accumulated fluences much less than the peak fluence. Consequently, post-irradiation data may be misleadingly pessimistic when applied without modification.

The probability of success of the swelling objective is no better than medium (50% to 70%). Extrapolations of current information to the fluences of interest are

highly uncertain, but indicate that swelling of 304 and 316 stainless steel may be considerably greater than 15%. Other alloys, however, indicate substantially less swelling. The penalties of higher swelling have not been estimated, and studies should be carried out to identify the sensitivity of power cost to steel swelling rate. It is thought likely that the cost penalties will be significant for each additional 5 or 10% of volumetric swelling.

The third objective has a probability of success similar to the swelling objective. If compatibility is defined as no more than 1 mil of attack, indications are that current cladding alloys may be incompatible even at temperatures considerably below 1300° F. If this is due to attack by the excess oxygen created during fuel burnup, remedies may be difficult to find. The interposition of a corrosion-resistant barrier, or the use of a getter for the oxygen, are possibilities. However, the successful development of a corrosion barrier that retains its integrity under thermal cycling and irradiation, that is not so thick as to penalize neutron economy, and that can be fabricated at reasonable costs, would not appear to have a high probability. The penalties of lowering cladding temperature are estimated as follows:

50°	~0.04 mill/kW-h
100°	~0.1 mill/kW-h
150°	~0.2 mill/kW-h

These penalties are largely the result of lowered system efficiency and higher heat exchanger costs. They do not assume that burnup can be increased, in spite of the possibility of higher cladding ductility at the lower cladding temperature, because the fuel-cladding compatibility problem is aggravated by higher burnups.

Carbide fuel cladding development will have objectives which correspond to

each of those discussed above for oxide, but may be different in magnitude. The study of carbide fuel was not sufficient to warrant suggestion of specific numerical objectives, and of course the cost penalties associated with their non-achievement are not available.

7.1.5 Task 7-10.1 Steady-State Proof Testing, Oxide

7.1.5.1 Definition, Functional Requirements, and Current Technology

This task area is defined as encompassing the steady-state proof testing of large numbers of pins of the most promising designs as determined by Task Area 7-6.

The functional requirements of the fuel pin and cladding have been discussed elsewhere. The functional requirement of this Task Area is to provide sufficient statistical data under prototypical conditions so that the design of the 1000 MWe LMFBF core can be determined in 1978 with reasonable expectation of achieving the performance goals. The testing of large numbers of pins in a single subassembly has been performed by UK in the Dounreay Fast Reactor (77 pins/bundle) and in several subassemblies of EBR-II (37 pins/bundle).

7.1.5.2 New Information Needed

In-pile testing of large subassemblies is most valuable in irradiating large numbers of pins fabricated under similar conditions and with any important variables being tested by many pins. This provides information on the uniformity of the fabrication process or the variation of behavior within pins of a single classification, for better appraising differences in behavior between pins incorporating different variables. In addition, pin spacer and support modes

can be investigated over an extended period, thereby learning effects of fast neutron-induced distortion, dimensional change and relaxation. This is of particular importance in this study because of the large number of pins per assembly.

Possible sites for bundle irradiation include FFTF and the Demonstration Plant or plants. Fuel bundle testing can be most fruitful if interim examinations can be conducted on selected rods, as was proposed for Task Area 7-6. A bundle fixture that incorporates a remote disassembly feature would be utilized so that individual rods could be removed, dimensioned and the same or replacement pin returned to the bundle. Approximately 10 assemblies of at least 127 pins each are recommended for this program.

7.1.5.3 Task Area Objectives and Effect of Non-Achievement

As indicated before, the objectives are:

- Obtain information to serve as a basis for final decisions on the design of the 1000 MWe LMFBR first core.
- Obtain sufficient statistical data to high enough burnups to provide a reasonable expectation that fuel element performance goals (including reliability) will be met.

The achievement of these objectives will depend, first, on the availability of facilities for irradiation tests. Either FFTF or a demonstration plant must be available in fiscal 1975, and must operate at full power, at a load factor of perhaps 60 to 70% average over three years. Given this assumption, the probability of obtaining the test data is very high. Whether or

not the test data will support performance goals is dependent on the probabilities of success in Task Areas 7-6, 7-7, 7-8, and 7-9.

If these tests fall short of obtaining enough data, or of achieving perhaps two-thirds of the design burnups, the risks involved in the core design will increase, but since the commitment for a 1000 MWe plant will have been made on the basis of the results of Task Area 7-6 in 1975, the program will proceed. The possibility that the first core will fail to achieve design goals will be increased, and the proof-testing will have to continue for a greater period of time.

Oxide fuel development to achieve an improved second core may be necessary. If the tests exhibit low fuel reliability, however, consequences could range from a need for more testing or for a design modification (schedule delay), to a need for development of a new fuel concept.

7.1.6 Task 7-10.2 Off-Normal Proof Testing, Oxide

7.1.6.1 Definition, Functional Requirements, and Current Technology

In addition to the need for proof-testing to establish steady-state performance, there is a need for proof-testing the 1000 MWe LMFBR fuel pin design, or designs very close to it, under abnormal conditions that can be expected to occur in service. Such conditions occur as a result of minor overpower transients, various conditions such as turbine trip or loss of station power that may cause scram and reduction of flow, and minor defects in fuel pins. Of principal concern in such abnormal conditions is the capability for continued operation of the fuel. The situations of interest range from one in which immediate replacement is required, through

those in which accelerated failure rates are a probability and careful monitoring is dictated, to those in which essentially no effect is expected. This Task is defined as covering the testing and analyses required to determine the capabilities and response of the fuel pin under the abnormal conditions, and to generate the criteria for fuel replacement or continued operation which will be used in licensing and which will become part of the operating procedure of the plant.

It is part of the functional requirements of the fuel pin design that those flow and power transients that can be expected to occur during an in-core residence time should not result in a requirement for fuel removal prior to normal goal burnups. Otherwise substantial, and perhaps intolerable, penalties will be paid in terms of fuel cycle costs and plant availability. This requirement is closely related to safety since some of the reasons for fuel removal may be related to decreased safety margins or loss of capability to withstand future transients. Testing under abnormal conditions is closely related to Task 10-2.4 of the Safety Element, and it is anticipated that knowledge of fuel failure limits and mechanisms gained there will be of great importance to this Task.

The purposes of this Task are primarily concerned with operational effects and economics, however, and only secondarily with safety. The tests conducted under this Task would not be expected to approach fuel failure conditions, except perhaps as a result of long-term deterioration or when initially defected pins are tested.

The current technology in the areas of interest to this Task is extremely limited. Fuel transient testing that has been carried out (see Task 10-2.4) has been oriented toward the immediate safety questions of fuel failure limits and failure mechanisms. Knowledge of these remains very imprecise. Definition of the short- and long-term

effects of transients which are far below the level of severity required to produce failure is virtually non-existent.

7.1.6.2 New Information Needed

The tests to be performed under this Task must be defined by the results of Dynamics and Control Studies under the Plant Design Element. These studies will identify power and flow transients for which it is desirable that the steady-state and transient capability of the fuel element be essentially unimpaired. In addition, the guidance studies of the Safety Element and the core design studies of the Core Design Element will define types of defects that might be expected to exist, and which should not significantly impair the operation of the fuel pin. Single pin tests will then be run to determine the response of the reference LMFBR fuel pins to these off-normal conditions. The effects must be determined for all conditions of interest, and in particular, for various fuel exposures. Although some pre-irradiated pins may be available from Task 7-6, it is assumed that additional pre-irradiations may be needed under Task 7-10.2. In general the conclusions of this Task will be drawn from careful examination of tested pins and comparison to other pins which have experienced only normal, steady-state operation. In some instances, however, it may be necessary to take pins tested under this Task and subject them to extended steady-state operation (in Task 7-10.1) or to severe transients (in Task 10-2.4) in order to verify by actual test that their operational capabilities are unimpaired.

7.1.6.3 Task Objectives and Effect of Non-Achievement

The objectives of this Task are to demonstrate, to the satisfaction of licensing authorities, that expected operational

transients and fuel defects do not require premature plant shutdowns and fuel replacements. Non-achievement is possible in two basic ways: the information acquired may not be sufficiently definitive, or the results may demonstrate unfavorable fuel behavior. In either case the penalties could be significant. Since the LMFBR core is highly power-flattened, a large amount of fuel operates at or near peak conditions, and if one assembly had to be replaced, a significant portion of the core would probably also be involved. The penalties in fuel costs and plant downtime may not be tolerable if premature removal and replacement of fuel is expected more than once or twice in a plant lifetime. To avoid this, it is likely that fuel steady-state normal operating conditions would be reduced, by reducing specific power, until the results of the expected off-normal conditions were negligible. The economic penalty of the reduced specific power would depend on the amount of reduction necessary, which cannot be predicted at this time, but the size of this penalty could be significant to the overall plant economics.

7.1.7 Task Area 7-15 Accommodation of Fission Gas

7.1.7.1 Definition, Functional Requirements, and Current Technology

This Task Area covers the analysis, out-of-pile testing, fabrication of vent devices for in-pile testing, and evaluation of test data from Task Area 7-6, with the responsibility for determining the recommended fuel pin design measures for accommodating released fission gas. It is anticipated that the recommendation will be a vented concept, but the possibility of a plenum is not precluded.

The functional requirements to be satisfied follow.

- a. Accommodate the fission gas released from the fuel at burnups up to 125,000 MWd/Te U + Pu without exceeding the strain or stress limits of the cladding.
- b. Prevent intrusion of significant amounts of sodium into the fuel.
- c. Seal the fuel pin during pre-irradiation handling.
- d. Minimize fuel pin internal pressure, to minimize opportunities for and consequence of fuel pin failures.
- e. Minimize release of activity to the primary system, in order to ease problems of failed fuel detection, maintenance, and gas leakage.

Requirements d and e are conflicting, since the practical way of reducing activity release to the primary system is to build in a long holdup time for the fission gas, and a long holdup time within the pin implies pressure buildup. The results of the 1000 MWe study appear to favor minimization of pressure as a more important objective than minimization of activity release, but this conclusion could be altered in the future.

The various alternative approaches to the handling of the fission gases include the following:

- a. A simple "diving bell" vent in which no specific attempt is made to provide a holdup of the gaseous fission products. As in the reference design, there is provision to prevent liquid sodium from reaching the fuel pellets. The reactor system in this case must be designed to accommodate a large activity level during reactor operation.
- b. An extension of the "diving bell" vent, in which provisions are made for retaining

at least a portion of the chemically active gaseous fission products such as Cs, I, or Rb. This would reduce the activity in the primary sodium system, but would transfer a part of the problem eventually to the reprocessing plant.

- c. A pressurized vent, where a significant holdup is accomplished before venting gas to the coolant stream. Since little useful holdup in the fuel pin can be accomplished by a simple vent (as in a above), it is necessary to store the gas under pressure. This is considerably more difficult to accomplish, and it must be assured that an excessive pressure is never reached within the fuel pin. One other problem is that there would be a possibility of releasing a significant amount of fission gas from the pressurized pin during the refueling process.
- d. The non-vented fuel pin. In this case the fuel pin must include a gas plenum of suffi-

cient size to store the entire production of fission gas for the life of the pin. A length of about 1.5 core lengths located below the core (at inlet temperature) would be required. In addition to the length, the pin wall must be about 50% thicker (0.015 inch instead of 0.010 inch), causing a sacrifice in neutron economy. An additional penalty would be the possibility of leaking the pressurized gas during the refueling operation, or sudden release in the event of a core over-power.

The non-vented pin is here considered chiefly as a backup in the event that vent development is unsuccessful.

The current technology of plenum fuel includes many tests to varying burnups, both in thermal and fast fluxes. The status of vented fuel includes at least the following five tests, run under PA-10.

<u>Designation</u>	<u>Peak Linear Power (kW/ft)</u>	<u>Burnup Reached, (MWd/Te U+Pu)</u>	<u>Fuel Length, (inches)</u>	<u>Top Blanket, (inches)</u>
B4A	23	16,000	5.1	~1 3/4
B4D	15	47,000	20	~12
B3F	15	23,500	23	~9
B3D	18	7,000	20	~12
B3E	18	9,000	23	~9

All of these tests were essentially successful, as far as vent operation is concerned. All utilized simple diving-bell vents, and were tested in a thermal flux, in NaK circulated by natural convection. A meltable plug was used to seal the vent prior to operation. In the case of B4A, a very small amount of sodium was detected on the upper surface of the top pellet of the fuel column. There were no evident ill effects from this sodium. In the case of B3F, the capsule filling procedure

evidently caused the meltable plug to be removed and the fuel pin was filled with NaK before the start of operation, so that B3F was in effect a NaK-bonded and vented pin. Nevertheless, the test was completely successful and no evidence of damage was observed. The available data from these tests indicated about a five-day average holdup time in the fuel meat itself. B3D and B3E are currently being stored and could be returned to the reactor for testing to goal burnups of 100,000 MWd/Te U + Pu.

7.1.7.2 New Information Needed

The limited number of tests thus far performed do not provide sufficient data on which to base commitment of a 1000 MWe core design, nor is the preferred vent design necessarily identified at this time. It is recommended that the two tests which are now in storage be reinserted into GETR for irradiation to goal burnup. A successful conclusion of these tests would go far to demonstrate that there is at least one vent design, the simple diving bell, that will satisfy the requirements. This design will then be a backup for any more sophisticated or modified designs that will be pursued. In addition, work should be carried in the following areas:

- Analysis and design to specify one or more of the most desirable and promising vent designs as candidates for development,
- Out-of-pile testing of features such as the meltable plug, and any provisions for axial fuel movement that are closely related to the vent design,
- Fabrication and in-pile tests of approximately five vented pins, and
- Specification of reference vent designs to be utilized in proof testing (Task Area 7-10).

7.1.7.3 Task Area Objectives and Effect of Non-Achievement

The objective of the Task Area is to establish, through testing and analysis, a vent design that appears to have a high probability of success, for incorporation in the proof tests. The vent design is to satisfy the functional requirements as described above. Failure to achieve this objective will result in the use of a plenum to accommodate the

fission gas. The financial penalty to the 1000 MWe LMFBR would be a minimum of 0.03 mill/kW-h, not including the costs of a longer reactor vessel, greater refueling cell height, and larger refueling mechanisms. These items could easily double the penalty. This minimum cost penalty occurs when the plenum is made sufficiently long so that 10-mil cladding can be used. A plenum-to-core length ratio of 3.25 is required. This design may not be found to be desirable or feasible when more detailed design is carried out. In addition, there would be significant safety implications of a pressurized fuel pin, including possibly greater tendencies toward propagation of damage after a pin failure, and the risks of handling after irradiation.

Based on successful experience to date, it is felt that this Task Area enjoys a high probability of success for its objectives.

7.2 TASK AREAS 7-16, 7-17, 7-18— CARBIDE FUEL DEVELOPMENT

7.2.1 Definition, Functional Requirements, and Current Technology

These Task Areas include the development necessary to commit a carbide replacement core for the 1000 MWe LMFBR in fiscal 1982. Included is study of fundamental behavior of carbide (Task Area 7-16), evaluation of alternate carbide designs (Task Area 7-17), and carbide proof testing (Task Area 7-18). Carbide cladding development is included under the general cladding development tasks, 7-7, 7-8, and 7-9. It will be observed that the carbide program contains essentially the same work areas as the oxide, and these areas bear a similar relationship to each other. This is due to the fact that most of the problems (e.g., fuel swelling, fuel-cladding compatibility, cladding ductility, cladding swelling) are similar to the oxide development needs, and that a similar degree

of understanding and assurance of performance is required before a commitment can be made for a core.

Since carbide fuel is not as well understood as oxide, and the uncertainties are greater, it is not as easy to be specific about development needs in individual areas such as fuel-cladding compatibility or cladding ductility. Furthermore, the studies of carbide fuel were not carried out in as much detail as for the oxide, and economic incentives were not as well identified for individual development goals. Therefore, the carbide fuel development program is not broken down here into its individual Task Areas, but the functional requirements, development objectives, and current technology are discussed for the carbide fuel pin viewed as a whole. The content of the individual Task Areas may be inferred from that of the corresponding oxide fuel Task Areas.

The schedule proposed for carbide is that Tasks 7-16, Fundamental Behavior, and 7-17, Evaluation of Alternate Designs, will begin at the same time as the corresponding oxide tasks, the beginning of fiscal 1973. However, the work will be at a somewhat lower level and will continue for two years longer, to mid-1977, because of the less advanced development status of carbide as compared with oxide. Proof testing is scheduled to begin in approximately 1977 and end in fiscal 1982.

The functional requirements for carbide are not as well defined as for oxide, since no optimization study has been performed. The functional requirements further depend on whether the carbide fuel is gas-bonded or sodium-bonded, but even this choice has not been made in the Task III studies. The primary functional requirement is that the carbide has a clear economic advantage over the oxide, sufficient to justify its development costs. The Task III studies indicated

a possible advantage of 0.1 mill/kW-h for the gas-bonded carbide and 0.2 mill/kW-h for the sodium-bonded carbide, under the restrictions imposed by requiring that the carbide core replace the oxide with minimal plant alterations. Assuming that these advantages are sufficient (which is a separate question and has not been analyzed), the corresponding gas-bonded and sodium-bonded carbide pin performances may be typical of minimum functional requirements for the carbide. These functional requirements for carbide fuel are summarized in Table 7-5.

Although the combinations of carbide fuel performance characteristics in Table 7-5 will give the economic advantage indicated, under the economic assumptions used, (see Task III report) it is improbable that these are the best designs possible to achieve a specified improvement in power cost. The following discussion of the current technology will indicate the uncertainties involved.

In order to achieve an average burnup of 100,000 MWd/Te, the peak fuel burnup approaches values as high as 150,000 MWd/Te. Concurrent with this, peak total cladding fluences approach 5×10^{23} n/cm². These are, of course, similar to the oxide exposure requirements. When this is compared to what has been tested for carbide fuels, the unknowns are numerous. Several combinations of carbide fuel stoichiometry, fuel-to-cladding bond, and cladding material are considered as candidate designs; however the most promising appears to be a slightly hyperstoichiometric, sodium-bonded, stainless steel cladding design where the fuel-cladding mechanical interaction is reduced (or hopefully eliminated) by provisions of a radial gap to accommodate fuel swelling. Elimination of the mechanical interaction is very desirable because cladding ductility may be reduced not only by irradiation but also by fuel-cladding chemical interaction (carburization). The understanding of these

TABLE 7-5

CARBIDE FUEL, ESTIMATED TYPICAL FUNCTIONAL REQUIREMENTS
SUMMARY FOR ACHIEVING MINIMUM ADEQUATE COST INCENTIVES

	<u>Gas-Bonded</u>	<u>Sodium-Bonded</u>
Peak Linear Power, Normal Operation, kW/ft	~16	~27
Average Linear Power, kW/ft	~9	~16
Estimated Central Fuel Temperature, °C	1365	1195
Average Specific Power, kW/kg U+Pu (requires fuel pin diameter in the vicinity of 0.25 inch)	~160	~235
Cladding Thickness-to-Diameter Ratio (to maintain economic advantage over oxide fuel with 10-mil cladding)	~0.04	~0.04
Burnup, MWd/Te U+Pu, Core Average	~100,000	~100,000
Cladding Temperature, °F	~1300	~1300
Fuel Smeared Density, % Theoretical	0.74	0.82
Neutron Fluence, Coolant Temperature, Reliability, Transient Capability	Similar to oxide - see Table 7-3	

factors is currently somewhat limited and the design of carbide fuel for a fast flux and high temperature environment requires considerable extrapolation to achieve 150,000 MWd/Te. The effect of fast fluence on stainless steels has been discussed in the oxide fuel section and need not be repeated here. However it should not be overlooked that carburization of the cladding might have some unforeseen effect or interaction with the radiation damage effects.

Out-of-pile compatibility tests have shown, in general, that stainless steels appear compatible with the mixed carbides except for slight carburization. Alloys high in nickel content appear to react with the mixed carbides with the resultant formation of uranium-plutonium-nickel alloys. The presence of sodium enhances these reactions, especially when the dicarbide phase is available. In-pile compatibility results have not always conformed to out-of-pile observations. This might be

anticipated because of the inherent temperature gradient which exists in-pile and can act as a driving force for carbon movement. It seems clear that the question of cladding-fuel compatibility remains unanswered at high burnups, particularly with sodium-bonded carbides at cladding temperatures of interest.

Another primary concern of carbide fuel design for high burnups is the fuel swelling which appears to be about an order of magnitude greater for carbide than for oxide. Several in-pile tests have been performed on both hypo- and hyperstoichiometric uranium carbide and hyper-stoichiometric mixed carbide with two types of cladding, stainless steel and niobium. These pins were gas-bonded and the flux was thermal. The stainless steel cladding of the hyper-stoichiometric fuel was not expected to have carburized, due to the absence of a sodium bond, nor was the thermal flux exposure expected to produce intolerable steel swelling or loss of ductility. However two

stainless-steel-clad pins exposed to 100,000 MWd/Te failed, as did one pin at 37,000 MWd/Te. One stainless steel pin did achieve 22,000 MWd/Te. The niobium-clad pins achieved burnups of 114,000, 113,000, 31,000, 24,000 and 22,000 MWd/Te without failure.

These in-pile tests appeared to indicate the swelling rate of hyperstoichiometric mixed carbide fuel is an approximately linear function of both burnup and fuel center temperature. At a fuel peak central temperature of 1400° C the burnup swelling rate appears to be 2 percent $\Delta V/V$ per 10,000 MWd/Te. Thus to achieve the target core average burnup of 100,000 MWd/Te, the peak fuel may experience a 30 percent volume increase. If the swelling is isotropic (expected if unrestrained) this means the peak burnup fuel pellets would expand about 10 percent in both the radial and axial directions. Recent data suggests that swelling may not in fact increase appreciably with center temperature above 1400° C. No conclusive evidence indicates whether lower density carbides would offer advantages in swelling behavior.

It should be noted that although hypostoichiometric carbide fuel does not appear to carburize stainless steel, its fuel swelling rate is considerably higher than the hyperstoichiometric carbide at 1400° C (2 to 5 times as much $\Delta V/V$) and appears to become catastrophic above about 1500° C.

Stoichiometric mono-carbide appears to have the advantage of both the hyper- (low swelling) and the hypostoichiometric carbide (no steel carburization). However its swelling is not expected to be lower than that of the hyperstoichiometric carbide, and stoichiometry may be expected to change during burnup. Additives such as chromium carbide and iron have been studied to attempt to stabilize the carbon and avoid the cladding-

fuel interaction, but very limited in-pile experience is available (a few pins in GETR).

In-pile tests in a fast flux (EBR-II) of carbide fuel are currently in progress; some results are reviewed below.

All the available fast flux data on carbide fuel was obtained with helium-bonded pins and thus does not answer the question of the in-pile behavior of the sodium-bonded design. The principal drawback of gas bonding is its high resistance to heat transfer and thus it may be more difficult to meet functional requirements for specific power and fuel density simultaneously without going to very small diameter pins or lower burnups to avoid possible increased swelling at higher fuel central temperatures.

The fast flux data has tended to confirm the thermal flux fuel swelling data for hyperstoichiometric mixed carbide. Three gas-bonded pins were tested in EBR-II for UNC to peak burnups of 30,000 MWd/Te U + Pu and at central temperatures near 1500° C. All were found to be intact. Three types of cladding were tried; vanadium with 20% titanium alloy, 316 stainless steel, and "pure" vanadium. The 316 stainless steel operated at up to 600° C and appeared to be ductile in a post-irradiation test. All the claddings were quite thick (viz: 23 mils on a 0.25-inch o.d.) and the available gaps were large enough to prevent fuel-cladding mechanical interactions according to the measured swelling.

Six gas-bonded pins tested by ANL in EBR-II have been inspected after burnups of up to 29,000 MWd/Te (three pins) and then 69,000 MWd/Te (three other pins). Again different types of cladding were tried (316 and 304 SS, vanadium, Hastelloy X, and niobium) and all utilized vibratorily compacted mixed carbide except for one pelletized fuel pin. All pins

were reported intact but cladding ductility from post-irradiation testing is not known at this date. The fuels that went to the burnup of 69,000 MWd/Te U + Pu were relatively low (80 to 85%) density powder, and cladding temperatures were between 650 and 700° C.

Additional discussion of the current technology may be found in the Appendix to the Fuels and Materials Element of the LMFBR Program Plan. The status of carbide fuel development in 1972 to 1973, when the development program recommended herein would begin, cannot be predicted, but it is assumed that most of the questions raised in the above discussion of the current technology will still require further work.

7.2.2 New Information Needed

In addition to the basic understanding of cladding behavior under irradiation, an understanding of the fundamental behavior of carbide fuel is needed in the following areas:

- The dependence of fuel swelling on density, void deployment, cladding restraint, operating temperature, burnup, and stoichiometry,
- The combined effects of irradiation and carburization on cladding properties, and
- The effects of additives, cladding temperature, fuel stoichiometry, and the sodium bond on the fuel-cladding chemical interaction.

Work to develop better cladding alloys may be required, depending upon the results of development in these areas. These cladding alloys must not have such high neutron cross-sections as to negate the potential cost benefits of carbide. Ex-

tensive additional testing is necessary to identify the most promising carbide fuel pin concepts. At present the experimental evidence seems fragmentary and inconclusive.

Finally, as with the oxide, proof-testing of significant numbers of pins to ascertain reliability will be required.

The carbide fuel development must be guided by continuing core design studies to ensure that the desired goals are updated according to current information, and that the development effort remains strongly directed toward specific design goals that are determined by economics. The core design studies for carbide are part of the Core Design Element.

7.2.3 Objectives and Effects of Non-Achievement

The objectives of the carbide development effort will be to demonstrate, by proof tests on significant numbers of fuel pins, that a reliable pin design exists which satisfies the functional requirements associated with a significant potential for cost savings. On the basis of the limited effort in this study, it appears that all of the following must be achieved:

- a. Core average burnup equal to or greater than 100,000 MWd/Te (U+Pu),
- b. Core average specific power of about 200 kW/kg (U+Pu),
- c. Core fuel smeared density equal to or greater than 80% theoretical,
- d. Net plant efficiency equal to or greater than about 40% (a relevant objective because of its close relationship to cladding temperature),
- e. Unit costs of fuel fabrication and fuel recovery approaching values similar

to those of the long-term predictions for oxide fuel,

- f. No significant additional capital costs over those of an oxide fueled LMFBR, and
- g. Effects of cladding in parasitic neutron absorption similar to those of oxide fuel pins; that is, about 10-mil cladding thickness, of a material with absorption cross-section not too much greater than stainless steel.

Table 7-6 presents the approximate response of a carbide-fueled 1000 MWe LMFBR to a deterioration in each of the above goals except the capital costs themselves (goal f) which has a more obvious effect (i.e., 10 \$/kWe $\approx$ 0.186 mill/kW-h). The magnitude of the parameter change is in most cases arbitrary simply to illustrate the effect of each while holding the others constant at the base values.

Note that no mention of a required linear power, fuel central temperature, type of fuel-to-cladding bond, or fuel swelling rate has been made. These combine in the manner described in the

Task III report (Section IV) to result in the primary goals listed. For example, the desirability of low fabrication costs makes large fuel-pin diameters attractive. This combines with core specific power to make high linear powers desirable but not absolutely necessary to achieve low energy costs. With a sodium bond, peak linear power generation rates of less than 30 kW/ft can meet the above goals, as shown in the Task III report.

It should be noted that some of the fundamental requirements are obviously dependent. For example if unit costs of fabrication and recovery are found to be lower than predicted, then core average burnup could be reduced for equal energy costs. The probability of achieving the overall goal of a significant reduction in the energy costs as compared to oxide fuel is probably medium to high.

7.3 TASKS 7-11.3, 7-12, 7-13.3, 7-13.4, 7-14—OTHER MATERIALS DEVELOPMENT

7.3.1 Definition

This group of Tasks encompasses all the materials development not connected with

TABLE 7-6
COST EFFECTS OF NON-ACHIEVEMENT OF
CARBIDE DEVELOPMENT OBJECTIVES

<u>Assumed Parameter Change</u>	<u>Cost Effect</u>
Burnup decrease to 70,000 MWd/Te U+Pu	+0.2 mill/kW-h
Specific Power decrease to 150 kW/kg (as compared with 200 kW/kg)	+0.07 mill/kW-h
Cladding Temperature decrease to 540° C from 700° C (net plant efficiency decrease by 3.5 points)	+0.30 mill/kW-h
Fuel Smeared Density decrease from 80% to 75%	+0.04 mill/kW-h
Fuel Fabrication Cost increase by 35% (core fuel from 180 \$/kg to 240 \$/kg)	+0.05 mill/kW-h
Cladding Material Changed to Niobium from stainless steel	+0.30 mill/kW-h

the fuel pin, except for work specifically associated with the steam generator, and work on structural materials where the only novel element is the sodium environment. Included in these Task Areas are absorber materials, structural materials for intermediate fluences, and bearing and facing materials.

7.3.2 Task 7-11.3 Absorber Materials

7.3.2.1 Definition, Functional Requirements, and Current Technology

This task covers the development of absorber materials, and cladding or containment materials for the absorbers, for the LMFBR control rods. Testing and development of the control rods themselves is covered under Element 3, Components, and the results of the work under Task 7-11.3 will be important input for the control rod development. Task Areas 7-7, 7-8, and 7-9, in Cladding Development, also may provide important information to Task 7-11.3, since the fuel cladding material may prove suitable for control rods as well. However, the assumption has been made here that this will not be the case, and that a separate effort is required on control rod cladding.

The functional requirements of the control rod material are related to two different types of rods, shim and scram. The boron carbide scram rods are positioned about 9 inches from the core during normal operation and are not inserted except when the reactor is shut down. The shim rods must operate within the core, at higher neutron fluxes than the scram rods. In spite of the out-of-core position of the scram rods, it is estimated that up to 20 liters of helium may be generated per rod (containing 19 pins) over a 2-year period. Two years is a minimum requirement for control rod life, since this is also the life of a fuel assembly, and the rods are designed so that replacement at the same

interval as the corresponding fuel assembly is convenient. Longer life might, of course, be desirable economically. The helium must be accommodated by space within the control material pins or by venting. In addition it is desirable that the control rod cladding be able to tolerate the peak sodium outlet temperature from the core (1275° F) since this removes any special cooling requirement for the rod. Since this temperature would occur at a location where control rod heat flux is very low, it would also be the approximate temperature of the boron carbide-cladding interface. At this temperature the boron carbide and any products of irradiation must be compatible with the cladding. The shim rods are tantalum in the reference design. They must withstand a maximum total neutron fluence of about 5×10^{23} n/cm² (5×10^{22} above 1 Mev). Although no gas generation problem is anticipated, the tantalum must not exhibit extreme swelling or other degradation of properties which would make design extremely difficult. Again, it is desirable that the tantalum, or its cladding, be compatible with sodium at temperatures in the neighborhood of 1300° F, to obviate the need for overcooling the control rod. The maximum temperature of the tantalum is estimated at less than 1500° F.

In case of failure of the control rod cladding, the sodium and control material should be sufficiently compatible that the damage does not tend to propagate to other pins.

Boron carbide is presently being considered for use in the General Electric 300 MW Demonstration Plant and the FFTF. Therefore, it can be expected that material areas requiring attention will be investigated prior to construction of a 1000 MWe LMFBR. However, it cannot be predicted at this time which particular development needs will be

satisfied, since the demonstration plants and the FFTF may utilize conservative designs to side step development needs.

A brief summary of the major areas presently requiring investigation to use boron carbide as a fast reactor control material follows.

- Effect of Fast Flux Irradiation On Swelling and Gas Release

The majority of the irradiation data on boron carbide was obtained in a thermal flux at temperatures below 550° F. Due to differences in the flux spectrum, operating temperatures, and temperature and burnup gradients expected in fast reactor control rods, the gas release and swelling rates may be significantly different than were observed in thermal flux reactors.

- Compatibility of Boron Carbide and Canning Materials

Boron carbide has been found to be compatible with stainless steels at temperatures below 1100° F. Overcooling of the control rod channel will be required to maintain the cladding below this temperature. This will create mechanical design problems. Development of alternate cladding materials, which are compatible with boron carbide and capable of operating at higher temperatures, may be an easier and more economical solution to this problem.

- Compatibility of Boron Carbide Reaction Products and Cladding Materials

An appreciable amount of lithium will be present in highly irradiated boron carbide since it is produced when the

boron atoms react with neutrons. Examination of thermal reactor control rods has not indicated that the lithium attacks the cladding. Since the control rods in a fast reactor will be operating at a much higher temperature (1000° F or greater as compared to ~600° F) the effect that lithium may have on cladding integrity requires investigation.

- Effect of Sodium on Boron Carbide Cladding Compatibility

Little work has been done in this area at the present time. Effects which irradiation and sodium impurities may have on the boron carbide-cladding compatibility need to be investigated.

- Failure Behavior of Boron Carbide Pins in Sodium

Due to the lack of experience with tantalum as a control rod material (in either thermal or fast flux spectra) little is known about the effects of radiation on this material as compared to boron carbide. Most of the information now available was obtained at low temperatures in thermal flux spectra. Tantalum has been found to have good resistance to attack by high purity sodium at temperatures up to 1600° F. In impure sodium the corrosion rate increases with increasing oxygen content. Although the LMFBR design specified that the sodium purity is to be less than 10 ppm oxygen it is possible that this level may be exceeded in actual operation. In this case cladding of tantalum may be required to assure successful long term operation. Work to date indicates that tantalum-tungsten alloys (~10% W) have a much greater resistance to attack by impure sodium. Development work is then required to determine the effects of sodium impurities on tantalum and tantalum-tungsten alloys and suitable cladding materials for tantalum. The

meager data available indicate that tantalum is compatible with stainless steels and high nickel alloys.

7.3.2.2 New Information Needed

The following information is needed for boron carbide.

- An irradiation proof-test of several rods of varying density and enrichment at operating spectrums, burn-ups and temperatures should be run to determine the effect of these variables on gas release, swelling rate, thermal conductivity and cladding compatibility. The compatibility of the reaction product lithium with cladding material should be verified in these tests. At least one of the test rods should be purposely defective.
- Because the compatibility of boron carbide and stainless steel may not be satisfactory at operating conditions, alternate materials should be sought or developed.
- A boron-carbide vented pin should be developed and proof tested.

The following information is needed for tantalum and tantalum-tungsten.

- An irradiation proof test of several of the more promising designs should be run in fast fluences beyond 10^{23} n/cm² and at temperatures ranging from 1000° F to 1600° F to determine the effects of these variables on swelling behavior, mechanical strength and ductility.
- Development work is required to determine the effects of sodium impurities on tantalum, tantalum-tungsten alloys and other suitable cladding materials.

- A series of long-term anneals ranging in length from 1000 to 10,000 hours should be carried out at temperatures from 900° F to 1400° F to screen the potential cladding materials for incompatibility.

7.3.2.3 Task Objectives and Effect of Non-Achievement

The task objectives are to develop and demonstrate adequate materials performance for the shim and scram rods. At present, this is taken to mean clad boron carbide for the scram rods, and clad or unclad tantalum, or possibly tantalum-tungsten, for the shim rod. The specific objectives correspond to the functional requirements described in Section 7.3.2.1.

With the possible exception of the high-temperature compatibility problem at the boron carbide-cladding interface, the objectives are thought to have a high probability of achievement. Non-achievement of high-temperature compatibility would require overcooling of the scram control rods. This would degrade the reactor mixed-mean outlet temperature slightly and would complicate the mechanical and thermal-hydraulic design of the core. Economically, however, the penalties would be small. If it should prove impossible to vent the helium generated in the boron carbide, moderate plenums or increased clad thicknesses, or both, will probably suffice with little cost penalty. If the control rods should prove to have a more limited lifetime (e.g., one year instead of two), the requirement for replacement would perhaps result in a small increase in downtime, and the additional cost of replacement, as compared to a two-year life, would be equivalent to approximately 0.02 mill/kW-h. If the tantalum or tantalum-tungsten should prove unsuitable, and if a boron-carbide vented pin is not developed, the shim rods would have to utilize either boron carbide with a

long and inconvenient plenum or some other, less efficient control material that would result in a less favorable core composition with more space for control.

7.3.3 Task Area 7-12 Structural Materials

7.3.3.1 Definition, Function Requirements, and Current Technology

This Task Area encompasses the specific work necessary to develop structural materials other than cladding for the LMFBR primary and secondary systems, with two exceptions. Structural material not subjected to significant neutron irradiation is included in the Sodium Technology Element because the testing and development is entirely related to the sodium environment. Materials development related specifically and solely to the steam generator, such as tube wastage tests, is included in the Components Element. Task Area 7-12 therefore includes structural materials development for items such as fuel assembly cans, core support plate, and reactor vessel, to the extent that information developed under the extensive cladding development effort is not also sufficient for these other structural components.

The LMFBR program plan, Task Area 7-12, contains an extensive tabulation of structural components and the conditions to which they are exposed. The discussion here will be restricted to the in-core assembly cans, core support plate, and reactor vessel; the range of service conditions and development needs for these three items is representative of the range of requirements for non-cladding structures exposed to significant neutron fluence.

The flux and temperature conditions seen by the in-core fuel assembly channels are similar to those experienced by the cladding; a total fluence of 56.4×10^{22} n/cm²

accumulated over a residence time of 2 to 3 years, and peak temperatures of 1250 to 1300° F. These channels must resist an outward-acting pressure differential due to the coolant pressure drop, and must take repeated thermal and stress cycling as the coolant flow or reactor power is changed. Significant plastic strains, perhaps 2%, must be tolerated at the corners without cracking, in spite of irradiation embrittlement. Swelling should be minimized to avoid seriously exacerbating the problems of core clamping and fuel assembly removal and insertion.

The core support plate, in the reference design, accumulates a fluence of about 7×10^{21} n/cm² > 1 Mev at the upper grid plate. Total fluence may be a factor of ten higher. This is over a period of 35 years, and at a temperature of about 800° F. The support plate will be designed to remain in the elastic regime of the stress-strain curve, and so should not have to withstand significant strains. It should further be noted that the current design employs projections from the grid plate, over which the fuel assembly end pieces fit. These projections are considerably closer to the core and will see correspondingly higher fluences, but are essentially unstressed compared to the support plate assembly.

The reactor vessel temperature is also 800° F, but the fluence is much lower: approximately 9×10^{21} n/cm² total, 10^{19} n/cm² > 1 Mev. As with the support plate, it will not be required to tolerate significant strains during normal operation. However, there may be special requirements for ductility under certain highly improbable accident conditions such as pressure pulses due to sodium boiling. Such unusual transient requirements have not been analyzed for the structural components considered here.

The current state of technology for the fuel assembly can material is essentially identical with that for cladding material. The

fluences and temperatures required for the cladding and assembly cans are of course far above those required for the support plate and vessel; the lower fluences should result in less radiation damage, but the lower temperatures will mean that the damage is not annealed out as rapidly. The 800° F inlet temperature also happens to be well into the range in which significant swelling is expected. There is reason to believe that cumulative damage may be rate dependent and therefore, flux dependent, and that lower fluxes are favorable. The balance of these factors is probably such that less radiation damage is to be expected at a given fluence in the core support plate than in the cladding, but knowledge of the quantitative influences of the different parameters is very poor.

7.3.3.2 New Information Needed

The information needed for the assembly can design is similar to that needed for cladding design and is expected to be supplied by the cladding program. The information needed for the other structural components is mechanical properties data at low and intermediate fluences (10^{19} to 10^{22} n/cm² > 1 Mev), relatively low temperatures (800° F), and low fluxes ($< 0.1\%$ of the peak in-core flux). The practical effect of the last requirement is to make it impossible to obtain the desired experimental data in time for the design of the 1000 MWe LMFBR. Reliance must be placed upon development of adequate models and theories for radiation embrittlement and swelling, which can be extrapolated to yield predictions about the conditions of interest. These extrapolations will not be verifiable by experiment in the case of the low fluxes, except over a very limited range of flux; perhaps a factor of ten decrease from peak flux can be experimentally investigated to low fluences, but a factor of a 1000 or more is desired.

7.3.3.3 Task Area Objectives and Effect of Non-Achievement

The Task Area objectives follow.

- a. Monitor development efforts in Task 7-5.2 (Cladding Behavior Mechanisms) to ensure that proper effort is being directed toward a theoretical understanding of radiation damage at low temperatures and fluxes.
- b. Monitor development efforts and test data from Task Areas 7-7, 7-8, and 7-9 to make maximum possible use of available information in supporting predictions of structural materials behavior at low temperature and fluxes.
- c. Supplement data from cladding tests by performing irradiation testing under conditions of low flux and temperature, where feasible and appropriate.
- d. Recommend materials properties assumptions to be utilized in design of non-cladding structural materials exposed to neutron irradiation.

All of these objectives have a high probability of success. The recommended materials properties for design may have varying degrees of conservatism, however, depending on experimental results and on the confidence placed in the theoretical understanding and extrapolation of radiation damage.

The consequences of not achieving objective d, which is the end result of this Task Area, or of having to make very conservative recommendations, will be changes in the design of the core, its supporting structure, and the vessel. The economic

effects of having to utilize free-standing assembly cans to minimize strain are discussed in the Core Design Element. They are estimated at 0.1 to 0.2 mill/kW-h. If the core support plate cannot be feasibly designed at a fluence of $\sim 10^{22}$ n/cm² > 1 Mev, the fluence will have to be reduced by the use of extensive shielding incorporated as part of the fuel assemblies. Length would be added to the reactor vessel and the fuel assemblies. The costs have not been estimated, but it can be noted that an extra foot of length on the reactor vessel costs on the order of \$100,000. If the vessel design is not feasible or desirable at a fluence of 10^{19} n/cm² > 1 Mev, $\sim 10^{22}$ n/cm² total, additional shielding can be added around the periphery of the core, which

will increase the size of the vessel. The costs associated with a given decrease in vessel fluence have not been calculated; a 1-foot increase in vessel diameter is estimated to cost about \$200,000.

7.3.4 Tasks 7-13.3, 7-13.4, 7-14
Bearing and Facing and Seal
Materials

These Tasks are in support of components development under the Components Element; particularly pump, control rod, and other mechanism development. Needs in these areas will in all likelihood be supplied by on-going programs at the Liquid Metal Engineering Center and Materials Technology, Incorporated.

SECTION VIII

ELEMENT 9 - PHYSICS

8.1 INTRODUCTION

The objective of the LMFBR Physics Development Program discussed in this section is to provide new theoretical and experimental information and adequate analytical tools that are required for confident design, licensing, and operation of a high performance 1000 MWe reactor, as described in the Task II and III Reports. Confident design in the Physics Element means that the plant physics parameters can be predicted with assurance of satisfying safety and performance requirements without design margins that impose an unreasonable burden on the plant economics.

Table 8-1 outlines the required physics development program. Task areas, task numbers, and titles are generally the same as those of the LMFBR Program Plan, Volume 9, except where a reasonable fit could not be found for the intended scope of a task in the present 1000 MWe development program. Major facilities required for cross section measurements and critical experiments, indicated in Table 8-1, have been constructed under on-going AEC-funded R&D programs; and it is assumed that the operating costs for the LINAC and other accelerator and nuclear data experimental facilities used for the cross section and related data measurements will be covered in such on-going programs. Several of the tasks listed in Table 8-1 are required for both the Demonstration Plant and the 1000 MWe plant development programs. Table 8-1 indicates where it is planned to fund such tasks under the Demonstration Plant Development Program. The dollar costs and manpower shown for such tasks in Table 8-1 covers only that portion to be funded by the 1000 MWe development program.

The scope of the 1000 MWe LMFBR Physics Development Program focuses on measurement of the most urgently needed cross section data requiring improvement, a series of four critical assembly programs needed for support of possible demonstration and 1000 MWe designs, improvement and in-house acquisition and adaptation of analytical methods needed in combination with refined nuclear data for accurate prediction of LMFBR physics parameters, and development and maintenance of in-house capability for continuous data evaluation and for interpretation of integral critical experimental information.

Table 8-2 lists major milestones for the 1000 MWe Physics Development Program. For Task Area 9-1, acquisition of nuclear data, major emphasis is placed on the near-term measurement of the highest priority data (needed for the Demonstration Plant design as well as the 1000 MWe design) to intermediate accuracy (Milestone 1). The highest priority data is limited in this development program to Pu-239 [$\sigma(n, f)$, $\sigma(n, \gamma)$, $\bar{\nu}$] and U-238 [$\sigma(n, \gamma)$, $\sigma(n, n')$] so that available facilities and manpower can be sharply focused on these most vital measurements. The ORNL LINAC facility should be utilized to the fullest extent during the next few years on high priority Pu-239 and U-238 cross section measurements to assure timely improvement of these data in accordance with Milestone 1. The description below of Task Area 9-1 indicates the intermediate accuracy requirements for the highest priority data and also the longer-term required accuracy for the highest priority and other important data (Milestones 2 and 3). In-house capability for continuous data evaluation and file maintenance (Milestone 4) is a vital task to assure that the

TABLE 8-1

PHYSICS PROGRAM OUTLINE

<u>Task</u>	<u>Title</u>	<u>Facility^(a) Requirements</u>	<u>Costs^(b) (1000\$)</u>		<u>Manpower (Man Years)</u>	<u>Schedule (Fiscal Year)</u>
			<u>Equipment and Facilities</u>	<u>Operating or Engineering</u>		
9-1.1	Cross-Section Measurements					
	1. Highest priority data to intermediate accuracy	ORNL LINAC ^(c)	(c)	(c)	(c)	1970 to 1971
	2. Highest priority and other important data to required accuracy	ORNL and RPI LINACS ^(c)	(c)	(c)	(c)	1972 to 1980
9-1.2	Cross-Section Data Evaluation and File Maintenance by Plant Designer	None		400 ^(d)	9 ^(d)	1974 to 1980

(a) Computer facilities are assumed to be available and thus are not listed under facility requirements.

(b) All costs in 1968 dollars. Cost of computation is included in engineering costs.

(c) Also supplementary use of ZPR-III and ZPR-VI critical facilities; and of other accelerator and nuclear data experimental facilities at ANL, BNL, LASL, etc., as required to achieve the milestones for important data measurements. Operation of such facilities, including the LINAC facilities, are assumed to be covered by on-going AEC programs for the entire period of required use in support of the 1000 MWe development program.

(d) Manpower and computer costs for plant designer only; other costs assumed to be covered by extensions of on-going AEC programs.

TABLE 8-1 (Continued)

Task	Title	Facility ^(a) Requirements	Costs ^(b) (1000\$)		Manpower (Man Years)	Schedule (Fiscal Year)
			Equipment and Facilities	Operating or Engineering		
9-2.2	Benchmark Critical Experiment					
	1. Demonstration Size*	ZPPR	(e)			1970 to 1971
	ZPPR operation and fissile material charges			(f)		
	Planning and following by plant designer			(f)		
	2. 1000 MWe Size	ZPPR	(e)			1974 to 1975
	ZPPR operation and fissile material charges			(f)		
	Planning and following by plant designer			70 ^(h)	2.0 ^(h)	
9-2.12	Mockup Critical Experiment					
	1. Demonstration Plant*	ZPPR	(e)			1972 to 1974
	ZPPR operation and fissile material charges			(f)		
	Planning and following by plant designer			(f)		
	2. 1000 MWe Plant	ZPPR	(e)			1977 to 1979
	ZPPR operation and fissile material charges			(g)		
	Planning and following by plant designer			125 ^(h)	3.5 ^(h)	

(e) Facility construction and materials acquisition assumed to be covered by on-going AEC programs.

(f) May be funded under on-going AEC program or under a possible Demonstration Plant Development Program, or a combination of both.

(g) Assumed to be funded under on-going AEC program.

(h) The analysis of the experiments is covered in Task 9-2.13.

* These programs are hypothetical only since no firm plans, schedules or funding sources have been authorized for Demonstration Plants.

TABLE 8-1 (Continued)

Task	Title	Facility ^(a) Requirements	Costs ^(b) (1000\$)		Manpower (Man Years)	Schedule (Fiscal Year)
			Equipment and Facilities	Operating or Engineering		
9-2.13	Analysis of Critical Experiments by Plant Designer	None				
	1. Demonstration Plant*			(f)		
	2. 1000 MWe Benchmark			90	2.0	1975 to 1976
	3. 1000 MWe Mockup			150	3.5	1979
9-2.11	Shielding Benchmarks	ORNL Shielding Facility, ZPR Facilities	(e)	75 ^(d)	2.0 ^(d)	1970 to 1976
9-3.1	Diffusion and Transport Theory					
	1. Development of efficient two- dimensional multigroup code capability	None		(i)		1970 to 1971
	2. Improvement of two-dimensional multigroup codes to required accuracy and efficiency	None		550	12	1972 to 1980
9-3.2	Methods and Codes for Space- Angle-Energy Synthesis	None		450	11	1970 to 1974
	Two- and three-dimensional diffusion theory synthesis code					
9-3.4	Burnup Physics and Fuel Cycle					
	1. Fuel burnup and management code with flexible features for refueling and operations planning, including fuel cost calculations	None		(i)		1970 to 1972
	2. Fuel burnup and management code with precise reactivity, power distribution, and control rod position computation during operating interval	None		350	8	1972 to 1976

(i) Assumed to be funded under on-going AEC programs and possibly a Demonstration Plant Development Program.

TABLE 8-1 (Continued)

<u>Task</u>	<u>Title</u>	<u>Facility^(a) Requirements</u>	<u>Costs^(b) (1000\$)</u>		<u>Manpower (Man Years)</u>	<u>Schedule (Fiscal Year)</u>
			<u>Equipment and Facilities</u>	<u>Operating or Engineering</u>		
9-3.5	Dynamics Improvement of reactor- kinetics codes and reactivity feedback models for accident studies	None		400	10	1970 to 1978
9-3.6	Group-Constant Averaging 1. Improved rapid adjustment of group constants	None		(i)		1970 to 1971
	2. Complete system for rapid adjustment of group constants	None		300	8	1971 to 1975
9-3.8	Development for Code Linking and Compatibility by Plant Designer	None		350	9	1970 to 1977
9-3.9	Computers, Computer Language and Code Center 1. Acquisition and adaptation of codes not developed by plant designer	None		300 ^(j)	8 ^(j)	1970 to 1979
	2. Set acceptable coding standards based on comprehensive study of computers	None		70	2	1970
9-3.10	Shielding Methods 1. Cylindrical removal diffusion theory code	None		(k)		1970 to 1972
	2. Evaluate removal diffusion, Monte Carlo, other methods and data for shielding analyses	None		75 ^(d)	2 ^(d)	1971 to 1973

(j) Effort and costs strongly dependent on achievement of acceptable coding standards (those which do not destroy code efficiency) in Task 9-3.9.

(k) Assumed to be covered by on-going AEC programs and their extensions.

TABLE 8-1 (Continued)

<u>Task</u>	<u>Title</u>	<u>Facility^(a) Requirements</u>	<u>Costs^(b) (1000\$)</u>		<u>Manpower (Man Years)</u>	<u>Schedule (Fiscal Year)</u>
			<u>Equipment and Facilities</u>	<u>Operating or Engineering</u>		
9-5.2	Criticality and Radioactivity Data	PNL Critical Lab; ZPR Facility	(e)	70 ^(d)	2 ^(d)	1970 to 1975
9-6	Interface with Other Projects	None		70 ^(d)	2 ^(d)	1970 to 1980

TABLE 8-2

PHYSICS MAJOR MILESTONES

Government Fiscal Year	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Task Areas 9-1 and 9-2		1	4					2			3
		5	9		6	7			8		

LEGEND:

1	Measure Highest Priority Data to Intermediate Accuracy	(Task 9-1. 1)
2	Measure Highest Priority Data to Required Accuracy	(Task 9-1. 1)
3	Measure Other Important Data to Required Accuracy	(Tasks 9-1. 1 and 9-1. 5)
4	Provide Designers with In-House Capability for Continuous Data Evaluation and Cross Section File Maintenance	(Task 9-1. 2)
5	Complete Possible Benchmark Critical Experiments to Guide Nuclear Design of Demonstration Reactor	(Task 9-2. 2)
6	Complete Possible Engineering Mockup Critical Experiments for Demonstration Reactor	(Task 9-2. 12)
7	Complete Benchmark Critical Experiments to Guide Nuclear Design of 1000 MWe Reactor	(Task 9-2. 2)
8	Complete Engineering Mockup Critical Experiments for 1000 MWe Reactor	(Task 9-2. 12)
9	Provide Designers with In-House Capability for Analytical Interpretation of Integral Data	(Task 9-2. 13)

TABLE 8-2 (Continued)

Government Fiscal Year	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Task Areas 9-3, 9-4, 9-5, and 9-6			10		12		13		11		
	20					16 15		18	14 19 17		

LEGEND:

- 10 Provide Efficient Computational Capability in Two-Dimensional Diffusion and Transport Theory (Task 9-3. 1)
- 11 Complete Improvement in Acceleration of Two-Dimensional Diffusion and Transport Theory Codes (Task 9-3. 1)
- 12 Complete Two- and Three-Dimensional Multigroup Synthesis Codes of Adequate Accuracy for Majority of Design Problems (Task 9-3. 2)
- 13 Complete and Routinely Apply Fuel Burnup and Management Codes of Required Scope and Accuracy (Tasks 9-3. 4 and 9-4)
- 14 Complete and Routinely Apply Improved Dynamics Codes for Excursion Studies (Tasks 9-3. 5 and 9-5. 1)
- 15 Achieve Complete Capability for Rapid Adjustment of Group Constants (Task 9-3. 6)
- 16 Capability to Apply Shielding Design Codes of Required Accuracy and Efficiency (Task 9-3. 10)
- 17 Acquisition and Adaptation of Required Codes by Designers (Task 9-3. 9)
- 18 Complete In-House Development by Designers for Code Linking and Compatibility (Task 9-3. 9)
- 19 Apply Feedback From SEFOR, Possible Demonstration Plant, and Other Reactor Operation to 1000 MWe Design (Task 9-6)
- 20 Set Acceptable Coding Standards on Basis of Comprehensive Study of Computers (Task 9-3. 9)

1000 MWe design effectively utilizes the extensive experimental cross section information generated.*

The critical experiment program, described below under Task Area 9-2, emphasizes the engineering mock-up assemblies and the somewhat "cleaner" benchmark assemblies that have a geometry and a composition reasonably characteristic of a preliminary engineering version of the actual reactor design (Milestones 5 through 8). Development of in-house capability for analytical interpretation of integral critical experimental results (Milestone 9) is an essential and high priority task in the 1000 MWe Development Program (see previous footnote).

It is vital that the plant designer play a key role in the planning, following and analysis of the above benchmark critical assembly experiments. These benchmark experiments are identified with the general critical assembly program in ZPPR covered under Task 9-2.2 of the LMFBR Program Plan, Volume 9. It is expected that several fast reactor organizations, in addition to the plant designer, will conduct analyses of the benchmark assemblies as a means of checking and refining their computational methods.

Task Area 9-3 described below emphasizes the development of analytical capability that is most urgent for confident design. Parallel paths of development are defined, with the basic computational tools forming a path which provides a compromise between accuracy and cost. These tools will be sufficiently accurate for the vast majority of

design computations yet efficient enough to be used routinely. The other path is that of precise, but computationally expensive, methods needed to test and calibrate the more approximate tools, as well as to provide accurate results for final computations and the analysis of critical experiments. While a broad program of methods development would aid the design effort, the methods identified as priority 1 in Table 8-7, under Task Area 9-3, are vital.

Acquisition and adaptation of computer codes developed by outside organizations (Milestone 17) is a vital task in Task Area 9-3, since in-house code development by the designer organization can be expected to provide only a fraction of the total analytical tools required to satisfy the accuracy objectives of the 1000 MWe design development program. Means for simplifying this adaptation without penalizing code efficiency must also be found. Developments to improve ease of use are strongly emphasized in the code work, since the ability to routinely use reliable codes is an important element in achieving accurate physics predictions.

Table 8-3 shows target accuracies for design prediction of 1000 MWe physics parameters. The development program outlined here is intended to achieve these accuracies which would satisfy the objective of the program stated in the opening paragraph of this element.

8.2 TASK AREA 9-1—NUCLEAR DATA

8.2.1 Task 9-1.1 Cross Section Measurements

8.2.1.1 Definition and Current Technology

Measured differential neutron cross sections are the primary information source for the nuclear data used in neutronics computations. Differential cross sections are

* Milestone schedules shown in Table 8-2 for Milestones 4 and 9 are for one sub-task only of each of the related tasks—namely, that of acquiring adequate analytical tools to conduct in-house data evaluation or in-house analytical interpretation of integral experiments. Actual conduct of the evaluation and interpretation work would carry beyond these sub-task milestone dates as indicated in Table 8-1.

TABLE 8-3

TARGET ACCURACIES FOR PREDICTION OF 1000 MWe PHYSICS PARAMETERS*

<u>Startup Criticality</u>	$\pm 0.5\%$ k	(~1% of fissile content as enrichment design margin)
<u>Power Distribution</u>		
Core peak-to-average in first core	$\pm 2.0\%$	(includes local and gross neutronics peaking factors)
Core peak-to-average after several batch refuelings	$\pm 3.0\%$	
Ratio of blanket power to core power	$\pm 3.0\%$	
Power ratio between zones of a 2-zone core	$\pm 2.0\%$	
<u>Control Rod Worth</u>	$\pm 5.0\%$	(for any individual rod, as well as the entire assembly of shim and/or safety rods)
	$\pm 10.0\%$	(for any specified pattern of rod insertions)
<u>Shim Control Requirements</u>		
Burnup	$\pm 0.2\%$ k	(for 1-year refueling interval)
Full Power to Shutdown	$\pm 0.3\%$ k	
<u>Reactivity Temperature Coefficient Components</u>		
Doppler	$\pm 10.0\%$	
Sodium density	$\pm 15.0\%$	
Fuel density	$\pm 10.0\%$	
Structure and cladding density	$\pm 15.0\%$	
Core height or radius	$\pm 10.0\%$	
<u>Reactivity Power Coefficient</u>	$\pm 15.0\%$	(for slow change near full power)
<u>Fuel Cycle (~100,000 MWd/Te)</u>		
Burnup	$\pm 4.0\%$	(for any single discharged core bundle)
Fissile Pu (239 + 241)	$\pm 2.0\%$	(for average of a discharged fuel batch)
Fissile Pu (239 + 241)	$\pm 4.0\%$	(for any single discharged core bundle)
Breeding ratio	$\pm 3.0\%$	
<u>Abnormal Configurations</u>		
Fuel slump in a fuel bundle	$\pm 0.05\%$ k	
Sodium void in a fuel bundle	$\pm 0.05\%$ k	
<u>Shielding and Gamma Heating</u>		
Heat rate of near-core structures	$\pm 10.0\%$	
Fluences in-core and for near-core instrumentation and mechanisms	$\pm 20.0\%$	
Biological dose rates in access areas (during operation or shutdown) from Na and other material activation, streaming in bulk shielding penetrations, etc.	exp(± 0.7)	(factor of 2 attenuation design margin in shield thickness)

* Includes the reduction of uncertainties due to running an engineering mockup critical assembly program prior to final specification of the core and fuel design.

collected and evaluated, and then processed into a form that can be directly used in computations of k_{eff} , breeding ratio, power distributions, Doppler coefficient, coolant loss reactivity and control worth. Since a number of different neutron reactions are important and since the cross sections are strong functions of both material and neutron energy, a large amount and variety of data are required for a single LMFBR neutronics computation. To collect all the necessary data, a multitude of independent measurements must be considered and evaluated.

Many differential cross sections which are important to the LMFBR have already been measured and the volume of data is currently very large. However, there are still gaps in the required measured results and, more importantly, some of the most vital cross sections have seriously contradictory experimental results. In such cases, average cross sections have been generated for use in LMFBR neutronics computations, and the uncertainty in these values is excessive.

8.2.1.2 New Information Needed

The differential cross section measurements required for the 1000 MWe design effort

can be broken into two groups. The higher priority group involves the more important fuel isotopes and the eventual data accuracy required is very high. However, because of technical difficulties in making such highly accurate measurements, shorter term, more intermediate, accuracy requirements for these isotopes are also defined. Both intermediate and high accuracy goals are listed in Table 8-4. Milestone dates for these measurements are given in Table 8-2.

The second group of measurement requirements have lower priority and less stringent accuracy needs. Foremost among them are $\sigma(n, f)$ of U-238, Pu-240, and Pu-241 with long term goals of $\pm 8\%$ accuracy, as well as $\sigma_{\text{absorption}}$ of fission products, which is needed to $\pm 15\%$. The $\sigma(n, \gamma)$ and $\sigma(n, n')$ of Pu-240 and structural materials are also required to an accuracy of $\pm 15\%$ over the long term.

8.2.1.3 Task Objective and Assessment of Effect of Non-Achievement

The objective of this task is to schedule, measure, and analyze differential nuclear data to the degree of accuracy required. To do this, it will be necessary to not only develop measurement techniques of higher precision, but also to

TABLE 8-4
INTERMEDIATE AND HIGH ACCURACY GOALS FOR
TOP PRIORITY NUCLEAR DATA MEASUREMENTS

<u>Quantity</u>	<u>Material</u>	<u>Required Accuracy (%)</u> *	
		<u>Intermediate</u>	<u>High</u>
$\bar{\nu}$	Pu-239	± 1	± 0.5
$\sigma(n, f)$	Pu-239	± 7	± 3
α	Pu-239	± 10	± 5
$\sigma(n, \gamma)$	U-238	± 8	± 4
$\sigma(n, n')$	U-238	± 15	± 5

* ($E_n = 0.5 \text{ keV to } 5 \text{ MeV}$)

perform a more complete and in-depth analysis and evaluation of the data presently available. Such analyses are covered under Task 9-1.2.

Non-achievement of the high priority accuracies listed in Table 8-4 may result in erroneous design of the 1000 MWe core and fuel and subsequent delay in getting the plant on line. Proof-test critical experiments provide the final check on quantities such as k_{eff} , power distributions and control worth, but high precision data are needed to minimize the probability that proof-test criticals will indicate necessary design changes. Changes resulting from proof-test criticals can substantially delay reactor start-up, with the subsequent loss of plant revenue.

Safety coefficients are not easily verified by critical experiments. Moreover, the computed sodium void reactivity, in particular, is highly sensitive to cross section errors. Non-achievement of the specified accuracies will force designers to use overly-conservative safety coefficients in the design of the 1000 MWe reactor, which, in turn, will penalize the fuel costs. With the present uncertainties in these data, this fuel cost penalty is estimated to be about 0.1 mill/kW-h.

8.2.1.4 Schedule

Milestones for the measurement of cross section data to the necessary accuracies are listed in Table 8-2. These measurements will require the use of facilities such as the ORNL and RPI linear accelerators. The measurement program will, for the most part, involve continuation and intensification of current on-going programs.

8.2.2 Task 9-1.2 Cross Section Evaluation

8.2.2.1 Definition and Current Technology

The broad range of uncertainty and inconsistencies which exist in measured differ-

ential cross section data require evaluations of the measured data in order to arrive at singular values of cross sections for use in reactor design. As new measurements are made and reported, these results must be quickly factored into the evaluations, making this activity continuous throughout the design period of a reactor.

A number of evaluated LMFBR nuclear data sets currently exist in the U.S. as well as abroad. The Evaluated Nuclear Data File/B resulted from a cooperative effort between reactor contractors, and its development was directed by the National Neutron Cross Section Center. This file provides all LMFBR contractors with a common set of evaluated data which can be used as a reference point in comparative design studies. Development of such cooperative data files is continuing. However, the mechanics for changing cooperative files are too cumbersome and slow to maintain the necessary up-to-date files needed for design calculations. At present, each LMFBR contractor maintains its own evaluated nuclear data files.

8.2.2.2 New Information Needed

Current programs for both design-based and cooperative data evaluations need to be continued. The cooperative evaluation program centered about CSEWG and NNCSC is adequate as planned. For design of the 1000 MWe LMFBR it will be necessary for the design organization to perform extensive data evaluations and file maintenance. Throughout the design phase of the 1000 MWe LMFBR, the effect that important new cross section data has on reactor design must be determined in a timely manner.

8.2.2.3 Task Objective

This task will provide nuclear data files based upon the most recent measured data available for use in 1000 MWe LMFBR neutronics calculations. To do this it will

be necessary that physicists close to the design program keep abreast of new measurements, conduct detailed evaluations (including analysis of critical experiments), and maintain the data files which are used for the neutronics calculations. Such files will take the ENDF/B format as well as various multigroup forms. The cooperative evaluation activities of CSEWG will provide strong support to this effort.

Failure to provide for extensive designer-based nuclear data evaluations and file maintenance will make it virtually impossible to take advantage of the high-precision measurements called for in Task 9-1.1. The design and cost penalties associated with being unable to capitalize on these measurements are discussed under Task 9-1.1.

8.2.2.4 Schedule

Designer-based nuclear data evaluations will be a continuing effort, running beyond the end of the 1000 MWe design program. This effort must be supported by the cooperative evaluation activities of NNCSC, as well as by the computational methods development described in Task 9-3, which is important to data evaluation.

8.2.3 Task 9-1.3 Nuclear Theory

Theoretical augmentation of measured cross section data may be needed for the generation and maintenance of nuclear data files if the high-precision measurements in Task 9-1.1 are not forthcoming. Continuation of the current nuclear theory program will provide the capability to do this.

8.2.4 Task 9-1.4 Measurements of Spectrum-Averaged Cross Sections

Measurements of cross section ratios and integral cross sections in neutron spectra which simulate those of the LMFBR provide

test results for the evaluation of nuclear data. In addition to such data obtained as part of the critical experiment program in Task Area 9-2, a number of additional measurements may be necessary as problems arise in the evaluation of specific cross section data. These measurements can be incorporated into the current programs of ZPR-III and ZPR-VI.

8.2.5 Task 9-1.5 Other Differential Nuclear Data

Differential measurements of delayed neutron and fission product yields are important to the LMFBR design. Uncertainties are large for both sets of yields for fission produced by neutrons of energies appropriate to fast power reactors. Knowledge of the delayed neutron yields is necessary for consideration of reactor dynamics as well as the analysis of material worth measurements. Fission product yields will become increasingly important as LMFBR design progresses and fuel cycle computations are more accurate. These data measurements have lower priority than any of the measurements listed under Task 9-1.1.

8.3 TASK AREA 9-2—INTEGRAL EXPERIMENTS AND THEIR ANALYSIS

This task area is defined as the planning, execution and analysis of experiments designed to provide the following:

- Experimental benchmarks for the nuclear design and analysis of the LMFBR, and
- Experimental confirmation of the nuclear engineering aspects of LMFBR core designs.

The experiments under this Task Area must be planned as a consistent progression from experiments on smaller size cores up to the 1000 MWe size. The design effort of

the 1000 MWe plant will utilize results obtained from other experiments such as those which might be conducted as part of a Demonstration Plant Development Program.

The facilities required for this Task Area already exist. The essential facility is the ZPPR; some supplementary experiments may be performed in the other ZPR assemblies. A remaining major requirement is the acquisition of plutonium of suitable isotopic composition. This is discussed in Task 9-2.1.

The schedule for Task Area 9-2 integral experiments and their analysis is shown in Table 8-5.

8.3.1 Task 9-2.1 Operation of Critical Facilities

8.3.1.1 Definition and Current Technology

This task is defined as the operation and maintenance of facilities, including provisions for a permanent operating staff, required for performing the experiments described in the subsequent tasks of this Task Area.

The essential facility is the ZPPR, including fissile and other materials requirements. An additional facility, such as ZPR-III, may be required part time for supplementary experiments.

The task requirements are discussed in detail as part of Task 9-2.1 of the LMFBR Program Plan, Volume 9, Physics. It appears that the task requirements, with two significant exceptions, are adequately covered as parts of the following existing ANL programs:

- ZPPR Operations and Analysis,
- ZPR-III Operations and Analysis,
- ZPR Fuel and Non-Fissile Material Procurement.

The first exception is the provision for expanding the ZPPR matrix size. The diameter (core and radial blanket) of the 1000 MWe

Reference Design presented in the Task II Report is 118.4 inches, and just fits the present 10-foot matrix size of ZPPR. This is a rather tight fit which does not even allow the inclusion of reflector material surrounding the radial blanket. Any design change which results in an increase of the core diameter will certainly call for the expansion of the ZPPR matrix size toward the full table size of 14 feet. Planning and provisions for this expansion should be included in the development program.

The second exception is the acquisition of plutonium of suitable isotopic composition. It is expected that the early LMFBR's will be fueled with plutonium discharged from the first cores of light water reactors. The Reference Design described in the Task II report is based on such plutonium with the following composition:

Pu-239	58.9%
Pu-240	25.7%
Pu-241	12.2%
Pu-242	3.2%

While the experiments need not be performed using the same plutonium composition, high priority must be placed on acquiring water reactor discharge plutonium from fuel with relatively high burnup. The amounts of fissile plutonium required for experiments representing the initial core loadings is about 1600 to 2000 kg for the 1000 MWe plant. Experiments representing the core in mid-cycle conditions will require about 2200 to 2600 kg fissile plutonium.

8.3.2 Task 9-2.2 Benchmark Critical Experiments

8.3.2.1 Definition, Functional Requirements, and Current Technology

This task is defined as the planning and execution of experiments on a simple, but

TABLE 8-5

SCHEDULE FOR TASK AREA 9-2, INTEGRAL EXPERIMENTS AND THEIR ANALYSIS

realistic, critical assembly model of the 1000 MWe LMFBR core. These experiments will form a logical extension of possible benchmark experiments on a Demonstration Plant size core, and of possible mock-up critical experiments on the Demonstration Plant.

The functional requirement of this task is to provide experimental information to be used during the final reactor analysis and design as measures of the accuracy of analytical methods and computer programs, and as experimental benchmarks to which calculated quantities can be adjusted. This functional requirement will be met by:

- The definition of a proper critical assembly model, and
- The performance of the proper series of experiments on this model. A 1-year experimental program is envisioned.

The critical assembly model will be established jointly by the reactor designer and the organization operating the critical facilities. The model will represent a balance between a desire for simplicity and the requirement that the model represent those design features of the core which are of practical usefulness in the design of the core. One such feature is the two-zone radial arrangement of the fissile material for power flattening.

8.3.2.2 New Information Needed

Experimental information is needed to guide detailed core design of the 1000 MWe plant. This information must be obtained from experiments on critical assemblies of representative geometry and composition. The high priority measurements are those which bear most directly on the economic and safety aspects of the design: criticality, power distribution, control worth, and reactivity effects and coefficients.

A more complete listing of parameters of interest, together with target accuracies is given in Table 8-3. An exhaustive listing of reactor parameters, estimated current accuracies and target accuracies is found in the LMFBR Program Plan, Volume 9, Physics.

The detailed planning of the experiments is part of this task.

8.3.2.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task are the following:

- Definition of the required critical assembly model with participation by the plant designer,
- Planning, with participation by the plant designer, of the series of experiments required to guide the detailed core design,
- Performance of the experiments,
- Following of the experimental program by the plant designer, possibly, through a site representative, and
- Complete reporting of the experimental results.

The probability of achieving the objectives of this task is high. The effect of not performing the benchmark experiments on the 1000 MWe model can not be completely assessed until the results of other similar experiments which may be performed for the Demonstration Plant or the FFTF have been analyzed.

It is expected that there will be sufficient differences between the Demonstration Plant and the 1000 MWe designs that a benchmark experiment series will be called for. The unavailability of benchmark experiment results will require:

- Larger design margins at the final design stage,
- A longer mock-up experiment series, and
- Possible design changes after the mock-up experiment.

These in turn will effect the economics and the schedule of the plant. An additional 10% design margin on power distribution, for example, will require at least 10% more fuel for the same power rating, leading to an approximately 0.5 mill/kW-h penalty.

8.3.3 Task 9-2.12 Mock-up Critical Experiments

8.3.3.1 Definition, Functional Requirements, and Current Technology

This task is defined as the planning and execution of experiments on a detailed critical assembly model of the 1000 MWe LMFBF core. These experiments will form a logical progression from the 1000 MWe benchmark experiments. Whereas the benchmark experiments will precede the bulk of the detailed core design effort in order to provide maximum guidance for it, the mock-up experiments will follow the main design effort and will be scheduled for completion before the fuel fabrication begins.

The functional requirements for this task are to provide experimental confirmation of the important core design aspects and basis for final fuel fissile specifications. The critical assembly model will be established by the reactor designer in cooperation with the operator of the critical facilities. A 1-year experimental program is envisioned.

There is no detailed experimental information available currently in the U. S. on plutonium fueled LMFBF cores of the size required by the 1000 MWe plant.

8.3.3.2 New Information Needed

Experimental information is required to confirm the important nuclear aspects of the core design and to provide a basis for fuel fissile content specifications. The main design aspects which will be investigated are: criticality, power distribution, control worth, and reactivity effects and coefficients. The detailed planning of the experiments is a part of this task.

8.3.3.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task are the following:

- Definition of the required critical assembly model,
- Planning of the series of experiments required to confirm important core design aspects and to set final fuel fissile content specifications,
- Performance of the experiments, and
- Following of the experiments by the plant designer.

The probability of achieving the objectives is high. The probability that these experiments will confirm the calculated core parameters of the reference design described in the Task II report, and the effects of not performing these experiments can not be well assessed now. This assessment will progress as information from possible Demonstration Plant or FFTF benchmark and mock-up experiments becomes available.

8.3.4 Task 9-2.13 Analysis of Critical Experiments

8.3.4.1 Definition, Functional Requirements, and Current Technology

This task is defined as the analysis of the 1000 MWe benchmark and mock-up critical experiments by the reactor designer.

The functional requirement for this task is to perform the analytical and interpretive work necessary to make the critical experiment results directly useful to the plant designer.

The current technology of the analytical methods required to perform this task is discussed in Task Area 9-3.

8.3.4.2 New Information Needed

The experimental information needed has been described in Tasks 9-2.2 and 9-2.12. The additional information needed from this task is the accuracy of the design methods, empirical adjustments required in the core design, and the definition of further needs in analytical methods development and experimentation.

8.3.4.3 Task Objectives and Assessment of Effect of Non-Achievement

The objectives of this task follow.

- a. Interpret the experimental data and cast them into a form directly useful to the designer.
- b. Establish the degree of agreement between analytical predictions and experimental results.
- c. Recommend improved analytical procedures and empirical adjustments for core design.
- d. Assess analytical methods development needs as indicated by results of item b above.
- e. Provide guidance for subsequent critical experiments.
- f. Report the results and recommendations.

This task is closely tied to Tasks 9-2.2 and 9-2.12. It is assumed, therefore, that

if those tasks are performed, this will also be performed. To ensure that the actual design methods are tested and that the experimental information is of direct and maximum benefit to the 1000 MWe plant, the plant designer must, himself, perform this task. It is recognized that other organizations may also perform similar tasks.

The probability of achieving the objectives of this task is high. Penalties associated with not accomplishing the objectives could include lack of confidence in design techniques, and this would in turn require increasing design margins with associated penalties. If it were necessary to increase the fuel inventory by 10%, for example, to allow increased margins for achieving a given burnup, the economic penalty would be from 0.03 to 0.05 mill/kWh.

8.3.5 Task 9-2.11 Benchmark Information for Shield Design

This task is defined as the performance of experimental and analytical work required to provide the shield designer with benchmark information. The need for information for core design on the accuracy of analytical methods and on possible empirical adjustments is discussed elsewhere in this task area. Similar need exists, at lower priority, in the area of shield design.

Current target shielding design accuracies, and the outline of a development program, are presented in the LMFBR Program Plan, Volume 9, Physics. It is assumed that extensions of current work in this area at Oak Ridge National Laboratory and at Argonne National Laboratory will achieve the task objectives. Participation by the designer of the 1000 MWe plant must be included in this task both in the planning and the analysis of shielding benchmark experiments.

The new information needed includes the following specific problem areas:

- Measurements of the absorbed energy from the decay of fission products produced in the fast fission of plutonium, as a function of irradiation time and cooling time,
- Gamma and neutron source measurements on exposed fast reactor plutonium oxide fuel for fuel transfer cask design, and
- Neutron attenuation measurements in shields which consist of absorbing rods surrounded by sodium or gas.

8.4 TASK AREA 9-3—THEORY AND COMPUTATION FOR REACTOR ANALYSIS AND DESIGN

The physics computational tools required for the design of the 1000 MWe LMFBR will take two parallel paths of development. The basic tools for design will result from the first path and will be characterized by their ease of use, and the compromise they present between accuracy and expense of use. These tools, primarily in the form of computer codes, will incorporate carefully chosen approximations which greatly reduce computational costs, but which still provide sufficiently accurate results to be used for 95% of the design calculations. As each computer code of this type is developed, extensive evaluations must be performed to determine its accuracy for design applications.

The second path of development is that of the highly-precise physics computational tools. This forms a parallel path to the basic design tools in that these precise methods will perform essentially the same computations, but the approximations will be known to introduce only insignificant errors. While the precise tools will be generally much more expensive to use, they are required for the following functions:

- To provide calibration points for the less-precise basic design methods.
- To permit a few final, highly-accurate check computations after the reactor design is frozen, and
- To aid in the analysis of fast critical assemblies to minimize computational errors when one particular effect is to be isolated.

Table 8-6 lists the total requirements in each of the two parallel paths of development for seven different basic types of physics computational capabilities. Achievement of the computational capabilities outlined in Table 8-6 will permit accurate calculations of all the physics quantities important to the design of the first 1000 MWe LMFBR, as well as the associated activities involved in the analysis of critical experiments. It is known that methods currently available already permit accurate computations of many of these quantities. The most notable exceptions are:

- A near-zero sodium loss reactivity (for which it has yet to be shown that accurate computations can be made with present methods),
- Fuel cycle effects due to control rod movement and details of the fuel management scheme,
- The effects of fuel meltdown and sodium voiding on reactor excursions, and
- Detailed effects in critical experiments.

Additional precise methods must be developed for these problems.

While many precise methods for computing physics quantities are now available, a large program is required to develop and evaluate

TABLE 8-6

PARALLEL DEVELOPMENT OF PHYSICS COMPUTATIONAL CAPABILITY

<u>Basic Design Approach</u>	<u>Type of Capability</u>	<u>Precision Approach</u>
1. ENDF/B Format	Basic Nuclear Data Files Used	ENDF/B Format
2. Generation of Multigroup Data Files for Rapid Data Adjusting.	Nuclear Data Processing and Adjusting	Multi-Region, Ultra-Fine Group Computation for Specific Reactor Conditions (Advanced MC ²)
3. Rational Approximation plus S _n Cell Computations	Heterogeneity Effects	Multi-Region, Resonance Self-Shielding Computation. Monte Carlo
4. Synthesis Diffusion Theory Computations in 2 and 3 Dimensions	Neutron Flux Solutions	Direct Solutions in 1 and 2 Dimensions, for both Diffusion and Transport Theory, Maximum Rate of Convergence.
5. Startup to Equilibrium Calculation, Using Approximations for Effects such as Control. Maximum Flexibility for Out-of-Pile Treatment, Makeup Sources, Complex Fuel Management Schemes. Includes Optimization Procedures.	Fuel Cycle	Detailed Calculation of Single Fuel Cycle, Including 3-Dimensional Effects of Control, Fuel Changes, and Blanket as Functions of Burnup. Link with 2- and 3-Dimensional Synthesis Flux Solutions.
6. Computation of Reactor Excursions Based on Selected Assumptions Regarding Core Meltdown. Space-Time Effects Approximated.	Dynamics	Space-Time Computations, Quasi-Static Approach.
7. Neutron-Gamma Ray Removal-Diffusion Theory Computations. Various Simplified Shielding Codes. Shielding Data Files.	Shielding	2-Dimensional Monte Carlo Calculations, Neutron and Gamma Ray S _n Codes in 1 and 2 Dimensions.

the more approximate tools for computing these same quantities as well as the other physics parameters of importance. In addition, a most vital part of the basic design capability path is the development of efficient and simple linking of the codes on this path, such that all use the same basic nuclear data, have consistent input and output and automatically produce the data needed to link them together.* Since it is not expected that the design contract will perform all the methods development tasks required, the capability to easily convert codes from one computer to another must be achieved. To arrive at a simple code convertibility without seriously hampering operating efficiency will require an extensive study of the characteristics of different computers. This study should be given top priority such that acceptable coding and computer standards can be set at an early date.

As already stated, some of the tools described in Table 8-6 are in existence while others are being developed in current on-going programs. Also, a large share of this development will be required for the Demonstration Plant efforts, although longer range capabilities in nuclear data processing, neutron flux solutions, fuel cycles, dynamics, and shielding will not be ready in time for that design program. Computational methods and tools developed for the demonstration and 1000 MWe LMFBR's will also provide the basic physics tools needed for long-range general LMFBR development. The physics design computational capability which still must be developed has been divided into major tasks, which are described in more detail below.

The schedule for Task Area 9-3 is given in Table 8-7.

*This is not to propose a large code system which automatically goes through the full gamut of design computations. It is necessary to stop the computation between major links to permit the physicist to study the results and change direction.

8.4.1 Task 9-3.1 Diffusion and Transport Theory

8.4.1.1 Definition and Current Technology

One- and two-dimensional flux solution computer codes using diffusion and transport theory form the basis for all physics design computations. These codes must be capable of performing the computations with enough precision to reduce all uncertainties to the insignificant level. To do this they must handle as many as 100 neutron energy groups, plus a large number of zones and several hundred mesh points in each dimension. Since large problems will be expensive, every effort must be made to minimize running costs.

A number of computer codes already exist which fulfill part of the requirements of this task. Examples of codes which handle multi-group problems are DOT (2-D Transport), ANISN (1-D Transport), and 2DB (2-D Diffusion).

8.4.1.2 New Capability Required

The principal initial effort needed is to test, understand and evaluate the codes currently available and then to modify particular codes to meet the requirements and to accelerate convergence. The study and application of improved convergence acceleration techniques is a longer-range development need.

8.4.1.3 Task Objectives

The objective of this task is to provide a precise and efficient calculational tool for computing neutron flux levels and associated quantities in the design of fast reactors and the analysis of fast critical experiments. The calculational capability achieved by this task will also be useful for shield design.

TABLE 8-7
SCHEDULE FOR TASK AREA 9-3

Task	Development Item	Priority	Fiscal Year									
			1970	1971	1972	1973	1974	1975	1976	1977	1978	1979
9-3.1	Two-Dimensional, Transport and Diffusion Theory Codes	1	Basic Codes*		Improved Acceleration							
9-3.2	Two- and Three-Dimensional Flux Synthesis Code	1	Code Development and Evaluation									
9-3.3	Special Methods for Evaluation of Nuclear Data and Critical Experiments	2			*							
9-3.4	Fuel Cycle and Management Codes	1	Flexible Management Codes*		Detailed Cycle Codes							
9-3.5	Reactor Excursion Codes	1	Simple Versions		*	Improved Feedback Mechanisms						
	Quasi-Static Code	2										
9-3.6	Rapid Cross Section Adjusting Codes	1	Early Version*		Complete System							
	Improved Detailed Data Averaging Code	2										
9-3.8	Code Linking and Compatability	1	Limited Version*				Inclusion of all Codes					
9-3.9	Computer Coding Standards Study	1										
	Conversion and Adaptation of Outside Codes	1			*							
9-3.10	Removal-Diffusion Theory Shielding Code	1		*	Evaluation							
	ENDF/B Data Processing Codes	2										

*Indicates also requirement for Demonstration Plant.

Failure to develop computer codes with these capabilities will make it impossible to perform physics design calculations with the accuracy necessary for the 1000 MWe LMFBR. Failure to achieve the desired accuracy could result in penalties similar to those described in Task Area 9-2.

8.4.1.4 Schedule

There are currently many independent activities aimed at developing multi-group, multi-dimensional diffusion and transport theory codes. With proper direction, this task can be completed with continuation of some of the present programs. Development of adequate codes is required for the Demonstration Plant in fiscal year 1972. Improvement in efficiency will be needed by fiscal year 1975 (see Table 8-7).

8.4.2 Task 9-3.2 Synthesis Techniques

8.4.2.1 Definition and Current Technology

Synthesis techniques applied to multi-group, diffusion theory provide less accurate but much less expensive computations of neutron fluxes than the normal multi-group solutions of Task 9-3.1. Synthesis may be applied to either, or both, spatial or energy distributions of the fluxes.

There are a number of computer codes available for performing flux synthesis computations. Most are limited to few groups. The multi-group code BISYN uses a simple space synthesis approach and has performed well in fast reactor computations. Other fast reactor synthesis codes are being developed.

8.4.2.2 New Capability Needed

Direct solutions of multi-dimensional neutron fluxes using diffusion or transport theory in codes developed under Task 9-3.1 will be too expensive for the majority of design computations, even after acceleration

techniques have been improved. Experience with spatial synthesis calculations indicate high enough accuracy with this method for most design computations. A computer code is needed which will perform either two- or three-dimensional synthesis calculations in up to 100 groups, with the option to also use energy synthesis. This code will then allow for a broad range of approximations such that different degrees of accuracy can be evaluated and the proper balance between accuracy and cost determined.

8.4.2.3 Task Objectives

The objective of this task is to develop and evaluate a flexible, two- and three-dimensional, diffusion theory code based on a flux synthesis principle. This code will be sufficiently accurate for use in nearly all two-dimensional and all three-dimensional design problems for the 1000 MWe LMFBR.

Non-development of this code will result in excessive costs for physics design computations, since the codes of Task 9-3.1 will have to be used for routine calculations. It may also result in design errors or poor design due to reluctance by physicists to perform extensive computations that may be required with the more expensive codes. These errors could result in economic penalties of the type discussed in Task Area 9-2. This task is needed to ensure complete coverage of all design possibilities and to properly evaluate design improvements as they are proposed.

8.4.2.4 Schedule

Completion of the code and its evaluation should be accomplished by mid-fiscal year 1974 (see Table 8-7).

8.4.3 Task 9-3.3 Other Analytical Methods

A number of special methods for computing neutron fluxes and cross sections will

be useful for calculating specific problems in the 1000 MWe LMFBR, in the evaluation of nuclear data and in the analysis of criticals. The development of these methods is of lower priority, with the most important being (1) a Monte Carlo, or other approach, for computing precise resonance self-shielding in heterogeneous systems and (2) miscellaneous cross section evaluation codes. Current on-going programs at Argonne National Laboratory, NNCSC, and elsewhere will provide the necessary codes in these two areas.

8.4.4 Task 9-3.4 Burnup Physics and Fuel Cycle

8.4.4.1 Definition and Current Technology

Computational methods for burnup and fuel cycle calculations take many forms. There are currently a number of computer codes which calculate changes in fuel isotopic content with burnup, cycling back to flux solutions in zero, one or two dimensions. Some of these codes have the added flexibility of various possible refueling schemes, control changes over the refueling interval and out-of-pile treatment. The complexity and breadth of the problems of fuel management requires that computational abilities be continually improved as experience is gained using the fuel cycle methods available to indicate preferred directions.

8.4.4.2 New Capability Needed

Fuel cycle and management codes will be developed as follows. Early codes will include maximum flexibility in choosing management schemes, such as type of recycle, source of makeup, out-of-pile treatment, and loading/shuffling pattern, but will approximate many of the space-burnup effects. These codes will follow recycling from startup to equilibrium and will include restraints set on k_{eff} , control worth, and power distribution. Work on these codes may also be carried out on a possible Demonstration Plant

Development Program; if so, the 1000 MWe effort will concentrate principally on detailed computations of single cycles in specific fuel management schemes. Eventually these codes will allow iterative burnup-flux distribution calculations in three dimensions by incorporating three-dimensional synthesis flux codes from Task 9-3.2. They will permit detailed studies of non-uniform burnup and refueling coupled with control rod movement effects.

8.4.4.3 Task Objectives

The objective of this task is to develop fuel cycle and management codes which will provide the capability of determining the optimum fuel management scheme for the 1000 MWe LMFBR. The fact that fast reactors are not encumbered by the severe local power peaking factors of thermal reactors makes it possible to determine a near-optimum fuel management scheme for the first LMFBR's prior to startup. This difference gives the first LMFBR's a chance to be on an equal level, or better, with thermal reactors in this respect, and may well be a factor in accelerating the growth of the LMFBR industry.

8.4.4.4 Schedule

The early code development will be completed in time for use on the Demonstration Plant. Since this type of code development is currently in progress, the first part of the task can be completed by continuation of existing programs on PA-10 at General Electric. Completion of the detailed codes in step 2 is required by fiscal year 1976; as shown in Table 8-7.

The probability of achieving the objectives of this task is high. Failure to achieve the objectives would result in a non-optimum fuel cycle, with consequent economic penalties. If, for example, the average burnup were off-optimum by 20,000 MWd/Te due to changes in the power peaking factor over a

number of cycles, the economic penalty would be in excess of 0.02 mill/kW-h.

8.4.5 Task 9-3.5 Dynamics

8.4.5.1 Definition and Current Technology

The physics contribution to the study of reactor dynamics involves the treatment of time dependent neutronics, including the effects of reactivity feedback. Two primary areas associated with reactor safety are identified as being in need of methods development for the 1000 MWe LMFBR. First, and most important, is the capability to compute reactor excursions through disassembly with meaningful feedback reactivity due to the Doppler coefficient, fuel and structure expansion, sodium-loss fuel redistribution, and finally disassembly itself. Meaningful feedback refers to reactivity change which is tied to a realistic physical mechanism. Codes of this type assume separability of space and time, and are currently in the early stages of development. The second area of development is in space-time effects. Computer codes which treat space-time effects in a more exact manner, but approximate feedback effects, will provide correction factors for the results from the excursion codes. The quasi-static calculational approach, being developed at Argonne National Laboratory, will satisfy the requirements in this area.

8.4.5.2 New Capability Needed

Safety considerations in the LMFBR design require the capability to study the effects of reactivity feedback on energy release in LMFBR excursions. The first codes developed will make use of reactivity tables and simple models for fuel redistribution and sodium voiding. Working in cooperation with development programs under Program Element 10, Safety, improved codes will be written which include more realistic simulation of the mechanisms producing feedback.

8.4.5.3 Task Objective

The task objective is to develop the computer codes required to perform safety analyses of the 1000 MWe LMFBR. These codes must culminate in the capability to compute energy release in simulated power excursions using realistic mechanisms for reactivity feedback.

Non-achievement of this objective will hinder the safety analysis of these large, low-leakage reactors and subsequently make it more difficult to be assured of their inherent excursion limiting capabilities. Such assurances are necessary for licensing purposes.

8.4.5.4 Schedule

The first, simpler computer codes for computing reactor excursions will be needed for the Demonstration Plant. Modifications of these codes plus the writing of new codes which incorporate more detailed feedback mechanisms will continue well into the design period of the 1000 MWe plant. Most kinetics methods development must be completed, including evaluations, by fiscal year 1978.

8.4.6 Task 9-3.6 Group Constant Averaging

8.4.6.1 Definition and Current Technology

Group constant averaging is the process of generating multi-group nuclear constants (primarily cross sections) from pointwise data for use in neutron flux solution and other neutronics codes. Fast reactor applications require a detailed averaging procedure, including the effects of resonance self-shielding, secondary neutron energy and angular distribution, and flux spectrum.

Two basic procedures of group constant averaging are now being developed. Both use pointwise data in the format of ENDF/B. The first, more accurate, approach generates

energy-group constants for specific reactor compositions directly from ENDF/B. This approach is that used in codes such as MC² and GAFGAR. The second, rapid adjusting, approach is less accurate but much more efficient. In this method, generalized group constants are computed which include factors for corrections due to composition. Codes such as ENDRUN and ETOX, which generate this type of group constant, are being used to test the accuracy of the approach. Results indicate high enough accuracy for most design computations.

8.4.6.2 New Capability Needed

The highest priority need is the expansion and completion of the rapid adjusting approach to include angular data and corrections for interactions between resonances of different light materials. Associated with this are computer codes which use the generalized group constants to quickly arrive at values for specific reactor description. An extensive methods evaluation program will also be necessary.

The present detailed averaging codes for specific compositions are only sufficiently accurate to serve as rough tests for the rapid adjusting technique. New code development in this area is also needed to perform highly accurate calculations of the sodium voiding reactivity, as well as for the analysis of critical experiments.

8.4.6.3 Task Objective

The objective of this task is to develop computational techniques and computer codes for generating multi-group nuclear data from pointwise data in the ENDF/B format. The basic technique for design computations will be the rapid cross section adjusting approach in which generalized multi-group data are used to obtain group constants for a specific reactor description.

This approach will be tested for accuracy by comparison with results from the more detailed, MC²-type approach.

Lack of development of a complete rapid adjusting technique will severely reduce the accuracy of the physics computations for the 1000 MWe design. Without the capability of this approach to quickly generate composition and temperature dependent data, significant approximations will have to be made throughout the design program. While rapid adjusting capabilities already exist, more accurate and complete methods must be available in order to make use of the computer codes developed under Task 9-3.1, 9-3.2, and 9-3.4 in all of their capacity.

8.4.6.4 Schedule

Early versions of the codes for rapid cross section adjusting will be available for the Demonstration Plant. The complete approach must be available at the time the final 1000 MWe design effort begins. (See Table 8-7.)

8.4.7 Task 9-3.8 Code Systems

8.4.7.1 Definition and Current Technology

The term "Code Systems" refers to the linking together of separate computational modules to form a flexible system for calculation. In its simplest form, it consists of independent computer codes which have coupled input and output such that large amounts of information coming out of one module is in the correct form to read directly into the next. In its most automated form, computation continues from one module to the next automatically.

Large and small code systems are now both in existence and being developed. There are no existing systems which meet the requirements for the 1000 MWe LMFBR design program.

8.4.7.2 New Capability Needed

The physics code system needed for the 1000 MWe design must have the following characteristics:

- All codes within the system must use the same nuclear data file. This will require the specification of a generalized nuclear data file format and the development of the cross section averaging codes described in Task 9-3.6.
- Transfer of data from one module to the next must be computerized. Only information which is additional will be required input for the next module.
- Input and output formats must be the same on all modules, to the extent possible.
- Computation must halt at the end of specified modules, requiring decisions by a physicist before continuing. Since it is nearly impossible to include enough flexibility into an automated system, a simpler system of this type provides the greatest capability.
- Input must be in a form which minimizes chance for error and maximizes usefulness and internal checking procedures must be used.
- Computational modules are to be patterned after the basic design approach in Table 8-6, including codes for data averaging, neutron flux solutions, fuel cycle calculations, and kinetics.

8.4.7.3 Task Objective

The task objective is to specify a generalized nuclear data file, to specify module linking and standardized input, and to write and modify codes to meet these specifications. The devel-

opment of such linking and standardization is necessary to provide the efficiency required to meet design milestones. Without the system, the physics design effort will be incomplete and the percentage of user errors increased.

8.4.7.4 Schedule

Specification of data file format and linking requirements will be completed in fiscal year 1970 and an early, simplified version completed in fiscal year 1972. Conversion of codes will continue until fiscal year 1977. (See Table 8-7.)

Since this task requires close integration with the design physicists who will use the code, it is required that the plant designer set the specifications.

8.4.8 Task 9-3.9 Computers, Computer Language and Code Center

8.4.8.1 Definition and Current Technology

Development of the computer codes needed for the physics of the LMFBR design will be a cooperative effort involving a number of different organizations. To make these codes fit into the code system of Task 9-3.8 will necessitate their conversion to the designer's in-house computer. This conversion will not be a simple task since computer hardware, software and philosophy vary from installation to installation. The specification of a language and computer configuration which is common to all the computers will result in unacceptably inefficient codes, since such specifications will be limited to the combined minimum capabilities of the computers considered.

Attempts have been made to standardize computer programming. These standards are only a start toward achieving meaningful standards for code development, and code conversion continues to be a difficult and expensive task.

8.4.8.2 New Information Needed

Computer programming standards need to be set which go much farther than the specification of an inefficient common language. Before establishing these standards, a thorough study must be made of the characteristics of all computers involved in LMFBR code development, and an attempt made to include efficient machine-language systems routines, with standard capabilities, in these specifications. The extent of effort needed to convert codes and the efficiency of the codes, themselves, depends upon achieving acceptable coding standards.

8.4.8.3 Task Objectives

The objectives of this task are to arrive at standards for computer coding which do not significantly reduce the efficiency of the codes and to provide the capability to convert acquired codes to the designer's in-house computer. Application of standards which reduce code efficiency will result in a 20 to 40% increase in computer costs associated with the physics computations for the 1000 MWe design. Conversely, a total lack of standardization will require code conversion costs which are a sizable fraction of the original development costs.

8.4.8.4 Schedule

A study to determine acceptable standards for computer coding must be started as soon as possible. Conversion and adaptation of outside codes to the designer's computer will continue until fiscal year 1980.

8.4.9 Task 9-3.10 Shielding Methods

8.4.9.1 Definition and Current Technology

Shielding computational methods are required to assure the nuclear adequacy of the LMFBR shield design without incorporating too

much costly overdesign. An extensive shielding methods development program has been carried out over the last 15 years in the ANP program, at Oak Ridge National Laboratory and in the U. K. In spite of this the methods are still inadequate for the design of the 1000 MWe LMFBR. Most of the methods available are for more specific problems, for smaller shields, or are too expensive for routine design calculations.

A computational technique which shows the promise as the major neutronics shield design method is removal-diffusion theory. This approach provides about the correct combination of accuracy and cost for all but special shielding problems, such as those which involve streaming effects. For these special problems existing two-dimensional Monte Carlo and S_n codes will fill the need. A removal-diffusion theory code, MAC, is operational, but is limited to slab geometry. Oak Ridge is currently developing a removal-diffusion theory code in cylindrical geometry.

8.4.9.2 New Capability Needed

The development of a one-dimensional removal-diffusion theory code in cylindrical geometry is required. In addition, accurate neutron and gamma-ray data files for shielding applications are needed, as well as the codes for processing these data. As nuclear shielding data becomes available in the new ENDF/B format, codes for processing data in this format must be developed, and an evaluation of the shielding calculational methods performed.

8.4.9.3 Task Objectives

The objective of this task is to provide and evaluate the computational tools needed to perform the nuclear shielding calculations for the design of the 1000 MWe LMFBR. This involves the development and conversion of computer codes to process shielding data and

to perform the shielding computations, themselves. Failure to develop these tools will require overdesign of the LMFBR shield in order to assure adequate shielding capabilities, which in turn, increases the building size and adds to the plant capital costs.

8.4.9.4 Schedule

Completion and evaluation of a cylindrical removal-diffusion theory code will be required for the Demonstration Plant in fiscal year 1971. Codes to process ENDF/B shielding data and evaluation of results will be needed by fiscal year 1975. (See Table 8-7.)

8.5 TASK AREA 9-4—ANALYSIS OF NEUTRONIC CHARACTERISTICS OF REACTORS AND FUEL CYCLES

8.5.1 Definition and Current Technology

Certain specific analyses of the neutronics characteristics of reactors and fuel cycles must be performed at a relatively early stage in the development of LMFBR's to provide guidance for the long range program. While such analyses will not necessarily lead to knowledge or capability which is of direct use in the first LMFBR designs, their results will permit evaluations of these first designs in relation to long-range LMFBR prospects. In some cases results of these studies may effect a change in the first LMFBR designs, particularly with respect to fuel design and fuel management philosophy.

The analyses covered under this task have the purpose of answering certain basic questions regarding alternate reactor designs and fuel cycle concepts. The extent to which these answers are final depends upon the degree of sophistication in the computational methods used for the analysis. A number of such analyses have been performed in the past, but the development of more sophisticated methods under Task Area 9-3, as

well as other programs, will require re-evaluation of the conclusions drawn from the earlier studies.

8.5.2 New Information Needed

Design studies and analyses are needed in three basic areas. The first and most important to the early 1000 MWe LMFBR designs is that of studies of fuel management. The analyses performed in these studies will require computational tools which are now only in the planning and development stage. Advanced fuel management studies may answer general questions regarding the economic consequences of operating practices such as fuel shuffling and control rod sequencing. Inasmuch as these general answers apply to the 1000 MWe design, they will directly benefit that project. The other two areas of study are the evaluation of alternate reactor concepts, designs and materials, and the sensitivity of long-range economics to data uncertainties and computational approximations. These studies, in addition to providing guidance for long-range LMFBR planning, will form a basis for measuring the adequacy of the first 1000 MWe plant as a prototype of advanced systems. The computation of direct effects of data and methods uncertainties on the design of the 1000 MWe LMFBR will be included as part of the design program itself.

8.5.3 Task Objective

The objective of this task is to provide a basis for evaluating the 1000 MWe design as a prototype of advanced LMFBR's and for incorporating, when possible, more advanced thinking into the design. This objective, as it relates to the 1000 MWe design, is secondary to the primary objective of such studies, which is guidance of long range LMFBR development. If these analyses are not performed, it will be more difficult to assure that the early 1000 MWe designs are consistent with the direction of long-range LMFBR development.

8.5.4 Schedule

Analyses of this type should be performed as the necessary computational tools become available, thus closely tying the analysis schedule to the development schedule in Task Area 9-3. Since the primary objective of these analyses falls outside the category of required development for the design of the first 1000 MWe LMFBR's, cost estimates are not included here.

8.6 TASK AREA 9-5—INTERDISCIPLINARY ACTIVITIES

8.6.1 Task 9-5.1 Physics of Safety

The physics contribution to the development of computational methods for reactor safety is described under Task Area 9-3, Theory and Computation for Reactor Analysis and Design. Task 9-5.1 serves merely to indicate the close cooperation required between physics and other reactor disciplines, particularly in the area of reactor safety. Since the actual development is covered in other task areas, additional costs or schedules are not necessary.

8.6.2 Task 9-5.2 Criticality and Radioactivity Data

8.6.2.1 Definition, Functional Requirements, and Current Technology

This task is defined as the experimental and analytical work required to produce information for planning the fabrication, shipping, and reprocessing of the LMFBR fuel.

The functional requirement of this task is to produce criticality and radioactivity data of about $\pm 20\%$ accuracy for typical LMFBR fuel compositions and geometries found during fabrication, shipping, and reprocessing.

Experimental criticality data on mixed plutonium oxide-uranium oxide fuel of LMFBR composition is essentially non-existent. The

FFTF Development Program is expected to begin supplying some of this information.

Basic data currently available on the radioactivity of plutonium containing significant amounts of isotopes other than Pu-239 are estimated to be accurate, at best, within a factor of two.

8.6.2.2 New Information Needed

Criticality data are required for mixed oxide fuel in the approximate range of 10% to 20% fissile content, in the form of powder, pellets, rods, and fuel elements.

Radioactivity data are required for plutonium oxide and mixed oxide with plutonium isotopic composition from the relatively "clean" ($\sim 88\%$ Pu-239, $\sim 12\%$ Pu-240), to the expected composition of high-burnup water-reactor discharge plutonium ($\sim 4 \times 10^{-6}\%$ Pu-236, $\sim 2\%$ Pu-238, $\sim 46\%$ Pu-239, $\sim 27\%$ Pu-240, $\sim 15\%$ Pu-241, $\sim 10\%$ Pu-242). The information must include the time dependence of radioactivity after fuel reprocessing due to the buildup of plutonium daughter elements and the daughters of U-232 which accumulates in the recycled uranium.

8.6.2.3 Task Objectives and Assessment of Effect of Non-Achievement

The objective of this task is to provide criticality and radioactivity information of such accuracy that the facilities and procedures designed to fabricate, ship, and reprocess LMFBR fuel are not burdened by undue design margins and economic penalties.

The probability of achieving the objective of this task is high. Fuel fabrication, shipping, and reprocessing were not studied as part of the 1000 MWe LMFBR Follow-On Program; thus, a quantitative assessment of the economic penalties due to the non-achievement of this task is not available.

It is expected that the FFTF Development Program and other programs at Pacific Northwest Laboratories and at Argonne National Laboratory will supply some of the required information. Extensions of these programs should be planned as required under the 1000 MWe Plant Development Program with appropriate participation by the plant designer.

8.7 TASK AREA 9-6—INTERFACE WITH OTHER LMFBR REACTOR PROJECTS

This task is defined as the work required by the designer of the 1000 MWe plant to make

the results of the other LMFBR reactor projects fully applicable to the 1000 MWe plant. The projects of special relevance are the SEFOR Program, the Demonstration Plant Development Program and the Demonstration Plant Operational Program.

While the analyses and the reporting of the results are part of these other programs, it is expected that the designer of the 1000 MWe plant must plan an effort to make these results applicable to his particular design and analytical methods.

SECTION IX

ELEMENT 10 - SAFETY

The safety development program projected is keyed closely to the needs of the 1000 MWe LMFBF reference design. Rather than attempting to provide comprehensive understanding of safety-related areas and phenomena, emphasis is placed on high-priority items which appear to be particularly important in terms of safety, licensability, or costs.

In some instances, such as transient fuel testing, the required development tends to be very specific to the needs of the LMFBF reference design. This is because the information desired is, for example, the behavior of a particular fuel pin design, and it is unlikely that data and extrapolation from more general and broadly-oriented development efforts will be adequate support for licensing, without incurring penalties for additional conservatism. Furthermore, the schedule for such testing must be integrated with the oxide fuels development tasks in Element 7, Fuels and Materials. In other cases, such as fuel-sodium interaction and the mechanical effects of sodium vapor pressures on core structure, the information obtained will be of more general applicability, and in fact, may have significant effects on the core design as it evolves. For these reasons, the schedule for such work usually begins earlier than for work which depends upon availability of a specific, well-defined design.

As in the other Elements, the existence and successful operation of a Demonstration Plant is assumed. It is not expected that the Demonstration Plant or its development program will make major contributions to the safety development requirements of the large plant, however. In many areas, such as fuel transient testing and containment blast capability testing, the Demonstration Plant tests must be just as highly directed toward its specific design as will be the 1000 MWe development

efforts. The results will probably not be directly applicable to the 1000 MWe, therefore. In some areas, the Demonstration Plant may well utilize conservative design solutions which are undesirable for the large plant. Finally, the much longer time interval available before large plant operation makes it possible to plan for significant changes in accepted approaches to safety, with increased use of probability techniques to quantify safety levels and aid in rational decision-making, whereas the Demonstration Plant must rely on precedent to a much greater extent.

The Demonstration Plant will eventually be of great value in providing operating experience and reliability on systems and components and will allow better utilization of a quantitative safety approach. The development associated with it will also provide important additional information in many safety areas, even though it may be difficult at this time to identify specific 1000 MWe informational needs that are satisfied by the Demonstration Plant program.

The Safety Element has important interfaces with almost all other elements, but the most closely related areas are perhaps Core Design, Fuels and Materials, Instrumentation and Control, and Plant Design. Decisions on development and design in all of these elements must take into account the safety implications based on information from the Safety Element. On the other hand the results of the development and design in those elements will result in a continual evolution of the reference design which defines the conditions of interest and the goals for the Safety Element. The problems of integration require that a semi-continuous program of design and analysis be carried out in the areas of Core Design, Plant Design, and Safety.

A very high priority area in the Safety Element is the studies, methods development, and data accumulation required to further a quantitative approach to safety which utilizes a probabilistic definition of acceptable risk levels. In addition, high priority is assigned to those areas of work closely associated with possible damage propagation in the core, since the ability to limit damage to a relatively small area of the core in all situations which have significant probability of occurrence is a necessity. The requirements for elimination of damage propagation may also be controlling for the instrumentation and control requirements of the core. The items that are closely associated with the damage propagation problem appear to be fuel-sodium interaction, fuel failure modes, blast effects on internal structures, and the behavior of failed fuel subsequent to failure under conditions of constant power.

Table 9-1 shows milestones for the Safety Element. Table 9-2 summarizes the program costs and facility requirements.

9.1 TASKS 10-1.1, 10-4.1.1, AND 10-4.2—GUIDANCE AND SPECIAL STUDIES

9.1.1 Definition, Functional Requirements, and Current Technology

This group of Tasks consists of analytical studies and plant safety analyses, methods development needed for application for a probabilistic, or quantitative, safety approach, failure-rate data collection and analysis, establishment of safety criteria, coordination efforts with the Core Design, Plant Design, and Instrumentation and Control Elements, and guidance of the remainder of the safety tasks. It will be carried out on a continuing basis, beginning in fiscal year 1970 and continuing until the start of detailed design on the 1000 MWe LMFBR, assumed to start in fiscal year 1977. The effort will be focused by at

least two safety analysis studies. The first of these will take place at the beginning of the task, in 1970, and will provide an initial basis for defining development needs in the context of a quantitative safety approach. This first study is actually funded under the Plant Design development which follows. This first study is actually funded under the Plant Design Element, where it is designated as Task 2-1.4. The second study will take place in 1976, in conjunction with updated core and plant design studies (see Core and Plant Design Elements), and will provide necessary guidance for the final design, particularly in defining the degree to which and the areas where a quantitative safety approach can be used in licensing. In addition, it will define any desirable further development to be carried beyond the 1981 (operational) LMFBR concept. These two studies will be based on the best plant design data available at the time. The first study will obviously be exploratory in nature, due to the greater uncertainties in the design, the generally earlier state of development, and lesser experience in application of probabilistic techniques to reactor safety analysis.

Subsequent to the completion of the first study, in fiscal 1972, efforts in methods development and criteria establishment will be carried through at least 1977.

The functional requirements of these tasks are implicit in their definition. No component or system development is to be carried out, but guidance and coordination is to be furnished to the remainder of the safety development, as well as to all areas of the 1000 MWe LMFBR program where safety may be an important influence in design or in development objectives. For example, in some areas there may be key components or subsystems which have a disproportionate effect on the overall probability of a particular accident chain. These items need to be identified, and consideration given to the potential for

TABLE 9-1
SAFETY MILESTONES

	Government Fiscal Year												
	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982
Tasks 10-1.1, 10-4.1.1, 10-4.2 Guidance and Special Studies	1		2		3		4			5			
Task 10-2.1 Initiating Accident Codes			6					7					
Task 10-2.2 Hypothetical Accident Codes			8					9					
Task 10-2.3 Sodium Boiling			10			11							
Task 10-2.4 Fuel Failure		12			13				14				15
Task 10-2.5 Post-Burst Phenomena			16			17 18		19					
Task 10-2.6 Sodium Fires								20					
Task 10-2.7 Fission Product and Pu Release								21					
Task 10-2.9 Oxide Fuel Properties								22					
Task 10-3.2 Containment Development				23					24				
Task 10-5.3 Transient Reactor Facilities	25		26		27								

MILESTONES

- 1 Begin 36-month probabilistic study of 1000 MWe LMFBR Safety.
- 2 Complete study; define development needs to implement a more quantitative approach to safety, and determine priorities on other safety development in a probabilistic context; accelerate required work on methods development and establishment of criteria.
- 3 Initiate second plant safety study.
- 4 Complete second plant safety study; define the degree to which a quantitative approach to safety can be used in licensing for the 1000 MWe LMFBR in 1981; define required further development.

TABLE 9-1 (Continued)

- 5 Contribute to obtaining operating license for the 1000 MWe LMFBR.
- 6 Initiate required initiating accident code development effort as defined by milestone 2, and not yet available from PA- 10, Demonstration Plant work, or ANL Programs.
- 7 Complete basic initiating accident code development for use in licensing of 1000 MWe reference design; continue with minor modification and liaison activities.
- 8 Initiate hypothetical accident code development, as in milestone 5 above.
- 9 Complete hypothetical accident code development for use in licensing of 1000 MWe reference design.
- 10 Begin experimental program on sodium vaporization under typical 1000 MWe design conditions and geometries.
- 11 Complete experimental program on sodium boiling.
- 12 Begin pre-irradiation for transient testing.
- 13 Begin transient testing of single pins in TREAT, SEFOR or other facility; begin transient tests of damage propagation in SEFOR (or other facility).
- 14 Complete oxide-fuel transient testing; recommend design assumptions for 1000 MWe LMFBR fuel pin transient capabilities and failure propagation behavior.
- 15 Complete carbide fuel transient testing; make recommendations as for oxide.
- 16 Initiate experimental studies of sodium vapor pressure and void propagation effects on core structure in alternative 1000 MWe LMFBR designs.
- 17 Complete experimental studies of sodium vapor effects on internal core structure; make safety recommendations to be factored into choice of core structure design.
- 18 Begin experimental program for tests of energy absorption by primary system or containment, or both.
- 19 Complete tests of energy absorption by primary systems; make safety recommendation for design of sodium tank, shield plugs, and component arrangement.
- 20 Required experimental data and analytical methods for sodium fires available.
- 21 Required experimental data and analytical methods for fission product and Pu release available.
- 22 Required physical and mechanical properties of oxide fuel available.
- 23 Begin containment development; design tests.
- 24 Complete containment development testing for 1000 MWe LMFBR.

TABLE 9-1 (Continued)

- 25 Begin study of required transient test facilities and possible modifications to SEFOR.
- 26 Begin facility modification.
- 27 Transient test facility for fast fuel assemblies or partial assemblies ready for operation.

TABLE 9-2
SAFETY PROGRAM OUTLINE

<u>Task(s)</u>	<u>Title</u>	<u>Facility Requirements</u>	<u>Equipment and Facilities</u>	<u>Operating or Engineering</u>	<u>Manpower (Man Years)</u>	<u>Schedule Fiscal Year</u>
10-1.1 10-4.1.1 10-4.2	Guidance and Special Studies	None	—	2400	64	1970 to 1980
10-2.1	Initiating Accident Codes	None	—	375	10	1973 to 1980
10-2.2	Hypothetical Accident Codes	None	—	250	7	1973 to 1977
10-2.3	Sodium Boiling	Out-of-pile loop; power and flow transient capability in-pile, single pins	200	1100	10	1973 to 1975
10-2.4	Fuel Failure*	In-pile power and flow transients, multiple pins, fast spectrum, from steady-state initial conditions	—	6000	30	1973 to 1982
10-2.5	Post-Burst Phenomena	Partial core mock-ups; scale models of tank and internals	—	1200	12	1973 to 1977
10-2.6 10-2.7 10-2.9	Sodium Fires Fission Product and Pu Release Oxide Fuel Properties	Covered by On-Going Programs				Complete by 1978
10-3.2	Containment Development					
10-5.3	Transient Reactor Facilities	Not determined; likely to include large in-pile loops for meltdown testing, and possibly scale mock-ups of containment concepts	3000**	5000**	30**	1972 to 1978
		Facility intended for needs of Task 10-2.4	5000	1200	30	1972 to 1978

*Costs are for oxide only; carbide costs would be similar in magnitude.

**Extremely uncertain. Efforts required could be very much less.

establishing their reliability at an early date by accelerated testing as a part of the development program. The tests themselves would be conducted within other elements, such as Components or Instrumentation and Control, but the Safety Element would provide guidance as to which items are amenable to such testing, and the relative incentives for specific component testing. Such techniques will obviously not be applicable to many large components and systems (reactor piping, heat exchangers, pumps, etc.), but may be quite useful in the development of items whose operating time is significantly less than continuous (control rod drives, isolation valves, control switches, relays, circuits, etc.). In some instances, other parameters such as cycling temperature or load might be substituted for time to simulate accelerated testing. Thus, an important function of this task is to provide active guidance to statistical testing, in addition to the passive collection of reliability data.

In addition, the use of probabilistic methods and quantitative safety criteria is to be furthered wherever possible, and a continuing assessment is to be made of the best licensing approach for the 1000 MWe LMFBR.

The current technology of the various safety areas is discussed under the other individual tasks. The current technology of the probabilistic approach to reactor safety is not well developed. It has not generally been used in licensing, although efforts to utilize probability to compare alternative designs or safety measures are increasing. Failure rate data, which is necessary to obtain the maximum benefits from a quantitative safety approach, is either non-existent or extremely sparse in numerous important areas. Some additional discussion may be found in the Task III report and under Task 10-4.2 of the LMFBR Program Plan.

9.1.2 New Information Needed

Other than providing guidance and coordination to the other safety tasks, the new information needed from these studies falls into two categories:

- Information on the safety implications of alternative design concepts, to assist the Core and Plant Design Elements, and specifically to provide a basis for important decisions on the safety features of the final design for the 1000 MWe LMFBR, and
- Information needed to further the quantitative, or probabilistic approach to safety, including adaptation of methods, failure rate data, and studies providing a basis for considering possible safety criteria stated in probabilistic terms.

Work has been initiated at the Liquid Metal Engineering Center to develop standard forms and procedures for reporting of relevant failure data, and to publish a Failure Data Handbook. This type of effort will not be duplicated here, but the emphasis in respect to data will be on taking advantage of available data from this and any other sources, assessing its significance for safety, and identifying priority needs for more data or improved reliability.

9.1.3 Task Objectives and Effect of Non-Achievement

The most important objective of these Tasks is to determine how to further the use of probabilistic methods and concepts, and to encourage such methods and concepts as approaches to fast reactor safety, to the greatest extent possible consistent with the 1000 MWe reference design schedule.

Additional objectives are to ensure that appropriate priorities are observed in the safety development work, to ensure that design decisions are made with maximum possible awareness of safety implications, and to identify the appropriate areas where reliability testing may be performed as a part of the components and instrumentation development. The objectives relating to a quantitative approach to safety have a low probability of complete achievement, but there is a high probability that the accepted approach to safety and licensing can be significantly influenced and modified. It is likely, therefore, that significant steps can be taken toward more rational safety requirements and risk assessments, but it is also likely that some important design measures will still be required for safety reasons without a good quantitative justification.

The consequences of not achieving acceptance and successful application of the probabilistic safety approach are inevitably an accretion of plant features, which appear on the surface to increase safety, but without any quantitative measure of the benefit. The effects on power costs may be dramatic, even to the extent of rendering the LMFBR non-competitive in some circumstances. No detailed estimates have been made of the costs of any particular safety features, nor is it possible at this time to predict what incremental safety features would be required. However, the comparative studies done in Task I of this study may be taken as illustrative of the possible penalties.

In this comparison, the economics of two cores were calculated. In one core, termed a "conservative" design, the following criteria were imposed:

- Complete voiding of the core and blankets shall result in a zero or negative reactivity effect.

- The Doppler $T\text{-}dk/dT$ with sodium voided from the core and blanket shall be -0.004 or greater in magnitude.
- Containment shall be furnished for a core disassembly accident. In the absence of detailed studies, 2000 MW-sec of available work was assumed to be representative of the amount of energy to be contained.

The first of these criteria will avert large sodium reactivity effects as long as voiding occurs at different times in different parts of the core. This should be the case in many circumstances that might be postulated. The second and third criteria are based on calculations of the effect of the Doppler coefficient on energy release.

It is clear, qualitatively, that the magnitude of the energy release in a core disassembly is dependent on the core design. A dependence on sodium void reactivity is expected because of the possible contribution of sodium voiding to reactivity insertion rates, either independently or in conjunction with reactivity changes due to fuel movement. Therefore, modifications of core design to increase the Doppler coefficient and to decrease the sodium void reactivity should, in principle, reduce the magnitude or the probability, or both, of core disassembly accidents. Previous conceptual designs have made significant compromises in core design in order to achieve this result and, in addition, have made provision for containing energy releases up to 1000 MW-sec of available work.

This combination of precedent and of a plausible argument for increased safety would make this kind of a core design a likely requirement under the current approaches taken to safety and licensing. However, in the absence of a probabilistic approach there is no way to:

- Assess quantitatively the degree of safety improvement that is accomplished through these criteria.
- Determine whether or not the plant is already safe enough without application of these limitations on the core design.
- Determine whether an equivalent improvement in safety might be accomplished in other ways at less cost.

The economic penalties attached to a core design that is limited by the above criteria were estimated by comparing it to a core optimized for economic performance. The physical characteristics of the two cores are described in the Task I report. The core design that meets the safety criteria must be compromised by shortening its height and incorporating beryllium oxide.

The fuel cycle costs represent the major cost penalty of the compromised design. Figure 9-1 shows the fuel cycle costs for both designs as a function of plutonium value. Because of the uncertainty in particular nuclear data and the uncertainty associated with the safety criteria imposed on the conservative design, alternative conditions were evaluated. For example, a more stringent sodium void criterion was applied to the conservative design. The base case shown on the curve corresponds to a maximum void reactivity criterion of approximately 2.65\$ and used the existing Brookhaven Evaluated Nuclear Data File (ENDFB) nuclear data. The lower boundary of the "reduced void criterion" curve corresponds to a maximum void reactivity of 1.00\$ and the upper boundary corresponds to 50\$, also with the existing ENDFB nuclear data file.

More recent British work and measurements performed by Oak Ridge National Laboratory at Rensselaer Polytechnic Institute have indicated that the plutonium alpha

value utilized for these calculations is probably too low in the energy range between 1 and 10 keV. An increase in the plutonium alpha value affects the conservative core more than the advanced core, since it not only decreases the breeding ratio and increases the required enrichment, but also increases sodium void reactivity. Thus, the conservative core must make further compromises in design to satisfy any given void criterion. The range of costs estimated for the increased plutonium alpha values corresponds at the upper end to the pessimistic side of the error bars obtained by Sowerby and Patrick at Harwell,⁽¹⁰⁾ and at the lower end to our own estimate of the most probable values.⁽¹¹⁾

It should be noted that if increased alpha values are combined with more stringent sodium void reactivity criteria, the economic effects are not simply additive, but tend to reinforce each other.

The economic criteria of Figure 9-1 are quite optimistic and should be considered as representative of the cost targets for a large and fully matured fast breeder industry. Figure 9-2 shows fuel cycle costs using less optimistic assumptions such as might be appropriate in the near term when fabrication and reprocessing plant throughputs are low. Again, the alternative safety and nuclear data effects are also considered.

From examination of both Figures 9-1 and 9-2, it is concluded that under optimistic circumstances, the conservative design shows only 0.2 mill/kW-h cost penalty. However, the potential for this difference to increase is high. Under some circumstances it is doubtful that the conservative concept could compete with other forms of power generation expected to be available in the 1980 period (the 1.5 mills/kW-h of fuel cycle costs for the reduced void criterion in Figure 9-2 is comparable to the fuel cycle costs of today's water reactors). The economically optimized design shows excellent fuel cycle costs under all circumstances.

FIGURE 9-1. COMPARISON OF ECONOMICALLY OPTIMIZED VERSUS CONSERVATIVE DESIGN FUEL CYCLE COSTS WITH OPTIMISTIC ECONOMIC CRITERIA

FIGURE 9-2. COMPARISON OF ECONOMICALLY OPTIMIZED VERSUS CONSERVATIVE DESIGN FUEL CYCLE COSTS WITH ALTERNATE ECONOMIC CRITERIA

It is of course not possible to say whether the conservative core design criteria used in this comparison would be deemed to be sufficient or inadequate under current safety and licensing procedures. Furthermore, the economically optimized design might require additional engineered safeguards to make it acceptable even under a probabilistic analysis, and these might reduce the cost differentials. However, it is felt at the current time that the real safety advantages of the conservative design are small or non-existent, and therefore not worth the cost penalty involved. This comparison could therefore well be representative of the possible penalties involved in failing to implement the more rational, quantitative approach to safety.

It will be seen from a perusal of the other Tasks in the Safety Element that emphasis on a quantitative, or probabilistic approach to safety has in no way reduced the needs for phenomenological understanding and predictive ability in accident analyses. In fact, these needs tend to be increased, even for very extreme, "hypothetical" accidents. For example, in order to establish goals for the probability to be allowed for core disassembly accidents, the consequences of such accidents should be well understood. To the extent that they are not well understood, the consequences will have to be conservatively estimated, (if a deterministic approach is used at this juncture) or the probability distribution for the accident consequences will have a very large variance. In any event, the result will be a requirement for a lower probability of occurrence, and this can only be achieved at some economic cost. It is altogether possible that a quantitative approach to safety will be of very limited economic benefit to the first large plants if uncertainties in accident analysis are too great. The incentives to place safety requirements in a more logical framework and thus avoid unneeded escalation in safety features in later plants would remain, however.

Recognizing that the degree of success of the quantitative approach to safety is dependent on the success of other safety developments, as well as on public opinion, on difficulties in altering procedures based on established precedents, and on the complexity of the complete plant safety analysis, it must be clear that the degree to which probabilistic techniques and criteria can be employed will be under continual review in these guidance studies. It is possible that de-emphasis of probabilistic approaches will be appropriate if further study shows a very poor prognosis for the usefulness of these techniques. If so, the remainder of the safety element would still continue with essentially the same work in most areas, with possibly some changes in priorities.

9.2 TASK 10-2.1—INITIATING ACCIDENT CODES

9.2.1 Definition, Functional Requirements and Current Technology

This Task will cover updating, expansion, and formulation of new codes for analysis of accidents up to the point at which core disassembly becomes possible, or the point at which the geometric model of the original core configuration is no longer very useful. Current accident codes under development in Project Agreement 10 are FORE II and FREADM. It is expected that these will be substantially improved or supplemented, or both, by continuing effort under PA-10, but it is also clear that continuing experimental programs will produce data that will require additional code development after 1972.

The functional requirement of the desired code or codes is to provide a feasible means of estimating the results of all accidents of interest up to the point defined above. This will include, but not be limited to, the following:

- a. Identification of the spatial and temporal pattern of fuel failure in the core,
- b. Estimates of the probable modes of failure,
- c. Estimates of sodium voiding patterns and rates,
- d. Estimates of fuel slumping patterns and rates,
- e. Estimates of reactivity changes due to items c and d above,
- f. Effects of the response of instrumentation and control systems, and
- g. Inclusion of the interaction between the primary system and the core.

A review of the state-of-the-art is given under Task 10-2.1 of the LMFBR Program Plan.

9.2.2 New Information Needed

Information on which to base an analytical description of the fuel failure, sodium voiding, and fuel slumping phenomena is needed. This information is expected to be available from Tasks 10-2.3, 10-2.4, and 10-2.5, and related work carried out by other organizations. In addition, information is required to correlate sodium voiding patterns and fuel clamping methods with reactivity additions, and to allow incorporation, if necessary, of the space-time dependence of power distributions. This information will be furnished by Task 9-3.5 of the Physics Element. The new information will be incorporated into a code such as FORE-II by the creation and successive incorporation of computational modules. The available codes will be reviewed at the time this task begins to determine the advisability of existing models versus creation of new, complementary codes. The question of a master code system versus individual packages will be considered.

9.2.3 Task Objectives and Effect of Non-Achievement

The objective of this Task is to provide computational tools that will satisfy the needs of the licensing and safety analyses for the 1000 MWe LMFBR; the code must therefore be available approximately in fiscal 1977. A secondary objective is to provide improved capability for accident analyses in the guidance studies (Tasks 10-1.1, 10-4.1.1, and 10-4.2) that are carried out before that time.

Given the required information from other Tasks, the probability of accomplishing the objectives is high. To the degree that the objectives are not accomplished, the capability for realistic accident analysis will be reduced. The effects will be increased costs in the Guidance Studies Tasks, and for the LMFBR licensing, because of the greater costs of utilizing less efficient methods, or increased conservative and capital cost in the design, because of lack of supporting analysis. Probably both effects will occur to some extent.

9.3 TASK 10-2.2—HYPOTHETICAL ACCIDENT CODES

9.3.1 Definition, Functional Requirements, and Current Technology

Use of probabilistic techniques in safety analysis and efforts to show that elaborate and expensive containments are not necessary for acceptable safety will increase, rather than diminish, the need for analysis of extremely improbable accidents. The influence of these accidents on design and on cost can be reduced if they can be shown to have sufficiently low probabilities, but the quantitative definition of "sufficiently low" is a function of the consequences of the accident. These consequences must then be estimated, with an effort to reduce conservatism and obtain realistic estimates wherever feasible. This Task will

cover code development to allow improved accident analyses of situations where the core is partially or completely disassembled. As in Task 10-2.1, it is expected that improvements taking place up until 1972 will still leave room for significant further improvements, and that in some areas new data will become available during this Task.

The functional requirements of the desired code or codes is to allow reasonable estimates of the consequences of the severe, improbable accidents described above. Improved capability of treating the following phenomena is extremely desirable:

- The partition of the available energy in a core disassembly accident into shock waves, quasi-static pressure, and sensible heat,
- The effects of interaction and heat transfer between hot fuel and sodium on the potential available work and damage to the system,
- The effects of sodium vapor pressure in the core on the disassembly behavior and energy generated,
- The energy absorption capabilities of the primary system,
- Behavior of molten fuel masses, and
- Radiation doses as a function of release and of meteorological conditions.

It is expected that better understanding and estimates of activity release which are also needed, will be available from Task 10-2.7.

A discussion of current technology is given in the LMFBF Program Plan.

9.3.2 New Information Needed

Additional data on the rates of heat transfer between molten or vaporized fuel and sodium, and on sodium vapor pressures during boiling is needed. Some information may be expected from work in Task 10-2.3; and it is expected that additional data may become available from programs carried out at National Laboratories and in other countries. However, the difficulty of performing these experiments and of determining the accident conditions that define what specific data is actually needed will mean that large uncertainties will continue to exist.

A similar situation exists for information on energy partition, the equation of state of the fuel, and the behavior of molten fuel masses. Any data that may become available will be utilized. Continuing uncertainties in these areas will reflect themselves in more conservative designs.

An increasingly sophisticated knowledge of the energy absorption capabilities of systems and components is to be expected as test data accumulates, first from Demonstration Plant development, and then from Task 10-2.5 for the 1000 MWe plant. These experiments will also provide some information on energy partition.

9.3.3 Task Objectives and Effect of Non-Achievement

The objective of this Task is to provide the best computational capability for very severe accidents that is feasible by about 1977. The probability of achieving this goal is high. However, the worth of the codes resulting will be determined largely by the available experimental data and the ability to determine that the experimental information is

actually relevant to the conditions of interest. A realistic assessment of the problems involved leads to the conclusion that the improvement over current capabilities may be substantial, but that great uncertainties and gaps in knowledge will remain. This will be manifested in design conservatism and higher costs. The choice of whether the conservatism will appear as additional redundancy in active engineered safeguards or possibly as some type of additional containment feature can potentially be optimized to minimize costs. The uncertainties in this area of analysis will be particularly important to the Guidance Studies and can be expected to have substantial effects on the basic approach taken to achieve acceptable plant safety.

9.4 TASK 10-2.3—SODIUM BOILING

9.4.1 Definition, Functional Requirements, and Current Technology

This Task includes experimental and analytical studies of sodium boiling phenomena under conditions specifically applicable to the 1000 MWe LMFBF. It will provide information to Task 10-2.1 and 10-2.2, to allow improved analytical descriptions of boiling phenomena, and it will provide direct experimental verification of calculations under conditions close to the reference design conditions, which will support licensing of the plant.

The objective of this Task is to obtain information on sodium boiling, and not directly to develop any particular item of hardware or software. There are consequently no functional requirements involved.

A discussion of current technology may be found under Task 10-2.3 of the Program Plan. There are two different types of boiling which are of interest. In the first, the fuel pin is intact, and the transfer of heat to the sodium is controlled by the normal paths through the fuel-cladding gap, cladding, and cladding-

sodium resistance. The amount of superheat that will exist before boiling begins is a key variable in this case, since the superheat will have a strong influence on the energy available during the initial vaporization and expulsion of liquid sodium. In the second type of boiling, the fuel pin fails and there is direct contact between either molten or solid fuel, and sodium. In this case, the rate of heat transfer from fuel to sodium, and therefore the size and condition of the fuel particles, is important. It is currently felt that the first type of boiling may prove to be of less concern. It is thought that under in-core conditions, (impurities in the sodium, vibration of the core components, etc.) achievable superheats may be low, resulting in small pressure pulses during vaporization and a low potential for damaging adjacent parts of the core. In the second type of boiling, the possible energy transfer from fuel to sodium and the resultant potential for damaging is not known to any extent, and concern about rapid propagation of damage in the core exists. Possible pressure effects during reentry of sodium after initial expulsion from the core must also be taken into account for both types of boiling.

9.4.2 New Information Needed

It is anticipated that on-going work at ANL, ORNL, Atomics International, as well as European programs, will result in large advances in the understanding of the boiling phenomena. In addition, analyses and possibly experimental work will be required in support of FFTF and the Demonstration Plant. It is assumed, therefore, that analytical models which partially or completely cover the situations of interest may be in existence by fiscal 1972. However, extrapolation of data may be necessary to cover 1000 MWe conditions. Furthermore, the type of boiling due to direct fuel-sodium contact is expected to be dependent on the size and type of fuel particles, and therefore on the failure mode of the fuel pin. It will probably be necessary to perform proof tests utilizing fuel pin conditions (design,

spacing, etc.) as close as possible to the eventual LMFBR situation.

Although reevaluation of the tests will be necessary before this Task begins, it is currently reasonable to project a need for some testing of both types of boiling. For the first type, out-of-pile tests would be run with emphasis on duplicating a representative in-core environment wherever feasible, and possibly including some testing of boiling promoters such as ultrasonic noise generators, to minimize superheat. On the order of ten tests might be envisioned. For the second type, in-pile single-pin capsule tests would be performed to produce fuel failure and rapid mixing of fuel and sodium. Both power and flow transients would be of interest. Power transients could be run in TREAT, but it might be necessary to delay at least some of the power transient tests until the availability of a fast transient test facility, presumed to be available in 1975, to ensure that fuel failure modes and temperatures are prototypical (see Task Area 10-5). Flow transients could be run in GETR possibly using an adaptation of the forced convection capsule developed under PA-10. Five tests of each kind might be reasonable.

These tests would provide useful design information for the subsequent fuel failure tests to be run in Task 10-2.4. It should also be noted that the fuel failure tests under Task 10-2.4 will provide some additional data on sodium boiling with fuel failure.

9.4.3 Task Objectives and Effect of Non-Achievement

The objective of this Task is to provide any experimental sodium boiling data necessary to supplement other programs in providing a secure basis for analytical predictions applicable to the specific operating conditions of the LMFBR. The probability of success is

dependent on the general progress made toward understanding sodium boiling before this Task begins. If a good general understanding has been obtained, the probability of success for this Task will be very high. If no progress is made beyond the current technology, the chance of success will be low. It is thought that some progress can be expected prior to 1972, leading to at least a medium probability of success for this task. If increased urgency were given to this area in the near future, chances for success would improve significantly.

Failure to understand and create the conditions that minimize superheat, or failure to show that the in-core damage potential of fuel-sodium reactions is not large, will mean that certain routes to extreme accidents will remain, which would otherwise have been eliminated. The size of the energy releases in the hypothetical accidents will also be more uncertain. Plant costs for safety systems and redundancy would increase, and the chances of minimizing special containment provisions could be greatly diminished.

9.5 TASK 10-2.4—FUEL FAILURE

9.5.1 Definition, Functional Requirements, and Current Technology

The Task encompasses the testing of fuel to determine failure thresholds, to determine probable fuel movements within the cladding or outside the cladding that would result in reactivity effects during the transient, and to determine the behavior of failed or defective fuel, particularly with regard to propagation of damage during abnormal conditions. Included are the in-pile tests, the analysis of the results, the responsibility for making recommendations of design directions to improve safety, and the responsibility for making recommendations for fuel failure behavior assumptions to be used in final design and safety analysis for the 1000 MWe LMFBR. The in-pile testing will be primarily

carried out in flowing sodium, and should include a facility that can produce the appropriate initial steady-state conditions, and then superimpose a transient. Both high- and low-burnup fuel should be tested. Some pre-irradiated fuel should be available from previous test programs such as PA-10. Additional pre-irradiations to obtain pins that are more typical of then current LMFBR designs is probably necessary. If so, it will be performed during the first three years of this Task, in EBR-II or possibly FERMI.

Results from Task 10-2.3 in-pile testing will aid in the design and analysis of the tests under this Task, and data from those tests which produce failure will provide additional information on fuel-sodium interaction for use by Task 10-2.3. The identification of conditions which result in fuel failure will be input for the code development effort under Task 10-2.1.

Both oxide and carbide fuel is covered, but no details are given here on the carbide transient testing. Its program would presumably be of similar magnitude to the oxide, and would cover approximately the years 1975 to 1982.

Although the functional requirements of this Task itself are largely concerned with the gathering of information on the behavior of fuel elements whose design is determined by the steady-state operating requirements, there are some functional requirements of the fuel pin which are associated with transient capability and failure behavior. It is expected that this Task will yield information that will indicate how to modify fuel pin design in such a way as to satisfy the general functional requirement of high tolerance for abnormal conditions. Reconciliation of safety incentives and economic incentives to arrive at a fuel pin design is the responsibility of the Core Design Element, in close cooperation with the Guidance Studies of the Safety Element. The functional requirements associated with safety that were assumed in the 1000 MWe LMFBR safety analyses were:

- If less than 50% of the cross section in the fuel meat reaches the melting point during a short transient, failures of the cladding will not occur,
- Short-time cladding temperatures of about 1600°F can be tolerated without fuel failure if fuel temperature rises do not result in significant fuel melting.
- The capabilities in the items above should be substantially retained even in the presence of fuel element failures or defects, if the failure or defect is of a type expected to exist in the core during normal operation.

These functional requirements may be expected to change as more information becomes available to better define what capabilities can be reasonably achieved. In addition, it will be necessary to differentiate carefully between high- and low-burnup fuel, and fuel operating normally at high and low linear powers. At this time, it is believed that these functional requirements may imply two things for high-burnup fuel with thin cladding. First it must be vented, or at least maintain very low pressures due to released gas under both steady-state and transient conditions. Second, the cladding must have a capability for significant amounts of strain at high strain rates at the end of fuel life, or as a possible alternative, ample space must be available to the fuel to accommodate thermal expansion, volume change due to melting, and local non-equilibrium released fission gas concentrations.

Fuel failure experiments have been and are being carried out in the TREAT reactor. These are reviewed in the LMFBR Program Plan, Task 10-2.4. Some of these tests have run oxide pins in sodium, under PA-10. These tests have all been run in a thermal reactor, from zero-power rather than steady-state conditions, in stagnant coolant, and have been devoted almost exclusively to severe power transient effects on isolated single pins. A

very small number of transients have been run on pre-irradiated oxide fuel. In general, the information available is suggestive, but by no means sufficient to draw systematic conclusions with any confidence. Current programs are directed toward remedying some of these deficiencies, and it is anticipated that at the time this Task begins, tests will have been run in flowing sodium, with both power and flow transients, and a small amount of additional data will be available on pre-irradiated pin performance. However, it is likely that power transients will still have been run only from a zero-power condition, that no fast flux testing will have been performed, that data will still be inadequate to allow formulation of design relationships, particularly on the 1000 MWe pin design (vented fuel, 10-mil clad, possibly a "tailored" cladding alloy).

9.5.2 New Information Needed

Sufficient tests are needed to estimate the failure thresholds for LMFBR fuel as a function of burnup, fuel smeared density, void deployment, and cladding thickness, in both power and flow transients. In addition, the effect of minor fuel defects and failures or failure thresholds, and on damage propagation tendencies during transients, must be assessed.

This will require an extensive series of tests of fuel that is as close as possible to the eventual LMFBR design, with such perturbations as may be of interest to determine possible desirable design modifications (e.g., different smeared densities). Since this information will be used in support of licensing, it is highly desirable that tests be as prototypical as possible. A fast flux transient test facility that can duplicate initial conditions, and that can run both power and flow transients on single pins or on bundles up to 19 pins, is essential if the greatest benefit is to be realized from this Task. The 19-pin capability

would be used to study damage propagation effects after failure. One may distinguish four types of tests; single-pin power transients, single-pin flow transients, multi-pin power transients, and multi-pin flow transients. In each type of test burnup will be a parameter of interest. In addition, there would be expected to be two or three other variables. One of these would be the "soundness" of the fuel pin; that is, whether it is defected, and if so, how. Others might be cladding thickness, smeared density, and void deployment. Although the number of tests would be uncertain, and in fact would be adjusted as the program proceeds, it would appear that a minimum of 10 to 15 single-pin power transients and 10 to 15 single-pin flow transients could be anticipated. Assuming that bundle tests would be delayed until pin design was better fixed, variables such as density and cladding thickness might be fixed, and four or five power transients and four or five flow transients could be sufficient. These numbers assume that no unexpected difficulties appear and that ability to plan the tests (which is dependent to some extent on the ability to guess the results) is high.

A supply of pre-irradiated pins will be available from previous testing in EBR-II. However, these may not be adequate for this purpose, particularly as far as providing pre-irradiated test specimens with thin cladding, representative alloy composition, and desired density ranges is concerned. Thus it is anticipated that pre-irradiation in EBR-II or FERMI, specifically for this Task, will be necessary. The pre-irradiation must begin in 1973, or earlier, and could involve between a third and a half of the total number of pins to be transient tested.

9.5.3 Task Objectives and Effects of Non-Achievement

The first objective of this task is to support the licensing of the 1000 MWe LMFBR by defining, as precisely as is feasible, the following:

- Failure thresholds of LMFBR fuel pins under power and flow transients, including the effects of minor fuel failures or defects, or both, existing prior to the transient,
- Transient behavior of the fuel pin (e. g., movement of the fuel within the cladding) for transients not producing failure, in order to include the appropriate reactivity effects in calculations, and
- Behavior of the fuel pin during failure, in order to assess the probability of sodium voiding, and damage propagation.

A second objective is to provide information on the dependence of failure thresholds and transient behavior on fuel pin design, so that the best transient capability consistent with economic requirements can be achieved.

As in the case of most of the objectives of the Safety Element, achievement here is a matter of degree. If the failure thresholds are not well determined, the assumptions used for licensing must be correspondingly conservative. A reduction in the assumed transient capability of the fuel pin may reflect itself in two ways. A corresponding reduction in linear power may be made, in order to maintain the margin between operating conditions and failure, or a change in design of the plant may be made in order to reduce the possible accident magnitude to be consistent with the reduced transient capability. An example of the latter approach would be to reduce the worth of the control rods (and increase their number), in order to reduce the potential reactivity insertion. Unfortunately, this approach may not be possible or practical for all types of accidents, and some reduction of steady-state operating conditions would be probable.

A reduction in the tolerable molten cross section from 50% to 25%, for example, would require a 475°F reduction in peak steady-state fuel temperature to maintain approximately the

same transient margin in terms of tolerable temperature rise. To maintain the tolerable reactivity insertion unchanged would require a slightly smaller temperature reduction, because of the temperature dependence of the Doppler coefficient. The reduction in fuel temperature could be achieved by a reduction in specific power from 157 kW/kg of metal to about 118 kW/kg of metal or by a reduction of fuel pin diameter from ~0.25 inch to ~0.22 inch. The former alternative would cost over 0.1 mill/kW-h, while the latter might cost only 0.02 mill/kW-h. However, the effect of diameter on fabrication cost could be greater than was assumed in these studies, and reduction of pin diameter has many qualitatively undesirable aspects.

Non-achievement of the other objectives would similarly require more conservative assumptions to be made in safety calculations. For example, the movement of fuel within the cladding might have to be assumed to be as unfavorable as would be credible, for calculating reactivity feedbacks. The fuel pin design would perhaps have less transient capability in fact, because of lack of information on how best to design it. The economic penalties could appear in a variety of areas, and have not been assessed. In the situation where not enough is known of fuel pin design effects on transient capability, the effects could be unfavorable for safety as well as economics.

9.6 TASK 10-2.5—POST-BURST PHENOMENA

9.6.1 Definition, Functional Requirements, and Current Technology

This Task includes experimental work and mathematical analysis of two phenomena. The first is the problem of possible high sodium vapor pressures within a fuel assembly, and the effects on the surrounding core structure. The concern here is to prevent or minimize the propagation of damage to adjacent assemblies. The second area is the effects

of relatively large-scale energy releases associated with core disassembly on the surrounding tank and primary system structure. Although the name of this Task as given in the LMFBR Program Plan has been retained here, it is somewhat misleading since only the second area would normally be thought of as "post-burst." Tests will usually be carried out out-of-pile using chemical explosives. In addition, a limited number of scoping tests might be run in-pile to establish the work and energy partition characteristics of sodium-fuel interactions that are to be simulated by chemical explosions, if information from Tasks 10-2.3 and 10-2.4 is not adequate for this purpose.

The results of this Task will provide improved mathematical models for use in Tasks 10-2.1 and 10-2.2, will aid in the evaluation of very severe accidents that must be done as part of the probabilistic analyses of the Guidance and Special Studies, will be of importance in evaluating alternative designs for core structure and for primary system arrangement (Core Design and Plant Design Elements) and will provide experimental evidence which will support licensing efforts.

The functional requirements of the fuel assembly can and of the primary vessel, tank, shielding, and refueling cell are involved in this Task. No specific requirements for the accident-resistance capability of these components were established in the reference design. In general it is simply desirable that these components be able to absorb as much energy as possible without failing or exposing remaining parts of the core or of the plant to damage. As the development program progresses and as the probabilistic analyses identify the possible tradeoffs for acceptable safety, the design will evolve towards features which incorporate a practical degree of resistance to explosive forces while avoiding intolerable cost.

A discussion of the current technology is contained in the LMFBR Program Plan. Several problems exist. The first is partition of explosive energy into shock waves and quasi-static pressure effects and heat, which is dependent on the time scale of the explosion. Because the time scale of the nuclear disassembly accident is in most cases long compared to that of chemical explosives such as TNT, the extrapolation of the damage effects of TNT explosions to the nuclear situation is incorrect. This also makes it difficult to perform experiments that will be directly applicable because of the lack of a suitable chemical explosive. Another problem area is that the capability of the structure to absorb energy is not predictable with much accuracy even when the loading can be well specified. The complexity of the primary system compounds this problem. Present practice is essentially to apply the results of tests using TNT in simple vessels.

9.6.2 New Information Needed

The information gained from Tasks 10-2.3 and 10-2.4 on the actual pressure pulses observed during sodium boiling and sodium-fuel interaction will be used to help design the experiments on fuel assembly cans, and may be supplemented by additional in-pile tests. Efforts will be made to utilize chemical explosives with the desired characteristics, or possibly to duplicate more closely the real situation by finding a method of injecting hot fuel or a similar material into sodium. Either full-size or scale-model testing may be utilized, with the test assembly can being surrounded by one or two rows of assemblies containing dummy fuel pins. Various assembly designs will be tested, possibly ranging from thick walled free-standing designs to "no-wall" or birdcage designs. By combining the results of these with appropriate analysis, mathematical models will be formulated to allow prediction

of conditions under which sodium vapor pressures will cause damage in adjacent assemblies. These tests will be relatively inexpensive; between ten and twenty experiments would be contemplated.

The tests on the effects of explosive energy release on the primary system will be delayed until the end of fiscal 1975, so as to make the tests on scale models as prototypical as possible. Input from Task 2-1.8, the containment study of the Plant Design Element, will also be available. These tests will be more expensive and time consuming than the single assembly tests, and would be limited in number to between five and ten. Their effect on design may be limited; the function of providing experimental evidence to support assumptions made in the safety analysis of the actual plant is relatively more important. As in the single assembly tests, efforts will be made to obtain sources of energy release that have appropriate characteristics of duration and heat capacity, and the experiment will be performed in sodium. In all the tests in this Task, however, it may be necessary to accept non-prototypical energy releases and to interpret the results to the best extent possible for their significance to the situations of interest.

9.6.3 Task Objectives and Effects of Non-Achievement

The objectives of this Task are:

- To provide experimentally-verified mathematical models for damage due to local sodium vapor pressures in the core, and
- To provide experimental support for assumptions made in calculating the effects of very severe accidents involving core disassembly.

These objectives have a high probability of success, but the precision and confidence that can be attached to the analyses that use these

experiments as supporting data could vary greatly, depending on the success experienced in mocking up the actual expected conditions. It is expected that there will be significant shortcomings in experimental capabilities, and that this may be reflected both in additional conservatism in the analyses, and in significant costs in the LMFBR design.

9.7 TASKS 10-2.6 AND 10-2.7—SODIUM FIRES AND FISSION PRODUCT AND Pu RELEASE

Better mathematical models, and the experimental data to support them, are needed to calculate the behavior of plutonium and fission products in sodium, the pressures and temperatures possible in sodium fires, and the behavior of resulting aerosols. On-going programs at Argonne National Laboratory and Atomics International, among others, combined with information that may be available from demonstration plant development programs, is thought to be adequate to provide the required information when needed for the 1000 MWe LMFBR.

9.8 TASK 10-2.9—PHYSICAL AND MECHANICAL PROPERTIES FOR ACCIDENT ANALYSIS

Determination of certain physical properties of mixed oxide fuel, should be given high priority from the standpoint of improved accident analyses. Of particular concern are the present major uncertainties in specific heat at the melting point, conductivity of molten fuel, and thermal diffusivity between 4000°F and the melting point. Significant uncertainties also exist in vapor pressure above the melting point, total enthalpy to the liquidus, and conductivity of solid fuel. It is assumed that on-going programs at the national laboratories, and elsewhere, will supply this information in time to be of use to the analytical studies outlined previously.

9.9 TASK 10-3.2—CONTAINMENT DEVELOPMENT

9.9.1 Definition, Functional Requirements, and Current Technology

Although the 1000 MWe LMFBR design as described in the Task II report has not generally incorporated the types of special features that have come to be associated with "containment," it is not to be inferred that containments are not significant in the context of a more quantitative approach to safety. A logical view of containment is that it is the capability of the plant to restrict the spread of activity, and thus limit the consequences, subsequent to a damaging accident. All plants, however designed, have some containment capability, although they may not always include large, expensive features whose only purpose is to increase their containment capability. It is thought, for example, that the 1000 MWe reference design probably has containment capability to cope with a meltdown of at least seven fuel assemblies, although special single-purpose containment provisions were minimized or eliminated.

As a minimum, the plant containment capability should in some sense be "adequate" to deal with those accidents which are sufficiently difficult to prevent, and/or whose consequences are sufficiently uncertain, that their probabilities are unacceptably high. Stating this in another way, if accident consequences are potentially serious and if prevention with a very high level of reliability is expensive, additional containment features may be the most efficient way of achieving an acceptable level of risk for the plant. Containment may also continue to be advisable as a form of insurance against accidents which are not envisioned during quantitative analyses. It is apparent that the needs for containment capability will be strongly dependent on the results of the other development tasks in the Safety and Instrumentation Elements, and ultimately dependent on

most of the elements in the Development Program. It is not possible, therefore, to define what development will be needed in the area of containment. This Task is defined, however, to include the analytical and experimental work that will be necessary to show that the containment capability of the plant is adequate, or to indicate the type of modifications needed. The functional requirements of the containment, which will determine the specific work to be performed in this Task, will be defined as other development proceeds. The guidance studies of the Safety Element, and Tasks 2-1.4 and 2-1.8 of the Plant Design Element, are the focal points for defining these needs.

The current technology of containment is not extremely sophisticated. A discussion of containments that have been employed is given under Task 10-3.2 of the Program Plan. The uncertainties in such areas as blast effects, behavior of molten or fragmented core material, and aerosols, and the expense and difficulty of making meaningful tests on large systems, have contributed to containment designs that are probably grossly conservative. It would be highly desirable to be able to tailor the containment specifically to suit the real needs of the plant, recognizing that a complete continuum of alternatives exists, but the present state of knowledge effectively prevents this.

An obvious interface exists between this Task and Task 10-2.5. In general, data on the response of the overall system to explosion effects will be generated in Task 10-2.5. Any other types of tests will be carried out under Task 10-3.2.

9.9.2 New Information Needed

As indicated in the previous discussion, the level of containment that will be provided cannot be determined at this time because it is a direct function of the success of other tasks and particularly the other areas in the Safety

Element. However, it is clear that analyses and certain model or proof testing will be required for any level of containment selected. Any containment concept selected must consider the following areas:

- a. Confinement of the damage from the accident of interest until the reactor is shut down,
- b. Cooling of the affected parts of the core following reactor shutdown to limit propagation of damage to the unaffected parts of the system to an acceptable level,
- c. Determination of the quantities and characteristics of any airborne fission products, and
- d. Selection of a barrier(s) to limit the amount of radioactive release to the public to meet the safety objectives established for the reactor.

The relative importance of each of these areas will be a strong function of the level of containment finally selected. The design described in the Task III report provides a good example. The objectives of the instrumentation in the design are to detect the failure of a single fuel channel and to shut down the reactor rapidly enough to limit damage to the affected channel and its neighbors (item a above). Task 10-2.5 specifically addresses itself to this problem. Cooling of the seven channels potentially affected is not covered under another task and would be considered as part of this Task. The specific objective would be to demonstrate that a core containing seven failed elements would be adequately cooled to prevent propagation and maintain the containment barrier (the primary system in this case). For this design, item c is relatively unimportant since the primary system which is also the containment barrier (item d) remains intact and any leakage from the primary system would be handled by the plant radwaste system. This

example provides an illustration of the kind of work to be performed under this task. Specific problems and level of effort will, as mentioned previously, depend on the success of the other tasks in the Safety Element.

9.9.3 Task Objectives and Effects of Non-Achievement

The Task objective is to develop the containment features appropriate for the plant, and to provide the data necessary to show that these features are adequate. In particular, sufficient knowledge of actual containment capability as a function of plant design should be developed so that gross conservatism in containment design and resultant cost penalties can be avoided. It is anticipated that much progress can be made over the current technology, but it is clear that the difficulties of development in this area are very great, and the degree of success will be limited. Thus, assessment of containment capability will probably remain quite conservative in the reference LMFBR, though both the degree of conservatism and the accident to be contained may be reduced as compared to current practice. The prospective cost penalties of utilizing an overly conservative assessment of its containment capability are impossible to identify at this time.

9.10 TASK 10-5.3—TRANSIENT REACTOR FACILITIES

9.10.1 Definition, Functional Requirements, and Current Technology

A need exists for a facility in which the fuel testing under Task 10-2.4 can be carried out. This Task covers studies of the suitability of SEFOR to supply some or all of the testing capability required; assuming that it is concluded that SEFOR should be utilized in this way, the Task covers the actual SEFOR modifications.

Capability for both power and flow transients, with sufficient isolation of the experiment to allow destructive tests, is required. Initial conditions of steady-state operation, with the transients superimposed, are needed. In addition, a fast neutron spectrum is highly desirable. For single pin tests, a thermal spectrum distorts the temperature gradient within the pin, particularly in power transients, and may result in non-typical failure modes. For bundle tests, a thermal flux creates undesirable power gradients across the bundle as well as in the individual pins. Since SEFOR is a fast reactor designed for both steady-state and transient operation, it appears to be uniquely qualified to supply at least some of these testing capabilities.

Currently available facilities are all thermal. For rapid power transients, only TREAT, which must initiate tests from a zero-power condition is available. For slower power transients, facilities such as Trail Cable at GETR provide some capability. Some improvements in this situation are scheduled; TREAT is to be given some capability for very short-term simulation of steady-state operation through the use of square-wave transients. Additional information may be found in the program plan.

9.10.2 New Information Needed

The information to be gained from this Task is 1) an evaluation of the potential of SEFOR as a transient test facility, and 2) the estimated cost and detailed design of the required modifications. Finally, the actual modifications will be performed under this Task.

9.10.3 Task Objectives and Effects of Non-Achievement

The objective of this Task is to improve the available facilities for fuel transient testing by development of the potential capabilities of SEFOR. It is felt that there is a high probability of success. If this is not successfully carried out, the fuel transient testing for the 1000 MWe LMFBR would have to be performed in inferior facilities, since it appears that modification of any other facility, or building a new facility for this purpose, would probably require significantly more time (and money) than would the modification of SEFOR. As a result, the value of the tests would be significantly lessened, causing increased difficulties in the licensing of the 1000 MWe LMFBR due to difficulties in extrapolation of failure data.

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