1000-MWe LMFBR FOLLOW-ON STUDY
— Task IV Final Report —
Research and Development
Requirements
Volume I

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Approved by B. J. Goulding
Project Engineer

M. W. Croft
Project Manager

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Prepared for
ARGONNE NATIONAL LABORATORY
by
BABCOCK & WILCOX
Power Generation Division
P.O. Box 1260
Lynchburg, Virginia 24505

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PREFACE

Babcock & Wilcox (B&W) is participating in the AEC's 1000-MWe Fast Breeder Reactor Follow-On Study Program under Argonne National Laboratory (ANL) Contract No. 31-109-38-1998. As stated by ANL,

"The over-all objective of the current 1000-MWe LMFBR Follow-On Study Program is to provide the AEC with information needed to establish research and development programs which will lead to the achievement of economic and safe LMFBR power plants by a viable nuclear industry. This information is to be developed through studies of 1000-MWe LMFBR power plant reference conceptual designs favored by the nuclear industry for further development and eventual sale to the electric utilities in the period of 1975 to 1985."

To achieve this objective, B&W's work under the Follow-On Study was divided into four major tasks. Task I, which involved the development of four basic plant concepts and the selection of a preliminary reference design, was completed in 1967; the results are documented in BAW-1316. The work under Tasks II and III, which included parametric and trade-off studies conducted in parallel with the conceptual design of a 1000-MWe LMFBR reference design, has also been completed and the results are described in BAW-1328. Task IV involved the preparation of research and development programs which would, when executed, enable the detailed design, construction, and successful operation of the contractor's optimized reference design.

This report describes the work performed under Task IV. Section 1 outlines the overall schedule and the basic approach that has been established to develop the reference design to the point where it achieves general commercial acceptance. Sections 2 through 10 contain individual R&D programs categorized in accordance with the nine major elements of the national LMFBR program plans. Volume I contains sections 1 through 6, and sections 7 through 10 are in Volume II.

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1. INTRODUCTION

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1. INTRODUCTION

In recognizing the need for fast breeders and selecting the LMFBR to fulfill that need, the LMFBR has been given top priority in the AEC's power reactor development program. The overall objective of the program, in simple terms, is to develop a safe, reliable, and economical LMFBR power plant for the utility environment. Consistent with the foregoing priority and objective, the general LMFBR development activities, which already have clearly established the technical feasibility of the concept, have recently been accelerated by both government and industry and oriented specifically toward the stated objective.

To continue the favorable trend toward developing the LMFBR in a planned and disciplined program, the specific technology required for the LMFBR to compete successfully with existing power plants must be clearly defined. Since these requirements could be described better if they were based on a specific LMFBR design, the AEC implemented a series of plant design studies by awarding five major plant design contracts under the 1000-MWe LMFBR Follow-On Study to potential fast breeder reactor vendors in the years 1966 and 1967. Babcock & Wilcox is participating in the Follow-On Study under Argonne National Laboratory (ANL) Contract No. 31-109-38-1998, and the work performed under Task IV of the Follow-On Study is described in this document.

This report summarizes the research and development programs that should be completed before the B&W LMFBR reference power plant could be designed, constructed, and then operated in electric utility systems in the mid-1980s. In addition, these programs inherently identify the most important LMFBR technology that needs to be developed and the problems or areas of uncertainty that must be resolved to meet the mid-1980 goal.

These research and development programs were prepared under Task IV of the Follow-On Study, as noted above, but they reflect R&D requirements that were identified continually throughout the Follow-On Study in accordance with the methodology implied in the overall objective given in the Preface. Some of the broad-based needs were identified in Task I during the development and subsequent evaluation of four preliminary concepts, before the reference design was selected. These were defined better, and additional specific requirements were established as the reference concept was developed and optimized by conceptual design studies and trade-off and parametric evaluations in Tasks II and III. Thus, the programs in this report reflect realistic requirements based on the specific features and design choices used in B&W's 1000-MWe LMFBR reference plant. Naturally, since much of the technology required stems from needs that are common to all LMFBR systems, these programs inherently include many items that are applicable to any conventional LMFBR plant.

Since the R&D programs are based on the requirements established in preceding tasks, the reports that document the earlier work are pertinent to the information presented here. The five volumes of BAW-1316 describe the Task I work which led to B&W's reference concept, an integral pool design incorporating a vibratory packed, mixed-oxide pin cluster, vented-to-coolant core, and an under-the-plug refueling system. Volumes 4 and 5 of BAW-1328 describe the major trade-off and parametric studies which were instrumental in establishing the final features and characteristics of the reference design; Volumes 1 through 3 contain a detailed description of the final reference conceptual design.

In the following parts of this section, the objectives of the Follow-On Study and the corresponding work scope are summarized briefly to place the subsequent material in the proper perspective; the basic philosophy, criteria, and assumptions that were used in preparing R&D programs are outlined, and the schedule and funding required to develop B&W's reference 1000-MWe LMFBR up to the time that a commercial contract is secured early in 1983 are summarized.

Subsequent chapters present the research and development programs which were prepared in accordance with the overall scope and criteria outlined in this section. These programs contain many separate

items or R&D tasks, which have been categorized and grouped to correspond with the nine major elements of the LMFBR program plan:

<u>Section</u>	<u>R&D Program Element</u>
2	Plant Design
3	Components
4	Instrumentation and Control
5	Sodium Technology
6	Core Design
7	Fuels and Materials
8	Fuel Recycle
9	Physics
10	Safety

Each R&D task includes sufficient information to indicate the basic objective, current status of technology, scope of work, and cost and schedule. Cost benefit information and the probability of successful execution of the task are also indicated whenever possible.

1.1. Program Objectives and Scope

The overall objective of the Follow-On Study as given in the Preface is to define realistic R&D needs by conceptual design, optimization, and evaluation of a specific 1000-MWe LMFBR plant, and to produce corresponding R&D programs that will, when executed, supply the technology needed to design, build, and operate LMFBR power plants in the 1975-1985 period. The overall objective was clarified in primary and secondary objectives which further emphasized the R&D-related aspects of the work:

Primary Objectives

1. To identify new information needed to permit the successful design, construction, and operation of the industry-favored reference conceptual designs.
2. To establish a program of research and development to obtain this new information (including redirection of existing programs wherever necessary).
3. To estimate the probability of success in all areas of the research and development program described above; and to determine how the various elements of power cost might be affected if work in these areas were unsuccessful or only partly successful.

Figure 1-1. LMFBR Development Schedule

4. To compare the value of the research and development specified in each area with the cost of the work, in terms of savings in power cost.

Secondary Objectives

1. To encourage the development of original approaches to the design of large-scale LMFBR power plants.

2. To gain a better understanding of the safety problems involving the design and operation of such plants.

3. To supply the industry with current estimates of future LMFBR system functional requirements, which will dictate the design of components and the specification of materials for such plants.

4. To provide the AEC with information to aid in the continuing formulation of the National LMFBR Program Plan.

To achieve these objectives, B&W's work under the Follow-On Study was divided into four major tasks as follows:

Task I – The development of four basic plant concepts based on the established ground rules and the estimated state of technology in 1980, and the selection of a preliminary reference design that incorporates the best features of each concept.

Task II – Detailed engineering work on the preliminary reference design, and, in parallel, the performance of in-depth engineering trade-off studies.

Task III – Parametric studies to identify and evaluate basic problem areas and to provide a basis for the selection of an optimized reference design.

Task IV – Preparation of research and development programs required for the successful design and construction of the 1000-MWe LMFBR reference design.

1.2. Summary Schedule

The overall schedule for development of B&W's 1000-MWe LMFBR reference power plant is shown in Figure 1-1. This schedule was established by evaluating the initial time span for each major R&D item against the need for introducing the breeder in the early 1980s and then rescheduling in an iterative process to reduce the total elapsed time;

the schedule indicates that construction of the reference commercial unit could begin early in 1983. It also reflects the need for constructing and operating a predecessor or demonstration plant before constructing the larger unit in a competitive commercial environment. Construction of the predecessor plant is scheduled to begin by the first quarter of 1976 in order to achieve two years of operation before the contract date for the 1000-MWe unit.

To a large extent, this schedule was established by the features and design choices incorporated in the reference design. In particular, the choice of vipac fuel in a vented-to-coolant core design influences the overall schedule. First, this selection requires development of the vented fuel to the point where a few assemblies can be placed in an early core of a demonstration plant before construction of the reference commercial unit is begun. Of even more importance, R&D related to plant design and performance with high fission product inventories must be performed successfully in time to incorporate provisions for operation with vented fuel in the demonstration plant.

Although the example above is a key factor in setting the schedule for the reference plant, it is not the only determinant in the timing. In developing the R&D programs presented in subsequent chapters, it was obvious that most of the items in each of the nine major technology areas are directly related and form vital links in the development, construction, and operation of the reference plant. Even though these interdependencies were examined in preparing this document to the extent allowed by the work scope, additional evaluation is needed to establish a schedule reflecting a thorough assessment of these interrelationships. To be effective, this activity should be performed in parallel with more concentrated plant design studies in certain crucial areas as outlined in section 2. In particular, the interdependent effect of design choices on R&D requirements needs to be assessed further and related in a continuous process to the overall LMFBR development schedule.

The schedule shown in Figure 1-1 is predicated on the assumption that the predecessor or demonstration plant would be a scaled-down version of the reference plant and, except for the core, would incorporate all the major features of the reference design. Except for reduced performance, reduced component sizes, and core design, the predecessor

plant would be designed to the same standards and criteria as would the reference plant. For instance, the pot features, the number of primary and intermediate loops, the refueling scheme, and the plant arrangement are identical. The initial core was assumed to be a conservative, non-vented pellet design, but the reactor internals and the plant were assumed to be designed to accommodate a vented-to-sodium, vibratory compacted core. This would permit actual in-reactor operation of several vented-vipac fuel assemblies to demonstrate the performance of these types of fuel assemblies and to demonstrate other associated plant features such as seals and cleanup systems.

Acknowledging the predecessor plant-reference plant relationship, the schedule and the R&D programs were developed to satisfy both plants. The philosophy used here was that all items of R&D that are required for the licensing and construction of the predecessor plant had to be completed in the 1976-1979 period. Beyond 1979 and through 1986, R&D that was directly related to the reference plant could be performed. In the area of components, this might mean that components designed for the predecessor plant could be used in the reference plant without change; or it might mean that upgrading and further proof-testing might be required to increase the size or improve the performance. In the fuels area, it means that two fuel development programs must be carried on simultaneously: a vipac-vented program and a pellet-nonvented program.

Within the restraints imposed by the design decision reflected in the reference plant, the adoption of this philosophy results in an orderly R&D program. It makes maximum use of both national and private facilities and should enable the reference 1000-MWe LMFBR to be designed, constructed, and operated in a safe and economical manner in the 1980s.

1.3. Estimated Costs

The estimated cost for developing the reference 1000-MWe LMFBR through the start of construction in 1983 is \$552 million. The estimate for each of the nine elements of the LMFBR program plan is shown by year in Table 1-1. Each element is further divided by task in subsequent sections.

Table 1-1. Summary Cost Estimate for Developing B&W's Reference
1000-MWe LMFBR

Program element	Cost by year, \$ × 10 ⁶															Total
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
Plant design	2.27	2.53	2.73	2.08	2.13	2.50	2.50	1.25	1.25	1.25	1.50	1.75	2.13	2.25	2.25	30.37
Components	1.70	2.23	2.84	4.04	4.57	4.19	4.93	5.06	3.46	1.01	--	--	--	--	--	34.03
Instrumentation	2.35	2.65	2.28	2.15	1.33	0.90	0.65	0.40	0.10	--	--	--	--	--	--	12.81
Sodium technology	3.00	3.00	4.40	7.35	10.20	15.30	9.30	4.00	--	--	--	--	--	--	--	56.55
Core design	2.20	3.40	3.88	1.95	1.55	0.61	0.26	0.22	0.16	0.13	0.13	0.18	0.17	0.14	--	14.98
Fuel	5.80	6.50	8.00	8.00	8.00	7.00	7.00	7.00	7.00	6.00	6.00	6.00	5.00	3.00	2.50	92.80
Fuel recycle	0.76	0.90	1.25	1.75	2.00	2.00	2.50	2.50	2.50	3.00	3.00	2.00	2.00	1.50	1.50	29.16
Physics	5.80	8.40	10.30	11.50	12.00	11.50	10.50	8.60	5.10	3.60	3.40	3.40	3.40	4.30	4.30	106.10
Safety	3.60	6.12	8.17	10.36	20.91	30.13	30.18	15.97	9.26	7.04	6.68	7.24	7.79	6.89	4.69	175.03
Total	27.48	35.73	43.85	49.18	62.69	74.13	67.82	45.00	28.83	22.03	20.71	20.57	20.49	18.08	15.24	551.83

In developing this estimate, it was assumed that facilities such as those planned at LMEC and the FFTF would be available as needed, but credit for the technology generated in the process of building these facilities could not be applied directly in terms of program costs. Thus, the estimates are based on complete development of the technology and capability to build a specific LMFBR plant, independent of information generated by others.

Unless otherwise noted in the individual programs, the funding includes an allowance for operating costs of major facilities during the time they are needed to perform the work outlined herein. However, cost for the construction of facilities is not included, since those facilities identified in Volume I, Appendix 3, of the LMFBR Program plans, were assumed to be available. Likewise, the estimate does not include any post-contract construction cost for either the demonstration plant or commercial plants, but, as indicated in section 2, it covers the design and engineering costs for both units to the point of contract for construction with electric utilities.

2. PLANT DESIGN

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2. PLANT DESIGN

2.1. Introduction

2.1.1. General

The Plant Design section is concerned with the development of the plant concept and related technology. The primary objectives of the plant design R&D plan are as follows:

1. To describe the approach for developing the B&W LMFBR plant design technology.
2. To provide the basis for coordinating the activities in this plan with those of the AEC.
3. To coordinate the various elements of the B&W R&D program, which deals directly with the end objective—a total plant.

From a design standpoint, the LMFBR "plant" is considered to be the entire central station, but the more conventional areas outside of the nuclear steam supply system have not been considered as developmental areas; however, in developing design parameters and economic, safety, and operating relationships among major systems, due consideration must be given to conventional areas.

These conventional areas outside the nuclear supply system may require improved reliability and availability, since the plant availability factor is directly affected by downtime for refueling and maintenance. In currently scheduled water reactors the postulated refueling outage period is optimized to 2 to 3 weeks scheduled on an annual basis (actual previous refueling periods have greatly exceeded these outage figures). In contrast, the refueling system for the LMFBR, specifically the B&W concept, offers rapid refueling so that a normal annual refueling time is less than the annual maintenance time required for the turbine, for example.

The primary objectives of the Task IV activities in the 1000-MWe LMFBR Follow-On Study are (1) to develop the program of research and development necessary for the detailed design, construction, and successful operation of the reference 1000-MWe LMFBR plant, and (2) to provide quantitative cost-benefit information on the proposed R&D program.

The B&W LMFBR schedule for reaching these objectives is shown in Table 2-1. The plan does not include plant improvements that might occur after the 1983 date for starting construction. The plan does include an evaluation to establish that the concept will be commercially competitive in the utility environment in sizes greater than 1000 MWe, since plant capacities of 1500 to 2000 MWe in the year 1990 are not inconceivable, and increased size considerations could influence the direction of the R&D program. The content and timing of the R&D program are also closely related to the growth of LMFBR sales, which is affected by plutonium prices and comparative-operating cost standards determined by future light-water reactor (LWR) capacity. This program will require revisions to reflect these considerations as they become established facts. Finally, changes in the industrial development of new methods and equipment will also necessitate re-evaluation of the program.

The course of action for the development of plant design aspects of B&W's reference concept LMFBR is directly primarily to the understanding and delineation of plant design engineering considerations. This will be accomplished by studies based on the concept derived in the 1000-MWe LMFBR Follow-On contract, followed by studies associated with the various target plant designs. This section contains no research and development requirements, because it deals with only engineering study work. However, as an output from these studies, the R&D program for other sections will be further identified in depth and requirement. This will be a continuing function as the R&D program needs more specific guidance.

All study areas must proceed in parallel at similar, reasonable rates to obtain a full understanding of all interplays, which will enable the end objective to be a comprehensive engineering design based on both safety and economic considerations and to include a large number

of trade-off studies. The interplay between these studies is important. An acceptable design can only be achieved in a short time period by parallel work efforts with a full interchange of information. A lack of information interchange can only result in iterations or extension of the studies, which would be time-consuming and inefficient.

The program plan states the assumption that at least three demonstration plants will receive major financial support, with several others receiving less assistance. The further possible need for second-generation plants, requiring significant financial support, cannot be ruled out at this time. This R&D plan provides the capabilities necessary for construction of the B&W predecessor plant on the same schedule as the third demonstration plant in the National Program. The predecessor plant could serve as the third demonstrator, or it could be a second-generation plant. It would be conservatively based on 1975 technology with emphasis on safety, reliability, and economy, and it would be capable of being scaled to at least a 1000-MWe LMFBR commercial unit.

There is a general feeling in the nuclear community that safety evaluations and licensing criteria for commercial LMFBRs will be a direct outgrowth of their predecessor demonstrators. While this is largely proper, unrealistic restraints or requirements may be placed on licensing of the larger, more competitively-oriented plant unless it is considered independently at this time. Complete safety criteria and requirements cannot be established with the smaller demonstrator plants, because they are likely to be designed under highly conservative safety criteria for less-stringent operating conditions. Furthermore, the demonstrator plants will probably have inherent safety features, such as high-leakage cores, which may not be readily achievable or desirable in the larger commercial cores. The large breeder reactors required for commercial operation must represent a careful compromise between inherent safety features and engineered safeguards, on the one hand and plant economics on the other. The engineered safeguards must compensate for some of the inherent safety characteristics that could drive LMFBR costs beyond acceptable limits.

There are, of course, many ways to obtain an engineering understanding of plant design problems. At one end of the spectrum

Table 2-1. Preliminary Manpower Cost Estimate for R&D

Task	Cost, \$ × 10 ³															
	Calendar year															Total
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
Nuclear Steam Supply and Plant Design Study	750	750	750	--	--	--	--	--	--	--	--	--	--	--	--	2,250
Supplemental Studies																
1. Refueling system	250	250	--	--	--	--	--	--	--	--	--	--	--	--	--	500
2. Reactivity control system	150	150	--	--	--	--	--	--	--	--	--	--	--	--	--	300
3. Intermediate system	225	225	--	--	--	--	--	--	--	--	--	--	--	--	--	450
4. Dynamic analysis and control	400	400	400	--	--	--	--	--	--	--	--	--	--	--	--	1,200
5. Plant maintainability	--	--	125	125	--	--	--	--	--	--	--	--	--	--	--	250
6. Containment and shielding	--	--	200	200	--	--	--	--	--	--	--	--	--	--	--	400
7. Effects of vented fuel on plant design	--	--	250	250	--	--	--	--	--	--	--	--	--	--	--	500
Plant Safety Analysis	500	500	500	--	--	--	--	--	--	--	--	--	--	--	--	1,500
Predecessor Plant Design	--	250	500	1250	1625	2000	2000	--	--	--	--	--	--	--	--	7,625
Reference Plant Design	--	--	--	--	--	--	--	500	500	500	750	1250	1625	2000	2000	9,125
Subtotal	2275	2525	2725	1825	1625	2000	2000	500	500	500	750	1250	1625	2000	2000	24,100
Advanced Development Studies	--	--	--	250	500	500	500	750	750	750	750	500	500	250	250	6,250
Total	2275	2525	2725	2075	2125	2500	2500	1250	1250	1250	1500	1750	2125	2250	2250	30,350

is trial and error consideration, which is usually not prudent and involves an extremely high risk factor (possibly resulting in cancellation because of inadequate depth of engineering). At the other end of the spectrum are continual research, development, and design (these result in paper studies only, which are also not practicable). An optimum engineering program is largely a product of judgment, and continual evaluation will be required throughout the effort. In quite a few cases, however, engineering studies enable problem items or areas to be designed out of the concept, thus reducing the research and development program. The plant design section takes this feature into account in considering that thorough engineering is essential if the R&D program is to provide the performance, safety, and reliability required for utility operation of the LMFBR.

This section of the program provides the master controlling center for most other tasks in the report. Initially, results and data from other sections must be continually fed into the plant design studies to enable the maximum use and interplay of information, and requirements must be fed from this section into other specific R&D areas to redirect the range of environmental investigation, if necessary. Continuous assessment and evaluation will direct all phases of R&D work to meet the goal of developing the plant concept and related technology in the required time and in the most economic manner.

2.1.2. Summary Budget and Schedule

The summary budget for the Plant Design section is shown in Table 2-1, which identifies each of the major tasks and supplemental studies. The manpower cost (in thousands of dollars) is estimated for calendar years from 1969 to 1983.

The estimated total R&D cost for the B&W reference plant design is \$24,100,000. Advanced concept studies to improve the safety, economy, and performance of the design are estimated to cost an additional \$6,250,000, giving an overall R&D cost for plant design of \$30,350,000. As can be seen from the table, the total R&D budget results in a relatively even engineering schedule during the first seven years, to the end of 1975, at which time the sale of the B&W predecessor plant is postulated.

The schedule is arranged so that certain tasks run in parallel and others run in sequence; this arrangement is reflected in the budget cost estimate. Rearrangement of the sequence of the tasks would probably result in changing both the funding and scope requirements.

2.2. Overall R&D Requirements

2.2.1. General

This section of the program plan is divided into six major sections, as follows:

1. Nuclear Steam Supply and Plant Design Study (2.2.2).
2. Supplemental Studies (2.2.3).
3. Plant Safety Analysis (2.2.4).
4. Predecessor Plant Design (2.2.5).
5. Reference Plant Design (2.2.6).
6. Advanced Development Studies (2.2.7).

Section 2.2.2 treats design development studies of the interfacial problem areas of the various system requirements and the determination of these trade-off effects, with due regard to safety, economy, and R&D requirements. The R&D requirements are continually delineated to enable early initiation of R&D for long-term items. This approach provides a compact, comprehensive LMFBR program for achieving economical and widespread use of fast breeder reactors at an early date. This part of the program requires considerable interplay between sections 2.2.3 and 2.2.4.

Section 2.2.3 consists of studies concentrating on specific systems that require a greater degree of in-depth engineering design than is allocated in the section 2.2.2 work scope. Some of these studies begin with the first work in section 2.2.2, and others begin towards its completion.

Section 2.2.4 deals with comprehensive safety analysis studies to demonstrate that the nuclear plant can be operated safely and without undue risk to the public. A document comparable to a preliminary safety analysis report (PSAR) will be prepared in this task; it will be submitted to the AEC for licensing consideration.

Section 2.2.5 describes the design studies required to integrate and optimize all previous studies into a comprehensive predecessor plant design demonstrating all the major features of the reference

plant. The plant is considered to be large enough to be readily scaled to a 1000-MWe plant.

Section 2.2.6 considers the re-evaluation of all previous work and studies and the incorporation of all knowledge gained into the reference design concept, resulting in an engineered reference design for a commercial reactor, which is compatible with contract negotiations and sale requirements.

These five tasks are considered necessary to meet the primary objective of the 1000-MWe Follow-On Study; they do not, by definition, extend into other LMFBR advanced system concept design or use areas considered necessary for an overall program evaluation. The tasks are based on a construction schedule for the B&W reference plant starting in 1983.

Section 2.2.7 describes the advanced development studies which, although they are in no way directly connected with the B&W reference design, are considered necessary and prudent for the continued development of LMFBR systems; they are included to obtain the reduced power generation costs postulated for the LMFBR systems.

This section of the R&D plan departs from the LMFBR Program Plan* primarily in the areas of general format, budget requirements, and the deletion of some general studies. The studies that have been deleted are listed below, along with the reasons for their deletion.

1. Plant Design Temperature Study: The near-term design approach appears to be adequately covered by the parametric studies associated with the 1000-MWe Follow-On Study. The long-term approach will be covered by the advanced development studies (see Table 2-1).

2. Plant Siting Study: This study is not considered a requirement for the B&W reference design, since the reactor designer was not specified as being a part of the implementing organization.

3. Development and Evaluation of Plant Economic Factors: This is considered to be a continuing task. The LMFBR system

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* LMFBR Program Plan, Volume 2, "Plant Design," WASH-1102.

must compete with the LWR systems in the market in order to become established, and this task should be an integral part of the work in section 2.2.2.

4. Specifications and Standards: This task is covered by the component R&D requirements.

5. User's Assessment of LMFBR Plants: This task is covered by the B&W R&D program requirement that utilities participate during the plant design study program.

6. Investigation of More Compatible Secondary/Steam Energy Systems: This study is not considered a requirement for the B&W reference design; however, it is covered in general terms in Task 2.2.7, Advanced Development Studies.

7. Elimination of Secondary System: Refer to item 6.

8. On-Line Refueling Study: Refer to item 6.

9. Investigation of Alternate Uses of the LMFBR: Refer to item 6.

10. Demonstration Plant Program: This task is covered within the scope of Task 2.2.5, Predecessor Plant Design, since the predecessor plant is fundamentally a demonstration plant having a capacity of less than 1000 MWe.

11. Compilation, Assessment, and Dissemination of Information: No separate task has been specifically allocated to the compilation and dissemination of plant design information or to the identification of requirements, as these are considered to be continuing functions.

12. Identification of Functional Requirements: This task is also considered to be a continuing and interrelated function.

13. Program Assessment: This task is covered by the requirements of the predecessor and reference plant design tasks.

This section of the R&D requirements has interfaces with all other sections in both technical, economic, and safety considerations. The development activities associated with the interfaces among all

sections must be established relatively early in the program to enable an interchange of information with other sections, which will enable the evaluation of the detailed R&D program.

All tasks are planned with a requirement for continuous evaluation and assessment of all LMFBR R&D needs. Cost benefits can be continually assessed, so that the R&D program can be redirected as necessary. In addition to the R&D program requirements, phase completion reports will be issued where necessary to guide and adjust the R&D program.

In most of the plant design studies, the services of utilities and the architect-engineer will be required. To avoid repetition, these services are only identified in specific work areas involving their heavy commitment.

Some of the areas that require an early assessment, especially in a "go" or "no-go" capacity, which could have an effect on the plant design, are listed below:

1. Nuclear Steam Supply System

- a. Engineering and technological considerations of the pot primary system; the effects of all system interfaces and trade-off considerations.
- b. Engineering and technological consideration of the intermediate system; the interrelationship and effects of layout and operation.
- c. Fuel handling.
- d. Reactor control.
- e. Shielding and containment.
- f. Normal and abnormal operating conditions.
- g. Maintenance and inspection/test requirements.

2. Components

- a. The effect of the steam generator.
- b. The effect of the intermediate heat exchanger. (The IHX and the steam generator requirements must be compatible.)
- c. The effects of other component requirements on plant design, either directly or indirectly.

by incorporation into other systems, e.g., refueling, control, etc.

3. Instrumentation and Control

- a. Detector and flow-measuring device requirements.
- b. Hardware and software computer requirements and their effects on plant design, basically in a "no-go" capacity.

4. Sodium Technology

- a. Mass transfer characteristics in the primary circuit.
- b. Cover gas cleanup requirements.

5. Core Design

- a. Vented plug concept requirements.
- b. Method of core support.

6. Fuels and Materials

- a. Fuel swelling effects.
- b. Cladding compatibility.

7. Fuel Recycle

- a. Method of receiving fuel.
- b. Method of dispatching irradiated fuel.

8. Physics

- a. Finalization of fissile loading.
- b. Finalization of core geometry.

9. Safety

- a. Coolant superheating and the resulting core voiding dynamics.
- b. Magnitude of energy generation for design-basis accident; the partition of energy and the mechanism involved.
- c. Analysis of credible accidents.
- d. Environmental and site analysis.
- e. Effects of safety criteria and requirements on plant design.

2.2.2. Nuclear Steam Supply System and Plant Design

2.2.2.1. Objective

The primary objective of this task is to determine the full extent to which the integral pot-type LMFBR concept can be developed for commercial use. The factors of design technology, safety, economics, materials, fabrication, erection, maintenance, and operation will be evaluated as they apply to the LMFBR plant. Although the evaluation and system profile design* will use the 1000-MWe LMFBR as a reference, the impact of each factor will be analyzed on extrapolated versions of the LMFBR plant size. The results derived from this work program will identify the potential of and the problems associated with the pool concept. The goal is to assess the general LMFBR pool concept plant from a practical, industry-oriented viewpoint through an objective engineering analysis, rather than to advance the design of a particular concept. This study has four specific objectives:

1. To select a pool primary system that will provide a firm basis for system and component design, taking into account all interfacial aspects of systems and components.
2. To evaluate the technical, safety, and economic factors that affect the pool concept LMFBR plant during design, construction, and operation.
3. To delineate, with engineering documentation for back-up, the safety and technical limitations and benefits of the pool concept.
4. To identify the required research, development, and component testing programs, including priorities and cost benefit calculations, for the pool concept LMFBR plant.

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* A system profile design is defined as the engineering design required (a) to establish the advantages, disadvantages, and limitations of a system concept; (b) to define the best technical approach aimed at optimizing the system under consideration, along with analyses of alternate technical approaches; and (c) to determine R&D requirements and identify those specific technical, economic, and safety factors that require further engineering resolution. A system profile design will require in-depth engineering of a specific concept as necessary to delineate the benefits, restrictions, and significant issues associated with the system being evaluated.

2.2.2.2. Summary of Current Technology

Several examples of pool systems exist, and industrial interest has been demonstrated by the number of studies that have utilized this concept. Three examples of special interest are EBR-II, PFR, and Phenix. The feasibility of the pool system has been demonstrated by EBR-II, which has been designed, built, and operated by ANL. Since going critical in 1963 it has operated quite successfully; the recent recurrent shutdowns for investigation appear to be primarily related to core design. The 250-MWe demonstration reactor PFR is under construction and is essentially on schedule; commissioning is scheduled for early 1970. With the satisfactory operation of the PFR, the British would appear to be committed to the pot concept. The 250-MWe French Phenix reactor has characteristics similar to those of the PFR; construction of the Phenix is scheduled to begin early in 1969. From the information available, the testing of components for the Phenix is well advanced. Recalling that the previous French sodium reactor, Rapsodie, was a spread-out system, it appears that the French are also committed to the pot concept.

The EBR-II reactor has demonstrated certain design and fabrication considerations, including design, fabrication, and erection of the top cover structure, together with on-site fabrication and testing to established standards; thermal expansion and alignment of the cover structure, tank and related internal components; and structural stability and integrity under normal loading conditions.

2.2.2.3. Scope of Work

Commencing with the broad LMFBR plant concept developed in the 1000-MWe LMFBR Follow-On Study, the pool concept will be subjected to a system engineering analysis aimed at exploiting advantages and defining limitations. The assistance of an architect-engineer will be obtained, as required, in various portions of this task. The end results of the program will be a system profile design based on the best technical approaches. In addition, the R&D program required to further advance the general pool concept LMFBR plant will be defined. At the conclusion of this program, a plant design will be established in sufficient detail to determine the full extent to which the pool concept

LMFBR plant can be developed for use in the near-term commercial environment.

Core Design — Review existing fuel assembly designs as they relate to near-term application in commercial-size plants that can immediately follow the demonstration plants, and define a reference design through more refined analysis and evaluation. Establish a reference core design for the "first round" commercial plants, and define the associated safety criteria and operating conditions.

Review the current core support, coolant plenum, and holddown features with regard to performance and safety considerations, and synthesize the core structures concurrently with establishment of the reference core.

Analyze fuel management schemes to establish relationships between operating limits, fuel cycle cost, and safety characteristics. Establish a reference fuel management method and generate detailed operating, safety, and economic performance data.

Design and Construction Technology — Apply the fabrication and maintenance requirements along with materials and cost data to determine the system arrangement and to establish functional requirements and parameters. The results from this task will directly benefit the system engineering and analysis, the nuclear steam design, and the safety analysis of the reference concept.

Review existing methods and computer programs for thermal and stress analysis to determine their applicability to the LMFBR reactor vessel system design. The methods necessary for the design, analysis, and generation of data for the plant safety analysis study will be developed.

Collect the existing data on industrial materials and integrate with the data available from the LMFBR National Program. The information will be utilized in establishing system material requirements and operating limits.

Analyze the construction methods and schedule for component and plant fabrication and erection, including an evaluation of on-site versus shop fabrication of the major components. Study

the requirements for quality control methods and procedures needed to manufacture a quality product.

Delineate the maintenance problems associated with the reactor vessel and the other major components, and compare them with available maintenance methods. Review the surveillance and inspection requirements, identify those required for the reactor vessel and other major components and systems, and relate these to the available surveillance and inspection techniques.

Systems Engineering and Analysis — Review the 1000-MWe reference design concepts developed in the follow-on studies and the results from the CURVE study at the Liquid Metal Engineering Center (LMEC); then develop a catalog of problem areas requiring specialized and/or detailed attention as part of the plant design.

Establish the site for this study, in conjunction with utility participation. Establish site design and engineering criteria and general ground rules to be used for the evaluation. The establishment of a utility-selected site should produce better understanding of general problems and a more meaningful appraisal for the PSAR.

Review the heat generation and heat transport system interfaces; study the effects of relative thermal expansion, differential pressures, and changes in pressure, temperature, and flow. Prepare functional design requirements compatible with the safety requirements in the "General Design Criteria for a Liquid Metal Cooled Fast Breeder Reactor."

Review the existing codes, standards, and specifications to determine their applicability, and recommend modifications where essential to the design of the reactor vessel and heat transport systems.

Develop preliminary cost trends and cost estimating information and procedures for major components, which are to be used in the system design task and in the final cost estimation for the system profile design. Devise methods for extrapolating the LMFBR plant size, and, where possible, determine break points in material, fabrication, erection, maintenance, and cost curves. The economics of scale will play an important part in the commercial introduction and acceptance of LMFBR plants in the 1980's.

Nuclear Steam Supply System (NSSS) Design —

Perform an overall system engineering analysis on the reactor vessel and other portions of the NSSS to ensure that the design meets the requirements established in the safety analysis study. Give special consideration to the thermodynamic and hydraulic effects on components located within the cavity formed by the reactor cover structure and the biological shielding.

Evaluate the effect on the core and heat transport systems of displacements caused by the physical weight of the components, their thermal distortion, pressure differentials, mechanical and flow-induced vibrations, seismic ground motions, and sodium bubble growth and collapse motions.

Review the design to define the best technical approach consistent with the goal of optimizing the NSSS for minimum power cost. Some analysis of alternate methods and approaches will be required to achieve this goal.

Complete the analytical design studies of the major components of the NSSS, including reactor vessel system components, the IHX, and the steam generators. These studies will include static and dynamic loading requirements and support methods which must account for thermal expansion and other displacements. Various methods of attaching the reactor vessel and other major components to the cover structure will be reviewed, together with the type and location of the insulation to be used on the cover structure.

Incorporate the results of the overall plant and systems design into the system design description. Include the design data that provide the basis for specifications for the reactor vessel, IHX, steam generators, and associated interfacial equipment. Tabulate the overall system cost data and analyze to obtain an overall capital investment and operating cost.

R&D Program — Define the R&D program required to ensure continuity in the development and advancement of the pool concept LMFBR plant and to provide component and system data for inclusion in the system profile design final report. Develop the cost-benefit ratios for the various elements in the R&D program, and incorporate these ratios in the R&D program document.

2.2.2.4. Schedule and Budget Estimate

This task is scheduled as a three-year program to begin in 1969 and end late in 1971. The following budget estimate covers only engineering services associated with plant design studies; the facilities and equipment R&D costs are covered in other sections of this report.

Year	1969	1970	1971
Engineering cost, \$ $\times 10^3$	750	750	750

The total budget estimate for this task is \$2,250,000. Figure 2-1 illustrates the program schedule in bar-chart form.

Figure 2-1. Program Schedule for Nuclear Steam Supply and Plant Design Study

2.2.3. Supplemental Studies

2.2.3.1. General

Areas with interfaces of particular importance to the B&W reference design are covered in this section, in more detail than is possible in the broader "Nuclear Steam Supply and Plant Design Study." Although other sections of the R&D plan cover the specific system-oriented development of some of these areas, the Plant Design

section must provide the overall evaluations and trade-off studies that guide the others toward the goal of an optimum LMFBR plant.

These studies are divided into two groups: the first group starts with the initiation of the main study and concentrates on systems directly associated with the primary system; the second group starts when the main study engineering work is basically complete and concentrates on the systems that need a maximum feedout of data to the studies. It is realized, of course, that in establishing this division, some of the interfacial requirements between the supplementary studies are lost (such as the relation between refueling and maintainability requirements). However, it was considered that refueling features have a predominant effect on plant layout, and hence, they should be established early, and maintainability requirements can be better established later.

A secondary advantage of this breakdown is a relatively even annual work level. The supplemental studies are scheduled as a four-year program to begin in 1969 and extend to the end of 1972. The budget estimate shown here includes only engineering services, since the facilities and equipment R&D costs are covered in other sections of the R&D report.

Year	1969	1970	1971	1972
Engineering cost, \$ $\times 10^3$	1025	1025	975	575

The total budget estimate for this task is \$3,600,000. The schedule is shown in Figures 2-2 through 2-8.

2.2.3.2. Refueling System Study

Objective — The general objective is to guide the development of an optimum refueling system in the components section by providing (1) a thorough engineering analysis, (2) an evaluation of how the plant design is affected by different approaches and facets of fuel handling, and (3) the relation of these effects to the safety, operation, performance, reliability, and plant economy.

The refueling system study has six specific objectives:

1. To review the various fuel handling mechanisms proposed for the LMFBR reference plant and to evaluate their application to the B&W primary pool concept.
2. To establish the design requirements.
3. To establish the relation of the fuel handling system to other plant systems.
4. To assess the influence of the fuel handling system on plant safety requirements.
5. To establish the limitations and the estimated cost of the fuel handling system.
6. To evaluate the R&D requirements.

Summary of Current Technology — The development of a satisfactory fuel handling scheme is essential to a successful LMFBR, since it provides the major benefit of a shorter refueling time (hence, higher plant availability factors).

The handling of fuel in sodium-cooled reactors has been successful in EBR-II, Fermi, Dounreay, BR5, Rapsodie, Hallam, and SRE. A number of fuel handling concepts have been used in these plants, but, in general, any mechanism must consider the following:

1. Fuel assemblies must be handled remotely without visual observation.
2. Precise initial alignment of equipment is required, as well as repeatability for many fuel handling cycles.
3. Precision mechanisms are required to operate in sodium, sodium vapor, or inert gas. Since they move through the gas space into the sodium and back, some parts are also exposed to wetting and drying cycles.

The B&W concept uses gas as subassembly coolant during fuel transfer. Although little American experience is available in this area, over half of the world's nuclear power generated to date has been produced in gas-cooled thermal reactors that use gas-cooled refueling machinery.

To date, most of the LMFBR fuel handling systems have emphasized performance and safety without particular concern for economy. Fermi is an exception, since economics was considered in its fuel handling system.

Scope of Work — The plant design aspects of refueling include all of the handling of fuel assemblies from the time they are received at the plant site through their delivery in shipping casks for transportation to the processing facility.

1. Current Fuel Handling Technology: Review the under-the-plug and through-the-plug refueling methods and catalog and compare various features. This review will include plants in operation, under construction, in design, and in study stages, both in this country and abroad. Evaluate both liquid and gas methods of cooling irradiated fuel elements.

2. Refueling Considerations: Identify the broad aspects of fuel handling that affect the B&W concept. This identification will cover the subassembly movement from fabrication to fuel processing.

Consider the effect of reactor refueling mechanisms on both on-site and off-site fuel reprocessing, and evaluate the effects and requirements for quick fuel turnaround.

3. Fuel Handling Influences on Plant Design: Identify the major refueling problems and evaluate their effects on the plant design while taking into consideration safety, operation, reliability, performance, and economics.

4. Near- and Long-Term Fuel Handling Designs: Establish refueling systems for near-term plants; i. e., the B&W demonstrator and commercial plants. Specify problem areas and recommend in detail the R&D considered necessary. Consider long range advanced refueling systems, assess their potentialities, and recommend R&D program activities to the degree practicable.

5. Preliminary Design of the Under-the-Plug Fuel Handling System: Establish the system functional requirements. Define relations with other plant items and delineate interfacial requirements. Assess safety aspects and define system limitations. Evaluate system maintenance and repair requirements as well as the facilities required for them. Establish the effect of fuel handling on plant availability. Estimate the cost of the refueling system and identify the major cost factors.

6. Preliminary System Design Description:

Incorporate the results of the overall system design into a system design description, including the design data.

7. R&D Program: Define the R&D require-

ments to ensure continuity in the development and advancement of fuel handling systems. Estimate R&D costs required for the program. Develop cost benefit information for the R&D program. The R&D will be covered in the "Core Components Handling Equipment" task of the Components section.

Schedule and Budget Estimate — This task is scheduled as a two-year program to begin in 1969 and be completed by the end of 1970. The following budget estimate includes only the engineering services associated with plant design studies; the facilities and equipment R&D costs are covered in other sections of the report.

Year	1969	1970
Engineering cost, \$ $\times 10^3$	250	250

The total budget estimate for this task is \$500,000. Figure 2-2 shows the schedule for the program in bar-chart form.

Figure 2-2. Program Schedule for Refueling System Study

2.2.3.3. Reactivity Control Study

Objective — The primary objectives of this study is to guide development of the reactivity control system in the Components section by providing a thorough engineering analysis, by determining how the plant design is affected by the different approaches and facets of reactivity control systems, and by relating these effects to plant safety, operation, performance, reliability, and overall economics. The study has five specific objectives, as follows:

1. To review the overall design of the various reactivity control systems proposed for LMFBR target plants and evaluate their application to this concept.
2. To establish the design requirements for the concept.
3. To establish the relationship of the reactivity control system to other portions of the plant.
4. To evaluate the ability of the reactivity control system to fully meet the plant safety requirements.
5. To estimate the cost of the reactivity control system.
6. To delineate the R&D requirements.

Summary of Current Technology — The development of a satisfactory reactivity control system is essential to a successful LMFBR. The system must "fail safe" and must be designed to effect reactor shutdown under normal as well as abnormal conditions.

The current status of reactivity control systems for large, sodium-cooled fast reactors can be summarized by saying that all known cases propose the use of conventional control rod schemes incorporating solid absorber rods actuated by a mechanism in the blanket gas environment. Thus, the control element operates in a liquid medium, as the LWR elements do, and the actuator operates in a gas medium similar to that of the actuator for gas-cooled reactors.

Due to the high radioactivity of the cover gas (due to vented fuel or postulated fuel leakage), removal of the actuators from the plant requires very special attention, which may not be readily accommodated by internal seals. The actuators may have to be removed

in a manner similar to spent fuel; that is, employing the shielded fuel transfer machine as is done in gas-cooled plants.

Scope of Work — The plant design aspects of the reactivity control study will be concerned with the engineering evaluation of the system and the method by which the components are removed and replaced, as well as the plant facility requirements for all on-site operations.

1. Reactivity Control System Current Technology: Review design information for reactivity control systems, and catalog and compare the various methods. This review will include plants in operation, under construction, in design, or under study, both in this country and abroad.

2. Criteria Investigation: Establish the need for a backup reactivity control system. If one is required, establish its operating criteria and functional requirements.

3. Evaluation of Design and System Engineering: Using the information cataloged above, evaluate the various aspects of system design. Establish the system design criteria. Perform studies to enable the identification and comparison of system aspects. Evaluate the differential thermal expansion effects of the system and devise techniques for compensating for them. Evaluate maintenance requirements and removal, handling, and maintenance equipment. Identify interfaces with other components, systems, and plant facilities. Establish the effects of partial load and transient conditions. Determine the system's operational reliability under casualty conditions. Prepare system layout drawings showing all relevant details of the system. Incorporate the results of this evaluation into a system design description, and estimate the cost of the system.

4. R&D Program: Define the R&D necessary to ensure continuity in the development of the reactivity control system. Estimate the cost of the R&D program, and develop cost-benefit information for inclusion in the R&D program. The R&D will be performed in the Reactivity Control System task of the Components section.

Schedule and Budget Estimate — This task is scheduled as a two-year program to begin in 1969 and be completed by the end of 1970. The following budget estimate covers only engineering services for plant design studies; the facilities and equipment R&D costs are covered in other sections of this report.

Year	1969	1970
Engineering cost, \$ $\times 10^3$	150	150

The total budget estimate for this task is \$300,000. Figure 2-3 shows the schedule for this program in bar-chart form.

Figure 2-3. Program Schedule for Reactivity Control System Study

Task	FY	1969	1970	1971
	CY	1969	1970	1971
1. Reactivity Control System Current Technology
2. Criteria Investigation
3. Evaluation of Design and System Engineering
4. R&D Program

2.2.3.4. Intermediate System Study

Objective — The main objective of this study is to evaluate the adequacy of the present design criteria for the intermediate heat transport system. The task has five specific objectives, as follows:

1. To evaluate the effects of changing temperature differentials and gradients associated with components in the intermediate system.
2. To identify the effects of these changes on components outside the system, with specific

reference to the core and the energy conversion system.

3. To identify the factors that influence the safety, operation, reliability, and economics of the plant design.
4. To establish system design parameters.
5. To identify R&D requirements.

Summary of Current Technology — The intermediate coolant system is an important transfer system between the reactor and the turbine-generator; as such, it must function reliably and economically. However, to date there appears to be a lack of studies that completely investigate the design considerations of this system. The full effect of an economic balance between such components as the IHX and the steam generator has not been achieved.

The largest intermediate system in use today is in the Fermi plant, where each loop is rated at only 156 MWt at 820 F; the reference design requirement is 817 MWt at a temperature of 1000 F. The current 1000-MWe LMFBR Follow-On Study provides a base for this task, which will enlarge and expand the Follow-On evaluation.

Scope of Work — For this study the intermediate coolant system is considered to include the IHXs, the steam generators, the reheater, the intermediate sodium pumps, the storage tanks, valves and piping, and the sodium purification equipment, gas system, and heating system.

1. Intermediate System Current Technology: Review design information and available data for intermediate sodium systems, including that on current and decommissioned plants in this country and abroad. The various design approaches will be assessed, and their characteristics will be cataloged.

2. Evaluation of Design and System Engineering: Using the information obtained above, evaluate the various aspects of the design for the intermediate system. Evaluate the functional requirements and design criteria. Evaluate techniques for compensating for the differential expansion of the system. Establish filling, draining, and venting requirements for the system. Determine the heating

requirements, method, and related control. Establish containment penetration and system isolation design features. Identify interfacial relationships with other components, systems, and plant facilities. Evaluate maintenance requirements. Establish the effects of partial load and transient conditions. Establish operational reliability under casualty conditions. Prepare a system layout drawing showing all relevant details of the system. Estimate the cost of the intermediate system and incorporate the results of this evaluation in a system design description.

3. R&D Program: Define the R&D needed for development of the intermediate system. Estimate the cost of the R&D program, and develop cost-benefit information for inclusion in the R&D program.

Schedule and Budget Estimate — This task is scheduled as a two-year program to begin in 1969 and be completed by the end of 1970. The following budget estimate covers only engineering services for the plant design studies. The facilities and equipment R&D costs are covered in other sections of this report.

Year	1969	1970
Engineering cost, \$ $\times 10^3$	225	225

The total budget estimate for this task is \$450,000. The schedule is shown in bar-chart form in Figure 2-4.

Figure 2-4. Program Schedule for Intermediate System Study

Task	FY	1969	1970	1971
	CY	1969	1970	1971
1. Intermediate System Current Technology				
2. Evaluation of Design and System Engineering				
3. R&D Program				

2.2.3.5. Dynamic Analysis and Control Study

Objectives — The broad objectives of this work program are to obtain a basic understanding of LMFBR behavior and to relate this behavior to instrumentation and control needs, system design, and operational requirements, such as permissible load changes. To accomplish these objectives a three-phase program has been defined.

1. Determine the dynamic response of an LMFBR plant to the transients typically imposed on utility plants.
2. Establish the functional specifications for the instrumentation and control systems, apply these systems to a plant simulation model, and demonstrate plant controllability.
3. Define the needed R&D for the instrumentation and control system.

The results of this work will provide a variety of useful information to engineers engaged in other areas of the LMFBR development. The output will include the following:

1. Information that may be used by the system designer for modification and/or improvement of plant response and operating characteristics.
2. Where possible, identification of transient behavior to the plant safety engineer.
3. Information to be used by the control designer for advanced control concept investigations.

Summary of Current Technology — The ability of the plant to respond to controlled power demands is of major importance in the successful operation of any nuclear power generation plant. The designer must know the plant performance during and following load-changing transients. The various operating parameters must be studied, so that the reactor, reactor coolant, intermediate coolant, and feed-water-steam systems are operated within their design limitations. Reactor plant safety must be maintained during all operations. The designer must also understand the operations of the entire plant during and following load transients.

In the current LMFBR programs, little work has been done that encompasses all the various aspects of this subject. However, an analytical model was developed at B&W during the Follow-

On Study, and the three major individual heat transport systems were analyzed first in an open-loop and then in a final closed loop simulation of the heat generation and intermediate heat transport system. The analysis proved the feasibility of the control arrangement used in these systems. This model used the balanced-system concept (single loops).

Scope of Work — In the development of a dynamic model for evaluating a system, it is essential to establish the bounds of the model. The model of the LMFBR plant to be developed for this work will consider operation over the normal power range (20 to 100% load) under normal operating conditions. Information on abnormal process behavior will be obtained by simulating a process abnormality within the bounds of the model and observing the direction and rates of change of the various process variables. Such transients will terminate when the model has run to its limit. This will allow the determination of the safety initiating conditions that result from abnormal process behavior. Such information is useful as input for detailed models designed for safety studies. The actual safety analysis considers the nuclear and thermal spatial dependencies of a nuclear reactor and must be carried out using more specialized codes, which are not required for the present study.

1. Component Mathematical Modeling: Apply the dynamic models developed in the instrumentation and control section to the specific plant components and systems. Analyze the performance characteristics of these component models over a range of normal and abnormal operating conditions.

2. System Modeling and Analysis: Perform a dynamic analysis of the plant, first using a single-loop plant analysis to determine performance under fully balanced load, then using a multi-loop model to simulate unbalanced plant conditions. The dynamic analysis will fit the component models developed above into a complete plant model, using techniques from the instrumentation and control section. Provide the system designer with the results of the analysis for possible modifications and/or improvements in design. Analyze abnormal operating conditions, failure modes, and related effects, and identify the conditions that affect the LMFBR plant safety analysis work.

3. Control Concepts and Instrument Specifications: Review suitable 1000-MWe LMFBR plant control concepts, and determine the control philosophy to be used with the system profile plant design. This review will emphasize the applicability, to this plant, of general techniques investigated in the instrumentation and control section. Determine the instrumentation and control dynamic requirements and specifications. Assess the feasibility and the incentives for using a digital computer to improve the LMFBR plant response and control. Determine the effects of the selected control concept and philosophy on the performance characteristics of the system profile plant design.

4. Research and Development Program: Review state-of-the-art instrumentation technology, and determine the suitability of existing instrumentation and control equipment for use in LMFBR plants. Define the areas of necessary instrumentation and control research and development, and incorporate these data in the R&D program document.

Schedule and Budget Estimate — This task is scheduled as a three-year program to begin in 1969 and be completed by the end of 1971. The following budget estimate covers only engineering services for plant design studies; the facilities and equipment R&D costs are covered in other sections of the report.

Year	1969	1970	1971
Engineering cost, \$ $\times 10^3$	400	400	400

The total budget estimate for this task is \$1,200,000. Figure 2-5 is a bar chart of the schedule for the program.

Figure 2-5. Program Schedule for Dynamic Analysis and Control Study

Task	FY	1969	1970	1971	1972
	CY	1969	1970	1971	1972
1. Components Mathematical Modeling		■	■		
2. System Modeling and Analysis			■	■	
3. Control Concepts and Instr. Specification		■	■	■	
4. R&D Program				■	

2.2.3.6. Plant Maintainability Study

Objective — The overall objective of this study is to evaluate and determine plant maintenance requirements for the reference concept to improve the pool concept LMFBR plant design for better availability and reliability. The specific objectives are as follows:

1. To identify the maintenance functions, both major and minor, that will be required for the plant, and to establish the relationships of frequency of maintenance, repair method, and maintenance time.
2. To recommend procedural or design changes that will improve plant availability and reliability while reducing plant maintenance.
3. To review and identify R&D requirements in this area.

Summary of Current Technology — To obtain a high plant availability factor, maintenance and repair considerations are extremely important. Not only must the plant have a high operational integrity, but maintenance must be readily accomplished. It is important that the full requirements for maintenance be understood in the early stages of plant evaluation in order to obtain an optimized economic design.

The LMFBR's maintenance operations must consider the opacity of the sodium, its radioactivity, and its chemical activity. Special atmospheres, temperature limitations, and special tools are often required. This consideration is compounded by the use of the additional intermediate sodium system, with the resulting increased number of components.

The LMFBR commercial-size plant outage cost can easily amount to \$50,000 per day; thus, not only is it important to minimize plant outages, but economics necessitates consideration of a substantial investment in maintenance facilities to reduce plant outage.

Water reactor plant maintenance is scheduled during refueling operations, since the refueling time exceeds the plant maintenance time. However, the LMFBR offers a short refueling time that could conceivably be less than the maintenance time.

The EBR-II, Fermi, Hallam, Dounreay, BR-5, and Rapsodie reactors have provided some information regarding repair

and maintenance. Each of these first-of-a-kind units had commissioning and operating difficulties that required maintenance; most of this experience points to the importance of adequate planning for maintenance.

The inspection and surveillance methods for the reactor pressure vessel must be evaluated to assess their effects on plant design, even though the pressure vessel inspection and surveillance criteria for LMFBRs have not yet been established. One major difference between the LWR and the LMFBR in this respect is that the former has a high-pressure reactor vessel and the latter a low-pressure tank.

Scope of Work — Starting with the LMFBR preliminary concept, the plant maintenance requirements and characteristics necessary for the 1000-MWe design will be evaluated. Utility participation will be required for this study.

1. Current Maintenance Technology: Review both existing plants and studies of maintenance requirements and practices for LMFBRs, both in this country and abroad, characterizing the requirements to determine the applicability to the reference concept. The cataloging of the characteristics will include environmental conditions and requirements for work areas, special equipment, and spare parts. Using operating experience and accident or incident information from existing LMFBRs, assemble maintenance information.

2. Maintenance Requirements: Determine the maintenance and inspection requirements for each portion of the plant. Establish the type and number of major spare part items. Establish the space requirements for inspection and maintenance, together with special tools, parts, and equipment needs. Evaluate the requirements for personnel shielding and safety features. Establish special maintenance procedures and assess the accompanying personnel training needs. Establish plant availability and the cost effects of plant downtime. Consider the need for special equipment to improve plant maintenance and economics, and re-evaluate the requirement for major spare part items. Determine the relationship between repair and replacement-part inventory for major cost components.

3. R&D Requirements: Define the R&D needed to ensure continuity and advancement of plant maintenance requirements and to improve plant availability and reliability.

Schedule and Budget Estimate — This task is scheduled as a two-year program to begin in 1971 and be completed by the end of 1972. The following budget estimate includes only engineering services associated with plant design studies; the facilities and equipment R&D costs are covered in other sections of this report.

Year	1971	1972
Engineering cost, \$ $\times 10^3$	125	125

The total budget estimate for this task is \$250,000. The schedule is shown in bar-chart form in Figure 2-6.

Figure 2-6. Program Schedule for Plane Maintainability Study

Task	FY	1970	1971	1972	1973
	C.Y.	1970	1971	1972	1973
1. Maintenance Current Technology
2. Maintenance Requirements
3. R&D Requirements

2.2.3.7. Containment and Shielding Study

Objective — The main objective of this study is to evaluate the adequacy of the current design criteria for reactor containment and shielding. Containment and shielding are combined here because of their close relationship for the design basis accident. The specific objectives of this study are as follows:

1. To review the containment and shielding requirements of the LMFBR and compare them with the present LWR technology and practice.
2. To develop new concepts that simplify the containment envelope and improve safety, reliability, and economics.

Summary of Current Technology — The containment practice for the LMFBR has yet to be established. One major difference between the LWR and the LMFBR in this respect is the high temperatures that the sodium-air reaction could produce in the LMFBR. As a result of this condition, a better understanding of plant design requirements is needed to determine the applicability of existing criteria and the necessity for new criteria. The AEC safety criteria issued in July 1967 are directed primarily to water reactors and are not necessarily applicable to LMFBRs; a review specifically directed at LMFBRs is required.

The trend toward larger reactors has created difficulties in the use of steel as a single construction material, because of the ASME Code requirement for all reactor containment vessel welds to be stress-relieved for plate thicknesses greater than 1.5 inches. The containment structures for the LMFBR require an in-depth review of the steel pressure vessel and the pre-stressed and post-stressed concrete pressure vessel concepts. Post-stressed concrete vessels (Wylfa Head) with an internal diameter of 96 feet have been erected to withstand an operating pressure of 385 psi. The advantages stated for large post-stressed concrete vessels over steel vessels are as follows:

1. The concrete structure containing steel tension members is estimated to be easier and less costly to erect (the weight of steel consumed is usually greater), and it offers good potential for increased size.
2. Since they are buried in the concrete, the stressed steel members are not exposed to high temperatures.
3. Scale model work has indicated that catastrophic pressure releases are not credible, and after the initial relief of pressure, the ruptured surfaces will reface.

Most of these advantages also apply to pre-stressed vessels.

The shielding required for an LMFBR may be thicker and more expensive than that for an LWR, since the induced

activity of the primary sodium coolant is greater and has a longer half-life. The shield cooling techniques also need additional evaluation, since a wide variety of cooling alternatives is available. Problems associated with post-DBA cooling also require evaluation, as well as the effects on other systems and the overall plant design.

Scope of Work — Beginning with the broad LMFBR plant concept developed in the 1000-MWe LMFBR Follow-On Study, the containment and shielding concept will be subjected to a system engineering analysis aimed at developing more explicit design criteria for the B&W primary pool concept.

1. Containment Technology Review: Review containment and shielding design requirements and practices for existing, planned, and design study plants that have been utilized for LMFBRs in this country and abroad. Compare these requirements with each other and with those used for LWRs. Evaluate the differences between steel and concrete containment vessels, various foundation designs (especially with regard to seismic effects), and the effects of small and large openings. State the requirements for leak-resistant penetrations. Review applicable codes, standards, and specifications to determine their applicability for LMFBR designs.

2. Shielding Technology Review: Review the available shielding materials, considering thermal expansion, long-term stability, and compatibility problems; evaluate the cost-benefit factors associated with the different materials. Review the functional requirements of the cooling system and identify their effects on plant design.

3. Design Criteria: Relate the AEC general design criteria of July 1967, to the LMFBR containment. Identify criteria that can be met by current technology, areas that require additional work (including cost-benefit factors where possible), and criteria that are not applicable. In the latter case a technical basis will be given for these conclusions.

4. Plant Design: Utilizing the reference plant design existing at that date, the following general areas will be selected for review, assessment, and further development:

- a. Location, materials, temperature limitations, size, surveillance, removal, maintenance, special features, and cooling of the neutron shield.
- b. Radiation streaming and resulting hot spots leading to equipment exposure and the need for personnel protection and local cooling.
- c. Requirements for and effects of the DBA, including energy absorption and post-DBA cooling.
- d. Containment criteria and means of accomplishing them, giving special attention to the leakage and energy absorption characteristics of containment penetrations.

5. Containment and Shielding Improvements:

Identify new concepts in containment and shielding arrangements that simplify the containment envelope and improve safety, reliability, and plant economics.

6. R&D Requirements: Obtain a better understanding of economic considerations to provide cost-benefit and R&D program evaluation.

Schedule and Budget Estimate — This task is scheduled as a two-year program to begin in 1971 and be completed by the end of 1972. The following budget estimate includes only engineering services for plant design studies; the facilities and equipment R&D costs are covered in other sections of this report.

Year	1971	1972
Engineering cost, \$ $\times 10^3$	200	200

The total budget estimate for this task is \$400,000. The program schedule is shown in bar-chart form in Figure 2-7.

Figure 2-7. Program Schedule for Containment and Shielding Study

Task	FY	1970	1971	1972	1973
	CY	1970	1971	1972	1973
1. Containment Current Technology			■		
2. Shielding Current Technology			■		
3. Design Criteria				■	
4. Plant Design			■	■	
5. Containment and Shielding Improvements				■	
6. R&D Requirements					■

2.2.3.8. Effects of Vented Fuel on Plant Design

Objective — The primary objective of this study is to provide a thorough understanding of the engineering considerations associated with using vented fuel in LMFBR power plants. The five specific objectives of this study are as follows:

1. To establish plant design and siting factors that may limit or control the use of vented fuel.
2. To establish fuel handling, shipping, and re-processing factors that may limit or control the use of vented fuel.
3. To assess the plant safety aspects of using vented fuel.
4. To obtain a better understanding of design, construction, and operation of reactor systems using vented fuel.
5. To fully establish the R&D requirements to accommodate vented fuels, and assess the cost-benefit factors relative to the LMFBR program.

Summary of Current Technology — The development of a successful LMFBR depends on achieving a low fuel cycle cost. To minimize fuel fabrication and reprocessing costs, high fuel burnup levels must be achieved while maintaining good nuclear economy. These considerations favor the use of vented fuel pins with controlled release of gaseous fission products. In addition, fuel venting may be advantageous because it eliminates the potential safety problem of instantaneous fission product release with subsequent fuel pin failures and gas blanketing of neighboring pins.

However, it should be noted that venting may not allow much higher burnup or reduced cladding thickness. At present no theory can adequately handle the fuel swelling caused by solid fission products and restrained gaseous fission products. Vented fuel also causes increased contamination of the blanket gas.

Most plants are designed to handle failed fuel elements, and hence, all existing plants have, at least indirectly, contributed to fuel venting experience.

The Dounreay experimental fast reactor is the only operating reactor that was originally designed with vented fuel. No unusual safety feature requirements were encountered, and this reactor has demonstrated the feasibility of vented fuel satisfactorily. Although some effort has been expended on the investigation of vented oxide fuel for LMFBR plants and on trade-off studies for vented versus non-vented fuel, it is considered that the data are not sufficient for meaningful plant design.

Scope of Work —

1. Current Technology: Review pertinent information on vented oxide fuel for both the loop and pool primary systems; catalog, compare, and evaluate all features. This compilation will include the results of studies as well as in-reactor and out-of-reactor data and experience.

2. Design Evaluation and System Engineering: Consider different types of venting mechanisms and evaluate the type, rate, and amount of release of fission products. Investigate and evaluate the factors that control the deposition of fission products in the

system. Identify requirements and techniques of equipment and system cleaning. Evaluate the effect of entrained fission gasses on plant design, performance, and safety. Consider the effects of vented fuel on the refueling system and on fuel shipping and reprocessing requirements. Define the requirements and methods for detecting failed fuel and the corrective action necessary.

3. R&D Program: Assess fuel effects on plant and power costs and develop cost-benefit information for inclusion in the R&D program. Define the R&D necessary to permit the use of vented fuel in the reference plant.

Schedule and Budget Estimate — This task is scheduled as a two-year program to begin in early 1971 and be completed by the end of 1972. The following budget estimate includes only engineering services associated with plant design studies; the facilities and equipment R&D costs are covered in other sections of this R&D report.

Year	1971	1972
Engineering cost, \$ $\times 10^3$	250	250

The total budget estimate for this task is \$500,000. The schedule for the program is shown in bar-chart form in Figure 2-8.

Figure 2-8. Program Schedule for Study of Vented Fuel Effects on Plant Design

Task	FY	1970	1971	1972	1973
	CY	1970	1971	1972	1973
1. Current Technology
2. Design Evaluation and System Engineering
3. R&D Program

2.2.4. Plant Safety Analysis

2.2.4.1. Objectives

A comprehensive safety analysis will be carried out to achieve the following objectives:

1. Demonstrate that the nuclear plant can be operated safely and without undue risk to the public by upgrading the reference design and adding necessary engineered safety features.

2. Develop a program for the R&D required for the engineered safety features and for methods development and verification. Perform cost benefit studies for the R&D program as it is developed.

3. Develop a document comparable to a PSAR for presentation to the AEC for licensing consideration. The content will be generally responsive to paragraph 50.34 of Title 10 CFR, Part 50, and to the "General Design Criteria, Liquid-Metal-Cooled Fast Breeder Reactor Using 1980 Technology" (see BAW-1328, Volume 3, section 14).

4. Develop a clearer definition of the design basis accident (DBA).

5. Following the licensing review, update the "General Design Criteria, Liquid-Metal-Cooled Fast Breeder Reactor."

2.2.4.2. Summary of Current Technology

The end product of the AEC's 1000-MWe LMFBR Follow-On Program is a reference plant design, including the reactor, heat transport system, steam turbine systems, auxiliary systems, instrumentation and control system, and engineered safety features. The normal operation of the plant and the reactor is assessed in that program, but the behavior of the reactor and the plant during abnormal situations is not studied in detail.

There are at least three levels of interest for any safety analysis associated with the 1000-MWe liquid-metal-cooled fast breeder reactor. To satisfy the first level, it must be demonstrated that the plant can be operated in a stable manner and that the control

actions themselves do not lead to fuel or core damage. This level has been partially achieved by the preliminary control analysis study.

The second level is related to equipment failure or malfunction. The plant design must ensure that these incidents are highly improbable and that, although limited core damage may be sustained in some cases, no general core damage should occur. Suitable protection will be engineered to minimize and localize the damage.

The third level includes cases in which significant damage to the core or the core vessel may occur. The plant design must render such events incredible. On this interest level, the most significant objective is the demonstration of the adequacy of protection of the public and the plant operators.

Many LMFBR characteristics are different from those of LWRs and are relatively independent of reactor or plant design. These differences will impose design requirements that have important effects on commercial acceptability of LMFBR plants. Some of the more important characteristics are listed below.

1. The fuel is not assembled in its most reactive arrangement.
2. Criticality can be achieved without any coolant in the core.
3. The core has a negative coolant density coefficient.
4. The effective fraction of delayed fission neutrons is small, about 0.004.
5. The prompt neutron lifetime is short, about 10^{-7} seconds.
6. The concentration of fission products can be high because of the potentially high fuel burnup.
7. The plutonium inventory is high.
8. The material irradiation levels are high.
9. The fission products may be vented to the core coolant.
10. The reactor core coolant and the intermediate heat transport fluid react vigorously and exothermically with both air and water.
11. The primary coolant is intensely radioactive and has a significantly long half-life.

12. The steam generator contains both sodium and water.

To date, five LMFBR facilities have been considered and reviewed by the Division of Reactor Licensing (DRL) and the Advisory Committee for Reactor Safeguards (ACRS). The FFTF most probably will be reviewed next. The review of these facilities has contributed significantly to our further understanding of reactor safety. Only one of these facilities, the 250-MWe Enrico Fermi reactor, has been licensed as a commercial power plant, reflecting the safety criteria, technology, and attitude of the 1950's.

The recent trend away from high-leakage cores in commercial-sized LMFBR designs reflects a change from pronounced caution to a more realistic approach to voiding and other safety-related questions, plus a stronger reliance on engineered safety features. The safety and design philosophy associated with this trend in core design requires review and debate by all parties. By conducting this review at an early stage of the National LMFBR Program, it will provide guidance for the demonstrator plant program and for future research and development activities.

The criteria, policies, and procedures associated with the safety and licensing of commercial LMFBRs must be established before LMFBRs are introduced to the electrical utility industry. Reactor manufacturers, the electrical utilities, and the Federal regulatory agencies must resolve the problems associated with the safety of the commercial-sized LMFBR, and they must establish specific requirements for licensing these plants.

There is a general feeling in the nuclear community that safety evaluations and licensing for commercial LMFBRs will be a direct outgrowth of their predecessor demonstrators. While this development will occur to a large extent, unnecessary restraints or requirements may be placed on the commercial product that is to follow, unless licensing of the larger, more competitively-oriented plant is considered independently at this time. Complete safety criteria and requirements cannot be established with the smaller demonstrator plants, because they are likely to be designed under conservative safety criteria for less stringent operating conditions. Furthermore, the demonstrator

plants will probably have inherent safety features, such as high-leakage cores, which may not be readily achievable or desirable in the larger commercial cores. The large breeder cores required for commercial operation must have safety and design characteristics that represent an acceptable and necessary compromise between inherent and engineered safety features on the one hand, and plant economics on the other. The engineered safety features must compensate for some of the inherent safety characteristics that would drive LMFBR costs beyond acceptable limits.

2.2.4.3. Scope of Work

1. Malfunction and Safety Problem Survey:

Catalog the conceivable malfunctions or faults that could involve the plant, the reactor, the core protection system, and the site and environment. Construct fault trees for these malfunctions to assess the qualitative response of the reactor and the protection system to each malfunction.

2. Definition of Initiating Conditions for Reactor Accidents: Analyze quantitatively the plant response to the malfunctions in order to determine the driving functions sensed by the reactor and protection systems as a consequence of the malfunctions. Perform a sensitivity analysis to demonstrate the effects of input constants and assumptions.

3. Analysis of Credible Reactor Accidents: Analyze the response of the core and the protection system to credible reactor accidents. A credible reactor accident is defined as any incident in which there is a single malfunction of the plant, the reactor, or the site or environment, plus a single failure of the protection system. Redefine the trip settings of the protection system to achieve the desired response by imposing the least severe protection system action required to maintain the basic integrity of the core.

4. Analysis of Design Basis Accidents: Analyze severe accidents that may result from reasonable independent malfunctions whose coincidence is not deemed credible; determine the consequences that could be injurious to the public and to the operator.

Develop functional requirements for the engineered safety features as required to prevent the accident, to mitigate its consequences, or to provide adequate protection for the public and the operator should such a design basis accident occur.

5. Environmental and Site Analysis: Evaluate the effects on the environment of wastes released from normal operation and from credible accidents. These evaluations will establish design requirements to ensure that the protection system is adequate to prevent the release of quantities of radioactive material great enough to be injurious to the public. Title 10, Part 20, of the Code of Federal Regulations will be used as a guide in this analysis.

Evaluate the effects of accidental release of wastes to the environment for those accidents which are considered to be incredible. Title 10, Part 100, of the Code of Federal Regulations will be used as a guide in this analysis.

Develop the functional requirements for the engineered safety features as required to prevent the accidental release of radioactive material, to mitigate its consequences, or to provide adequate protection for the public and the operators should a release occur.

6. Criteria Development and Assessment: Review, modify, and expand the "General Design Criteria for Liquid-Metal-Cooled Fast Breeder Reactors," and incorporate the changes resulting from B&W, utility, and AEC reviews. Analyze the general design criteria for LMFBRs produced by other 1000-MWe subcontractors, and recommend appropriate changes in the General Design Criteria. Review, as necessary, the General Design Criteria at appropriate times during the safety analysis and following reviews by utility, AEC, and ACRS personnel.

7. Preparation of the PSAR: Incorporate the information available on the reactor and plant design and the information obtained in the safety analysis studies into a Preliminary Safety Analysis Report (PSAR). Present this document for review to the utility, the AEC regulatory staff, and the ACRS. Interpret the response from the review in terms of the design changes that would be required to license

the reactor, the feedback to the General Design Criteria, the design requirements for the plant engineered safety features, and the R&D program.

8. Methods: Provide minor modifications in existing codes as used in LMFBR safety studies, to keep current. Obtain computer codes that are released by contractors developing methods for the AEC and by the national laboratories; from these, develop operational programs.

9. R&D and Test Program: Review current R&D and test efforts appropriate to the LMFBR, starting with the R&D and Test Program developed during the 1000-MWe LMFBR Follow-On Study. Delineate areas where there is a lack of confidence in the safety analysis because of the lack of appropriate data or methods. Identify areas in which economic gain might be realized by the availability of nonexistent data or by an improved understanding of the physical mechanisms or processes.

Develop appropriate R&D and test programs to meet the needs of the ACRS and those requirements established by liaison among the AEC, utilities, and B&W.

Analyze the R&D and test programs on a cost basis; assess the effect on the economics of the reference design.

2.2.4.4. Schedule and Budget Estimate

This task is scheduled as a three-year program to begin in 1969 and to be completed by the end of 1971. The following budget estimate includes only engineering services for plant design studies; the facilities and equipment R&D costs are covered in other sections of this report.

Year	1969	1970	1971
Engineering cost, \$ $\times 10^3$	500	500	500

The total budget estimate for this task is \$1,500,000. The schedule for this program is shown in bar-chart form in Figure 2-9.

Figure 2-9. Program Schedule for Plant Safety Analysis

2.2.5. Predecessor Plant Design

2.2.5.1. Objectives

The primary objectives of this study are to further the development of plant design technology, making full use of the previous engineering study data, and to establish the plant design requirements for the B&W predecessor reactor to a stage that is compatible with contract negotiation and sale requirements. The predecessor is expected to be larger than the currently conceived demonstrator plant (200 to 500 MWe) and smaller than the 1000-MWe reference plant. Although the previous evaluations and system profile design will use the 1000-MWe LMFBR as a reference, the impact of each factor will be scaled to the predecessor plant. Consistent with this function, the objectives of the predecessor plant are as follows:

1. To demonstrate that a commercial-size reactor can be built within the scope of time and money restraints.
2. To show that a commercial-size reactor can be operated safely, reliably, and economically.
3. To provide meaningful focal points while validating technical and economic results of the R&D program.

2.2.5.2. Summary of Current Technology

The commitment to build the predecessor plant will depend on the engineering and economic assessment and the willingness of manufacturers, utilities, and the AEC to participate and accept the risks involved. The answer to this commitment cannot be established at this time, since the willingness of all parties to participate will depend on several factors:

1. The degree and success of the earlier plants in the USA and abroad.
2. The economic incentive of the fast breeder reactor, as compared with thermal reactors or other competitive energy sources existing at the evaluation date.
3. The different types of LMFBR systems planned or in operation.
4. The capability of the industry.
5. The availability of funds.

The demonstration plants, while showing the potential of LMFBRs to serve as economical power sources for the utility industry, may not in themselves demonstrate the economics or safety features of a large reactor plant. Safety criteria and requirements for the large plants cannot be directly taken from the demonstrators, which are likely to be designed under conservative safety criteria for less stringent operating conditions. Furthermore, the demonstration plants have inherent safety features, such as high-leakage cores, which may not be readily achievable or desirable in the larger commercial reactors.

The larger breeder cores must have safety and design characteristics that represent an acceptable and necessary

compromise between inherent and engineered safety features on one hand, and plant economics on the other. In addition, because of the early demonstrator commitment date and the generalized lack of technology, the demonstrator plants are likely to have a "belt-and-suspenders" over-engineered approach. The increase in reactor size will depend on many features, and the change from the demonstrator to the target commercial plant may take several steps to fulfillment of achievement.

2.2.5.3. Scope of Work

Commencing with the LMFBR system profile plant design developed in the previous task for plant design studies, this study will be aimed at exploiting the advantages obtained, resulting in a design for a predecessor plant based on the best technical approaches obtainable at that date.

1. Major Feature Selection: Review all available data regarding LMFBRs built, under construction, or planned in this country, and fully assess specific requirements to justify the selection of the size and major features of the predecessor plant.

2. Core Design: Review existing fuel element designs as they relate to the near-term application in the predecessor plant, and define a reference design by more refined analysis and evaluation. Establish a reference core and define the associated safety and operating criteria and conditions. Review the current core support, coolant plenum, and holddown features with regard to performance and safety considerations, and synthesize the core structures concurrently with the establishment of the reference core.

Analyze fuel management schemes to relate operating limits, fuel cycle cost, and safety characteristics. Establish a reference fuel management method and generate detailed operating safety and economic performance data.

3. Systems Engineering and Analysis: Review the 1000-MWe reference design concepts, plant design studies, and supplemental studies developed in the LMFBR research and development program; incorporate the best results into the reference plant concept.

Develop a catalog of problem areas that require specialized and/or detailed attention as part of the scaling down of the plant design.

Review the heat generation and heat transport systems to evaluate the component and subsystem interaction requirements. Study the effects of relative thermal expansion, differential pressures, and changes in pressure, temperature, and flow to establish an improved set of functional design requirements. Relate these functional requirements to a set of principal design criteria by considering the imposed safety requirements.

4. Design and Construction Technology: Apply the fabrication and maintenance requirements, along with the materials and cost data developed in previous studies, to determine the system arrangement and to establish the functional requirements and parameters. Review the existing methods and computer programs for thermal and stress analysis to determine their applicability to the LMFBR design.

Analyze the construction methods and scheduling for component and plant fabrication and erection, including an evaluation of on-site versus shop fabrication of the major components. Study the requirements for quality control methods and procedures needed to fabricate a quality product.

Identify and catalog the maintenance problems associated with the reactor vessel and other major components, and evaluate the ability of available maintenance methods and procedures to solve the problems.

Review the surveillance and inspection requirements, identify those required for the reactor vessel and other major components and systems, and relate these to available methods and techniques.

Review the existing codes, standards, and specifications to determine their applicability. Obtain assistance from an architect-engineer, as required, in various portions of this task.

5. Nuclear Steam Supply (NSS) Design: Perform an overall system engineering analysis on the reactor vessel and other portions of the NSS to ensure that the design meets the functional

design requirements. This analysis will include data from the plant scale model testing in the Components section.

Give special consideration to the thermodynamic and hydraulic effects on components in the cavity formed by the reactor cover structure and the biological shielding.

Evaluate the effect on the core and heat transport systems of displacements caused by the physical weight of the components, their thermal distortion, pressure differentials, mechanical and flow-induced vibrations, seismic ground motions, and sodium bubble growth and collapse motions.

Define the best technical approach to the goal of optimizing the NSS for minimum power cost. Some analysis of alternate methods and approaches will be required to achieve this goal.

Complete the analytical design studies of the major components of the NSS, including reactor vessel system components, the IHX, and the steam generators. These studies will consider static and dynamic loading requirements and various methods of support, which must allow for thermal expansion and other displacements. Various methods of attaching the reactor vessel and other major components to the cover structure must be reviewed, as well as the type and location of insulation used on the cover structure, to determine that the methods chosen are adequate.

Based on the results of the overall plant and systems design, prepare system design descriptions for the reactor vessel, IHX, steam generators, and the associated interfacial equipment. The overall system cost data will be tabulated and analyzed to determine the capital investment and operating cost.

6. R&D and Test Program: Define the R&D and test program required to ensure continuity in the development of the predecessor and reference plant designs to provide component and system data for inclusion in the PSAR. Develop the cost-benefit ratios for the various elements in the R&D and test program, and incorporate these ratios in the R&D and test program document.

2.2.5.4. Schedule and Budget Estimate

This task is scheduled as a six-year program to begin in 1970 and be complete by the end of 1975. The following budget estimate includes only engineering services for plant design and engineering studies; facility and equipment R&D costs are covered in other sections of this R&D report.

Year	1970	1971	1972	1973	1974	1975
Engineering cost, \$ $\times 10^3$	250	500	1250	1625	2000	2000

The total budget estimate for this task is \$7,625,000. The schedule for the program is shown in bar-chart form in Figure 2-10.

Figure 2-10. Program Schedule for Predecessor Plant Design

2.2.6. Reference Plant Design

2.2.6.1. Objective

The objective of this study is to improve the predecessor plant design developed during the previous task, making full use of the completed plant design and supplemental studies, to bring the design to a stage compatible with contract negotiation and sale requirements.

2.2.6.2. Summary of Current Technology

The commitment to build the reference plant will depend primarily on the success and achievement of the predecessor and other LMFBR plants. It will also depend on the engineering and economic assessment existing at the time, together with the requirements of the utility to purchase.

The outstanding conceptual difference between the predecessor and reference plants is that the fuel for the predecessor is non-vented, and the fuel for the reference plant is vented-to-sodium. Thus, prior to the use of the vented fuel concept in the reference plant, its feasibility and reliability must be demonstrated.

Since it is considered that the reliability features can only be demonstrated in a reactor, it is proposed that the predecessor plant will be designed to contain a limited number of vented fuel assemblies. In addition, a two-year period has been allowed between the operation date of the predecessor plant and the sale of the reference plant to confirm and demonstrate reliability.

2.2.6.3. Scope of Work

The scope of work in this subtask is similar to that for the predecessor plant; see section 2.2.5.3.

2.2.6.4. Schedule and Budget Estimate

This task is scheduled as an eight-year program to begin in 1976 and be completed by the end of 1983. The following budget estimate covers only the engineering services for plant design studies; the facilities and equipment R&D costs are covered in other sections of this report.

Year	1976	1977	1978	1979	1980	1981	1982	1983
Engineering cost, \$ × 10 ³	500	500	500	750	1250	1625	2000	2000

The total budget estimate for this task is \$9,125,000. The schedule is shown in bar-chart form in Figure 2-11.

Figure 2-11. Program Schedule for Reference Plant Design

2.2.7. Advanced Concept Studies

2.2.7.1. Objective

The primary objective of these studies is to further improve the reference plant design by development related to the safety, economics, and performance of the LMFBR system. An additional objective is to establish and evaluate the use of LMFBRs for applications other than central-station power production.

2.2.7.2. Summary of Current Technology

General experience has shown that, although first-of-a-kind units have higher capital costs than predecessors, the predecessor-modified and -improved designs result in a continuing reduction of capital cost when related to a constant dollar value equivalent. There is no apparent reason why the LMFBR system should not also follow a similar trend, which is consistent with a competitive market. The general lack of sodium technology in industry, and specifically the electric utilities, will probably lead to conservatively rated plant and equipment. Thus, as experience with sodium is obtained throughout industry, improvements in plant safety, economics, performance, and erection should be obtainable.

For the immediate future, the principal application for the LMFBRs will be for central-station electric power

generation. However, with the rapidly expanding economy and the established use of LWRs for process heat, technology requirements may well dictate other uses for the LMFBR system.

The possibility of using LMFBRs for dual-purpose requirements appears to be more attractive than using other reactor systems. The projected fuel costs are low, the output temperature is high, the power generation unit is not too complex, and the system can produce fuel in excess of its own needs.

It appears unlikely that the LMFBR will be suitable for small units in the near future. Both the operating characteristics and the use of sodium technology demand a relatively highly trained and qualified staff. This should not, however, be looked upon as a permanent feature, and it could be that with further development and operating experience with sodium circuits, design and operating procedures could make the system sufficiently simple and standardized for smaller, isolated reactor sites.

2.2.7.3. Scope of Work

Commencing with the broad LMFBR engineering knowledge developed and obtained during the first three years of the plant R&D studies, these studies will be aimed at improving and exploiting the further advantages that can be achieved by LMFBRs.

1. LMFBR Review: Review all available data regarding the LMFBR with specific reference to major different features and novel designs or proposals. Evaluate these data and prepare a program of work.

2. Central-Station Plant Design Improvements: Work in this area will be directly associated with the consideration for improvements to the central-station plant. The work task to be considered will include but not be limited to the following:

- a. On-line refueling.
- b. Investigation of a more compatible intermediate coolant.
- c. Elimination of the intermediate coolant system.
- d. Methods to improve safety.

e. Methods to reduce capital and indirect costs.

f. Methods to improve plant economics.

In addition, all major systems will be reviewed to identify areas where improvements could be considered.

3. Evaluate the LMFBFR for Other Uses:

Work in this area will be concerned with the evaluation of LMFBFRs for uses other than as central-station power plants. The work tasks to be considered will include isolated reactor sites, power and water storage schemes, power and process heat production, power and water purification, small- and medium-sized power reactors, and agricultural-industrial complexes for underdeveloped areas.

4. Advanced Concepts: Work in this area will be concerned with LMFBFRs that do not use sodium as the primary coolant and whose fuel is not necessarily of circular section or assembled in pins.

5. R&D Program: Define the R&D requirements to ensure continuity in the development and advancement of the LMFBFRs. Estimate the R&D costs required for the program, and develop cost-benefit data for inclusion in the R&D program.

2.2.7.4. Schedule and Budget Estimate

This task is scheduled as a nine-year program to begin in 1972 and be complete by the end of 1980. The following budget estimate is for only the engineering services associated with advanced concept studies.

Year	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
Engineering cost, \$ × 10 ³	250	500	500	500	750	750	750	750	500	500	250	250

The total budget for this estimate is \$6,250,000. Figure 2-12 shows the program schedule in bar-chart form.

Figure 2-12. Program Schedule for Advanced Concept Studies

[illegible]

2.3. Cost Benefit

The expansion of America's capacity to generate nuclear energy (installed and on order) far exceeds the most optimistic projection of only a few years ago. Attempts to revise projections may continue to underestimate user demands and the impact of air pollution controls. Indeed, the recent EEI report on nuclear power growth,¹ published in April 1968, estimates a 1980 capacity of 195,000 MWe, compared with the earlier AEC projection of 120 to 170 MWe. The same EEI report projects a capacity of 1,000,000 MWe in the year 2000. This postulated rapid expansion of nuclear power, if met only by light water reactors, will place heavy demands on uranium supplies. Using the EEI projection for the year 2000 and assuming a light water industry with plutonium recycle beginning in 1976, the cumulative U_3O_8 requirements will be approximately 2,300,000 short tons. This requirement vastly exceeds the AEC's 1968 estimate of uranium ore available at less than \$15 per pound. It is expected that current and projected exploration will locate substantial additional ore reserves, a portion of which may be within the \$10 to \$15 per pound price range. However, the free world nuclear power industry is expanding at a rate similar to that of the United States, so that competition for ore is expected to develop on a world-wide basis.

If the annual ore requirement continues to increase rapidly, as projected, it will necessitate continuous development, investment, and construction of new mines, mills, and diffusion plants. This investment by the mining industry to meet the uranium demands through 1980 could exceed \$1 billion.^{2,3} Fast breeder reactors greatly reduce this requirement for continuing to increase uranium ore production.

The most important single incentive to develop fast breeder reactors is their potential for achieving lower power generation costs. Other incentives for developing the breeder reactor are to protect the capital investment in light water reactors against the increase in fuel costs that would result from the continuing rise in uranium ore costs, and to conserve and make more efficient use of our national resources.

The power generation costs of the fast breeder reactor must be compared to those of the light water reactors. Light water reactors appear to be in a reasonably advanced stage of development. It is expected

that future improvements will be primarily the result of refinements that require only nominal research and development costs, compared with previous investments.

This is not the case with the fast breeder, which will require considerable research and development funding to attain a competitive position. The R&D cost may well exceed \$1 billion, which must be justified by anticipated savings in power generation costs. To compete effectively, LMFBF fuel costs must be lower than LWR fuel costs, since capital costs are expected to be higher for the first-round LMFBFs than for the equivalent-sized LWRs. However, the postulated total power generation cost for the LMFBF is expected to equal that of the LWR in 1985 and to be approximately 0.9 mill/kWh lower in the year 2000.¹ On the basis of this projection, the postulated LMFBF power generation cost will be 0.5 mill/kWh lower than the equivalent LWR power cost in the year 1993, which corresponds to savings of \$3,500,000 per year for a 1000-MWe plant operating at an 80% capacity factor.

B&W's long-range forecast estimates that sales of LWRs to utilities will begin to decrease in 1987 because of LMFBF competition. The added LMFBF capacity is expected to grow rapidly, reaching about 25,000 MWe per year in 1990 and about 100,000 MWe per year by 2000. Using the EEI cost savings estimates cited above, the annual operating cost savings of LMFBFs coming on-line in a single year is expected to pass \$100,000,000 by 1995. Cumulative operating cost savings compared with those of LWRs would reach \$1 billion by the year 2000.

These cost benefits are based on the assumption that the R&D necessary for optimum-design LMFBF plants is pursued vigorously. An operating cost increase of only a few tenths of a mill per kWh, caused by a reduced or delayed R&D effort, would have significant economic effects. Both the savings per LMFBF plant and the number of LMFBFs sold (instead of LWRs) would decrease, thus doubly reducing the total savings and increasing power generating costs.

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3. COMPONENTS

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3. COMPONENTS

3. 1. Introduction

3. 1. 1. General

Successful, economical, and safe operation of the large sodium-cooled fast breeder reactor will depend to a great degree on the reliable performance and the availability of all plant systems. These systems are made up of components, and thus the performance and availability of the systems are directly dependent on the functional adequacy of the components, their individual reliability, and their requirements for and ease of performing maintenance. The LMFBR Follow-On Study clearly showed the reliance of component on component, component on system, and system on system. It is very important to emphasize this interdependence, so that prescribed programs for the development of components can be accomplished with full cognizance of this interplay. Success or failure, and the degree of such results, must be factored against system requirements and against offsetting results that may occur in developing other components of other systems.

The objective of each of the projects in the components R&D plan is to perform the analytical and experimental efforts that are necessary to permit fabrication, assembly, and operation of the components and their systems. The overall objective is to develop all components and all systems to permit the construction and economical and safe operation of the LMFBR plant. The basis for the prescribed work is the B&W conceptual design established in the Follow-On Study. Design descriptions, design requirements, functional requirements, principles of operation, and safety precautions for the major components in this design were discussed in the Task II and III report for the study.¹

Component design is affected by the many environmental and performance differences between the LMFBR and the water reactor.

In some cases, these differences may be solved analytically, but most require development of mathematical models and analysis techniques that are not available now. Experimental confirmation of these results may be required. In other cases, detailed engineering and design work will clarify the degree of problems and possibly solve some uncertainties. In still other cases, fabrication of models and nondestructive and even destructive testing will be required before the degree of confidence is raised to a level that will justify safe and economical operation of the LMFBR plant.

For the component development programs described in this report, it is assumed that the B&W conceptual design has been advanced to the preliminary design level; that is, components have been studied and engineered to establish the design as optimum, subject to successful completion of the R&D efforts. For example, it is assumed that the reactivity control rod drive proposed in the conceptual design has been studied in considerable depth concurrently with a study of alternate design possibilities; that a trade-off study has been performed to establish that this concept design is the best; that all of the major parts of the drive have been investigated sufficiently to establish that they conform to all requirements, subject to final development; and that all engineering has been completed to a state that will permit detailed design and fabrication of test components to proceed. The prescribed programs include the study efforts necessary to establish the detail requirements for research and development; also included are the efforts required to design models, test specimens, and prototypes. The programs provide analytical support for the development of methods, testing procedures, facilities, and equipment.

Of major importance in this plan are the components that must function in the sodium coolant or in a sodium vapor environment. High temperatures, temperature fluctuations, radiation effects, corrosion, sodium deposits, and the effects of fission product plateout are all considered. Also important are the components that have special functional requirements; safety is paramount, and therefore, reactivity control devices head this list. Availability, reliability of performance, length of life, maintenance, and economics are important, and the development programs consider these factors as well.

Significant advances have been made in sodium technology (see section 5 of this report) and in the development of components that are suitable for operation in the environment of the LMFBR, but there are still areas where research, development, and testing require further attention. Additional work is now being performed under AEC sponsorship, e. g. , work at the Liquid Metal Engineering Center (LMEC) on the development of large sodium pumps and steam generators, and their testing work to determine material behavior and wear characteristics in a sodium environment. The results of these efforts, along with the concentrated preliminary design efforts and other related programs, are important prerequisites for this R&D program.

The components R&D plan is divided into eight sections, each of which discusses a major component area and describes the R&D tasks to be performed. Two additional sections are added: one describes a program to determine surveillance requirements and to establish equipment and techniques for its accomplishment; and a second deals with cost benefits or the quantitative and qualitative incentives for carrying out the components R&D plan.

3.1.2. Summary Budget and Schedule

Table 3-1 and the tabulation below summarize the budget and schedule needed to develop the new data and information required for successful design, construction, and operation of the LMFBR reference plant components. Table 3-1 gives a total dollar breakdown by calendar and fiscal year. The tabulation is a summary of required manpower, facility, and equipment cost estimates. The estimated costs do not include construction of any major testing facilities, but they do allow for adaptation and rental of existing facilities, such as those at LMEC. It is assumed that these facilities can provide all of the testing space required; no effort was made to compare total demand on the facilities with available space or schedule.

Summary Budget Estimate

Manpower	\$15,737,000
Facilities	6,235,000
Equipment	12,040,000
Total	<u>\$34,012,000</u>

Table 3-1. Summary of Overall Schedule and Estimated Budget
for R&D Work on LMFBR Plant Components

Task	FY CY	Estimated cost, \$ × 10 ³										Total
		1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	
		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	
Components												
1. Steam generator		1700	1600	1800	1200	600	--	--	--	--	--	6,900
2. Liquid-metal to metal heat exchanger		--	--	230	530	420	460	560	100	100	--	2,400
3. Sodium valves and piping		--	--	--	350	700	1225	825	600	100	100	3,900
4. Large sodium pump development		--	--	--	600	800	800	2100	2200	1500	--	8,000
5. Core components handling equipment		--	--	--	--	445	710	640	847	495	115	3,252
6. Misc handling equipment and facilities		--	--	--	--	--	--	95	395	410	230	1,130
7. Reactivity control system		--	270	270	770	840	310	130	490	520	410	4,010
8. Reactor vessel and related components		--	360	490	540	610	380	380	230	130	--	3,120
9. Component surveillance		--	--	50	50	150	300	200	200	200	150	1,300
Total		1700	2230	2840	4040	4565	4185	4930	5062	3455	1005	34,012

The budget provides estimates for the engineering and design efforts required to define and produce models, test specimens, and prototypes needed for detail development. It includes the manpower for engineering, test planning, test operation, and test analysis. It also lists the cost of manufacturing the required equipment, test apparatus, and instrumentation. Testing of components in a sodium environment is budgeted where necessary, but maximum use of non-sodium research facilities for assembly and testing is planned wherever possible.

The R&D plan and its schedule are based on making maximum use of design, fabrication, testing, and operational data that will be generated in other LMFBF programs; in addition, the plan considers the required uses of the development results. Note that the steam generator project is a continuation of current programs. Most of the other component development projects are more dependent on information from the FFTF and the first demonstration plant, which is scheduled for operation in 1976. Generally, these projects terminate in 1977 and 1978, which will permit demonstration of their results in the B&W predecessor plant before they are required for the commercial LMFBF. In reviewing the latter schedules, it should be noted that they also depend on concentrated preliminary design efforts. The plant design projects in section 2 of this report should be referred to for cost estimates and near-term schedules for these efforts.

3. 2. Overall R&D Requirements

3. 2. 1. Steam Generator

3. 2. 1. 1. General

Programs for the development of technology for liquid-metal-heated steam generators have been pursued actively for more than 15 years. Steam generators have been built and operated in power plant systems on the submarine "Seawolf," EBR-II, Sodium Reactor Experiment, LAMPRE, Hallam Nuclear Power Facility, and Enrico Fermi Power Plant in the United States, as well as in the Dounreay Fast Reactor in Great Britain and KNK in Germany. Some of the steam generators have operated essentially without trouble, and some experienced difficulty during initial operation; however, they have operated

quite well since the troubles were corrected. A few, e. g., Fermi, have had recurring problems that have prevented the plant from demonstrating reliable operation.

Under contract to the USAEC since 1963, Babcock & Wilcox (B&W) has been carrying on a program to develop a safe, reliable, economical sodium-heated steam generator. The program has included the preliminary design of a full-size steam generator, supporting research and development work, preliminary design of a prototype steam generator, and detailed design and fabrication of the prototype.

Some engineering tests were performed on the first helical-coil prototype steam generator, but tests with sodium have not materialized. The prototype program has been re-evaluated, resulting in the decision to eliminate the stainless steel in the superheat section and replace it with Croloy 2-1/4. This decision is compatible with the temperature conditions that occur in the B&W reference design.

The development program described herein is concerned only with the development of a single-bundle steam generator. The reheater will not be mentioned, because successful steam generator development will mean that reheater problems are also solved.

3.2.1.2. Sodium-Water Reaction Studies

Objective — These studies are to be carried out to study the effect of a large sodium-water reaction on the B&W helical-coil steam generator selected for the reference design 1000-MWe LMFBR.

Summary of Current Technology — A great concern to the designers of sodium steam generators is the possibility of a leak in which high-pressure water and/or steam mixes with the sodium. The results of such a reaction are difficult to predict in detail.

Information gained by APDA in their program of sodium-water reactions indicates that the effects of mixing are probably a function of both the sodium and water temperatures, the flow rate of the water into the sodium, and the tube ligament.

Small leaks do not generally cause a pressure transient, but they do result in contamination of the sodium and the accompanying evolution of hydrogen gas.

Large leaks produce severe pressure transients which are a function of the inlet water flow rate, the temperature, and the time required to actuate the rupture discs. The actual kinetics of the sodium-water reaction with the steam generators are not well known. Some theoretical formulations have been applied to experimental observations, resulting in a number of qualitative conclusions. Much work remains to be done in determining the effects of sodium-water reactions and in translating these effects into useful design information.

Technical Justification — The problems associated with the development of heat exchange equipment are many and varied; they relate to performance, reliability, maintenance, lifetime, and economics. However, it is not common to use two fluids that react as violently as sodium and water in the quantities anticipated in a 1000-MWe plant. With the present state of the art, it has been demonstrated that large sodium-water reactions can be accommodated safely. The questions that have not been resolved are those involving minimization of damage after a reaction, design features that will prevent the propagation of leaks from tube to tube without excessive cost, and design features that will minimize the probability of small leaks. Generally speaking, design features that appear to satisfy the requirements for a steam generator, relative to the sodium-water reaction, tend to make it uneconomical. For example, APDA has shown that damage will not propagate if there is a great enough distance between tubes.

The R&D outlined under this task is required for a better understanding of the effects of a sodium-water reaction in order to eliminate the costly conservatism of current steam generator design.

Scope of Work — The study of the effects of a large sodium-water reaction will be divided into three tasks: (1) an investigation of reaction fundamentals, (2) a dynamic stress analysis of the steam generator, and (3) an experimental evaluation of the properties of candidate materials under transient loading conditions. In Task 1, a method for predicting the shock and pressure wave characteristics and their propagation in a large-scale sodium-water reaction will be developed. This will be accomplished by a systematic investigation of the

mixing process, experiments in small-scale equipment, development of scale-up theory to analyze the large sodium-water reactions in commercial steam generators, and verification of the theoretical work by a small number of large-scale experiments. Task 1 is divided into the following four phases:

Phase I — Preliminary studies which will

1. Ensure maximum utilization of existing information.
2. Establish the capability for measuring shock characteristics in liquid sodium.

Phase II — An investigation of shock and pressure waves in small-scale sodium-water reactions, which will

1. Establish a sodium-water reaction vessel configuration that will permit pressure transducers to function reliably.
2. Result in preliminary data on the effects of injection conditions on shock magnitude.
3. Determine whether or not measurable shock waves can be produced by a sodium-water reaction in small-scale equipment.

Phase III — An analytical and experimental investigation of scaling laws to

1. Develop a theory for extrapolating from small to large leaks.
2. Verify theory by larger-scale tests.

Phase IV — An investigation to

1. Develop a method of predicting the effect of sodium temperature and pressure on the sodium-water reaction.
2. Develop a method for predicting the effect of water temperature and pressure on the sodium-water reaction.
3. Develop data required for predicting the effects of size and/or geometry on the sodium-water reaction.

Task 2, the dynamic stress analysis of shell and internal parts and verification, includes development of a method for predicting the motion of the steam generator shell under the transient pressure loadings defined in Task 1. This will be accomplished by

1. Applying the available methods in the literature for elastic and plastic analysis of fast loading of structural parts to a steam generator undergoing a large sodium-water reaction. Asymmetrical loadings at various locations on the shell and on internal parts, such as tubes, tube supports, shell liners, etc., will be analyzed to determine the degree of deformation of the respective components.

2. Confirming the analytical methods developed in part 1 using data obtained from experiments utilizing explosives and solid propellants to introduce controlled quantities of energy. Pressure-time and volume-time characteristics will be in the range of those expected for large sodium-water reactions.

In Task 3, the mechanical properties of interest for candidate materials under fast loading will be obtained at three temperatures representative of those existing in the steam generator. The experimental program will

1. Determine strain-hardening effects.
2. Determine whether materials behave in a brittle manner.
3. Determine Young's modulus for Croloy 2-1/4.
4. Determine ultimate tensile strength and change in yield point.

3.2.1.3. Post-Accident Component Recovery

Objective — This effort is directed toward the development and demonstration of a procedure for returning the helical-coil steam generators or reheaters to their initial condition of integrity and cleanliness in the shortest possible time after a sodium-water reaction incident.

Summary of Current Technology — The occurrence of a leak and its accompanying sodium-water reaction will necessitate the removal of the steam generator system from service. Assuming that the leak was detected and that the steam generator was isolated and shut down safely, the plant operator still faces the problems of (1) assessing the extent of damage, (2) locating the damaged areas and making repairs with assurance that all significant damage has been corrected, (3) cleaning up reaction products from the whole secondary sodium system,

and (4) returning the system to operation safely in the shortest possible time. Neither expedient methods for determining the extent of damage to a steam generator or reheater nor methods for returning the unit to service have been formulated to date. However, the recovery technique will utilize many commercially proven processes and inspection techniques. The remaining processes and techniques will be developed in this program.

Technical Justification — There are three major incentives for developing an efficient recovery technique for the steam-raising equipment: (1) economic penalties resulting from a loss of generation capacity (\$15,000 to \$50,000 per day for the whole 1000-MWe plant), (2) the high cost of spares, and (3) the necessity for conclusively proving that a repaired unit is as safe or safer than a new unit of identical design.

Scope of Work — The program objectives will be accomplished in four tasks, as follows:

Task 1 — A model sodium system will be designed and built in this task.

Task 2 — Non-destructive testing methods will be developed in this task to locate and determine the extent of damage after the tube bundle has been removed from the system. Testing methods under consideration include such well established methods as ultrasonic and eddy-current techniques. Development will include applicable scanning and traversing patterns. The techniques developed in this phase can conceivably be modified to permit future operational scanning.

Task 3 — The effect of intermediate loop shutdown procedures on the extent of steam generator damage and system cleanup will be investigated in this task, using the model developed in Task 1.

Task 4 — In this task, solvents or combinations of solvents for the products of a sodium-water reaction will be tested for rapidity and completeness of dissolution and for compatibility with materials of construction.

3.2.1.4. Vibration Investigation

Objective — The vibrational characteristics of the B&W helical-coil steam generator will be analytically and experimentally investigated to identify potential problem areas and to suggest design modifications, if needed, to reduce or preclude the possibility of fatigue failures.

Summary of Current Technology — Mathematical analysis and application of theory in the field of vibration have been utilized in both fossil-fired and nuclear steam generators for many years. Computer codes are available to guide preliminary designs of the models used to check out the vibrational characteristics of the unit. Since vibrational characteristics are a strong function of the configuration, test results are used to modify the analog and digital codes for a specific application.

Technical Justification — Several operational heat exchangers, including sodium-heated steam generators, IHXs, and some water-to-water heat exchangers, have suffered tube failures caused by tube vibration and wear or fatigue. One Fermi steam generator suffered such severe wear of the tubes near the sodium inlet nozzles that holes were worn in the tubes and a large sodium-water reaction occurred. Some techniques for analyzing heat exchangers for vibration exist, but they do not give adequate assurance that every location in a steam generator is free of damaging vibration due to any forcing function. A great deal of additional work is needed to bring analytical methods to an acceptable level of accuracy and to realistically evaluate the B&W design.

Scope of Work — An analytical and experimental vibration investigation will be performed initially on a general-type structure and then applied to the particular structural design of the sodium-heated steam generator to ensure its safety and integrity. The study will include the following types of forcing functions:

1. Repeated simple harmonic vibration transmitted to the structure from rotating machinery, such as the pumps, motors, turbines, etc., in the system.

2. Flow-induced vibration caused by the coolant flowing through the tube banks at certain velocities. Two modes of vibration are induced by flow: one is the vortex-excited vibration when the flow is perpendicular to the tube, and the other is a shear type vibration when the flow is parallel to the tube.

3. Mechanical shock loading, which is transient. The response may reach a certain peak and then gradually attenuate with respect to time. However, the peak response may cause the structure to fail and thus cause rupture of tubes in the sodium-heated steam generator.

4. Earthquake loading, which is of a random nature and depends on geographic location. The importance of this investigation can never be overemphasized.

The vibration program will be divided into the following four tasks:

Task 1 — Structural Modeling: A mathematical model of beam-type construction will be established to represent the structural components. Either a lumped mass technique² or the receptance method³ can be used to determine the natural frequencies and corresponding normal modes. This frequency information can be used by the designer to make the necessary modification to ensure the isolation of the system from the possible harmonic vibration source.

Task 2 — Description of Forcing Functions: For simple harmonic vibration, the main concern is the resonance of the system. If a problem exists, it will be solved by modifying the design to shift the natural frequency of the system and thus avoid the resonance. Two sources of flow-induced vibration will be considered: parallel flow and transverse flow. Several investigators have considered parallel-flow-induced vibrations.⁴⁻⁶ Paidoussis also supplemented his analysis with some experimental work.^{7,8} Methods discussed in the cited reference will be extended to handle problems in the B&W steam generator. Several methods are also available for predicting the vibrations induced by transverse flow.⁹⁻¹¹ Some of these techniques apply to high-velocity flow, but modification will extend coverage to the low velocity range as well. Of specific concern in the shock loading of the steam generator is the shock

front wave propagation. In order to solve the dynamic response of structures, it is necessary to specify the forcing function in some mathematical form. Shock wave propagation is described by the amplitude attenuation and energy dissipation method in reference 12. Some shock loadings are specified using shock spectrums;¹³ however, for most mechanical shock loading, we can specify the loading as a function of time. The analysis of structural response can be obtained simply by the transformation technique (either Fourier transformation or Laplace transformation). The description of earthquake loading has been extensively discussed in reference 14. Since the earthquake loading is of a random nature, it would be more appropriate to describe it by a spectrum form. The design based on this loading form would be conservative, since the loading description is the envelope of the maximum peak response.

Task 3 — Dynamic Structural Response and Strength Analysis:

In Tasks 1 and 2, a mathematical model of the structure, as well as a mathematical description of the loading, will be established, and in Task 3 the dynamic response of the structure to the forcing function will be established. For loadings of a stochastic (probability variable) nature or those given in spectra forms, the procedure for coupling the forcing function and the structural characteristics has been outlined in reference 15. For loadings specified as a time function of a sinusoidal nature, the transformation technique will be applied to solve a set of differential equations. The corresponding stress and strain distribution will then be calculated on the basis of the dynamic response of the structure.

Task 4 — Experimental Verification: The details of the experimental verification program cannot be established until the basic formulation of the design model is completed. However, tentative plans are to use the original tube bundle from the B&W 30-MWt prototype sodium-heated steam generator for the experimental studies. In this task, experimental studies will be conducted to verify the analytical procedures and techniques developed under the preceding tasks. The experimental studies will include deflection, stress, strain, and frequency measurements for the various loading conditions discussed previously.

3. 2. 1. 5. Thermal Shock Study

Objective — The objective of this study is to develop analytical methods for estimating the maximum thermal stresses that will occur in the structural members of a B&W steam generator undergoing temperature transients in power maneuvers.

Summary of Current Technology — The current design procedure to determine the maximum thermal stress in a member involves successively solving for the thermal stresses at discrete multiple times. The designer uses his own judgment to determine an initial time during the temperature transient to calculate the thermal stresses. He then calculates the thermal stresses at times before and after this initial guess to obtain a stress gradient. If a continuing decreasing trend or a continuing increasing trend is evident, this process is repeated at some other time in the transient until peaking (indicating a maximum stress) occurs.

Technical Justification — Any equipment in a liquid metal system exposed to the flowing sodium stream can be exposed to very rapid temperature changes. Reactor scrams or the loss of sodium pumps can cause changes of several hundred degrees F in the sodium temperature in a few seconds. These very rapid changes in sodium temperature can cause high thermal gradients and high thermal stresses in thick metal parts. There are techniques for analyzing the temperature profiles throughout a solid at any time during a transient, but establishing the location and time of a "worst case," on which detail stress analyses should be made, requires laborious hand calculations which consume an impractical amount of time. The need for computer programs to perform these analyses becomes evident when time, technical precision, and economics are considered as important factors in the design phases of the program.

Scope of Work — Transient thermal stress analysis work will be divided into three phases, as follows:

Phase I — A preliminary study will concentrate on evaluating existing analysis techniques to obtain program orientation.

Phase II — The most promising analytical techniques found in phase I will be refined and applied to simple geometric shapes as a validity check.

Phase III — Promising techniques developed in phase II will be applied to typical complex geometrical shapes from the B&W steam generator. Conclusions and recommendations will be prepared when this phase is complete.

3.2.1.6. Prototype Program

Objective — The prototype program is being carried out to prove the performance, reliability, and safety of the B&W helical-coil steam generator design.

Summary of Current Technology — The sodium steam generators now operating and the ones that have operated in the past have yielded a great deal of operational data. However, the general consensus is that there is much to be gained from actual operation in sodium of scaled version of a new concept.

The Alco unit now at LMEC is awaiting completion of testing. The B&W prototype will be installed in the SCTI when it is completed. In addition, the prototype of the modular concept will then be tested.

Technical Justification — The information gained from the prototype test program will be used by the designers of the full-size units. The essential design information relative to stress, evaluation, vibration, and performance cannot be obtained by any other means.

Scope of Work — Task 1, the prototype test program, is divided into the eight phases outlined below.

Phase I — Field Tests: A hydrostatic pressure test using water will be conducted on the prototype steam generator to provide strain data to evaluate the computation methods used to estimate the pressure stress.

Phase II — Prototype Shakedown: Shakedown tests with sodium will be performed in this phase to determine whether

the prototype is operational. Before testing, all instrumentation will be calibrated and methods will be established to check the calibration at specified times during the testing at SCTI.

Phase III — Stress Evaluation: Experimental stress data will be obtained at lower-than-design temperatures to verify stress calculations. Lower-than-design temperatures are specified because of the unreliability of strain gage data at temperatures approaching 1000 F.

Phase IV — Design Performance: Data on the prototype steam generator operating at design conditions will be obtained in this phase; these data will be used in the following ways:

1. To verify component temperatures at such locations as the superheater outlet.
2. To determine the functional steady-state performance and the adequacy of design procedures. These data will be valuable in confirming the overall design procedures, but they cannot be used to evaluate design procedures in specific areas, such as nucleate boiling region, film boiling, etc.
3. To check for maldistribution of sodium- or steam-side flow.
4. To determine the pressure and thermal stress levels and the accuracy of temperature profiles in critical stress areas.
5. To determine minimum load for the steam generator.

Phase V — Analog Simulation Transfer Function: The dynamic response characteristics of the prototype steam generator, when it is subjected to stepped variations in operating parameters, will be determined. The results will be subsequently used to modify the analog computer simulation.

Phase VI — Transient Tests: A comprehensive transient testing program, encompassing all parameters of engineering interest, will be conducted in this phase to assemble data on structural reliability and dynamic performance.

Phase VII — Maximum Steam Generation Tests: The performance and stress levels when the prototype is operating at its maximum capability will be ascertained in these tests.

Phase VIII — Reduced Steam Pressure Tests: Design data at reduced steam pressures will be obtained in these tests.

Task 2 includes prototype data analysis and test reporting. Test data will be transmitted to B&W for reduction and analysis as they are generated, so that the results can influence the direction of the program. Since most of the data will be acquired on automatic equipment, computer programs will be written to reduce and analyze them efficiently. A detailed report covering all test work will be written. Topics in this report will include but will not be limited to (1) methods of data reduction, (2) methods of analysis, (3) results of analysis, and (4) the relation of the results to the prototype design.

Task 3 covers destructive analysis of the prototype steam generator. After test work at the SCTI has been completed, the prototype steam generator will furnish valuable information to the designer via destructive analysis. The prototype will be removed from the test area after testing, carefully disassembled, and selected parts cut apart for further detailed analysis. Some preparatory work involving original material and "as-built" dimensions will be done before the unit is tested. Tentative steps in the destructive analysis program are as follows:

1. All spare material, shop records, and test records will be identified and retained in preparation for the destructive tests. "As-built" dimensions will also be verified.
2. The prototype will be removed from the test stand and transported to the inspection site.
3. The upper head and the tube bundle will be removed from the shell.
4. A detailed chemical, metallurgical, and hydrodynamic analysis will be performed on selected parts.
5. A report will be written to describe the results of the destructive analysis.

Budget Estimate —

Manpower	\$2,900,000
Facilities	2,000,000
Equipment	<u>2,000,000</u>
Total	\$6,900,000

Table 3-2 is an itemized schedule and cost estimate summary for this task.

Table 3-2. Summary of Schedule and Estimated Costs
for R&D Work on the Steam Generator

Task	FY CY	Estimated cost, \$ × 10 ³						Total
		1970	1971	1972	1973	1974		
		1969	1970	1971	1972	1973	1974	
Steam Generator								
1. Sodium-water reaction studies		500	800	1000	500	--	--	2800
2. Post-accident component recovery		--	100	200	200	--	--	500
3. Vibration investigation		100	100	100	--	--	--	300
4. Thermal shock study		100	100	--	--	100	--	300
5. Prototype program		<u>1000</u>	<u>500</u>	<u>500</u>	<u>500</u>	<u>500</u>	<u>--</u>	<u>3000</u>
Total		1700	1600	1800	1200	600	--	6900

3. 2. 2. Liquid-Metal to Metal Heat Exchanger

3. 2. 2. 1. Objective

The objective of this program is to develop information to be used in designing the full-size reference plant sodium-to-sodium heat exchanger.²⁷ The information developed will ensure that the resultant design can be fabricated and operated successfully with a minimum of conservatism by reducing the magnitude of the safety factors normally applied.

3. 2. 2. 2. Summary of Current Technology

In general, sodium-to-sodium heat exchangers have now reached a state of development wherein concern for the heat

transfer fluid has diminished, and the designer has increased his attention to the problem areas common to all heat exchangers, i.e., flow distribution, temperature transients, economics, and the like.

The design, construction, and operation of certain heat exchange units are of specific interest to the B&W plant because of their similarity to the reference plant unit. These units are used in the EBR-II, the Enrico Fermi, and the PFR reactors.

The EBR-II heat exchanger is located within a tank of sodium, and thus it represents environmental problems that are similar to those of the reference plant. The others have forced circulation similar to the reference plant in that they utilize gravity-flow on the reactor coolant side.

The PFR, which is now under construction, is very similar to the reference plant in that the heat exchangers are located in the sodium and operate under a gravity head on the primary side. The performance characteristics for EBR-II were very close to the predicted values and otherwise offered satisfactory operation.

The Fermi heat exchangers did not fare as well. The predicted values of UA for these heat exchangers were almost twice the measured values.¹⁶ However, the thermal duty for all three heat exchangers was off by only 12% of the predicted values, because the operating LMTD was 112 F instead of the predicted 80 F.

It appears that the most probable cause of underperformance is maldistribution of the sodium on the shell side. Another very probable cause is overestimating the effective heat transfer area. Undoubtedly, other causes that acted in concert to reduce performance were fouling, flow bypassing, non-wetting, and cover gas entrainment.

Of equal interest to the thermal performance is the fact that in over 30,000 hours of operation on the primary side, these units have proved the adequacy of their structural integrity and the effectiveness of the tubesheet seals and the leak detection system.

Construction of the PFR units is proceeding, and a limited amount of design information is available.¹⁷ Heat transfer coefficients are given for shell-side parallel flow for triangular and square pitched tubes as a function of the Peclet number, tube pitch, and tube diameter.

On-line heat transfer coefficients on the shell sides of tube bundles are also given by O. E. Dwyer.¹⁸ These are derived from a semi-empirical relationship. The comparison of the two methods of calculating the heat transfer coefficients is somewhat different. Dwyer's correlation is supported by others.¹⁹

Pressure drop calculations are more of a problem than heat transfer, especially in the areas of cross flow. The accuracy of pressure drop calculations is especially significant in gravity flow IHX's, since the geometric configuration of the system is fixed, and an error here could cause a significant drop in plant performance. Pressure drop analysis is also important in that it is related to flow distribution in the shell side. Generally, flow distribution devices tend to consume the available driving head and, consequently, to reduce the flow. More effort is required to develop better correlations for the low-pressure-drop systems.

The greatest concern in the area of fabrication lies with the tube-to-tube sheet weld. The present method of attaching heat exchanger tubes to tubesheets is to expand the tube into the tube-sheet hole using an internally applied roller. This results in a residual compressive stress in the tube wall, which holds it against the tube-sheet.²⁰ To minimize relaxation of the residual stress at elevated temperatures and to ensure a no-leakage seal, the junction is welded on the front side of the tubesheet. A gas test is used to test the welds for leakage.

The disadvantages of this type of weld are (1) the lack of complete penetration and the resulting high stress concentration at the junction, (2) the inability to perform NDT to check the depth of penetration and the completeness of the weld, and (3) the potential problem area caused by a deep crevice behind the weld.

One way to improve the reliability of the tube-to-tubesheet junction is to make a full-penetration, totally inspectable weld between the tube and tubesheet. This has been done on the B&W prototype steam generator by machining a weld preparation on the back side of the tubesheet and then welding the tube to it.²¹ A small x-ray source was designed to be placed axially along the inside of the tube;

film is wrapped around the outer diameter of the tube and a 360-degree picture of the weld is made. The main disadvantages of this type of weld are as follows:

1. Cost — welds must be fabricated and inspected individually.
2. Shrinkage — weld shrinkage makes its use on straight-tube bundles difficult.
3. Maintenance — it is necessary to remove the outer tubes to reach one at the center.

A new technique which has been quite successful is to expand and weld the juncture and then back-braze the crevice to obtain a strong bond along the tubesheet thickness.²⁰ This requires a braze material that is compatible with the requirements of a sodium system and a procedure for applying the braze to many joints simultaneously.

Another approach is to explosive-weld the tube into the tubesheet. The result is a strong metallurgical bond between the two. This method also could be used for plugging tubes when desired. However, additional development and testing are required before this method is advanced to the production stage.

3.2.2.3. Technical Justification

Because of the very complex interrelationship between the configurations of the IHX and the reactor vessel and the cover structure, it is necessary to know precisely the characteristics that influence the design of the heat exchanger. For example, the diameter of the heat exchangers directly influences the diameter of the vessel, the cover structure, and the sodium inventory; the height of the tube bundle influences the height of the reactor vessel, not only because of the height itself, but because of the difference in the elevation of the hot and cold leg required as a driving head. The height of the reactor vessel thus influences the sodium inventory. Increased height also increases the height of the reactor building required to clear the tube bundle for installation and removal.

From the paragraph above it can be seen that, in order to optimize the entire reactor system to promote the most economical design, certain information on performance and operation is required. Of equal concern we must be assured that, after the system has

been designed and constructed, it meets the design conditions. As stated earlier, the flow through the heat exchanger depends on a fixed geometry, and therefore, it requires a precision much greater than a heat exchanger whose driving head is supplied by a pump.

Research and development is also essential to ensure the successful design of a unit that is so large that a full-size prototype cannot be built and tested (as with the reactor coolant pumps) before installation in the plant.

3.2.2.4. Scope of Work

Task 1 — Review of Design Requirements: B&W's IHX design (relative to the performance, environment, reliability/availability, maintenance, operational safety, standards, and interfacial requirements) will be reviewed to determine compatibility with materials and state-of-the-art design techniques as well as probable future developments. In addition, a literature survey will be conducted, and the thermal and hydraulic designs of similar heat exchangers will be compared with the proposed reference design. If more attractive design features are uncovered in the review, they will be investigated to determine their effect on the plant as a whole.

Task 2 — Analytical Methods Development: The objective of this task is to develop improved analytical methods that are applicable to IHX design to supplement present techniques^{22, 23} for analyzing pressure losses, vibrational characteristics, and thermally induced stresses. Of paramount importance is the development of improved analytical tools that will initially guide the hardware program toward meaningful progress. In addition to initial guidance, the analytical tools must provide semi-empirical correlations with a confidence factor that is adequate for the design of large units. These methods will also be used in providing error analysis information.

Task 3 — Prototype Design: The objective here is to design a reduced-scale IHX (1) which accurately simulates the thermal, hydraulic, and structural characteristics of the B&W 1000-MWe reference design, (2) which is adequately instrumented so that valid engineering data are produced for all parameters of interest, and (3) which

is economically compatible with the national testing facilities located at the LMEC.

The thermal, hydraulic, and structural problems of an IHX can be simulated with reasonable accuracy with a scale model using present engineering techniques. The key to success in this area lies in focusing attention on problem simulation rather than in trying to optimize the design of the prototype. Experience on a number of IHXs has shown that thermal performance can be predicted reasonably well, except for the Fermi unit. Because failure to obtain the predicted IHX performance in a 1000-MWe plant could have serious economic repercussions, prototype testing is recommended. As a byproduct of the tests, a great deal of information concerning installation, removal, and decontamination of the selected design will be assembled, and the results can be factored into the design of the full-scale unit.

A reduced-scale IHX will be designed on the basis of what is thought to be required to ensure the generation of valid design information. Included in the design will be mocked shielding, gas seals, insulation as required, and the like.

Since the reference plant contains six IHXs, and since each one is supplied from a common source, it can be seen that the cold leg can be maintained at a constant temperature as long as each heat exchanger is operating properly. In the event of the loss of intermediate flow, the reactor coolant would continue to flow through the shell side, causing a rise in temperature in the cold leg. Under these conditions, the desire to operate the plant at partial load must be accompanied by shutting off the high-temperature sodium. Therefore, it will be necessary to incorporate into the prototype a reliable device for accomplishing this objective. Because this device will be in the primary flow circuit, it will be necessary to investigate the pressure losses and flow symmetry. Most likely the model size will be compatible with the LCTL at LMEC. The design of the prototypal model will be based on state-of-the-art engineering data and the design requirements enumerated in a previous task.

Task 4 — Prototype Fabrication and Test: The scope of work covers the fabrication of the prototypal IHX discussed in Task 3 and experimental determination of its thermal, hydraulic,

structural, and operational performance. Facilities are available at the LMEC to test prototypal IHXs. Test procedures for B&W's IHX will follow generally accepted methods developed in test programs for other similar equipment.

A detailed test procedure will be written for the prototype. Information on all four inlet and outlet sodium temperatures and flows will be obtained to determine performance as well as the feasibility of obtaining meaningful outlet temperatures on the shell side for control purposes. Axial and radial temperature distributions will be obtained both inside and outside of the tubes to confirm heat transfer coefficients and to determine whether stagnation exists.

The test program will include measuring ΔP and ΔT across the tubesheets to determine the existing stresses that will ultimately yield the operational limits as well for the basis for design correlations.

If the design calls for a gas space below the upper tubesheet, then the effects of level changes and gas entrainment will be measured as functions of flow.

The test program will be written so that it obtains information at both transient and steady-state conditions. These data will then be analyzed.

In general, the basic information required to develop temperature, flow, pressure, pressure drop, level, and leak detection instrumentation will be developed under section 4 of the R&D plan. Nevertheless, an effort must be expended to design, fabricate, and test the high-temperature experimental instruments required to obtain information from the prototype. Refinements in test information will be needed to ensure accurate data. In particular, improvements will be needed in strain gages and accelerometers for use in a high-temperature sodium environment. It will be a major effort to customize the sensing instruments to be compatible with the intents and purposes for which the prototype was designed. Further, the design of the prototype must accommodate these sensing instruments, and therefore, it must receive input from an instrument test effort.

The instruments will be designed to respond to the transients that are expected in the prototype as well as to steady-state conditions.

Maintenance and handling procedures will be demonstrated with the prototype and will yield useful design information, even though the sodium is not radioactive. During this phase of the program, the handling equipment will be mocked up as if the sodium was radioactive.

Under this task, inspection techniques will be developed to determine fabrication integrity, tube leaks, and the general conditions of the vital parts after operation. This portion of the program will also produce methods of data reduction and application of the results to design and test modifications.

Task 5 — Model Tests: The objective of the model tests is to validate (1) the shell side hydraulic performance, (2) the tube side bundle inlet performance, and (3) the hydraulic performance of the primary flow-blocking mechanism by simulating the full-scale IHX with the flowing water of a reduced-scale model.

Water flow modeling will also be used to screen candidate flow configurations. Water-model simulation of sodium flow problems is discussed in considerable detail in a recently published LMEC report.²⁴

From an assessment of the reference design heat exchanger, it appears that flow imbalances could exist on both the primary and secondary sides. Experimental verification is required to develop better correlations for the low-pressure-drop systems and to ensure that there will be flow balance.

In order to ensure satisfactory tube side sodium distribution, studies will be performed to develop an optimum design for the lower head. In addition, the development of a primary inlet-flow distributor arrangement and a shutoff device that results in satisfactory flow distribution without excessive pressure drop will also be pursued. Three water models will be required to accomplish these objectives. One of these would simulate the secondary side of the IHX. Simulation will begin at the upper end of the center downcomer and will include a removable lower head. The lower tubesheet will be simulated, followed by a divided

plenum designed to separate the tubesheet into approximately 32 equal area segments on four diameters. Each of these segments will have an outlet tube equipped to measure the flow through the individual segments. In this manner a 32-point, four-diameter map of flow distribution can readily be obtained.

It is expected that a 1/4-scale model would be employed to study primary side distribution. This would be a transparent model using plastic rods to simulate the heat exchanger tubes. Data would comprise visual observations and sufficient measurements to map flow distribution at selected planes. A 1/4-scale model would also be used to study shutoff devices. Scaling will include both geometric and kinematic similarity comparisons. The former require that the dimensional ratios be similar for all portions of the shape involved. Similarly, kinematic ratios must be equal and the same as the geometric ratio. This is accomplished by the use of dimensionless parameters derived from a comparison of the forces acting on the fluid.²⁵ If equivalence of relationships such as the Euler, Reynolds, and Froude numbers is attained, then the model is equivalent to the full size component.

Task 6 — Tube-to-Tubesheet Attachment: The objective is to develop a superior tube-to-tubesheet bond by exploring two advanced techniques: explosive welding and back brazing. Tube-to-tubesheet attachment is the most important design and fabrication problem encountered in the construction of an IHX for LMFBR service for these reasons:

1. Thermally induced stress concentrations are the highest at this junction.
2. The large number of joints increases the statistical probability of having defects.
3. The effects of tube vibration are more pronounced in this area.
4. The unit must be removed to repair junction failures.
5. Contamination of the intermediate system cannot be permitted for safety reasons.
6. Gross leakage of intermediate sodium into the reactor vessel would present a serious radioactive sodium disposal problem.

The reliability of the IHX can be improved by minimizing the crevice behind the tube-to-tubesheet junction. Three ways to accomplish this have been mentioned previously: back-side welding, back brazing, and explosive welding. The latter two would require the following development before they are put to commercial use:

1. Back Brazing — A braze material must be developed which is compatible with liquid sodium and is not adversely affected by mass transfer; fabrication procedures must also be established for this process.

2. Explosive Welding — A means must be developed to ensure weld integrity by distinguishing between a mechanical and a metallurgical bond.

All three methods of fabrication will then be fully tested under operating conditions and compared to determine which approach best satisfies the LMFBR program objectives. Small sodium loops will be satisfactory for this purpose.

Task 7 — Tube Vibrations: Although the sodium velocities through the shell side are expected to be quite low because of gravity flow, and since cross flow is expected to occur only at the entrance and exit of the shell, an investigation must be completed to ensure that (1) the cross-flow Reynolds number is below the transition value,²⁶ and (2) the natural frequency of the tubes is greater than the maximum frequency of the vonKarman vortices. If it is found that problems exist because of these factors, then a full-scale model of a portion of the tube bank will be designed and constructed to simulate actual conditions. Model testing will ensure that previous engineering is correct and that self-excited vibrations do not occur.⁴ If results are unsatisfactory, corrective action will be taken.

Budget Estimate —

Manpower	\$1,200,000
Facilities	900,000
Equipment	<u>300,000</u>
Total	\$2,400,000

Table 3-3 is a summary of estimated costs and schedule for work in this area.

Table 3-3. Summary of Schedule and Estimated Costs for R&D Work on Liquid-Metal to Metal Heat Exchangers

Task	FY CY	Estimated cost, \$ × 10 ³								Total
		1970	1971	1972	1973	1974	1975	1976		
		1969	1970	1971	1972	1973	1974	1975	1976	
Liquid-Metal to Metal Heat Exchangers										
1. Review of design requirements		50	--	--	--	--	--	--	--	50
2. Analytical methods development		30	30	30	30	30	--	--	--	150
3. Prototype design		--	200	200	100	--	--	--	--	500
4. Prototype fabrication and test		--	--	--	300	500	100	100	--	1000
5. Model tests		50	200	140	30	30	--	--	--	450
6. Tube-to-tubesheet attachment		50	100	50	--	--	--	--	--	200
7. Tube vibrations		<u>50</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>50</u>
Total		230	530	420	460	560	100	100	--	2400

3.2.3. Sodium Valves and Piping

3.2.3.1. Sodium Valves

Objectives — To develop reliable 14- and 20-inch sodium flow control valves capable of varying the flow rate from 20 to 100% and to develop reliable, very-low-leakage, sodium blocking valves of 26-, 28-, and 32-inch diameters for use in the intermediate coolant loops of the B&W 1000-MWe LMFBR power plant.

Summary of Current Technology — Experience with large sodium valves is virtually nonexistent. On the other hand, most of the experience with small- and medium-sized valves has been traumatic (see the LMFBR Program Plan for a discussion of sodium valve experience). The Allis-Chalmers Company, under contract to the AEC, reviewed LMFBR valve requirements and evaluated state-of-the-art designs for this service. They concluded that common designs

would not be acceptable and that advanced designs would be needed. B&W is generally in agreement with the conclusions and recommendations of Allis-Chalmers; however, the choice of a pot reactor concept and the particular intermediate loop configuration somewhat mitigates the valve development problem for the B&W reference design.

Technical Justification — The development of large, reliable isolation valves is mandatory from a safety standpoint because of the possibility of compromising reactor building integrity via the intermediate loop pipework. This type of valve will also be used to isolate systems for maintenance and to limit the consequence of intermediate pipework failure.

Because the steam generator and reheater will be operated in parallel, a throttling valve will be needed to proportion the sodium flow to the reheater for all turbine loads. An additional throttling valve will be located in a bypass line to achieve the desired control accuracy and stability. This bypass leg would be needed regardless of whether or not a sodium-heated reheater was included.

Scope of Work — The following valves must be developed for the B&W 1000-MWe LMFBR power plant:

<u>Type of valve</u>	<u>Diameter, in.</u>	<u>Number required in reference plant</u>
Isolation	26	15
	28	6
	32	3
Throttling	20	3
	14	3

The sizes included in each group are within a range that will not require the development of a separate valve for each size.

Budget Estimate —

Manpower	\$ 700,000
Facilities	800,000
Equipment	<u>500,000</u>
Total	\$2,000,000

Table 3-4 is a summary of the estimated costs and schedule of work on sodium valves and piping.

Table 3-4. Summary of Schedule and Estimated Costs for R&D Work on Sodium Valves and Piping

		Estimated cost, \$ × 10 ³								
	FY	1970	1971	1972	1973	1974	1975	1976		
Task	CY	1969	1970	1971	1972	1973	1974	1975	1976	Total
Sodium Valves and Piping										
1. Sodium valves		100	200	500	500	500	100	100	--	2000
2. Sodium piping		250	500	725	325	100	--	--	--	1900
Total		350	700	1225	825	600	100	100	--	3900

3.2.3.2. Sodium Piping

Objectives — The main objectives of this program are as follows:

1. To develop the analytical methods required to design sodium piping, mixing sections, and slip joints.
2. To verify these methods experimentally as required.
3. To develop the technology required for fabrication of these components.

Summary of Current Technology — There has been relatively little experience with large-diameter thin-walled pipe operating in a sodium environment. Consequently, much work must be done before the engineer can accurately predict the behavior of the reference design piping system. Only one sodium plant (Fermi) has used piping comparable to that now shown in the reference design. That plant used a section of pipe 30 inches in diameter with a wall 3/8-inch thick. The designers of the Fermi plant prepared their own requirements and standards, because the limits of operation were beyond the scope of the present-day codes.

The mechanism of failure for large-diameter, thin-walled pipe in the sodium environment needs investigation; present piping codes may not be adequate to define the mode of failure. One of the more important problems that faces the piping designer is the instability of large-diameter, thin-walled piping. The piping in the B&W 1000-MWe design ranges from 14 to 36 inches in diameter and has a reference wall thickness of 3/8 inch.

Many varied piping flexibility programs exist today, and the means are available to analyze almost any large, complex, high-pressure, high-temperature, thick-walled piping system. These programs must be modified for large-diameter, thin-walled sodium piping. The same analysis regarding flexibility, moments, stresses, reactions, and the like may still apply. The difference is in the additional considerations that arise from sodium as a fluid and the high-temperature, low-pressure, 36-inch-diameter, 3/8-inch-thick pipe being investigated.

It is clear that R&D is required to develop the additional analytical tools that are needed to ensure confidence in the design of sodium piping systems for large power plants like B&W's 1000-MWe reference plant.

Mixing tees with diameter ratios of 3.0 or more are quite common in steam and high-temperature pressurized water systems. The methods for the design and analysis of these mixing chambers are also well-known. Chambers with diameter ratios of less than 3.0 are not as common, and the methods of design and analysis are not as refined as they are for the more conventional mixing tees. Furthermore, the temperature differences now being accommodated are considerably less than those for the B&W reference LMFBR plant, which has some mixing tees with a diameter ratio of about 1.25 subjected to a temperature difference of 173 F and others with a diameter ratio of about 2.6 subjected to a temperature difference of 300 F. The present state of the art for liquid-metal mixing tees cannot handle these design problems. There are two reasons for this:

1. Large mixing tees, which promote good turbulent mixing of sodium, have never been fabricated.
2. Present designs create large thermal stress problems because of stratification of the high-temperature sodium caused by improper mixing.

Further development is required to advance the technology of mixing tee design for small diameter ratios and large temperature differences.

The design of slip joints is not a new art, but some problems exist in the B&W plant that affect the design of the slip joints in the reactor coolant piping and require R&D to prove out the design. The areas in question are the required leak-tightness of the joint in an elevated-temperature sodium environment and the moment and shear carrying ability of the slip joint design. In addition, the maintenance and inspection requirements need to be determined.

The state of the art for economic fabrication technology for piping of this size, as applied to the power plant, will need investigation. Many methods of fabrication may be feasible for construction of the reference design piping system. The merits and practicality of cold springing, extrusions, welded rolled plates, centrifugal casting, explosive forming, and ribbed pipe should be evaluated. The most economical or practical way to manufacture this large-diameter, thin-walled pipe has not been clearly defined within the industry.

Some research is needed to advance the state of the art in welding technology. Tests should be run on typical sections of piping to determine methods for controlling weld shrinkage and subsequent distortion. In general, the state of the art in fabrication of piping of this kind is not well-developed. However, the methods may be available from a combination of techniques used in various industries.

Scope of Work — The reference plant piping is predicated on using type 304 stainless steel with slip joints to accommodate thermal expansion in the reactor coolant system. The intermediate coolant system uses Croloy 2-1/4 piping and expansion loops to account for the thermal growth of the piping. In addition to thermal expansion, the intermediate system piping is required to mix sodium streams of different temperatures without deleterious structural effects. The basis for the selection of stainless steel piping in the reactor coolant system was to maintain a preponderance of a single material in contact with sodium. The basis for the selection of Croloy 2-1/4 for the intermediate piping was to reduce costs. Reduced costs were achieved by selecting a material that not only costs less, but which also has a much lower

coefficient of thermal expansion, which, in turn, reduced the length of the expansion loops, the size of the buildings, and the sodium inventory.

The scope of work required to develop the information necessary to pursue the reference plant piping design includes the following items:

Task 1: An economic evaluation will be required to determine whether Croloy or stainless steel is the more satisfactory material for the intermediate piping. This study must compare intermediate piping designs of Croloy 2-1/4 and type 304 stainless steel, using the data acquired in this program as outlined in task 3 below. Only then can a meaningful economic comparison be made.

Task 2: Known failure theories, such as maximum principle stress, maximum shear stress, and strain energy, will be reviewed to determine their applicability to the unique problems of designing large-diameter, thin-walled piping. The reviews will consider the effects of instability (such as elastic buckling, local plastic instabilities, creep buckling, and creep rupture) when determining the failure theory applicability. The recommended failure theory criteria and its accompanying mode of failure will be verified experimentally.

Experimental investigations will consist of subjecting sections of large D/t piping to the expected loading conditions to produce the recommended failure mode. As testing proceeds, strains and deflections will be recorded until failure occurs. The results will be correlated with the recommended failure theory and modifications will be made where necessary.

Task 3: Analytical studies aimed at determining the static and dynamic behavior of LMFBR piping systems will be conducted. These studies would produce piping flexibility programs that would consider elastic-plastic design, creep, mechanical and thermal fatigue, instability, strain concentration, dead weight and thermal expansion loading, vibration and seismic loading, local loads, corrosion, and erosion.

To provide input for these programs, material properties tests are required to obtain long- and short-term axial and bending creep and creep rupture data; cyclic fatigue data; tensile data; and

strain-hardening information. These tests will be performed under the Fuels and Materials Program. Corrosion and mass transfer information will be obtained from the Sodium Technology Program.

Task 4: Studies will be carried out to determine whether scale-model testing can satisfactorily predict the behavior of full-sized piping systems. If the results of the studies are favorable, the dimensionless parameters required to adequately mock up the piping systems must be determined. Using these scaling techniques, pipes and pipe fittings will be subjected to various loads and combinations of loads to determine applicable stress distributions and fatigue limits. In addition, the flexibility and stress intensification factors for pipe fittings will be determined in these tests. Of particular interest in this regard is the primary system piping from the pump to the core inlet plenum. The piping run has very few bends and is connected at one end to a rigid structure and at the other end to a controlled-leakage slip joint. The slip joint must be tested to determine its load and moment carrying ability. The results of these experiments will provide input to, and confidence in, the flexibility programs derived in task 2 above.

Task 5: Develop a mixing tee design capable of withstanding a temperature difference of 300 F between the mixed fluids. A program consisting of the following phases is recommended:

1. An analysis to determine the magnitude of the problem and possible solutions. This would tie in with the cost benefit studies (3.3.3) as well as with the results of tasks 1 and 3 above.
2. A water model simulation phase to develop mixing tee configurations.
3. Proof tests of a reduced-scale model in a sodium loop. The objective of the water model tests would be to develop a configuration that produces a uniform temperature rapidly through turbulent mixing and protects the walls of the tee from cycling temperatures and high reversing stresses. Since this approach would essentially ignore the conduction factor, results should be conservative when applied to sodium. At the present time it appears that a model for this purpose would probably be about 1/6 scale. Instrumentation would consist of sensitive

thermocouples to measure inlet conditions and to measure fluid temperature distribution at one or more downstream planes. The apparatus would also be equipped with flow meters and controls on each inlet line as well as means for controlling relative stream temperatures.

Budget Estimate —

Manpower	\$1,000,000
Facilities	300,000
Equipment	<u>600,000</u>
Total	\$1,900,000

Table 3-4 is a summary of the estimated costs and schedule of work on sodium piping and valves.

3.2.4. Large Sodium Pump Development

3.2.4.1. Objective

The objective of this effort is to develop an efficient and reliable variable-capacity pump (41 to 100% of full flow) capable of delivering up to 37,000 gpm of 800 F sodium against a 121-foot head, while operating in the radioactive environment of the reactor coolant system of B&W's 1000-MWe LMFBR. The pump should reflect practical and economical engineering solutions to such problems as maintenance, handling, and radiological safety hazards.

3.2.4.2. Summary of Current Technology

The successful operation of a number of mechanical liquid-metal pumps has proven the validity of generally accepted hydraulic design techniques, the potential reliability of sodium-lubricated bearings, and the acceptable performance of dynamic seals for reactors operating with unvented fuel. A more complete discussion of sodium pump experience can be found in section 3 of the LMFBR Program Plan.

The advent of the vented fuel concept has greatly increased the problem of providing adequate dynamic seals for the primary pump shafts. Compounding this problem is the fact that seals must be located so that the highly radioactive blanket gas cannot administer harmful radiation doses to the occupants of the reactor building. Briefly speaking, acceptable solutions to the problems of seal location, performance, lifetime, and replacement for LMFBRs with vented fuel have not

been found. However, there is reasonable assurance of success in this area if the proper research and development effort is applied.

Experience with large hot-charge pumps (4000 to 7000 hp) in the petroleum industry has shown that thermal pump problems cannot be reliably predicted. Thermal problems are not the same on identical units; i.e., the cure for thermal expansion troubles on one pump will not guarantee problem-free operation on a second, apparently identical, unit. These facts clearly indicate a definite need for proof testing every large sodium pump destined for reactor use.

3.2.4.3. Scope of Work

The scope of the LMFBR pump development program depends a great deal on the basic reactor concept. B&W has selected the pool configuration and proposes the following program to make efficient, reliable, compatible, and economical primary pumps available on a timely basis.

Task 1: A preliminary parametric pump analysis will be conducted to define the range of interest for bearing and seal development.

Task 2: Seals and bearings will be developed early in the program, because the degree of success in these areas will greatly influence the remainder of the development program.

Task 3: A pump design program with more emphasis on interfacial requirements will be initiated. The program will include consideration of the entire life cycle of the reactor coolant and intermediate coolant pumps for a pool reactor concept. Attention will be addressed to, but not limited to, the following areas:

1. On-site versus off-site initial pump cleaning.
2. Pump transportation and installation during plant erection.
3. Initial "dry" testing of pump and support equipment.
4. The role of the pumps in plant preheating.
5. Wet pump testing prior to nuclear operation.
6. Post-irradiation primary pump removal.

7. Handling equipment for contaminated reactor coolant pumps.
8. Intermediate pump removal and support equipment.
9. Post-irradiation primary pump decontamination.
10. Reactor coolant pump decontamination facility.
11. Post-irradiation pump repair.
12. Post-irradiation pump installation.
13. Post-irradiation faulty pump disposal.
14. Pump driver type and configuration.
15. The possibility of sharing support equipment with other components, i. e. , maintenance interface.

Study depth should be sufficient to permit the following:

1. Selection of a pump design with due regard for all the factors that influence plant safety and economics.
2. Identification of any problem areas requiring additional development that would affect the feasibility of a particular reactor design.
3. Realistic development of design criteria, quality assurance programs, and pump standards.

The findings of the 1000-MWe LMFBR pump design study will be compared with the design requirements for the FFTF and LMFBR demonstrator plants. It may be possible to eliminate any development testing on pumps for 1000-MWe plants if enough similarity exists between the systems and if a reasonable degree of success is achieved on the first units. This does not mean, however, that the need for proof testing of pumps for 1000-MWe service can be eliminated.

Task 4: If development testing is required to ensure the availability of successful pumps for the 1000-MWe plant, then a program will be initiated on the basis of the data and designs generated by the joint effort of the reactor system designer and the pump designer.

3.2.4.4. Budget Estimate

Manpower	\$1,500,000
Facilities	500,000
Equipment	<u>6,000,000</u>
Total	\$8,000,000

Table 3-5 is a summary of the estimated costs and schedule of work on large sodium pump development.

Table 3-5. Summary of Schedule and Estimated Costs
for Large Sodium Pump Development

Task	FY CY	Estimated cost, \$ × 10 ³							Total
		<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>		
		<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>	
Large Sodium Pump Development									
1. Preliminary parametric pump analysis		100	--	--	--	--	--	--	100
2. Seals and bearings		200	500	500	300	200	--	--	1700
3. Pump design		300	300	300	300	--	--	--	1200
4. Pump fabrication and development testing		--	--	--	1500	2000	1500	--	5000
Total		600	800	800	2100	2200	1500	--	8000

3.2.5. Core Components Handling Equipment

3.2.5.1. General

Objective — The general purpose of this effort is to provide the analytical and experimental information needed to accomplish the design, assembly, operation, and maintenance of the 1000-MWe LMFBR core components handling equipment. In general, this will involve design and/or experimental investigations to determine how the equipment is affected by the following:

1. The high radiation level and toxicity of the vented fuel.
2. The radioactivity and chemical activity of the sodium.
3. The radioactivity of the cover gas.

4. The relatively high operating temperatures.
5. The decay heat rate of the fuel.
6. The inability to make visual observations.
7. The close spacing of the fuel assemblies in the core.
8. The criticality hazard.

These major pieces of equipment are involved:

1. Rotating shield plugs, bearings, seals, drives, and locking devices.
2. Reactor vessel access plugs.
3. Refueling mechanism.
4. Rotating fuel storage drums.
5. Refueling port plug.
6. Fuel transfer machine.
7. Transfer tubes.
8. New fuel cell apparatus.
9. Spent fuel cell apparatus.

The operations and interrelationships to be developed for this equipment are shown schematically in Figure 3-1.

The development program will have to be carried out in a manner that will demonstrate that, in addition to moving new and spent fuel and radial blanket assemblies to all their required positions in the reactor complex (see Figure 3-1), the core components handling equipment is adaptable for handling other reactor components, such as control rods, radial neutron shields, and similar replaceable components. The specific purpose of the program is to identify and verify the accuracy and applicability of design information that can be used with confidence in designing mechanical equipment to meet the exacting requirements placed on it by a 1000-MWe LMFBR.

Summary of Current Technology — Considerable experience with core components handling equipment in sodium-cooled reactors has been gained in operating the SRE, Hallam, Enrico Fermi, EBR-II, and several foreign reactors, e.g., Dounreay and Rapsodie. The technology and experience compiled will be valuable background for designing systems for the LMFBR plants, as will the experience of the

Figure 3-1. LMFBR Fuel Handling System,
Flow Diagram

water reactor fuel handling systems and near-term plants such as FFTF. None of these systems, however, appears to fulfill all of the LMFBR requirements without considerable adaptation and (therefore) development. In many cases, existing systems have apparently circumvented problem areas by resorting to complex and sometimes makeshift designs and procedures. To arrive at the optimum LMFBR system, detailed design must combine the best features of existing technology into the safe, reliable, and economical configuration that fulfills all the functional requirements. The design effort must be complemented by a thoroughly planned and executed R&D program.

Some component parts of the B&W reference handling system fall within existing technology; the design of the new and spent fuel cells is probably a good example. These are essentially "hot cells," and their design is based on the direct use of technology established in the numerous hot cell facilities throughout the country. Much of the equipment required in these cells may also be provided directly or with very little modification. The internal shipping cask container top-cutting tools and welding tools probably are in this class. The use of these tools will complicate inspection procedures and quality assurance requirements and presents new problems. To prove the safety and reliability and to ensure economical operation of all the core handling systems, considerable design, modeling, and testing are mandatory.

The Program Plan covers in adequate detail the existing technology for many of the sodium-cooled plants.²⁸ This plan also recognizes that the LMFBR systems will require new equipment, new combinations of components, and new systems. The shipment of spent fuel assemblies in a sodium-filled cask is of special importance.^{29,30} This puts new requirements on the core components handling systems. Economy and safety are the driving forces behind the outlined programs.

Scope of Work — Before the major mechanical components of the fuel handling system, such as the gas-cooled fuel transfer machine and the under-the-plug refueling mechanism, can be designed with any degree of confidence, it is considered mandatory to verify the accuracy and applicability of the analytical design calculations

to be used with experimental data. The major areas of concern are the gas cooling capabilities of the fuel transfer machine, the operating characteristics of the fuel assembly grippers and their actuators, the seals and bearings of the rotating plugs, the fuel transfer machine/reactor vessel interface, and in general, the operation of mechanisms in sodium, sodium vapor, and dry inert gas.

The design and fabrication work required to attain the objectives of this program consists of designing and building suitable models of the following components:

1. Fuel transfer machine (excluding shielding).
2. Refueling mechanism.
3. Reactor access port plugs.
4. Rotating fuel storage drum (partial model).
5. Refueling port plug.
6. Fuel assembly.
7. Rotating plug bearings, seals, drives, and locking devices.

These models must be designed and constructed to exhibit operating characteristics comparable to those of actual operating equipment.

The models can be tested at LMEC facilities or at B&W's Alliance Research Center (ARC). In addition to the equipment available at these facilities, the design and fabrication of an environmental test vessel or the modification of an existing test vessel will be necessary. The vessel must be capable of retaining sodium and a cover gas at temperatures and pressures equivalent to those expected in the reactor vessel during normal and abnormal operation. It must be adequately instrumented and large enough to allow meaningful and comprehensive testing of the models using air, water, inert gas, or sodium. The tests will be performed in the following order:

1. Air Tests — These tests can take place either in the fabrication shop or in the test vessel, whichever is appropriate; they will serve as an initial check on the operability of the mechanisms and the validity of their design.

2. Water Tests — These tests can take place in the test vessel; however, a transparent tank or a tank with viewing ports would be

better, since it would permit direct observation of the operating mechanisms and would allow the direction of convective cooling currents through a mockup fuel assembly to be determined using dye or some other traceable ingredient without resorting to periscopes and mirrors.

3. Inert-Gas Tests — The inert-gas tests will be carried out in the test vessel. In addition to determining the operating characteristics of the mechanism in an inert atmosphere at temperatures equivalent to those expected during actual operation, the leakage rate across the gas seals associated with these mechanisms can be obtained.

4. Sodium Tests — All sodium tests will be carried out in the test vessel and will be performed last to minimize sodium cleanup efforts.

In conjunction with the individual R&D programs that follow, a study will be conducted to determine and clearly define all the required functions and functional environments of the core components handling equipment. In addition, the overall performance of the assembled models will be monitored while the individual R&D programs are being conducted.

3.2.5.2. Gas Cooling

Objective — The objective of this program is to demonstrate the feasibility of using gas to remove the decay heat from spent fuel assemblies discharged from the 1000-MWe LMFBR. The problems of cooling an assembly during handling cannot be completely evaluated until the design of the fuel assembly and its handling equipment have been established. The design of the cooling system appears to be a matter of determining its required capacity and physical configuration, except for the question of the effects of sodium deposition on the electrical and mechanical parts. This may be a serious problem, since sodium-rich gas from a fuel assembly may be piped all through the system with subsequent condensation of the sodium on the cooler parts.

In addition to determining the degree of the problem, the need for, and the methods of preventing sodium deposition on critical system parts, this program will provide the information needed to determine the flow and pressure drop required to (1) adequately

cool a spent fuel assembly using gas coolant and (2) to determine the methods of providing the required flow while a fuel assembly is in transit between holding stations.

Summary of Current Technology — The major problem areas associated with gas-cooling the fuel assemblies in existing systems for sodium-cooled reactors are as follows:

1. Cleaning and maintenance have been extremely difficult.
2. Due to inleakage of air and subsequent reactions with sodium, sodium oxide has accumulated in critical areas.
3. Sodium vapor and entrained sodium in cooling gas streams have condensed on handling machine internals and resulted in clogging and jamming.
4. Vapor traps and gas purifiers have been inefficient.
5. Safe fuel assembly temperatures could be exceeded if the ability to maintain gas circulation should be lost or if a transfer operation should be interrupted at a sodium-gas transition position.

Solving these problems for the 1000-MWe LMFBR appears to be a matter of design and provision of adequate gas flow; however, there is a need for experimental data on which to base the design and prototype testing to verify the adequacy of the system.

Scope of Work — The environmental test vessel and the models of the fuel transfer machine, the reactor vessel access port plugs, and the fuel assembly will be used during the tests described below. In addition a fuel assembly guide tube will be installed in the test vessel to simulate the guide tube that extends downward from the reactor vessel access port to a point just above the rotating fuel storage drums. Before testing, and with the model port plugs installed, the test vessel will be purged with inert gas and filled to the desired level with liquid sodium, which will be held at a temperature of 800 F during testing. The testing will serve to check the heat transfer, flow, and pressure drop calculations used to determine the required gas flow through the assembly. During this same test, we can obtain data concerning the amount of sodium

carried out of the fuel assembly and deposited inside of the fuel transfer machine by the flowing gas. Using the fuel transfer machine, the fuel assembly will be lowered into the 800 F sodium in the test vessel. The heat transfer and pressure drop data for the fuel assembly will be determined under heat addition conditions after it is removed from the 800 F sodium and placed in the fuel assembly holding and cooling position in the machine. Preliminary calculations show that, after a 70-day decay storage period, the decay heat rate for an LMFBR fuel assembly will be approximately 17 kW and that the fuel cladding temperature can be limited to 500 F with a cooling gas inlet temperature of 100 F and a flow rate of 1341 lb/hr.³¹ Since these values are preliminary, and since there are uncertainties regarding their accuracy, the test equipment will be adapted to give a range of values. During testing, the heat rate will be varied in steps, and for each heat rate established, the cooling gas inlet temperature and flow rate will be adjusted to obtain the desired temperatures in the fuel assembly. After several of these tests have been performed, the internals of the fuel transfer machine, such as the vapor traps, gas purifiers, the grapple tool drive system, and the heat exchanger, will be inspected to determine how much sodium is deposited in or on them.

Technical Justification — The critical fuel handling operation in B&W's reference design may be the cooling of spent fuel assemblies during their transfer from the rotating storage drum to the sodium-filled shipping cask. With a minimum of 70 days' decay storage and a resulting heat generation rate of about 17 kW from the spent assembly, the importance of developing the cooling system cannot be underestimated. Because of the consequences of cooling system failures (the minimum might be fuel meltdown), the capabilities, reliability, and safety aspects must be thoroughly proved in programs specifically designed for this purpose.

Relationship to Current or Planned Programs — The specific requirements for cooling spent fuel assemblies are a function of the fuel assembly design, the reactor exposure, the decay storage period, and the handling system; the B&W design proposes a particular combination of these components. No programs appear to exist that can provide

the information required; however, it is recognized that other reactor designers could propose programs that might solve many or all of the problems.

Major Facilities and/or Equipment Required —

A major facility is required to accommodate the environmental test vessel, the model fuel transfer machine, and other appropriate equipment. This facility will require a sodium system and could be the large component test loop (LCTL) at LMEC, adapted for this purpose, or another comparable facility.

Budget Estimate —

Manpower	\$210,000
Facilities (rental and adaptation)	200,000
Equipment	340,000
Total	<u>\$750,000</u>

Table 3-6 is a summary of estimated costs and a schedule of the work to be done on core components handling equipment.

Table 3-6. Summary of Schedule and Estimated Costs for R&D Work on Core Components Handling Equipment

Task	FY CY	Estimated cost, \$ × 10 ³							Total
		1970	1971	1972	1973	1974	1975		
		1969	1970	1971	1972	1973	1974	1975	
Core Components Handling Equipment									
1. Gas cooling		70	230	225	225	--	--	--	750
2. Mechanisms		--	30	60	92	85	--	--	267
3. Fuel transfer machine/ reactor vessel interface		150	105	20	20	80	30	--	405
4. Transfer tube		--	--	--	65	60	55	--	180
5. Spent fuel cell		145	135	100	155	155	30	--	720
6. New fuel cell		--	110	105	150	115	--	--	480
7. Rotating shield plug bearings, seals, drives, and locking devices		80	100	130	140	--	--	--	450
Total		445	710	640	847	495	115	--	3252

3.2.5.3. Mechanisms

Objective — The objective of this program is to develop fail-safe mechanisms that can be used to remove fuel and blanket assemblies and other components, contaminated with radioactive sodium and fission products, from the reactor vessel environment.

Summary of Current Technology — The need for a development program in this area exists primarily because information concerning the factors that affect long-term reliability cannot be adequately determined from analyses or prior experimental results. Some of the factors are operating temperatures and loads, thermal transients and shock, loads imposed by mechanical shock and vibration, corrosion, wear, self-welding, and radiation intensity.

Scope of Work — The general design goals for the fuel handling system mechanisms program are simplicity, ease of operation, operational dependability, prevention of damage to reactor components, and ease of maintenance. To ensure the attainment of these goals, it will be necessary to test prototype devices at environmental conditions that simulate operational conditions. This requires operating the test mechanisms in liquid sodium, sodium vapor/inert gas, and dry inert gas. Careful attention must be given to materials selection to achieve high reliability for mechanisms subjected to this environment (see section 7).

The major mechanisms to be tested during this program are the fuel assembly grippers (grapple tools) of both the fuel transfer machine and the refueling mechanism. Included in these tests will be the drives, guides, bearings, and seals required for gripper operation. Some of the necessary details which must be investigated are as follows:

1. Optimum operating speeds.
2. Accurate, reliable devices for position and force indication.
3. Wear, especially in areas where sliding motions occur.

4. The binding effects of sodium oxides or other insoluble contaminant compounds.
5. Useful service life.
6. The maximum allowable clearances consistent with function and reliability.
7. Methods of replacement or repair that do not require reactor shutdown.
8. Configurations that can be effectively drained and cleaned of liquid sodium, thereby minimizing corrosion and the formation of sodium oxide.

The information gained from this program should be applicable to the design of mechanisms located at the other fuel handling stations within the reactor complex, since those mechanisms will operate in an environment that is less severe than that considered by this program.

Technical Justification — The problems associated with the development of LMFBR mechanisms are many and varied. They relate to performance, reliability, safety, maintenance, and economics. Solving these problems is dependent on detailed design, which must incorporate current technology and experience to the highest possible degree. Defining all the requirements is another very important necessity. Proof-testing is of equal importance in producing optimum systems; therefore, mechanisms must be modeled and tested under conditions simulating actual reactor conditions. The mechanisms considered in this program are those whose malfunction, inaccuracy, or complete failure could result in either unsafe or uneconomical performance. Testing in a facility where the sodium environment conditions can be simulated is recommended for the following reasons:

1. The reliable performance of the mechanisms can be impaired by sodium deposits. The severity of the problem must be demonstrated under test conditions.

2. The reference concept proposes mechanisms that can be inserted into the reactor sodium environment, perform their function safely and reliably, be removed from the reactor in a gas environment, and be used again at the next required occasion with only a minimum of cleaning and inspection. It is also proposed that component parts, such as grapple tools, be remotely replaced in the equipment storage pit, if

necessary, and then returned to service with only remote inspection and checkout. All of these proposed procedures must be confirmed by testing.

3. Testing of the gas cooling system (section 3.2.5.2) is considered necessary in a sodium facility. Little additional penalty is estimated for designing such a test facility and its equipment to permit testing the mechanisms.

Relationship to Current or Planned Programs —

The B&W reference concept has specified mechanisms whose performance requirements are probably quite similar to what will be proposed by other designers. Many of these requirements may be solved in planned programs. However, no programs appear to exist that will solve the specific combination of performance, handling, and maintenance problems that have been proposed. To arrive at the optimum systems, we believe a combination program is necessary.

Major Facilities and/or Equipment Required —

It is proposed that the facilities of the Alliance Research Center will be used for initial assembly and checkout of all mechanisms. These facilities will also be adapted and used for all testing that must be performed in either air or inert gas.

The mechanism tests in sodium will be conducted using the same environmental test vessel, model fuel transfer machine, equipment, and facility as are used for testing the gas cooling system (section 3.2.5.2).

Budget Estimate —

Manpower	\$147,000
Facilities (see section 3.2.5.2 also)	70,000
Equipment	50,000
Total	<u>\$267,000</u>

Table 3-6 is a summary of estimated costs and a schedule of the work to be done on core components handling equipment.

3.2.5.4. Fuel Transfer Machine/Reactor Vessel Interface

Objective — The goal of this program is to develop components that meet the fuel transfer machine/reactor vessel interface requirements. The major components to be considered here are the reactor vessel access port plugs with their seals and locking devices; the fuel transfer machine closure valves; and the fuel transfer machine extension bellows with their seals, actuators, and locking devices. Since fuel assemblies are transferred into and out of the reactor vessel while the reactor is operating, and the fuel transfer machine becomes an extension of the reactor vessel boundary during transfer operations, these components must function properly when subjected to the reactor vessel operating environment and must be able to withstand all conditions to which the reactor vessel boundary will be subjected, including all design basis accidents.

Summary of Current Technology — It is questionable whether sufficient information is available to accurately determine the severity of the environment and the resultant requirements at the fuel transfer machine/reactor vessel interface under normal and abnormal conditions. The difficulty of predicting the normal operating environment at the interface arises from the use of vented fuel and high operating temperatures. The vented fuel will release highly radioactive gases into the sodium coolant cover gas, thereby creating a requirement for absolutely leak-tight seals between the fuel transfer machine and the reactor vessel. The high operating temperature can cause radioactive sodium vapor to enter the cover gas and be available for deposition on sealing surfaces and mechanisms at the interface. This will increase maintenance problems and impair the ability of the seals to meet the requirements of leak-tightness. The uncertainties concerning what constitutes a credible design basis accident, as well as the lack of accurate analytical methods to predict the conditions that result from design basis accidents that have been postulated, lead to uncertainties in determining the structural requirements at the interface.

Scope of Work — The scope of work for this program depends on the amount and relevance of environmental and

operational data made available from operating test reactors related to the 1000-MWe LMFBR program and from R&D programs within the program. Thus, this effort will consist mainly of collecting and evaluating the applicable information from these sources and from existing literature. The information required for proper design of the components at the fuel transfer machine/reactor vessel interface is listed below:

1. Maximum pressure.
2. Maximum temperature.
3. Radiation level.
4. Allowable leakage rate.
5. The rate of deposition of sodium, sodium oxides, or other insoluble contaminants on components and their effect on component performance.
6. The condition of fuel assemblies upon removal from the core.
7. The magnitude, duration, and rate of change of forces, deflections, and temperature during accident conditions.
8. The adhesion, wear, friction, and corrosion characteristics of candidate materials.
9. The adverse effect of items 2 and 3 on material properties.
10. The type, configuration, and material of seals needed to obtain a leakage rate equal to or less than that established in item 4.
11. The temperature, pressure, velocity, and time required to effectively purge interspaces of dangerous contaminants.

If this evaluation should reveal that there is insufficient information to determine these characteristics or the reliability of the components at the fuel transfer machine/reactor vessel interface, then an independent R&D program (based on the evaluation) can be initiated to determine what information is lacking.

The components for the transfer machine/reactor vessel interface area will be designed on the basis of information obtained during the work phase described above. The objective of this effort will be to provide model components to be used with the environmental test vessel and fuel transfer machine mockups (see sections 3.2.5.2

and 3.2.5.3). Testing of the components under simulated normal conditions will be accomplished in combination with the referenced work. Abnormal and accident condition testing requirements will be determined during the investigative phase of this program. After all other testing is completed, the temperature and pressure within the test apparatus will be varied to simulate the abnormal and accident conditions as nearly as possible. The absolute limits will be governed by the safety requirements of the test facility. The apparatus will be instrumented so that test results can be correlated with analytical predictions.

Technical Justification — To design and proof test the proposed fuel assembly transfer concept, a concentrated investigation effort must be devoted to establishing the conditions that the fuel transfer machine and interfacial components will witness. The definition of these conditions will then be used by designers to establish the components and test requirements necessary to ensure the safety and reliability of the procedures. Some actual testing must also be accomplished to prove component design and to check the predicted analytical results.

Relationship to Current or Planned Programs — Considerable work is planned or is already underway to investigate the effects of the design basis accident. Also planned is considerable sodium technology work. The results of these efforts, along with experience from other sodium reactor systems, will serve as bases for establishing the information required.

Major Facilities and/or Equipment Required — Alliance Research Center facilities and LMEC facilities will be used in this work (see 3.2.5.2 and 3.2.5.3).

Budget Estimate —

Manpower	\$360,000
Facilities	10,000
Equipment	<u>35,000</u>
Total	\$405,000

Table 3-6 is a summary of estimated costs and a schedule of the work to be done on core components handling equipment.

3.2.5.5. Transfer Tube

Objective — The objective of this program is to demonstrate the operability, maintainability, reliability, and leak-tightness of the transfer tubes. In the B&W reference concept, both vertical and horizontal transfer tubes have been proposed.³² The vertical tubes for transferring both new and spent fuel are essentially identical; they consist of top and bottom valves separated by a straight length of gas-tight pipe. The valves must provide positive operation and sealing, and these functions must be very reliable. The design must provide for mating with the fuel transfer machine at the operating floor level and for positive sealing and locking to the machine. The valve actuators must be electrically and/or mechanically interlocked with the machine operation to prevent accidental opening or closing. The horizontal transfer tube is very similar to the vertical tubes, except for its orientation and length.

In this program, it is planned to design, model, and test only the spent fuel transfer tube design. This tube will witness the worst conditions of temperature and environment, and a demonstration of its capabilities should prove the acceptability of all designs. The testing will involve both vertical and horizontal operation.

Summary of Current Technology — Fuel transfer tubes are being used extensively in all power reactors with varying degrees of success. Malfunctions of closure valves have occurred, but this has usually been traced to improper material selection, to geometrical arrangements with clearances that were inadequate to prevent jamming by contaminants, to poor alignment, or to inadequate quality control.

Scope of Work — The initial task will be to determine as accurately as possible the environmental and functional requirements for the transfer tube components and to determine what facilities are available for simulating these requirements. In the second task, the environmental data defining the temperature, pressure, and corrosiveness of the environment will be used to select the materials for test models. The third task will be to use the data concerning the functional requirements, e. g., leak-tightness, loads, alignment, speed

of operation, maintenance, control, and monitoring, to determine the configuration and scale of the models. The fourth task will involve determining the necessary tests, performing the tests, interpreting the results, and presenting the data obtained in a usable form.

Technical Justification — The reference concept proposes a vented fuel assembly design and the handling of spent fuel assemblies after a minimum decay storage period. This imposes conditions of temperature and environment not previously encountered in any reactor design. Because of the consequences of malfunction, all of the spent fuel handling systems, including the transfer tube, require detailed development programs to ascertain performance, reliability, and maintenance procedures before incorporation in an actual plant is considered.

Relationship to Current or Planned Programs — Although valve development programs have been planned,²⁸ none of these appear to include the required scope of work.

Major Facilities and/or Equipment Required — Preliminary assembly and operation of the transfer tube design can be accomplished at good research facilities, such as Alliance Research Center. If the early phases of this program establish that environmental conditions, such as sodium vapor, must be simulated, it is proposed that the environmental test vessel and model fuel transfer machine setup at LMEC be adapted for this work (see section 3.2.5.2).

Budget Estimate —

Manpower	\$125,000
Facilities	25,000
Equipment	<u>30,000</u>
Total	\$180,000

Table 3-6 is a summary of estimated costs and a schedule of the work to be done on core components handling equipment.

3.2.5.6. Spent Fuel Cell

Objective — The goal of this effort is to develop the spent fuel cell equipment to a point of high reliability and confidence

in its use. The spent fuel cell of the reference design must be equipped to install spent fuel assemblies into a shipping cask.³³ The equipment must be operated remotely; therefore, viewing apparatus (such as leaded-glass windows and television cameras) must be provided for monitoring remote operations. The following major systems and pieces of equipment are to be developed:

1. A lifting device to handle the inner container of the shipping cask.
2. An orbital welder to seal-weld the container.
3. A metal cutter to open the container.
4. Measuring devices to check the temperature and level of the sodium in the container.
5. A cooling and heating system for the container.
6. A draining and filling system for the container.
7. A leak detector to ensure that the container is properly sealed.
8. Transfer tubes (see section 3.2.5.5).

Summary of Current Technology — The spent fuel cell is essentially an inerted hot cell for handling radioactive components that have been removed from the core for shipment off the reactor site. Therefore, experience and knowledge from existing hot cells, along with information from the mechanisms program (see 3.2.5.3) and the gas-cooling program (section 3.2.5.2) should provide an adequate design basis for the equipment of this cell. However, to determine the operating characteristics, the interrelationships, the maintenance procedures, and the remote control procedures involved will require mockup and testing of this equipment.

Scope of Work — To accomplish the aim of this program, it will be necessary to mock up the equipment of the spent fuel cell, to test the individual components for operability, and to test the assembled components in their desired configurations to ensure that the operation of placing a fuel assembly in the shipping cask inner container and sealing the container can be accomplished reliably using remote-controlled equipment.

Since spent fuel assemblies are to be shipped in sodium, the inner container of the shipping cask will be filled with

sodium in the spent fuel cell. The equipment for filling the container with sodium and the methods for determining and maintaining the sodium level and temperature will have to be developed during this program.

The experimental effort will consist of test, evaluation, and modification of the basic operating mechanisms and controls, with emphasis on function, reliability, and the development of remotely maintainable equipment and techniques for producing consistently good seal welds for the sodium-filled shipping containers.

Technical Justification — As is the case for other components in the proposed spent fuel handling system, mockup and testing of the spent fuel cell is considered absolutely necessary. Although much of the equipment may be considered state of the art, the proper functioning of the equipment is extremely important, and modeling and testing must be accomplished to demonstrate its operability, reliability, and maintainability.

Relationship to Current or Planned Programs —
None.

Major Facilities and/or Equipment Required —
It is proposed that the preliminary assembly and operation of the equipment for the spent fuel cell be accomplished at research facilities, such as those at Alliance Research Center. The development of critical equipment and instrumentation can progress rapidly in this kind of atmosphere. Final proof testing is proposed at a hot cell facility where sodium can be used and where the actual remote conditions can be simulated. It is further proposed that, for these final tests, the mockup fuel assembly used in the gas-cooling development program be used (see section 3.2.5.2). This assembly is to be designed for simulating the decay heat and will be valuable in duplicating actual conditions. The hot cell facility selected for final testing must be adaptable to provide the essential services, test apparatus, and instrumentation required.

Budget Estimate —

Manpower	\$435,000
Facilities (rental and adaptation)	150,000
Equipment	135,000
Total	<u>\$720,000</u>

Table 3-6 is a summary of estimated costs and a schedule of the work to be done on core components handling equipment.

3. 2. 5. 7. New Fuel Cell

Objective — The objective of this program is to develop the new fuel cell equipment and the assembly removal and inspection procedures. The equipment must be operated remotely to permit the handling of assemblies containing reprocessed plutonium fuel; therefore, leaded-glass viewing ports and viewing apparatus must be provided to monitor the remote operations. The following major pieces of equipment are to be developed:

1. A mechanism to extract a fuel assembly and shipping container unit from the horizontal transfer tube and raise it to a vertical position.
2. A mechanism to open the shipping container and permit the removal of a fuel assembly.
3. Viewing and inspection equipment.
4. Transfer tubes (see section 3. 2. 5. 5).

Summary of Current Technology — The new fuel cell is a shielded, inert-gas-filled clean room for handling assemblies loaded with recycled plutonium fuel. At this time, complete fission-product cleanup does not appear possible; therefore, hot cell operational techniques are required. Technology from existing hot cells and laboratories, as well as information from programs for the development of spent fuel cell equipment (section 3. 2. 5. 6), should provide most of the information necessary for designing equipment in this cell.

The critical procedure, however, will be inspection. It is proposed that new assemblies will be shipped to the reactor site sealed in a container. This container is removed from the bulk shipping transporter, passed through a horizontal transfer tube, raised to a vertical position, and opened. The only on-site inspection proposed is accomplished as the assembly is raised from the container and lifted upward (vertically) into the fuel transfer machine. The inspection techniques and equipment must be demonstrated to ensure that the new fuel assembly is structurally sound and clean. That is, these procedures must ensure that the assembly meets the specifications

established for reactor installation. To accomplish these goals, it is apparent that such techniques as ultrasonics may be required in addition to visual techniques. For this application, the available information does not appear to be sufficient, and therefore, development is necessary.

Scope of Work — The first phase of this program is an effort to establish fuel assembly inspection requirements and methods. The fuel assembly design must be reviewed to establish its modes of failure under the proposed conditions.³⁴ State-of-the-art equipment and techniques must then be reviewed to establish candidate equipment and the requirements for adaptation and new development. Based on the results of this work, the necessary modeling, testing, and facility requirements will be established, and design of the models and equipment will be accomplished. After fabrication, individual components will be tested for operability, reliability, and ease of maintenance. All components will then be assembled to simulate their operating positions. Testing and further development will be carried out to ensure that the operations of removing a new fuel assembly from its shipping container and inspecting of the assembly can be performed reliably and efficiently.

Technical Justification — The inspection and handling of new fuel assemblies at the reactor site are critical procedures. The consequences of installing damaged or unclean assemblies in the reactor core could lead to very dangerous conditions or, at least, very costly downtime. Even though proven equipment and techniques are proposed for most areas, the complete system must be modeled and, with simulated conditions, developed to a high degree of reliability.

Relationship to Current or Planned Programs — No reactor design appears to have incorporated handling and inspection procedures comparable to the proposed system. We have discovered no current or planned programs that will accomplish the proposed objectives.

Major Facilities and/or Equipment Required — Although the proposed system involves hot cell techniques and equipment, only good research-type facilities are considered necessary to accomplish this program. Facilities like B&W's Alliance Research Center and

Nuclear Development Center can mock up testing equipment and the required new fuel cell equipment to simulate the hot cell isolation and can perform all of the required testing.

<u>Budget Estimate —</u>	
Manpower	\$375,000
Facilities	--
Equipment	<u>105,000</u>
Total	\$480,000

Table 3-6 is a summary of estimated costs and a schedule of the work to be done on core components handling equipment.

3.2.5.8. Rotating Shield Plug Bearings, Seals, Drives, and Locking Devices

Objective — The objective of this R&D program is to develop the bearings, seals, drives, and locking devices of the 1000-MWe LMFBR rotating shield plugs to the point where their operability, reliability, and maintainability have been satisfactorily proven.

Summary of Current Technology — Although the technology for rolling element bearings is well-established, the methods for calculating the bearing loading conditions imposed by the large-diameter rotating plugs of this design are not accurate enough to warrant fabrication and operation without testing. Since the bearings are protected from the reactor environment, allowing conventional lubricants and materials to be used, their design depends on the establishment of their loading condition. The specific bearing design (balls or rollers) will most probably be dictated by manufacturing capability and the possibility of Brinelling under static loading conditions.

It must be recognized that the many variables and unknowns in these large-diameter bearing applications make it difficult to accurately predict the minimum expected bearing life. Inaccuracies and deflections in the structure supporting the bearing may produce localized load conditions which are very difficult to calculate.

Since the plug bearing loads are high and not central, a computer program which is available to size the bearings may be used. The program enables the manufacturer to determine the elastic compliance of a system of any number of rolling element bearings under

all possible loading conditions. The solution not only defines the loading and attitude of each rolling element, but determines the displacement of each inner ring with respect to its outer ring. The code also determines area dimensions, contact element load, contact angle change and stresses at each rolling element position, the locations of pressure ellipse extremities, the non-linear spring rates of the bearing in all modes, friction and spin torque due to rolling, and fatigue life.

In general, the rolling-element bearing will function properly if there is little or no wear or surface degradation of the rolling element or races. Surface damage of any kind will result in rough operation, which further accelerates wear and could lead to complete failure. Ball- or roller-type bearings require a hardened race or an equivalent design, such as non-hardened races with nylon balls. Roller bearings have higher rated capacities than ball bearings of the same size, series, and manufacturing method; this is because the cylindrical rolling elements spread the load over greater areas and produce less stress and deformation for a given load.

It is possible that the small and intermediate plug bearings can be manufactured from forged rings; then conventional balls and separators may be used. Proper preparation of the forgings for machining may be accomplished by the forging supplier or by the bearing manufacturer, depending on the size of the piece and other factors. The large-diameter plug bearing can possibly be manufactured using the wire-race-segment approach. The wire-race antifriction bearing is one in which the rolling elements roll on inserted hardened steel wire raceways. This bearing can be manufactured in single-piece rings up to 14 feet in diameter and possibly in segmented rings up to 30 feet or more. The round or profiled wires lie in grooves of the surrounding rings. The separation of the raceways from the rings has many advantages:

1. A single bearing is capable of transmitting any combination of radial thrust and tilting moment loading.
2. The bearing can be mounted directly to the cover structure and rotating plug with bolts; no auxiliary mounting clamps or rings are required.

3. The bearing may be renewed by replacing wires and balls at much less cost than replacing solid-race bearings.
4. Internal clearance may be adjusted to suit requirements.
5. This bearing type has low, uniform, frictional torque and high resistance to damage from shock loading because of resilience in the wires.
6. The bearing requires no maintenance except lubrication.

The wire-race bearing has been proven in applications, including nuclear reactors, for many years. Since the surrounding rings are not subjected directly to rolling stresses, they may have virtually any cross section and may be made of a great variety of materials.

The use of dip-ring seals is a proven method of sealing rotating plugs to a reactor cover structure; these seals have been used or designed into several sodium reactor systems. The sodium reactor experiment (SRE) and the Hallam reactor are examples where bismuth-tin-alloy-filled dip-ring seals were used to seal the rotating plugs required for the installation and replacement of moderator can assemblies. In both instances, only relatively minor problems have been reported. The major complaint appears to originate from the excessive time required to melt the seals before rotation could be accomplished. The degree of underdesign is questionable because the design was based on very little use; not for refueling procedures, but rather only for conditions requiring change of moderator assemblies. The bismuth-tin-alloy-filled dip-ring seals are also incorporated in the EBR-II, Dounreay, and Fermi reactor designs. In these cases, the seals require melting for refueling the reactor, although in the last example, the seals have never been filled. Successful use has been reported in both EBR-II and Dounreay.

The EBR-II design approach has been used in the conceptual 1000-MWe design—a combination of molten-frozen seals. In this approach, the portion of the sealing alloy exposed to the reactor cover gas is kept in the molten state at all times to maintain the gas seal. The portion of the alloy on the outboard side is frozen during reactor operating periods to prevent seal metal loss in the event of a large pressure differential across the seal. The entire seal is melted to permit rotation

of the plug. Early reports from EBR-II had reported some difficulty with the seal design: primarily indicated were problems of solid material forming in the trough which made rotation difficult. These problems have apparently been solved, and recent reports have not indicated further trouble. The 1000-MWe dip-ring seal design will utilize the EBR-II design and established state-of-the-art technology.

The use of mechanical seals to seal cover structure components and installed equipment in sodium reactor systems has now been well-established. Fermi is a good example of successful use eliminating the need for dip-ring seals designed and incorporated into the cover structure. The size, number, and operational requirements of these seals are greater in the 1000-MWe reactor environment.

The rotating plug drive system is an assembly of state-of-the-art components peculiar to this reactor design. Many system designs available from the heavy equipment industry can be adapted for use as the drive mechanism for these plugs.

Three basic locking methods have been included in the design of the cover structure for retention of the structure and the components installed in it: (1) simple mechanical fasteners, such as bolted and welded joints; (2) manual or machine-operated bayonet locks; and (3) a special pin-locking device to be operated in conjunction with the fuel transfer machine. All three methods are considered state of the art. Only the third requires more than analytical support. This is a special assembly of parts associated with the reactor vessel access port plugs; it is installed so that operation can be initiated sequentially by machine action. A model will be required to proof test its operation and to establish its reliability and adequacy.

Scope of Work — Research will be required to establish candidate bearing designs and material combinations. The proposed candidates are ball or roller bearings with either segmented or one-piece, hardened and ground races; or non-hardened races with hardened and ground wire races for conventional balls. After the bearing designs and material combinations have been selected, models must be designed and constructed for testing to confirm the adequacy of the design. The models should be tested under conditions simulating the

loading movements and temperature that the actual bearings will experience during reactor operation. The possibility of Brinelling must be investigated by subjecting the models to long-term (6 months) static loading with vibrations which simulate, as closely as possible, the vibrations and resultant dynamic loads that will be imposed on the operating bearing by running pumps and other plant-forcing functions. Subsequently, the bearings must be tested for operating capability. Cleanliness is a prime consideration for these bearings; hence, they must be protected from airborne particles, as well as the expected source of contamination—the reactor vessel environment. Methods of accomplishing this are incorporated into the design; however, the shields to keep the lubricant in and the contaminants out will have to be modeled and tested along with the bearings.

The dip-ring seal development effort involves research into the EBR-II dip-ring seal design and operational data to update present information with actual performance data. In addition, an analytical effort will be necessary to size heating and cooling systems and to design models for testing. The models will be constructed and tested to obtain sufficient data to define availability, reliability, and maintainability.

The rotating plug mechanical seal development effort involves determining the seal materials, performance, and life expectancy. The choice of seal depends on all the operating conditions of the application, such as temperature extremes, type of media (air, inert gas, sodium vapor), whether the seal is dynamic or static, pressure differentials, and lubrication. The main performance criterion for these mechanical seals is leak rate. It can be seen that the allowable leak rate for the seals is extremely small by observing that there are approximately 50 major mechanical seal areas in the present design of the cover structure, and that the total allowable leak rate for the reactor vessel cover gas is approximately 25 cc per day, assuming vented fuel is used. To establish the performance and life expectancy of the selected seals will require prototyping and testing of each seal configuration and the surfaces to which it must seal under conditions simulating those expected during actual reactor operation. For the dynamic seals involved,

this includes long-term (6 months) static testing at the required temperatures and pressures with subsequent movement and leak tests. Where appropriate, these tests should be conducted in the presence of sodium vapor to determine its effect on the ability of the seals to keep the leak rate within design limits.

Technical Justification — This program is needed because the large size and the environment of the bearings, seals, drives, and locking devices of the rotating shield plugs (which are critical to the successful operation of the reactor) make it difficult to predict their minimum expected operating life.

Relationship to Current or Planned Programs — The basic information required to design the individual components of the items to be studied in this program will come from other programs; however, the performance and life expectancy of the items cannot be determined without modeling and testing the specific combinations of the design.

Major Facilities and/or Equipment Required — Most of the tests required by this program can be carried out at LMEC.

Budget Estimate —

Manpower	\$220,000
Facilities	150,000
Equipment	<u>80,000</u>
Total	\$450,000

Table 3-6 is a summary of estimated costs and a schedule of the work to be done on core components handling equipment.

3.2.6. Miscellaneous Handling Equipment
and Facilities

3.2.6.1. General

Objective — The overall goal of this program is to provide the technology required to design, fabricate, and operate the special equipment and facilities needed at the reactor site for decontamination and/or remote maintenance of equipment that has been exposed either directly or indirectly to the reactor vessel environment.

Summary of Current Technology — The removal

of large equipment items from radioactively contaminated sodium has entailed special handling to prevent reaction of the sodium, the release of radioactive material, and the exposure of personnel. Maintenance systems have included gas locks for handling equipment that is to be removed from sodium systems. Typical design features for equipment of this type are as follows:

1. Provisions for keeping the component in an inert atmosphere during and after removal.
2. Sealing provisions for the sodium system during and after component removal.
3. Adequate crane capacity and control.
4. Provisions for moving the component to a cleaning or maintenance area.
5. Provisions for shielding and cooling where required.

Gas lock handling apparatus have been used to remove sodium pumps in the EBR-II, Fermi, and Hallam reactors. The EBR-II and Fermi pumps were removed into a large cylinder connected to a valve section which was sealed to the pump nozzle. A piston with inflatable seals was used to seal the interior of the cylinder from the atmosphere; it also acted as a lifting adapter for the pump. The valve section provided a seal for the opening created by removing the pump. The Hallam pumps were removed using a plastic bag. In all three cases, an argon gas atmosphere was maintained about the pumps during the removal operation.

Remote handling equipment and techniques have been adapted to handling equipment contaminated with radioactive sodium. This has been accomplished, for the most part, with only minor changes. It appears that most remote handling facilities, e. g. , hot cells, could have been designed to handle radioactive equipment contaminated with sodium by a more careful selection of materials and by adding an inert gas supply with appropriate controls and instrumentation. This may require a decrease in facility leakage, since most cells operate at a slight negative pressure and would require excessive gas if facility leakage could not be kept relatively low.

Various methods and procedures have been developed and are being used to clean equipment contaminated with radioactive sodium; procedures for disposing of waste sodium and sodium-contaminated waste have also been developed. The principal cleaning agents have been water, steam, liquid ammonia, methyl alcohol, and oil. No definite rules have been established for the selection of one method over another, and a combination of methods has often been used to obtain the desired cleanliness. Steam cleaning was chosen for the reference design, since it is especially suitable for large, complex components, and it has the advantages of lower cost, non-flammability, and a complete reaction is more likely because the sodium is in a liquid state at steam-cleaning temperatures.

Although the design of the remote maintenance and decontamination equipment and facilities of the reference design is considered to be within the scope of existing basic technology, the R&D programs described in the following subsections are considered to be necessary because of the physical size of the equipment involved and to ensure that the specific combination of operations proposed can be carried out safely, reliably, and economically.

3.2.6.2. Valve Box

Objective — The purpose of this program is to develop a remotely operable valve box that can be used to remove large components, contaminated with radioactive sodium and fission products, from the reactor vessel environment. The major items to be handled will be IHXs and primary pumps. This program must establish the purge systems, components, materials, and procedures needed to fabricate and remotely operate a valve box that will remove and install large components in an efficient, reliable, and economical manner. The valve box consists of two tandem-mounted units, as described in section 7.2.3.2 of the CSDD.¹ The lower unit seals to the reactor operating face and is essentially a large, motor-operated valve with a thin sheet gate that will be wound onto rollers to provide an opening approximately 9.5 feet in diameter. The upper unit is similar to the lower one and must mate and seal to it. The valve box design requires that a flexible bag approximately 9.5 feet in diameter be attached and sealed to the top side of the upper unit

and to a component lifting adapter. The flexible member must be extendable and collapsible to permit the removal and installation of the components being handled, yet it must seal the reactor vessel reliably during the handling operation. With its flexible member fully extended, the valve box will be approximately 50 feet tall. This necessitates a guide cage to keep the flexible bag circular and to prevent its over-extension as it extends to accept a component being removed from the reactor. In addition, the guide cage will act as a stabilizing member during transfer operations and as the carrier for the upper unit of the valve box.

Summary of Current Technology — The state of technology for this item is summarized in section 3.2.6.1.

Scope of Work — Model design and performance testing must be carried out. Testing conditions should simulate operating conditions whenever possible. The environmental conditions to which the internal surfaces of the valve box will be exposed are radioactive sodium droplets at 400 F, contact with a pump or IHX covered with a radioactive sodium film and plated fission products at 400 F, and argon gas containing sodium vapor and nuclear fission products. The external environment will be ambient air. This design and test program must be carried out to establish such things as fabrication techniques, installation, inspection, and operation.

The following should be considered during the R&D program to ensure the development of a safe, reliable, and workable unit.

1. Perform research to establish a material that will meet the operational and environmental requirements of the flexible extension member.
2. Since the valve box is not designed to stop gamma radiation, procedures for remote operations will have to be developed, i.e., methods of attaching and detaching the tandem valve units, and operation of the lifting adapter which has to engage both the rotary crane hook and the component being lifted.
3. Determine seal materials and configuration and conduct performance tests to establish their reliability. Inflatable seals may

prove to be the best type for sealing between the tandem valve units and for sealing the sheet-type closure gate to the valve body.

4. The ability to effectively seal the 9.5-foot opening of the valve box using the sheet-type closure gate must be demonstrated. In addition, life testing of the closure gate and its associated rollers, seals, and drive system must be performed.

5. The instrumentation for monitoring the operation and conditions of the valve box must be determined during the design and performance testing phase. Measurements of temperature, pressure, nuclear activity, and the force required to lift a component being handled are essential.

Technical Justification — This program is necessary to establish the feasibility of fabricating and operating an extendable and collapsible valve box unit that is large enough to permit the removal and installation of large reactor components (IHX, pumps), yet seal the reactor vessel reliably during the handling operation.

Relationship to Current or Planned Programs — The technology developed for other components under other sections of this R&D program will be investigated and applied where appropriate. Material data from the fuels, materials, and sodium technology sections of the program will be used. The instrumentation and control section will provide the information required to design controls and monitoring instrumentation for the valve box. Basic requirements and the space limitations for the valve box will be supplied by the plant design section of the program.

Major Facilities and/or Equipment Required — The tests in this program that involve sodium can be performed at LMEC or at ARC. The remaining operational and leak tests can be performed at any test facility or shop that has adequate measuring instruments and gas supply available.

Budget Estimate —

Manpower	\$ 75,000
Facilities	100,000
Equipment	<u>65,000</u>
Total	\$240,000

Table 3-7 is a summary of estimated costs and a schedule of work to be done on miscellaneous handling equipment and facilities.

Table 3-7. Summary of Schedule and Estimated Costs for R&D Work on Miscellaneous Handling Equipment and Facilities

Task	FY CY	Estimated cost, \$ × 10 ³					Total
		1973	1974	1975	1976	1977	
		1972	1973	1974	1975	1976	
Miscellaneous Handling Equipment and Facilities							
1. Valve box		--	45	195	--	--	240
2. Decontamination vessel and facilities		--	--	150	320	155	625
3. Equipment storage pit		--	50	50	90	75	265
Total		--	95	395	410	230	1130

3. 2. 6. 3. Decontamination Vessel and Facilities

Objective — The objective of this effort is to develop the decontamination vessel and the facilities, equipment, and procedures associated with its operation to the point that plant equipment contaminated with radioactive sodium and requiring maintenance or disposal can be cleaned safely and efficiently to the degree dictated by the maintenance or disposal procedures.

Summary of Current Technology — The state of the art with regard to decontamination methods is discussed in section 3. 2. 6. 1. A development program in this area is necessary primarily because of the size and complexity of the reference design components that may require cleaning. In this design, articles contaminated with radioactive sodium and requiring cleaning are to be placed in an inert-gas-filled vessel, where steam, followed by a water rinse, will be applied to effect cleaning. Whether sodium in relatively inaccessible areas of the large, complex-shaped reactor components will be reacted by such a process and whether the reaction rate can be kept within

acceptable limits cannot be adequately determined by analysis or from the results of prior experimental efforts.

Scope of Work — Articles contaminated with radioactive sodium can be cleaned safely and effectively by placing them in an inert atmosphere and applying saturated steam followed by a water rinse. Although this method has been selected for use in the reference design, the literature surveys and analytical studies of this program will cover other cleaning methods, such as the hot oil bath and the use of commercial solvents with an alcohol base. This additional coverage will aid in determining whether the proper decontamination method was chosen; if it was not, the program will provide data for making the proper choice.

To ensure that sodium in relatively inaccessible areas is reacted by the cleaning process, it is mandatory that the physical configuration of each piece of equipment be simulated, contaminated with sodium, and subjected to the process. This will be accomplished by conducting literature surveys and by instituting a combined analytical and experimental program to design models of decontamination and plant equipment configurations that can be used to determine the degree of decontamination that can be attained for actual plant components. These models will also be used in analyzing the effects of the decontamination progress on the structural materials.

The program should generate the information and develop the procedures required to design, assemble, and operate a decontamination vessel that can be used to remove radioactively contaminated sodium from complex-shaped reactor components such as IHXs, pumps, refueling mechanisms, emergency core coolers, control rod assemblies, purification equipment, instruments, and special equipment. The vessel into which the pieces to be cleaned will be placed must be capable of containing an inert gas and the cleaning medium and must be adequately sealed, vented, and drained. Since this vessel will be used as a transfer container and no shielding is incorporated in its design, remote methods of handling and sealing will have to be developed. In this design, steam cleaning in an inert atmosphere is to be used. The techniques are well-established; however, special precautions are required to prevent the release of radioactive material and exposure of

personnel. During the actual decontamination process, the vessel will be placed in a shielded pit, which is essentially a service station for the decontamination vessel. Provisions for inert gas, steam, and water supply, and for discharge of the gaseous, liquid, and solid products of the decontamination process are to be an integral part of this pit.

Technical Justification -- A development program is necessary in this area to determine whether radioactive sodium can be removed from relatively inaccessible areas of complex-shaped reactor components, such as IHXs, pumps, the refueling mechanism, purification equipment, and reactivity control apparatus.

Relationship to Current or Planned Programs -- The prototype IHX to be designed and built for the "Liquid-Metal to Metal Heat Exchanger" program (section 3.2.2) may be used during this program, provided the testing schedule can be adjusted, and provided the model is not damaged beyond usefulness during those tests.

Information on the basic mechanisms that govern sodium-water reactions generated during the steam generator sodium-water reaction studies will probably provide useful input for this program.

Major Facilities and/or Equipment Required -- Most of these tests can be performed at the LMEC. Structural tests required to determine the effects of the decontamination process on structural properties can be performed at any test facility having the necessary equipment.

Some test models may be salvaged from other programs, e. g., refueling mechanisms from the core components handling equipment program (section 3.2.5), IHXs from the liquid-metal to metal heat exchanger program (section 3.2.2), and pumps from the large sodium pump development program.

Budget Estimate --

Manpower	\$295,000
Facilities	130,000
Equipment	<u>200,000</u>
Total	\$625,000

Table 3-7 is a summary of estimated costs and a schedule of the work to be done on miscellaneous handling equipment and facilities.

3. 2. 6. 4. Equipment Storage Pit

Objectives — This program has two overall objectives: The first is to determine and clearly define the environmental and operational requirements of the storage pit and its equipment. The second is to define and carry out a test program simulating these requirements as nearly as is required to develop the equipment of this facility to the point that recourse to after-the-fact design will not be necessary. Since the equipment storage pit is essentially a remote maintenance, pre-operational checkout, inspection, and handling area rather than a place of storage, the specific objective of the program is to provide the technology required to design the pit equipment to remotely handle radioactively contaminated reactor or plant components that need routine maintenance or have malfunctioned.

Summary of Current Technology — The equipment storage (and remote maintenance) pit is essentially a hot cell for handling, inspecting, and maintaining radioactive reactor components. Experience and knowledge from existing hot cells, in conjunction with information from the mechanisms program (section 3. 2. 5. 3), the fuel transfer machine/reactor vessel interface program (section 3. 2. 5. 4), and the spent fuel cell program (section 3. 2. 5. 6), should provide the basic design data needed to design the equipment required in this pit. However, detailed design of appropriate models and testing of the equipment involved may be necessary to develop the remote operating procedures and to ensure that the variety of operations proposed for the pit can be accomplished without compromising the safety, reliability, maintenance, and economic aspects associated with the pit or other interfacial systems.

Scope of Work — The scope of work for this program includes efforts to definitively establish the design, safety, and performance requirements that must be met by this cell and all its equipment. These requirements will be established, to the degree possible, by conducting literature surveys and analytical studies to reduce time-consuming and costly experimental determination to a minimum.

The establishment of the remote inspection and maintenance procedures that will be performed in this pit requires the definition of all inspection and maintenance requirements for the items to be handled.

The major plant equipment items for which inspection, checkout, storage, or maintenance procedures and apparatus must be developed are as follows:

1. The refueling mechanism.
2. The fuel transfer machine internals.
3. The control rod drive unit (including drive, extension rod, and control rod).
4. Primary pumps.
5. Intermediate heat exchangers (IHXs).

On the basis of the requirements dictated by these components, the necessary models, tests, and cell requirements will be defined and the required design work accomplished. After the individual components have been fabricated and tested, they will be assembled in their simulated operating positions to ensure that they will function reliably and efficiently as a system. It is anticipated that only the assembly and disassembly procedures associated with maintaining the refueling mechanism and the fuel transfer machine internals will require any testing other than preoperational proof testing.

Technical Justification — Inspection and maintenance of reactor equipment at the reactor site is mandatory for economical plant operation; therefore, definitive information on the environmental and operational requirements of the equipment storage pit is essential. The detailed design and fabrication of the special equipment required for remote inspection, storage, checkout, and maintenance work would not be prudent without mockup and testing of critical equipment and instrumentation.

Relationship to Current or Planned Programs — Some of the information from the spent and new fuel cell programs (sections 3.2.5.6 and 3.2.5.7), the mechanisms program (section 3.2.5.3), the gas-cooling program (section 3.2.5.2), and existing hot cells should be applicable to this program. It should help to determine the operational

procedures, maintenance procedures, interrelationships, and remote control procedures of the pit.

Major Facilities and/or Equipment — The equipment and procedures proposed in this program can be developed in any research facility having hot cell techniques and equipment available, e.g., the Alliance Research Center or Nuclear Development Center can simulate hot cell isolation and can perform all the required testing and evaluation.

Budget Estimate —

Manpower	\$240,000
Facilities	--
Equipment	<u>25,000</u>
Total	\$265,000

Table 3-7 is a summary of estimated costs and a schedule of the work to be done on miscellaneous handling equipment and facilities.

3.2.7. Reactivity Control System

3.2.7.1. General

Objectives — The objectives of this program are to develop the design and engineering data necessary for the detailed design and fabrication of the 1000-MWe LMFBR reactivity control system components and to demonstrate their functional adequacy.

The program will consist of three interrelated major phases: (1) control rod drive development, (2) control rod development, and (3) prototype reactivity control system design and demonstration. The first two phases will be performed in parallel and will result in engineering and design data necessary for the detailed design and fabrication of the prototype reactivity control system. The third phase is closely related and will require parallel efforts during the late stages of the first two phases.

This approach emphasizes the availability of critical design data early in the program to ensure timely input to the designer during the preliminary design phases. In addition, the critical components and subassemblies will have been identified, evaluated, tested, and optimized prior to the final design of the system. Thus, the

approach emphasizes individual component and subassembly development in the initial phases of the program to identify and solve critical problem areas, followed by evaluation and demonstration testing of a prototype reactivity control system.

Summary of Current Technology — Reactivity is controlled in most of the existing LMFBR facilities by top-mounted drive systems. All of these have used electromechanical drives with electromagnets to facilitate quick rod release. In general, all components above the control rod extension have been protected from the sodium and inert cover gas by seals. Control rod position indication has utilized both direct and indirect means—Dounreay uses direct indications, and Fermi and EBR-II use indirect position indication. Environmental conditions have caused problems such as galling, self-welding, and wear in bearings.

Reactivity control drives affect plant and core designs as well as fuel handling equipment and procedures. The drives in EBR-II and Fermi are mounted on the rotating refueling plugs, which necessitates disconnection during refueling operations. Dounreay avoided having to disconnect drives during refueling by adding a mechanism and a cantilevered extension between the control members and the drives.

3.2.7.2. Control Rod Drive Development

Objective — The objective of this phase is to develop the design and engineering data necessary for the detailed design of the prototype control rod drive and to demonstrate the functional adequacy of the drive components and subcomponents.

Summary of Current Technology — A number of control methods (reflector control, movable fuel, and movable absorbers), control materials (B_4C , rare earths, and tantalum), and reactivity control drives (mechanical, electromechanical, and hydraulic) have been proposed for or used in fast reactors. The state of the art, however, is limited to mechanical and electromechanical systems that depend on magnetic releases as the prime method of releasing rods for insertion into the core during scrams. The major problem areas for existing LMFBR reactivity control system drives have originated in the

lack of sufficient data to determine environmental conditions and the deleterious effect this has on materials and mechanisms.

Scope of Work — In conjunction with an in-depth evaluation and analysis of the proposed control rod drive concept, the existing control rod drive technology will be reviewed to determine its applicability to the reference concept. The review of technology will be an on-going activity to ensure that relevant information generated by commercial and national LMFBR research programs is available during the development program. This review will include the determination of optimum design features and evaluation of design deficiencies in existing LMFBR drive systems and components. The preceding evaluation, analysis, and review will consider such things as materials, seals, gears, actuator, position indicators, instrumentation, tolerances, loadings, fabrication methods, maintenance procedures, and failure and safety analyses.

The results of the evaluation, analysis, and review described above will be used to establish functional, operational, and design criteria for the prototypal components and mechanisms required to attain the objectives of this program. These results will also be used in identifying critical problem areas and selecting specific components for further evaluation. After the critical problem areas and components have been identified, the original concept will be reevaluated to determine whether the problem areas can be eliminated by modifying and/or upgrading the design. The problem areas that cannot be eliminated will then be clearly defined, and the various experimental parameters and tests required to solve the problems will be established. Following this, design work on test apparatus will begin.

The testing program for the control rod drive will consist of subassembly and component tests followed by modular unit tests. The reference control rod drive comprises the three main modular units listed below.

1. The upper actuator housing, which contains the control rod drive motors, differential gear trains, brake, and flywheel.
2. The lower actuator housing, which contains the electromagnet, drive screw, ball nut, antirotational guides, magnetic power leads, upper and lower drive limit switches, and position indicators.

3. The combined snubber/seal housing, which contains the pneumatic snubber, linear motion seal, static backup seal, extension rod linear motion guide bearings, control rod extension, limit switches, and bearing leak detector.

A significant part of the tests for these units can be accomplished in a non-sodium environment, because they will be protected from the sodium by the linear motion seal at the lower end of the combined snubber/seal housing.

Subassemblies and components of the three modular units will be evaluated and tested to optimize their design and establish their reliability. At the conclusion of the subassembly testing and evaluation, final design and fabrication of prototype subassemblies and mechanisms will be accomplished. These subassemblies will be assembled into the modular units of the control rod drive system, which will be tested and evaluated in turn.

The upper actuator housing module will undergo tests to evaluate its electrical system. These tests will evaluate the holding and lifting operation of the module and ensure its electrical-mechanical compatibility and adequacy. Items to be investigated include (1) module response to insertion, withdrawal, and hold demands (including inertia effects); (2) feasibility and compatibility of non-lubricated, impregnated, and/or prelubricated wear surfaces; (3) simulated abnormal conditions; (4) module response to electrical input demands; (5) evaluation of electrical system in normal and abnormal modes; and (6) compatibility of electrical and mechanical characteristics.

Testing and evaluation of the lower actuator housing module will involve (1) ball screw-ball nut tests for wear, lubrication, and material compatibility; (2) tests of the accuracy and repeatability of the position indicator; (3) tests on the magnetic clutch to investigate such items as electrical operation, failure modes, spurious electrical feedback effects, etc.; and (4) antirotation guide tests for wear, torque characteristics, and material compatibility.

The testing and evaluation required to develop the combined snubber/seal housing module will include in-depth technology surveys of existing snubber assemblies, seals, linear motion

bearings, limit switches, and leak detectors to determine their applicability to this design. Information gained from these surveys will also be used to define the test procedures and apparatus required to establish the functional and operational adequacy of the module.

After a detailed test program has been defined and the test apparatus has been designed and fabricated, the snubber/seal module will undergo extensive testing to evaluate its functional, safety, and operational characteristics. This testing will include such items as snubbing characteristics in various snubbing mediums (air, buffer gas, cover gas), the effect of misalignment and clearance on snubbing characteristics, the effect of wear (snubbing characteristics versus life), the evaluation of material compatibility and the effects of abnormal operation. The experimental results will be compared with analytical models developed and design improvements will be made as required. Also included in this phase will be extensive testing of the linear motion and static seals; this will include evaluation of wear life, material compatibility, effect of misalignment, clearances and tolerances, and leakage investigations. At the conclusion of these tests, design modifications will be made, followed by short-term tests of these improved designs to ensure their satisfactory performance.

Technical Justification — The 1000-MWe LMFBR will have higher power levels and a larger core than existing LMFBR plants. Therefore, R&D is required to identify and clearly define the various experimental parameters for solving the problems of these more severe operating conditions.

Relationship to Current or Planned Program — The reactivity control rod drive development program has a functional interface with the core design program and depends on the fuels and materials and sodium technology programs to supply information for the proper selection of materials. The instrumentation and control section will provide assistance in developing specific instruments for monitoring the operation and condition of all the drive components. The safety-related functional requirements of the drive will be delineated in the safety program.

Major Facilities and/or Equipment — None.

Budget Estimate —

Manpower	\$2,010,000
Facilities	--
Equipment	500,000
<hr/>	
Total	\$2,510,000

Table 3-8 is a summary of estimated costs and a schedule of the work to be done on the reactivity control system.

Table 3-8. Summary of Schedule and Estimated Costs for R&D Work on the Reactivity Control System

Task	FY CY	Estimated cost, \$ × 10 ³										Total
		1970	1971	1972	1973	1974	1975	1976	1977	1978		
		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	
Reactivity Control System												
1. Control rod drive		270	270	770	840	310	50	--	--	--	--	2510
2. Prototype reactivity control system design and demonstration		--	--	--	--	--	80	490	520	410	--	1500
Total		270	270	770	840	310	130	490	520	410	--	4010

3. 2. 7. 3. Prototype Reactivity Control System Design and Demonstration

Objective — The objective of this phase is to demonstrate the performance and reliability of the prototype reactivity control system for the 1000-MWe LMFBR reference design. In this phase, the prototype control system will be designed, fabricated, and subjected to demonstration and proof testing, which will consist of full-size prototype testing in water and sodium test facilities. The final demonstration of reliability will be accomplished through performance testing of the system in a test or demonstration reactor.

Summary of Current Technology — Current technology for reactivity control systems cannot be directly extrapolated to cover the 1000-MWe LMFBR system design, since the conditions

imposed by the 1000-MWe design are more severe. The state of the art has been discussed earlier and summarized adequately in section 3 of the LMFBR Program Plan.

Scope of Work — The results of the engineering and experimental efforts of the preceding tasks will be used in the detailed design and fabrication of the prototype reactivity control system. This task also includes the detailed definition of the performance and proof-test programs. The required test facilities will also be designed and constructed.

The reactivity control system will be tested in an "open air" test facility to evaluate general performance, lift and load performance, scram evaluation, and the like; these tests will permit a relatively quick evaluation of the performance of the assembled components. The open air tests will be followed by performance testing in a static-water autoclave facility. The performance tests of the entire reactivity control system will include load and lift tests, scram testing, etc., under normal and abnormal conditions (e.g., power loss, misalignments). After the static-water autoclave testing, the performance of the system will be evaluated in a static-sodium autoclave test facility to correlate in-water and in-sodium test results.

To demonstrate operational adequacy under flow conditions, a dynamic water test facility similar to that being used for the development of PWR control rod drives and fuel assemblies can be used. Included will be stroking and scram testing under both normal and abnormal conditions such as misalignment, simulated credible incidents, and accidents. Removal, maintenance, and handling methods will be investigated. The results of these tests should provide the engineering data needed for the design of the final prototype. The necessary engineering modifications will be performed, and the system will be assembled for proof testing in a dynamic sodium facility.

The reactivity control system will be subjected to extensive testing in the dynamic sodium test facility. Evaluation of the final prototype design in this facility will include life testing, scram operation, and the like under normal and abnormal conditions. With these tests, the adequacy of the reactivity control system to operate in the high-temperature, flowing-sodium environment will have been demonstrated.

The final task will be to install the prototype reactivity control system in a test or demonstration reactor to demonstrate its reliability and performance under actual in-core and reactor environmental conditions. The demonstration will verify the adequacy of the system under conditions like irradiation and thermal gradients, which cannot be achieved in out-of-core facilities. The previous out-of-core development and testing should have demonstrated the functional adequacy of the system. The in-core performance testing is included in anticipation of the requirements of customers and/or regulatory bodies.

Technical Justification — The reliability of the reactivity control system cannot be demonstrated without performance and proof testing in a sodium environment.

Relationship to Current or Planned Programs — The design and operating relationship between the reactivity controls and the overall plant will be defined by the plant design program. The special measuring devices needed for R&D and operation of the reactivity control system will be developed by the instrumentation and controls program. Basic material information, e.g., compatibility data and structural properties for material applications where radiation and corrosion damage is anticipated, will be supplied by the fuel and materials and sodium technology sections of the program. Radiation effects and the degree of component activation will be determined by the physics R&D program. Safety-related functional requirements of reactivity control, including shutdown, will be delineated by the safety program.

Major Facilities and/or Equipment — The out-of-reactor performance and proof-test program can be carried out at the LMEC or similar facilities using a modified version of the environmental test vessel proposed in the core components handling equipment program (section 3.2.5). The preliminary "open air" test of the assembled components should be performed at the facility where the individual component tests are conducted, e.g., Alliance Research Center. The demonstration testing under actual in-core and reactor environmental conditions could be accomplished in a test or demonstration reactor system such as FFTF.

Budget Estimate —

Manpower	\$ 600,000
Facilities	500,000
Equipment	<u>400,000</u>
Total	\$1,500,000

Table 3-8 is a summary of estimated costs and a schedule of the work to be done on this phase of the reactivity control system program.

3.2.8. Reactor Vessel and Related Components

3.2.8.1. General

Objective — The general objective of this program is to optimize the design of the reactor vessel and related components, such as the cover structure, the core vessel, the internal support bridge with its attendant tension support members, the core support and holddown grids, the meltdown pan, and the shield tank. The specific objectives are to obtain from existing technology or determine from experimental tests the following:

1. Creep data, both axial and bending, from short-term and long-term testing to determine creep rates and establish primary creep laws.
2. True stress-strain information for elastic-plastic analysis and for fatigue analysis.
3. Strain hardening coefficients as a function of strain rate.
4. Cyclic fatigue data to determine cumulative damage; this includes the cyclic rate, relaxation, cyclic creep, hold time, and failure criteria.
5. Long-term, high neutron flux irradiation effects.
6. Loading conditions imposed by design basis accidents.
7. Reliable and economical methods for field fabrication and/or assembly of the reactor vessel and related components.
8. A method for nondestructive testing of individual and assembled components.
9. A configuration to prevent flow maldistribution within the reactor vessel, the core vessel, and the core itself.

10. Suitable analytical models and methods for accurate analysis and establishment of acceptable structure deformations, stresses, and vibrational frequencies.
11. Candidate materials and configurations for insulation.
12. The requirements for cooling.
13. The type and configuration of a machine capable of field machining large diameters to close tolerances.

Summary of Current Technology — The size of the reactor vessel (52 feet in diameter by 49.5 feet in depth) dictates field fabrication, because a vessel of this size is too large for transportation as an assembled unit. This generates fabrication and inspection problems. Although the technology of vessel fabrication with austenitic stainless steels is well-developed, the field fabrication technology for vessels of this size needs to be expanded to attain the desired tolerances, freedom from distortions, and the required 30-year life in a sodium environment with neutron flux irradiations. The state of the art of nondestructive methods (radiography, ultrasonics, and dye penetrant tests) for inspection of field-fabricated vessels needs to be supplemented by developing advanced equipment and techniques.

The feasibility of using flat, built-up reactor cover structures has been proved in several designs. However, the specific design of the 1000-MWe LMFBR cover structure will require an expansion of existing technology with concentrated effort to establish analytical methods to predict its behavior under conditions of static and transient loads and temperatures.

Existing design practice may be uneconomical and/or unconservative for the reactor vessel internals. Existing analytical methods for predicting the effects of creep and relaxation on deformation, creep buckling, and thermal buckling are inadequate.

3.2.8.2. Reactor Vessel

Objectives — The overall objective of the work described herein is to provide the necessary design information to ensure that the reactor vessel functional and safety requirements are met.

A portion of the R&D required to design the reactor vessel is discussed in the Fuel and Materials and Safety sections of this report. This information, the basic design data for material properties and accident criteria (e. g. , DBA), is classified as "go-no-go" R&D. Specifically, the following information is required:

1. Materials data (sodium environment and 30-year life)
 - a. Creep data, both axial and bending, initially from short-term testing and ultimately from long-term testing.
 - b. Minimum creep rates and primary creep laws.
 - c. True stress-strain information (for elastic-plastic analysis and preliminary fatigue analysis).
 - d. Strain-hardening coefficients as a function of strain rate.
 - e. Low-cycle fatigue data, including the effects of cumulative damage, cyclic rate, relaxation, cyclic creep, effect of hold time, and failure criteria.
 - f. Long-term, high neutron flux irradiation effects.
2. Loading conditions imposed by DBA, including pressure and shock values.

The specific objectives of this program are as follows:

1. To establish a reliable and economical method for field fabrication of the reactor vessel.
2. To determine methods for inspection, including nondestructive testing techniques.
3. To determine requirements and methods for testing the complete assembly.
4. To develop a configuration that will ensure against flow maldistribution in the reactor caused by inlet or outlet plenum geometry effects.
5. To determine the flow distributions under conditions of single and multiple primary pump failures.

Summary of Current Technology — The technology of vessel fabrication processes for the austenitic stainless steels is well-developed. However, existing technology for the fabrication of large structures with relatively thin walls, such as this reactor vessel, has not been developed to the degree required for a 30-year life in a sodium environment. The thin walls of the vessel make it difficult to attain the desired tolerances and freedom from distortion.

At present there are only a few nondestructive inspection methods for fabrication inspection. Methods such as radiography and ultrasonic and dye penetrant tests are limited, which makes the need for advanced equipment and techniques important.

Considerable experience has been gained in the empirical approach to determining flow distributions in large reactor vessels where the fluid is water. No large-scale sodium tests have been conducted, but it is felt that water tests should give acceptable accuracy in determining the flow distribution.²⁴ Complete three-dimensional velocity maps can be determined experimentally by measuring velocities using noise analysis techniques.

The arrangement of the inlets and outlets of the IHXs provides conditions that are conducive to unbalanced and unstable flow in the vessel. These characteristics make it difficult to obtain a consistent and desirable flow pattern within the vessel.

In the development of the distribution system for the PWR, considerable experience has been acquired that can be applied to this design to minimize the problem of obtaining the desired distribution characteristics.

It is recommended that the reference design be examined in the light of PWR experience to develop a preliminary design for the inlet plenum. The adequacy of this design should then be confirmed and necessary refinements made through flow model studies.

Scope of Work — The development program for reactor vessels falls entirely under section 3 of the LMFBR Program Plan. Specifically, the following items will be investigated:

1. Develop a reliable and economical method for field fabrication, which will provide the same degree of confidence and quality

control now attainable with shop-fabricated vessels. Included in this program will be an evaluation of both partial and full on-site fabrication, evaluation and development of welding procedures and the development of models, if necessary, which can be assembled so that erection criteria and procedures can be established.

2. Develop techniques for inspection of the reactor vessel based on present nondestructive testing equipment. Included in this program will be techniques for weld inspection and tolerance verification.

3. Perform a thorough study to establish analysis methods and testing requirements for the complete assembly. Develop methods to accomplish the required testing.

4. Develop advanced equipment and techniques for nondestructive inspection, which will replace or supplement present state-of-the-art methods. The objective is to increase the statistical reliability of inspection methods, thereby decreasing the need for post-installation testing.

5. A scale model of the prototype primary flow path will be tested. The fluid will be water at temperatures from 200 to 250 F. Smaller model tests in flowing sodium will aid in determining the importance of maintaining the 800 F sodium dynamic or kinematic viscosity in the water tests. Fluid flow rates (scaled) will range from approximately 50 to 125% of rated core flow and from approximately 50 to 200% of the flow expected under conditions of natural convection core cooling. Complete fuel assembly mockups will not be required, i. e., only the fluid resistance need be simulated. The thermal shield and leakage flow paths will not have to be included, since the two flows combined represent 1.5% of the total.

The pseudo-steady flow distributions in the vessel with one or two primary pumps inoperative is of interest. Scaled flow rates in the remaining primary loops (for the test condition) will range from 100 to 125% of the design value.

Required test data include the fluid velocities at the fuel assembly entrance nozzles for each test condition. These data will serve as the measure of design success. Intermediate data,

which will serve to pinpoint development problems (the actual cause of flow maldistribution of the fuel nozzle entrances) will be visual flow observations and photographs, as well as noise analysis and velocity measurements at selected locations.

It is expected that low-temperature water could be used for these studies, and that the fuel assemblies could be simulated by tubes providing the proper flow resistance and having integral flow meters to determine flow distribution. This model would also incorporate the radial shield portion of the reactor system and provide the means for confirming flow calculations in this area.

Technical Justification — Inspection and fabrication methods need to be advanced to ensure that the reactor vessel is built to fulfill LMFBR requirements. Critical items are welding, forming, chemical cleaning, and their effects on material properties. Definitive criteria are needed to accurately predict performance and for the establishment, evaluation, or verification of the analytical methods used in the design.

Relationship to Current or Planned Programs — The material information essential to the reactor vessel development program should come from the fuels and materials and sodium technology sections of the LMFBR Program. In addition, the Combined United Reactor Vessel (CURV) study being conducted by the LMEC should provide information regarding the treatment of the DBA, as well as information about seismic effects, material considerations, containment philosophy, inspectability, maintainability, and the applicability of the ASME Pressure Vessel Code to the LMFBR concept. Some of the models developed to carry out the flow tests described in this program can be used during the reactor vessel internals program that follows.

The technology and equipment required to field machine the support flange of the reactor vessel will be developed by reactor cover structure program (section 3.2.8.3).

Budget Estimate —

Manpower	\$1,300,000
Facilities	200,000
Equipment	<u>100,000</u>
Total	\$1,600,000

Table 3-9 is a summary of estimated costs and a schedule of the work required on the reactor vessel and related components.

Table 3-9. Summary of Schedule and Estimated Costs for R&D Work on the Reactor Vessel and Related Components

Task	FY CY	Estimated cost, \$ × 10 ³									Total
		1970	1971	1972	1973	1974	1975	1976	1977		
		1969	1970	1971	1972	1973	1974	1975	1976	1977	
Reactor Vessel and Related Components											
1. Reactor vessel		150	230	240	350	200	200	150	80	--	1600
2. Cover structure		100	80	230	190	140	180	80	50	--	1050
3. Reactor vessel internals		110	180	70	70	40	--	--	--	--	470
Total		360	490	540	610	380	380	230	130	--	3120

3.2.8.3. Reactor Cover Structure

Objective — The overall objective of the R&D described herein is to provide the necessary design information to ensure that the reactor cover structure functional and safety requirements are met. This R&D is planned on the present reference design; however, it is recognized that a good deal of engineering is required on the reactor vessel and cover structure before the concept can be fixed for preliminary design analysis. This engineering may significantly affect the concept layout and consequently, different interfaces with other functional parts of the reactor coolant system may be created. The work may also uncover new R&D requirements needed to complete the development of the design.

The designer's needs in the area of R&D may be grouped into two broad categories—basic design data (e.g., materials data) and performance data from the development and proof-testing of specific designs. Within each of these categories two classes of information exist: that without which the requirements of the design could not be met, and that which leads to a more economical design.

A portion of the R&D required to design the reactor cover structure is discussed in other parts of this document. This

information is basic design data for material properties and accident criteria (e.g., DBA) and is classified as go-no-go R&D. Specifically, the following information is required:

1. Material Data

- a. Creep data, both axial and in bending, initially from short-term testing and ultimately from long-term testing.
- b. Minimum creep rates and primary creep laws.
- c. Cyclic fatigue data for 30-year life.
- d. True stress-strain information for elastic-plastic analysis and preliminary fatigue analysis.
- e. Strain-hardening coefficients as a function of strain rate.
- f. Low-cycle fatigue data, including the effects of cumulative damage, cyclic rate, relaxation, cyclic creep effect of hold time, and failure criteria.
- g. Long-term, high neutron flux irradiation effects.

2. Loading conditions imposed by DBA, including pressure and shock values; the specific objectives of this program are as follows:

a. Structure

- (1) To develop suitable models and methods to permit accurate analysis of the structure and the establishment of acceptable structure deformations.
- (2) To determine the combined thermal and mechanical stresses, including creep effects, so that member sizes may be chosen.
- (3) To find member and structure natural frequencies and compare them with acceptable values.
- (4) To specify displacement boundary conditions and analyze elastic-plastic problems.

b. Insulation

To determine the candidate materials and configurations for the reflective and/or bulk insulation required for the structure, and to determine the requirements for cover cooling.

c. Fabrication

- (1) To develop and proof-test a machine capable of field-machining large diameters to close tolerances.
- (2) To develop methods for dimensional control and field adjustment of interfacing components of the structure.

Summary of Current Technology —

Structure: Flat, built-up reactor cover structures have been used and proved in several designs; the reference 1000-MWe structural design is an expansion of existing technology. The cover is a massive assembly, 50 feet in diameter by 10 feet thick. It has many large and small penetrations to permit the installation of major system components. The cover contains sufficient shielding material (serpentine aggregate and steel shot) to reduce radiation levels at the operating floor to the required limits and to protect installed equipment. The cover structure provides the support needed to maintain the complete assembly in its proper functioning order against all but major accident (DBA) conditions.

The structure consists of an outer fixed section and inner rotating plugs. Basically, each section is made up of top and bottom plates separated by stepped, cylindrical ring beams and internal structural members. Each access port is formed by a stepped, cylindrical liner welded into its respective structural cover section. The assembly contains shear rings and locking devices to ensure retention of the plugs and all installed equipment during normal, abnormal, and accident conditions.

These characteristics are essentially state of the art. The design details of the specific items require expansion of this technology with a concentrated effort to establish mathematical models and analytical methods to accurately predict behavior. The cover assembly is larger than existing designs and has more, larger rotating plugs. The conditions of reactor operation, especially those resulting from accident conditions (DBA), impose structural requirements on the assembly that increase the emphasis on the accuracy of analytical methods.

The finite-element method of analysis has recently been applied in analyzing general plane stress and plane strain; it has more recently been extended to the analysis of general plate, shell, and frame problems. The analysis involves the derivation of appropriate element stiffness matrices which approximate the state of strain within the element, the superposition of the element stiffnesses to obtain the stiffness of the assembled continuum, and the solution of

the equations representing force equilibrium at the nodal points to obtain the nodal displacements. The stress distribution may be obtained from the strain conditions within the elements after the nodal displacements have been found. The approximate member and structure natural frequencies can also be obtained.

Insulation: No existing technology is available to completely cover the requirements for the insulation in the reactor cover structure. However, these requirements may be fulfilled by the use of a combination of items of known technology. An alternate solution may be a cooling system. The insulation requirements of the reference 1000-MWe LMFBR design are as follows:

1. An insulation package or cooling system to reduce the reactor temperature to a maximum of 800 F at the lower face of the serpentine shielding material.
2. The insulation must perform its function reliably over the 30-year life of the plant. It must function in the sodium vapor environment or be protected from it. Its performance must not be reduced by the collection of sodium solids or fission product solids on its face. Its performance must not be jeopardized by nuclear interactions, nor shall such interactions result in unsafe reactor conditions or conditions that might cause failure, decreased reliability, or maloperation of the cover structure or reactor equipment.
3. The alternate cooling system may be completely effective alone, or it may be used in combination with insulation. The requirements for either system are identical to those for insulation material.

Some current technology is available for use in designing metal reflective insulation which makes use of parallel plates assembled with intermediate non-conducting spaces. Each surface of the plates acts as a heat-reflecting surface, and the assembly is engineered to allow for surface expansion or contraction and still maintain the enclosure, in order to provide a very effective heat barrier. The reflective plates and supporting structure can all be of high-strength stainless steel to withstand exposure to high-intensity nuclear radiation with negligible physical deterioration and, therefore, provide

a long insulating life expectancy. This reflective insulating structure is relatively unaffected by thermal shock and vibration.

Because the entire supporting structure and insulation is constructed of austenitic stainless steel, this product has better resistance to earthquake and shock than other types of insulation.

For the present cover structure configuration, bulk insulation may be a candidate, but since insulation may be needed on the sodium side or the bottom of the structure (which has a temperature of 1100 F), R&D will be needed to find a suitable material for this environment.

Some bulk materials, such as Kaolin Wool, which consists mostly of alumina-silica, have been tested at 850 F in an inert atmosphere, dipped in sodium for 10 minutes, and then preheated to 840 F for 16 hours. The results were a complete charring and deep sodium penetration. The same tests performed at 500 F and 350 F showed no sodium penetration.

The present configuration, which consists of a bottom steel plate with the insulation above, could still incorporate alumina-silica insulations into the design, but methods of accommodating differential thermal expansion would have to be investigated.

Some new types of insulating material with thermal conductivities less than half that of conventional insulations will also be investigated for this application. The insulation development was predicated on the theories of mechanisms of heat transfer which are of prime importance insofar as high-temperature insulations are concerned. One mechanism is molecular gas conduction, in which heat is transferred by collisions or interactions between air molecules having different energies, and another mechanism, thermal radiation, in which heat energy is transferred as infrared or visible electromagnetic radiation. By utilizing high-density insulating materials, it has been possible in some instances to reduce the radiant-energy component of heat transfer to a low value, thereby reducing the thermal conductivity to about that of still air.

The technology for a gas cooling system is well established, but gas distribution methods will require proof testing.

Fabrication: The reactor cover is substantially larger than existing designs and has many penetrations of various sizes.

The cover is 59 feet in diameter and 10 feet thick; the largest rotating plug has a diameter of over 32 feet. Although methods and machinery exist for fabricating and erecting structures somewhat similar to the cover, further development work is needed to satisfy the specific needs of the structure.

The structural members are all type 304 stainless steel, which complicates both field welding and machining. Reliable machinery is not now available to machine the necessary diameters to the desired tolerances, but the technology for developing such machinery does exist. Similarly, the basic technology may exist for controlling roundness and concentricity, but the selected methods must be confirmed experimentally to ensure the reliable performance of the cover structure.

Scope of Work — The R&D areas which are peculiar to the reactor cover structure are outlined below. These efforts fall entirely under section 3 of the LMFBR Program Plan. In order to attain the overall objective of fulfilling the structure's functional and safety requirements, the following efforts are essential:

1. Perform engineering and analytical work to accurately define the requirements and environment of the reactor cover structure, and establish a mathematical model and methods for accurately predicting the behavior of the structure under normal and accident loading conditions.
2. Since the number of variables and structural complexities associated with the cover structure would take an unreasonable amount of time to solve by hand calculation, the development of a suitable computer program will constitute a major part of the analytical effort. The method for structural analysis will probably be a program based on the finite-element method.
3. Develop the necessary field fabrication machinery for successful construction of the reactor cover structure. Proof-test the equipment, using full-sized sections representative of pieces to be field-machined, such as the sealing and bearing surfaces for the large rotating plug.
4. Develop methods for controlling roundness during the welding of liners to large structural members. Investigate controlled

weld-shrinkage techniques, intentional out-of-round machining prior to welding, and the feasibility of final machining after complete installation of the liners. Develop methods for maintaining the concentricity of mating parts during erection.

5. Perform preliminary design and proof-testing of methods for providing field adjustment of critically aligned components. Investigate methods that would allow adjustment after initial heatup of the reactor and the potential hazards of the system, such as increased leakage through the cover structure. Proof-test selected designs, including the effects of any welding or large loadings that may occur after final adjustment of the components. The methods should supplement the field machining to be developed under item 1 by providing closer alignment of components, as well as permitting unanticipated adjustments in the field. The methods developed must be proof-tested.

6. Determine the cooling requirements of the cover structure, and design and test a model of the gas coolant distribution system.

7. Select candidate insulating materials by conducting a literature survey, and test these materials for compatibility with the requirements and environment established under item 1.

Technical Justification — More data are needed to predict the structural behavior of the cover structure with its attendant shielding, cooling, and insulation. Further development of the requirements of the cover structure components is needed to permit material selections and configurations to be made.

Relationship to Current or Planned Programs — The field-machining methods and equipment developed for this program will be applicable to the reactor vessel and possibly to some of its internals. In any case, the machining methods should prove useful in the field fabrication of all large reactor vessels and cover structures and other large field-fabricated equipment that requires machining.

The basic material properties, such as allowable stress, creep rates, fatigue life, etc., required for the completion of this program will be supplied by the sodium technology and fuels and materials programs. The safety-related functions of the cover structure

will be specified by the safety program, and the influences of fuel handling and plant operation will be covered in the plant design section.

Major Facilities Required — None.

Budget Estimate —

Manpower	\$ 750,000
Facilities	--
Equipment	<u>300,000</u>
Total	\$1,050,000

Table 3-9 is a summary of estimated costs and a schedule of the work required on the reactor vessel and related components.

3.2.8.4. Reactor Vessel Internals

Objective — The overall objective of these activities is to develop the reactor vessel internals to perform as a system in the reference LMFBR plant. The program will include all of the engineering design and analytical work, all the basic R&D effort, and all the model, prototype, and life testing required to advance the conceptual design to the state where fabrication, installation, and operation in the reference plant can be accomplished successfully. These activities must be performed on a schedule that is compatible with the overall LMFBR Program and, more specifically, to the schedule of the demonstration and predecessor plants as given in the schedule accompanying this R&D report. To fulfill these objectives, the reactor vessel internals must conform to the following conditions:

1. An expected operating lifetime of not less than 30 years is required.
2. Internals must be compatible with the LMFBR environment.
3. The design bases of the internals must be consistent with code methods (i.e., ASME) for determining allowable design stresses.
4. The reactor vessel internals must be compatible with all reactor systems and with the refueling methods.
5. Internals must be consistent with the economical power generation goals of the plant.

The reactor vessel internals are a part of the reactor vessel system, as described in detail in Volume 2, section 2 of the 1000-MWe LMFBR Follow-On Study report, BAW-1328.¹ The activities required for thorough development of the reactor vessel internals are discussed in the following paragraphs. For this discussion, the identical breakdown given in the description is used:

1. Internals support bridge.
2. Core vessel.
3. Meltdown pan.
4. Grid plate.
5. Shield tank.

In this report, the descriptive data for these components will be limited to that necessary to clarify the R&D requirements. Reference should be made to BAW-1328, Volumes 2 and 3, where further description is required.

Summary of Current Technology — The technology for fabrication, assembly, testing, and use of the reactor vessel internals is essentially within the existing state of the art. The major technology gap is the lack of analytical methods that predict the requirements that will be placed on the assembled units and allow accurate analysis of the structures.

The design of the shield tank is outside existing technology; this is because it is an assymmetric shell under creep conditions, and existing technology provides no way to model creep deformation accurately.

Scope of Work — The scope of work for the reactor vessel internals includes formulating analysis methods to predict the effects of creep and relaxation on deformation, creep buckling, and thermal buckling. The effects of radiation and sodium on materials (see section 7) must be assessed and factored into the design.

Analytical models for accurate analysis of the reactor vessel internals will be established through literature surveys wherever possible and through tests where necessary.

Model design and testing will be required to confirm analytical calculations, to evaluate fabrication techniques, to simulate load distribution, and to establish flow patterns.

The analysis and testing of the internal support bridge and the core vessel will involve establishing analytical models for accurate analysis and mockup to simulate mating and local conditions.

The effort to accurately define the shield tank will include developing analytical methods and building and testing models to determine load distributions and establish coolant flow paths. The modeling will include mockups of the IHX and upper neutron shields to establish flow patterns into shield tank blisters and IHXs.

Suitable analytical models will be developed to permit accurate analysis of the grid plate and core holddown assembly structure and flow patterns. Analysis will establish orificing size and flow characteristics through the structure and the flow and temperature effects on the structure.

The particular technique will have to be tailored to particular temperature conditions and the possibility of thermal buckling must be investigated. It is felt that a structural model of the structure would be justifiable. Although the grid plate structure is similar to the core holddown assembly, the latter will require a more extensive analysis, because it is in the temperature range where creep deformation may occur. At this time there is no way to model creep deformation; therefore, creep analysis will have to be based strictly on analytical procedures and on supplementary experimental verification on simpler models.

Technical Justification — To compensate for deficiencies and uncertainties in the existing technology, adequate analytical methods, based on experimental testing, will have to be developed to support the design and to predict the performance and the reliability of the reactor vessel internals.

Relationship to Current or Planned Programs — The range of member sizes, materials, material thicknesses, configurations, and operating conditions of the reactor vessel internals generates the need to review practically all R&D programs associated with sodium reactors. Theoretical methods for predicting the dynamic behavior of structures developed during other programs may cover the effects of live loads caused by fluid velocity, pressure surges, and the like, and by seismic disturbances or induced vibrations.

Some of the models developed to carry out the flow tests given in the reactor vessel program can be used during this program.

Major Facilities Required — None.

Budget Estimate —

Manpower	\$420,000
Facilities	--
Equipment	<u>50,000</u>
Total	\$470,000

Table 3-9 is a summary of estimated costs and a schedule of the work required on the reactor vessel and associated components.

3.2.9. Component Surveillance

3.2.9.1. Objective

The overall objective of this task is to determine the surveillance requirements for component systems of the LMFBR plant. Particular attention is given to components that are to be designed for the expected 30-year plant life. The specific components considered are those of the B&W 1000-MWe reference plant, although the results of the proposed program will be of overall benefit to all LMFBR designers. The specific objectives of the program are as follows:

1. To determine the surveillance requirements for the major components of the primary and secondary sodium systems.
2. To establish the criteria for accomplishing the surveillance; that is, to determine what information is desired, what portion of the component must be examined, and the frequency of the examination.
3. To develop the necessary instrumentation and/or procedures to accomplish the required surveillance.

3.2.9.2. Summary of Current Technology

There are almost no specific requirements for the surveillance of LMFBR components, nor are there programs specifically designed to determine these requirements. The primary reason

for this deficiency is the fact that the National Program objectives have been concerned mainly with obtaining basic physics, safety, sodium technology, and breeding data. The LMFBR Follow-On Studies will do much to clarify surveillance requirements, but existing and near-term fast reactor designs will still be aimed primarily at the objectives above, in addition to gaining valuable fast reactor operating experience.

To date, the following fast reactor design approach has been generally used:

1. To design sodium-cooled and fast reactors to obtain physics, safety, sodium technology, and operating data.

2. To consider plant availability for the production of electric power and plant economics only as secondary objectives, and therefore, to use conservative and even ultra-conservative component systems, if necessary, to minimize the requirements for inspection or surveillance.

3. To consider the lives of the plant and components also as secondary goals, as opposed to the possible economic requirement for a 30-year life for the LMFBR.

4. Where inspection of components was deemed necessary, it was usually considered more expedient to plan for reactor shutdown to permit such inspection. Generally, state-of-the-art equipment and instrumentation have been used, even if periodic replacement was considered necessary, rather than waiting for the design and development of long-lived items. Using similar reasoning, it has also been considered better to plan shutdowns for extracting components for inspection, or even draining of the sodium, to allow the use of inspection techniques that are more nearly routine.

This design approach is not debated—in fact, it has permitted and will continue to permit the collection of valuable experience and data. Sodium technology and radiation effects will be especially important. Good or bad, these results will permit LMFBR component designers to approach their task with a higher confidence level. Inspection or periodic surveillance procedures can be planned, if required, and the component design will consider factors that will

permit these procedures with a degree of safety and reliability not now possible. It will also be possible to determine the effects on plant economics.

The Program Plan discusses the technology developed at the SRE, Hallam, EBR-II, Fermi, and Dounreay reactors.²⁸ These reactors have now compiled many hours of operation and have experienced relatively little trouble with components that are subjected to the sodium environment. The problems that have been encountered apparently have been due to faulty fabrication, such as the pipe weld in Dounreay. The Program Plan points out that extrapolation of this technology to the sizes required for the commercial plant—1000 MWe and larger, a 1200 F sodium temperature, and a 30-year life—is not prudent. The B&W reference 1000-MWe design is, however, quite similar in some aspects to the fast reactors that have been built. The operating conditions of the reference reactor vessel, primary pumps, and IHXs may be especially important. Consideration of the larger sizes is a problem, but the operating conditions of temperature and neutron fluence are quite similar in many cases to those of the EBR-II, Fermi, and Dounreay reactors, and experience with these reactors will do much to help determine the surveillance requirements of the B&W design.

Considerable work is also being accomplished in the light-water reactor programs to develop surveillance and inspection techniques and equipment. Although much of this effort seems to be priority at the time, indications are that emphasis is on the use of ultrasonics techniques to inspect such items as vessels, piping, and nozzles. The information developed will prove valuable to the LMFBR program as its requirements are established. Among programs for which information has been published, the under-sodium ultrasonic scanner system may prove especially applicable.³⁵ If this program is completed successfully, some of the surveillance problems of the LMFBR may be alleviated.

3.2.9.3. Scope of Work

The component design used in existing and near-term fast reactors was called conservative or even ultra-conservative. The effort described in this task will define the programs that are needed

to support the continued use of such conservatism, or, alternately, to specify the equipment and techniques required for inspection. The various means of ensuring component reliability must be compared to determine which method is the most desirable in each situation. That is, remote surveillance, periodic inspection, provisions for adequate safety (such as double containment), or a combination of these are all available to the system designer. The relative merits of each scheme must be evaluated for each area in question. Part of the purpose of this program is to provide the information necessary to make such an evaluation. The specifics of the program are as follows:

1. Review the following components to determine surveillance requirements:

- a. Reactor vessel and internals (excluding core).
- b. Reactor cover structure.
- c. Steam generator and reheater.
- d. IHX and NaK-to-air heat exchanger.
- e. Pumps.

Analyze and evaluate design data to determine the critical areas requiring surveillance or monitoring. Perform a failure-mode analysis of the components, assessing the potential causes of failure, critical areas of plant performance and safety, and the potential for detecting failures or impending failures. Items such as cracking, stress corrosion deterioration of the mechanical and chemical properties of materials due to environment, thermal expansion, flow size propagation, changes in stress patterns, etc., will be considered in the evaluation of surveillance requirements.

2. Utilizing the foregoing analyses and evaluations, determine the criteria for accomplishing the required surveillance. If more than one indication of incipient failure could occur in a given location, determine which are the most easily detected, and whether these alone are sufficient for surveillance. Identify the specific areas of concern, the information that could be gathered from these areas to indicate incipient failure, and the frequency at which information should be gathered to ensure detection before failure.

3. Evaluate existing inspection and instrumentation methods, including such environmental considerations as materials compatibility, temperature limitation, corrosion/erosion effects, and the like. Perform experimental and screening tests to determine the suitability of candidate instrumentation.

4. Based on the results of other tasks in this project, determine the requirements for developing the new equipment, instrumentation, and techniques necessary to fulfill the surveillance requirements. This task will include the design and engineering efforts required, along with the fabrication and testing of model or prototype equipment. Of major importance to this program will be the progress attained in the "Viewing Through Sodium" project proposed in section 4.2.2.9 of this report (Instrumentation and Controls). In conjunction with the development of postulated surveillance equipment requirements, only a moderate adaptation of the viewing apparatus has been estimated.

3.2.9.4. Technical Justification

The need for component surveillance stems from the lack of data on the long-term environmental effects of sodium and radiation on material properties. The designer cannot be sure that his design is truly conservative without the basic material data, and therefore, until these data become available, component surveillance will be necessary. Secondary justification stems from the lack of complete confidence in fabrication technology. Problems with existing fast reactors—and for that matter, with thermal reactors—have usually been caused by faulty fabrication and equally faulty quality assurance techniques. This situation may be corrected, but component surveillance may be mandatory even after the confidence level in these techniques is extremely high.

3.2.9.5. Relationship to Current or Planned Program

At this time, no existing program is apparently underway that will accomplish the prescribed objectives. The Program Plan (Task 3-9) has outlined the problems of the reactor vessel and has discussed a program to solve these problems.²⁸ To our knowledge, this work has not been started. Two programs have already been mentioned

that have a direct relationship to the program outlined; the first is the APDA project,³⁵ and the second is the proposed project from section 4.2.2.9. The efforts being performed for surveillance of PWR components and the programs which will undoubtedly be funded to determine FFTF vessel surveillance requirements are also of potential importance to this work.

3.2.9.6. Major Facilities and/or Equipment Required

No major facility is required for development of equipment to perform nondestructive testing of the reactor vessel (pot). It is anticipated that a facility like the Alliance Research Center or the Nuclear Development Center will serve this purpose. The development of equipment to inspect the cover structure can also be accomplished at such a facility. For equipment required to function within the sodium, however, a facility such as those available at LMEC will be required. Preliminary work is planned at ARC or NDC, but final development and calibration will be done in a sodium loop system where reactor conditions can be simulated.

3.2.9.7. Budget Estimate

Manpower	\$ 875,000
Facilities	200,000
Equipment	<u>225,000</u>
Total	\$1,300,000

Table 3-10 is a summary of estimated costs and a schedule of the work required for the component surveillance program.

Table 3-10. Summary of Schedule and Estimated Costs for R&D Work on Component Surveillance

Task	FY CY	Estimated cost, \$ × 10 ³									Total
		1970	1971	1972	1973	1974	1975	1976	1977		
		1969	1970	1971	1972	1973	1974	1975	1976	1977	
Component Surveillance											
Task I		50	50	75	50	--	--	--	--	--	225
Task II		--	--	--	50	--	--	--	--	--	50
Task III		--	--	75	100	--	--	--	--	--	175
Task IV		--	--	--	100	200	200	200	150	--	850
Total		50	50	150	300	200	200	200	150	--	1300

3.3. Cost Benefit Studies

3.3.1. Steam Generators and Reheaters

The present reference design is characterized by three steam generators and three sodium reheaters which generate steam at 2400 pounds and 950 F and also reheats it to 950 F. Assuming that the steam temperatures and flows are attainable, but that the intermediate sodium temperature (1000 F) cannot be reached, economic penalties result. The economic repercussions are explained in the following paragraphs.

Experience in analyzing the "fixed charges" portion of the total power generation costs for a reheat cycle has shown that, within a limited range of steam generator hot end temperature differences, the capital cost remains relatively constant. Calculations also show that for these conditions, the optimum sodium temperature leaving the steam generator remains fairly constant.

The temperature of the sodium entering the steam generator and reheater is 1000 F. If it is assumed that only 975 F is attainable while maintaining the reactor bulk outlet temperature at 1100 F, then the hot end temperature difference for the IHX will rise from 100 F to 125 F.

Optimization studies have shown that the temperature of the sodium leaving the reheater for minimum power generation cost is related to the intermediate sodium temperature difference by the equation $\Delta T_{RH} = \Delta T_{IHX} - 125$. Using this relationship, it can be seen that the optimum sodium temperature leaving the reheater will remain at 825 F if the inlet temperature to the reheater is decreased from 1000 to 975 F. Thus, for a given thermal duty, decreasing the inlet sodium temperature to the steam-raising equipment leaves the cold end temperatures of the intermediate loop relatively constant, increases the hot end ΔT for the IHX, allows the cost of the steam generator to remain relatively constant, and increases the total intermediate flow.

The major economic effect takes place in the reheater, where the LMTD decreases from 113 to 88 F, while the IHX LMTD is increased from 100 to 108 F. Since cost trends for the reheater were based on a constant overall heat transfer coefficient and a constant cost

per square foot of heat transfer area, the total capital cost can be expressed by the following equation:

$$C_{RH} = HTA_{RH} \text{ \$k/ft}^2 = \frac{Q_{RH}}{U_{RH} \Delta T_{MRH}} \text{ \$k/ft}^2$$

This equation can be reduced to

$$C_{RH} = HTA_{RH} \frac{\Delta T_{MRH}}{\Delta T_{MRH_2}}$$

where

$$\begin{aligned} C_{RH_2} &= \text{new capital cost of the RH} \\ C_{RH_1} &= \text{reference plant cost of the RH } (\$7.1 \times 10^6) \\ \Delta T_{MRH_1} &= \text{reference plant LMTD (113 F)} \\ \Delta T_{MRH_2} &= \text{new LMTD (88 F)} \end{aligned}$$

Other effects that tend to change the fixed charges portion of the plant are minor and interdependent and thus generally tend to cancel each other out. For example, the increased LMTD of the IHX tends to decrease the capital cost of the IHX. But the increased flow of the intermediate system (due to the decreased IHX temperature difference) increases the pumping power and, hence, the capital cost of the intermediate pumps, increases the cost of the piping (because of the increased diameter), and increases the cost of valves (because of increased piping sizes).

The cost of the reheaters in the reference plant is $\$7.1 \times 10^6$. The cost of a reheater with a 975 F sodium inlet temperature would be

$$C_{RH_2} = \$7.1 \times 10^6 \frac{113}{88} = \$9.1 \times 10^6$$

The increased cost in fixed charges amounts to approximately 0.03 mill/kWh for a decrease of 25 F in the intermediate sodium system.

3.3.2. Intermediate Heat Exchanger (IHX)

The reference plant arrangement includes six gravity-flow IHXs within a reactor vessel. The level differences between the hot and cold leg provide the driving force for the reactor coolant.

The geometric configuration was determined by considering the duty, the heat transfer, and the hydraulic characteristics of the unit in conjunction with temperatures and the effect of the heat exchanger cost on total power generation. Other vital relationships, on which no quantitative values were available, were also considered; these factors were resolved qualitatively. For example, it was known that, as the diameter of the heat exchanger was allowed to increase, the vessel increased in diameter, the cover structure increased in diameter, the sodium inventory increased, and the reactor building diameter increased. On the other hand, if the heat exchanger diameter were allowed to decrease, then the reactor vessel increased in height, the reactor building increased in height, and the reactor coolant pump shafts increased in length.

Two points are noteworthy in these considerations. First, there is an economic trade-off, and second, there is an increase or decrease in the probability of success in component development.

The design study has revealed that it is not so much a question of the credibility of developing a heat exchanger, but whether or not the heat exchanger's prerequisites change the configuration of its surroundings. For example, should it be discovered in the course of the R&D that the reactor coolant stagnates as it proceeds through the shell side of the IHX, any proposed solution that either adds pressure drop to the system or changes the diameter of the heat exchanger will, in turn, change the configuration of the reactor vessel and some of its interfaces.

The considerations that affect the development of an economical and reliable IHX, as related to cost benefits, fall into three broad categories:

1. The development of the heat exchanger itself, i.e., the solution to fabrication, structural, performance, and maintenance problems.
2. The effects of partial attainment of item 1 on the rest of the system in terms of economics

and the probability of successful development of components so affected.

3. The effect of partial attainment of the analytical relationships that will ensure the predicted performance, economics, and reliability of large units.

The following is an assessment of these three categories:

3.3.2.1. Heat Exchanger Development

1. Item: Tube-to-Tube-Sheet Attachments

Partial Attainment — This will affect reliability and, hence, downtime in the neighborhood of \$15,000 to \$50,000 per day. In addition, it could mean the production of contaminated coolant from the nonradioactive intermediate sodium. Further, it presents the possibility of contaminating the intermediate system. Lack of reliability will force the investment of expensive spare parts.

2. Item: Maintenance

Partial Attainment — The reference design heat exchanger must incorporate the means necessary for its installation and removal in a reasonable amount of time, giving due consideration to the problems associated with a radioactive sodium system. Should this prove to be impractical (due to the cost of downtime), a change in maintenance philosophy may be required, such as repair in place by remote means. This, of course, would mean a new design concept accompanied by a probable increased plant capital cost and an increased R&D cost.

3. Item: Reactor Coolant Shutoff Device

Partial Attainment — This would downgrade plant efficiency in the event that the plant would require operation with less than the full complement of heat exchangers because of excessive leakage due to closing problems.

Non-Attainment — Would prevent the plant from operating with less than six IHXs.

4. Item: Vibration Effects

Partial Attainment — This could require the plant to operate at partial loads to prevent excessive damage.

5. Item: Maldistribution

Partial Attainment — Maldistribution will cause a reduction in plant operating capacity by affecting performance.

6. Item: Heat Transfer Coefficients

Partial Attainment — Within certain limits, the overall heat transfer coefficient can be increased by increasing the tube-side flow. This will increase the pressure drop and pumping power on the intermediate side, resulting in increased capital cost for the pumps and a somewhat downgraded plant efficiency. The shell-side coefficient can be increased by increasing the cross flow or reducing the ligament. The former causes an increased pressure drop, which will change the height of the vessel; the latter could cause either a change in the required pressure drop or a change in vessel diameter.

7. Item: LMTD

Partial Attainment — The reference design hot and cold end temperature differences were established at 100 F, resulting in an LMTD of 100 F. The results of an analysis of a higher and lower LMTD are shown below.

Case	T_{RI}, F	T_{RV}, F	$LMTD_{IHX}$	HTA, ft^2	wt_{IHX}, lb	Cost/IHX \$	Fixed charges, mills/kWh	Total, mills/kWh
1	800	1150	125	8,385	131,100	598,000	2.205	3.432
2 (Design point)	800	1100	100	10,388	146,900	670,000	2.199	3.438
3	800	1050	50	21,844	209,100	953,000	2.224	3.479

The tabulation above indicates that, for the six heat exchangers, the total capital costs for Case 3 is \$1,700,000 higher than for Case 2. The increased total power generation cost for this comparison is +0.041 mill/kWh. The power generation cost for Case 3 is greater than that for Case 2 because of the higher cost of the heat exchangers and the greater capital cost associated with the increased core flow. Because of the complex changes that must occur when the LMTD is changed, there is no cost incentive for going to a higher LMTD (125 F), as demonstrated in Case 1.

8. Item: Prototype Test Instrumentation

Partial Attainment — The conclusive verification of all performance and structural integrity lies in the successful operation of the prototype. Partial attainment of the test instrumentation requirements for the prototype would result in uncertainties in the performance and structural integrity of the full-sized unit. Overdesign (conservatism) to offset these uncertainties would entail additional capital cost.

3.3.2.2. Effect of IHX Performance on the Plant

How the rest of the plant is affected by partial attainment in the development of the heat exchangers has been discussed. However, further discussion of nonattainment of the required IHX performance is required.

If it is assumed that the performance requirements can be met by an increased pressure drop through the shell side, then the difference in height between the hot and cold leg would increase. Since the reactor coolant pumps are located in the cold leg, this situation would lead to an increased pump shaft length and diameter and the attendant difficulties in seal development. The major difficulties can be characterized as those which relate to thermal expansion and to both radial and thrust bearings.

The need for a greatly increased pressure drop to meet performance may dictate the requirement for a head on the reactor with piping to channel the flow to the heat exchangers. This, of course, would mean a different kind of fuel handling system.

If it is assumed that required performance can be attained by employing an increased LMTD, holding a reactor outlet temperature of 1100 F, then the intermediate coolant temperature, and hence the throttle steam temperature, would have to be reduced, with a probable decrease in steam pressure. This would undoubtedly mean a decrease in net efficiency of at least a percentage point. Further, it would mean an increased capital cost for the turbine island and an increased fuel cost.

3.3.2.3. Analytical Methods Development

The development of analytical methods for predicting performance and structural integrity mainly influences the direction and cost of the R&D required to develop large units from data on smaller units. Here again, the confidence in design information developed from empirical relationships decreases, resulting in increased capital costs due to overconservatism.

3.3.3. Piping

The piping in the intermediate system is Croloy 2-1/4; its size is based on a sodium velocity equal to 25 fps. The reference design employs expansion loops to account for thermal growth. This combination of material and velocity provides a piping design that has reasonable moment and thrust values.

Among the many parameters that require investigation to establish design information, erosion is one of the most important. It is a function of velocity, and as such, sets the diameter of the pipe. Partial attainment of the 25-fps velocity for the design flow rate will require (1) large pipe diameters and increased support problems, (2) longer expansion loops, (3) more insulation and more heating lines, (4) a larger sodium inventory, (5) a more complex, double containment within the reactor building, (6) a less compact plant (i.e., a larger building), and (7) larger valves and the like. The increase in diameter caused by partial attainment of the velocity may dictate the use of more than three intermediate loops. The anticipated R&D will provide an insight to these uncertainties.

The reference design requires the outlet sodium streams from both the steam generator and the reheater to join and give a uniform-temperature sodium stream to the pump suction. The exit sodium temperatures from these two components create a temperature difference of 173 F at the junction. The line leaving the reheater is 26 inches in diameter, and that leaving the steam generator is 32 inches in diameter. These lines join to form a 36-inch line for the intermediate pump suction.

Partial attainment of the R&D needed to achieve a 173 F temperature difference at full load will require a decrease in the reheater sodium outlet temperature and an increase in the sodium outlet temperature

from the steam generator, producing a decrease in the difference between the outlet temperatures from the two units.

This situation will result in a decrease in the capital cost of the steam generator and an increase in the capital cost of the reheaters. The net result will be a total increase of the fixed charges part of the total steam generator cost. Figure 3-2 is a plot of the temperature difference between the sodium streams entering the mixing sections (ΔT_{MIX}) versus the change in power generation costs for off-design-point values. This curve includes only the effect of changing capital costs of the steam generator and reheater. It does not include any changes that may occur in piping size, pumping power, valve sizes, and the like. These changes are considered to be minor.

Although Figure 3-2 shows that the reference design is at the optimum, the cost incentive to remain there should be re-evaluated in the light of actual R&D costs as well as a more detailed analysis of reheater costs.

Figure 3-2. Power Generation Costs Vs Reheater Sodium Outlet Temperature

3.3.4. Fuel Handling Procedures

This discussion is concerned primarily with the procedures of loading and unloading fuel assemblies from the reactor storage drums and shipping the spent fuel. The reference 1000-MWe LMFBR design proposes the following method:

1. Refuel one third of the core (fuel and blanket) each year.
2. Load new fuel assemblies into the storage drums over a period of 30 days just prior to the scheduled refueling, with the reactor either at full power or at some partial power level (off-peak load), but with no penalty for reactor shutdown.
3. Refuel the core (reactor off-line) by removing spent assemblies and placing them in the storage drums and then taking new assemblies from the drums and placing them in the core.
4. Retain spent fuel assemblies in the storage drums (reactor on-line) for a minimum of 70 days' decay.
5. Remove spent assemblies from the storage drums and place them directly into sodium-filled shipping casks (one per cask) and ship over a period of 80 days. Again the reactor is at full power or partial load, as discussed in item 2.
6. Receive new assemblies (possibly four per container) and store in the receiving area or (possibly) move them directly into the reactor building and into the storage drums.

If the loading and unloading of the storage drums cannot be accomplished with the reactor at power, then full use of the shutdown periods required for other purposes should be programmed. At this time, such an alternate does not appear to be completely feasible; therefore, we will consider cases that require reactor shutdown to zero power specifically for the loading and unloading operations. Within the limitations of the contract budget, all cases cannot be considered nor can absolute numbers be generated for each, but the trend and approximate cost benefits for the successful completion of the research and development will be given.

The reference refueling scheme calls for the removal of 130 fuel and blanket assemblies on an approximately annual basis. Additional ground rules and assumptions are as follows:

1. Time required to reach hot shutdown condition and return to full power, hours ³⁶	6
2. Time required to insert a fresh assembly or remove a spent assembly from the pot storage drum, hours	1
3. Number of shipping casks and transport vehicles available for use	24

The economic ground rules are those used in the fuel cost calculations.³⁷

Three alternatives to the reference refueling scheme are considered. In the first case, the reactor is shut down as many times as required to remove the spent fuel assemblies from the storage drum; in the second case, additional storage facilities are provided outside the pot; and in the third case, enough extra shipping casks are purchased to store all 130 assemblies so that all the spent fuel can be removed from the pot during a single shutdown.

3.3.4.1. Case 1

With the reactor in a hot shutdown condition, all 130 fresh assemblies are placed in the storage drum; this takes place just prior to scheduled refueling. The refueling procedures are used in the exchange of fresh and spent assemblies between the core and the storage drums. After the spent fuel decay period, the reactor is again shut down and 24 assemblies are loaded into the available 24 casks. Thus, six shutdowns would be required to load all the spent fuel assemblies. Although these shutdowns would probably be scheduled for weekend lulls, in this analysis it is assumed that they occur when the casks have all returned from the previous trip. Therefore, the shipping time would be the same as for the reference case, which assumes the removal of spent fuel with the reactor on-line. The lost time can be tabulated as shown below:

Shutdown to load fresh fuel, hr	6
New assembly loading time, hr	130
Shutdown to remove spent fuel ($\times 6$), hr	36
Spent fuel removal time, hr	<u>130</u>
Total, hr	302
Days	12.6

With the additional downtime, the capacity factor becomes 0.836, and the reference capacity factor³⁷ is 0.86. The decreased capacity factor causes the following cost penalties (in mills/kWh):

Fuel cost penalty	0.015
Fixed cost penalty	<u>0.102</u>
Total penalty	0.117

3.3.4.2. Case 2

In this case we consider the addition of a secondary sodium-filled storage pool outside the reactor building but accessible directly through the spent fuel transfer tube. Again, new fuel loading of the storage drums is assumed to be as described in Case 1. Spent fuel unloading is to be accomplished in one reactor shutdown, but the spent fuel will be transferred to the secondary sodium pool, and the 24 casks are again assumed to accomplish shipping in the 80-day shipping period.

1. Estimated time for shutdown, startup, and new fuel loading, hr	136
2. Estimated time for shutdown, startup, and spent fuel unloading, hr	136
3. Total penalty for shutdown, days	11.33
mill/kWh	0.106
4. Estimated capital cost of secondary Na storage pool and equipment (\$1,267,000, based on estimate in reference 2, Table C-4, item F), mill/kWh	0.028

Total penalty for Case 2, mill/kWh 0.134

It should be noted that since the capital cost estimate in item 4 is not based on detailed safety or operational analyses, it may not be realistic. A case using a water-filled secondary storage pool has not been considered

here; this is justified in the Shipping Cask Trade-Off Study,²⁹ where a large additional penalty was estimated because of the much longer decay storage period required.

3.3.4.3. Case 3

This case would involve the purchase of an additional 106 shipping casks to provide space for all 130 assemblies, which could then be removed from the pot storage drum during one shutdown. Using the value of \$62,500 per cask (from reference 29), the additional capital cost would be \$6.625 million—a figure considerably higher than the cost estimated for the secondary sodium-filled storage pool in Case 2.

3.3.4.4. Summary

If it cannot be demonstrated, with the reactor at power, that (1) fresh fuel can be inserted in the storage drums and (2) spent fuel can be removed, then the power cost for the reference LMFBR will be increased significantly. This increase would occur regardless of the refueling method selected—reference, Case 1, Case 2, or Case 3.

On the basis of this study, the more straightforward approach of shutting down the reactor to remove fuel as the shipping casks become available (Case 1) is the preferred alternate. The penalty is estimated to be 0.117 mill/kWh or \$890,000 per year for a single 1000-MWe plant. The assumed 6-hour shutdown time for this case seems reasonable, since no major physics tests would be necessary for simply inserting or removing fuel from the storage drums. However, if the shutdown time had to be increased, the advantage of Case 1 disappears; i. e., at 15 hours per shutdown, the costs of Cases 1 and 2 are equal. On the other hand, if we have underestimated the cost of the secondary storage facility in Case 2, then the cost differential between the cases given in the estimates would be greater, and Case 1 would still be the preferred alternate even if the shutdown time were increased.

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4. INSTRUMENTATION AND CONTROL

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4. INSTRUMENTATION AND CONTROL

4.1. Introduction

4.1.1. General

The objectives of the Instrumentation and Control R&D plan are to develop sources of supply for reliable hardware that will meet the instrumentation, control, and protection requirements of the B&W predecessor and reference LMFBR designs; and to develop the analytical techniques necessary to predict the performance of these plants and to design protection and control systems for them. In meeting both of these objectives it is necessary, as part of the development effort, to show by test and analysis that the hardware and methods are satisfactory.

LMFBRs present many environmental and performance differences from the well-developed water reactor technology that affect instrumentation and control. Radiation detectors and their cables are exposed to temperatures and in-core radiation levels (particularly fast neutrons) well above proven capabilities. Neutron detectors must also accurately measure fluxes of neutrons over a very broad energy range, whereas water reactors entail only the limited and simpler measurement of thermal neutrons. Sodium instrumentation is in need of some temperature up-grading, and major improvements are needed in its capability for reliable on-line performance. New types of instrumentation must be developed for some measurement functions, such as water/sodium and sodium/gas leaks. In fact, there are virtually no measurement functions associated with the primary and intermediate sodium in the B&W reference design that can be performed reliably and satisfactorily with existing instrumentation.

Because the LMFBR core must remain submerged in optically opaque sodium, operators cannot visually observe such operations as fuel handling, nor can they inspect the core or its supporting

Table 4-1. Cross References of Task Numbers in B&W's
R&D Plan and AEC's Program Plans

Task	B&W Task	Closest AEC Program Plan Task	
		August 1968	January 1967
Neutron detector - in-core ion chambers	4.2.1.2	4-7.1	1
Neutron detector - in-core advanced methods	4.2.1.3	4-7.1	1, 3
Neutron detector - out-of-core	4.2.1.4	4-7.2	2
Fission product measurement	4.2.1.5	4-3.3	16
Circuitry	4.2.1.6	4-2.1	4
Temperature measurement	4.2.2.2	4-7.3	5, 6
Flow measurement	4.2.2.3	4-7.5, 4-7.6	9, 10
Pressure and Δ pressure measurement	4.2.2.4	4-7.8	13
Level measurement	4.2.2.5	4-7.10	15
Purity measurement, on-line	4.2.2.6	4-3.2	18
Sodium-water reaction detection	4.2.2.7	4-3.6	20
Sodium-to-gas leak detection	4.2.2.8	4-3.7	21
Viewing through sodium	4.2.2.9	4-3.5	30
Electrical connector, gas	4.2.3.2	4-5.4	25
Electrical cable	4.2.3.3	4-5.3	26
Metals - cladding for detectors and cables	4.2.3.4	4-6.1	27
Electrical insulation	4.2.3.5	4-6.2	28
Sensing and control devices for actuators	4.2.4.2	4-3.8	29
On-line computer	4.2.5.2	4-4.2	35
Simulation and analysis	4.2.5.3	4-4.1	37

structure. The development of "viewing" devices to offset this disadvantage must begin almost from scratch.

The complexity of a large LMFBR plant having multiple series and parallel heat transfer paths, a large vessel of primary sodium, variable flow rates and transport delays, high temperatures and large temperature differentials, and fast core response presents an analytical problem that can only be approached with existing techniques and equipment. A transition from almost purely analog analysis to new hybrid and digital techniques is required; these new techniques are generally not available and must be developed. Similar problems arise with respect to control and protection, where major advances are needed in the application of digital computers.

Although the R&D plan described herein is specifically related to the B&W predecessor and reference plants, its scope is sufficiently general that it applies to the overall LMFBR Program Plan. The converse, however, is not true: several tasks in Program Plan 4 are omitted, and a number of others are reduced in scope. The omitted Plan 4 tasks are not required in the B&W reference design, although in some cases minor design improvements could be made if the effort were successfully completed. The omitted Plan 4 tasks also include those which are applicable only to special test instruments. The area of pipe and vessel surveillance is excluded from this plan but is included in the components R&D plan. Scope reductions usually consist of narrowing a broad effort or eliminating backups and are described in detail under the heading "Relation to Program Plan" in the individual task descriptions following. Although these task descriptions had already been written before the August 1968 issue of Program Plan 4 was received, all references herein apply to that issue. For convenience in reviewing, Table 4-1 provides a cross reference for the B&W R&D plan and both issues of the ANL Program Plan 4.

Two other aspects of the R&D may differ from the needs of other LMFBR plants: First, in the B&W reference design, considerable care is taken to facilitate detector maintenance and replacement by isolating them from sodium using wells and thimbles; other designers may take other approaches which may require longer-lived detectors with correspondingly higher R&D costs. Second, the B&W reference

Table 4-2. Instrumentation and Control R&D Cost and Schedule Summary (Thousands of Dollars)

Task	Total	Calendar year								
		1969	1970	1971	1972	1973	1974	1975	1976	1977
Instrumentation and control (overall)	12,800	2350	2645	2280	2150	1325	900	650	400	100
Radiation measurement	2,200	600	595	505	400	100	--	--	--	--
Neutron detector - in-core ion chamber	700	200	200	150	150	--	--	--	--	--
Neutron detector - in-core advanced methods	800	200	200	150	150	100	--	--	--	--
Neutron detector - out-of-core	250	50	75	75	50	--	--	--	--	--
Fission product measurement	350	100	100	100	50	--	--	--	--	--
Circuitry	100	50	20	30	--	--	--	--	--	--
Sodium instrumentation	4,950	1025	1200	950	900	450	300	150	--	--
Temperature measurement	550	100	200	150	100	--	--	--	--	--
Flow measurement	1,150	400	350	200	200	--	--	--	--	--
Pressure and Δ pressure measurement	200	100	25	25	25	25	--	--	--	--
Level measurement	200	50	50	50	50	--	--	--	--	--
Purity measurement, on-line	700	--	200	200	200	100	--	--	--	--
Sodium-water reaction detection	450	100	100	50	50	50	50	50	--	--
Sodium-to-gas leak detection	100	25	25	25	25	--	--	--	--	--
Viewing through sodium	1,600	250	250	250	250	250	250	100	--	--
Parts and materials	1,050	250	275	175	175	175	--	--	--	--
Electrical connector, gas	275	50	75	50	50	50	--	--	--	--
Electrical cable	250	50	50	50	50	50	--	--	--	--
Metals - cladding for detectors and cables	175	50	50	25	25	25	--	--	--	--
Electrical insulation	350	100	100	50	50	50	--	--	--	--
Actuator instrumentation and control	200	25	25	50	75	25	--	--	--	--
Sensing and control devices	200	25	25	50	75	25	--	--	--	--
Computer technology	4,100	450	550	600	600	600	500	400	300	100
Computer, on-line	2,800	250	350	400	400	400	400	300	200	100
Simulation and analysis	1,300	200	200	200	200	200	100	100	100	--
General upgrading, FFTF feedback	300	--	--	--	--	--	100	100	100	--

design places more emphasis on digital computers for reactor protection than may be the case with other designs; in this event, it is likely that the R&D cost herein may be relatively high.

The Instrumentation and Control R&D plan is divided into five areas, each containing from one to eight individual tasks. Radiation Measurement (4.2.1) has five tasks covering in-core and out-of-core nuclear instrumentation as well as failed fuel detection. Sodium Instrumentation (4.2.2) includes eight tasks devoted to individual sodium-associated measurements and observations (including viewing under sodium). Parts and Materials (4.2.3) comprises four tasks broadly applicable to most LMFBR instruments: electrical connectors and cables, and cladding and insulation materials. Actuator Instrumentation and Controls (4.2.4) consists of a single small task providing devices for control rod, pump, and valve instrumentation. Computer Technology (4.2.5) contains two large tasks: The on-line computer system used for protection, control, and data logging; and the off-line computer capabilities for plant simulation and analysis.

4.1.2. Summary Budget and Schedule

Table 4-2 summarizes by task the overall budget and schedule. The estimated total cost reflects the plan that no major facilities will be constructed nor complete computer systems purchased; allowance is made for preparing test locations in existing facilities and for computer rentals. It is also assumed that existing facilities can provide all of the testing space required; no effort was made to compare the total demand on test facilities with the available space.

The schedule is accelerated to the maximum extent believed practical in order to provide as much capability as possible for the FFTF, which is now scheduled to begin operation in 1974. It is noted that this acceleration maximizes the accumulation of operating experience on final design units in the FFTF, which is expected to be highly beneficial to subsequent reactors. For most tasks, the most important milestone is the completion of final hardware designs by early 1972, and proof testing is generally planned for completion by the end of 1972. Departures from these two dates, and other significant milestones, are discussed in the following paragraphs.

Advanced types of in-core neutron detectors (section 4.2.1.3): Final design late 1972, completion of proof testing late 1973. This effort supplements section 4.2.1.2, and those portions needed for the FFTF can be counted on (except for final proof testing) by mid-1972.

On-line purity measurement (4.2.2.6), sodium-water reaction detection (4.2.2.7), and sodium-to-gas leak detection (4.2.2.8): Proof testing of basic devices (plugging meter, gas chromatograph, flame photometer) will be completed in 1972. Proof testing of more sophisticated devices will be completed by the end of 1973, except that improvements in the monitoring of sodium-water reactions will continue through 1975. All of these areas should be completed in time for the first demonstration plant to begin operation in 1976.

Viewing through sodium (4.2.2.9): A device should be available for experimental use in the FFTF by 1974, and improved equipment may be available for the first demonstrator plant; however, these predictions are speculative, and it is doubtful that the desired objectives will ever be completely reached.

All of the Parts and Materials (4.2.3) and Actuator Instrumentation and Controls (4.2.4) work will continue one year beyond the 1972 target dates indicated above. However, the additional effort will be devoted primarily to detailed evaluation of the test results and to devices not needed for the FFTF. These tasks will provide satisfactory output for the FFTF by 1972.

In both computer technology tasks (4.2.5.2 and 4.2.5.3), it will be necessary to extract information for the FFTF and the first demonstrator before the task completion date. The FFTF has already performed considerable work in these areas, and extensive digital protection system equipment apparently is not planned for the FFTF; thus, it is merely expected that these tasks will supplement an existing project effort. Both efforts will complete initial go-rounds by 1974, in time for use in the first demonstrator (although less than optimum). By the time the construction of the B&W predecessor plant begins in 1976, both tasks will be complete except for liaison with projects.

The last entry in Table 4-2, "General upgrading, FFTF feedback," provides for correcting equipment defects discovered during the first few years of FFTF operation. No attempt is made at this time

to allocate this effort among the various tasks, and the estimated cost is pure guesswork, but the need for the effort should be recognized.

4.2. Overall R&D Requirements

4.2.1. Radiation Measurement

4.2.1.1. General

The radiation measurement R&D effort supplements current technology to provide the instrument systems necessary for all required radiation measurements in an LMFBR plant. Included in this effort is R&D for all nuclear instrumentation and failed-fuel monitoring detectors and circuitry.

The B&W reference design includes three types of radiation measurements that require development efforts: (1) In-core nuclear instrumentation is used for power and flux shape information and control. Existing detector types (ion chambers) must be substantially improved, particularly with respect to temperature, to withstand the LMFBR environment. Recently developed circuitry (MSV) must be optimized to the improved detectors and upgraded from experimental to on-line grade. New types of detectors should be developed to facilitate installation design and to permit measurements not possible with conventional thermal neutron detectors. (2) Out-of-core nuclear instrumentation is used for basic protection and control of the reactor. Existing detector types (fission chambers) must be upgraded in temperature, and must be matched to cables and circuitry (particularly MSV and pulse) developed for use with the improved detectors. (3) Failed-fuel detection warns of excessive fuel element defects. Again, temperature upgrading is the major improvement needed; the detector and pulse circuitry from the out-of-core nuclear instrumentation can be adapted for use here. Additional basic information is also needed for a better understanding of the use of failed-fuel monitors.

This effort is divided into five individual tasks. The largest tasks cover in-core ion chambers and advanced methods of in-core neutron detection, since it is in these areas that the LMFBR represents the largest extrapolation from present technology. The smaller efforts cover out-of-core neutron detectors, circuitry, and failed-fuel

monitoring, all of which represent relatively straightforward extensions of current technology.

4.2.1.2. Neutron Detector, In-Core Ion Chamber

Objective — Detectors must be developed for measuring neutron flux at local points within the reactor core. These measurements are used for power flattening, for information, and for planning. The reference design requires ion chambers with a diameter of 3/16 inch or less and the ability to operate in a dry in-core thimble at 1400 F for several years with MSV (see 4.2.1.6) circuitry. The major improvements needed over existing detectors are (1) temperature upgrading from a present limit of about 600 F, (2) possible improvement in resistance to fast neutron radiation damage, and (3) selection of an optimum neutron-sensitive material and improvement of the signal-to-noise ratio.

Summary of Current Technology — In-core ion chambers are being applied extensively to pressurized water power reactors for measuring thermal neutron flux at temperatures of approximately 500 to 600 F, and they have been tested to about 1000 F. These detectors usually contain enriched uranium-235 (or for short-lived applications, boron-10) as a neutron-sensitive material. Fairly rapid sensitivity reduction (some 5 to 15% per year), caused by ^{235}U depletion, led to the development of regenerative fission chambers.¹ In these detectors, various combinations of fissile and fertile materials were used. Most manufacturers coat the neutron-sensitive material on the detector electrode(s) by electroplating or some other means, and one manufacturer usually alloys uranium with the structural metal. Various structural metals and ceramic insulation materials are available.

In-core ion chambers are operated almost exclusively in the d-c mode, while the reference design requires MSV operation (see section 4.2.1.6 regarding MSV circuits).

Some information on design optimization techniques for MSV operation is available,² and it is believed that additional proprietary information may exist. MSV operation requires low-impedance cable (the standard metal-sheathed, ceramic-insulated cable appears

to be less than optimum); again, reference 2 reports development effort on the cable and its attachment to the detector.

Scope of Work — The scope is divided into five phases, as follows:

1. Establish detailed specifications for environment and performance.
2. Perform preliminary design, including some parts testing.
3. Fabricate and test preliminary units.
4. Perform final design, factoring in (a) the results of preliminary tests (b) improvements from cable, insulation, and body material tasks, and (c) updating of specifications.
5. Fabricate and test final units.

In the first phase, the environmental and performance requirements on the detector will be formulated with as much detail and accuracy as possible at the time. This phase will establish quantitative goals for the remainder of the task. For example, specification of the neutron energy spectrum, gamma level, temperature, range of acceptable signal level, size, minimum signal-to-noise ratio, and other factors are necessary for selection of the type and quantity of neutron-sensitive material, electrode spacings, and many other design parameters. This first phase necessarily will be performed by a central agency.

The second phase consists of preliminary design work by one or more vendors working independently but subject to the supervision and coordination of the central agency. The design effort will be based on the first phase specifications and the vendor's experience as a starting point. Wherever process development or fundamental testing is required, central facilities may be employed.

For the third phase, each vendor will fabricate a few preliminary design units. These units will be commonly subjected to temperature tests and combined temperature and radiation tests by the central agency, which will furnish detailed test data to the vendors. The vendors will be permitted to participate freely in the tests. When these tests are completed (after perhaps six months), the preliminary

units will be examined as necessary to identify improvements needed in the final design phase.

The fourth phase consists of establishing the final design, based on (a) the preliminary unit tests; (b) improvements that become available from the cable, insulation, and body material tasks; and (c) updated specifications.

In the fifth phase, units representing the final designs will be fabricated and subjected to common testing. Since the reference design permits replacement at power, only 12 to 18 months of in-reactor testing is necessary; information on lifetime can then be obtained from operating experience.

Relation to Program Plan — This task is similar to Task 4-7.1 of Program Plan 4, with the following major differences:

1. The Program Plan scope includes considerably more effort on basic R&D applied to "parts," such as the neutron-sensitive material.

2. The Program Plan places a larger part of the task with the central agency and less with manufacturers.

3. The Program Plan calls for 1200 F sodium-immersed detectors up to 1/4 inch in diameter, rather than 1400 F dry units of 3/16 inch maximum diameter and calls for slower development of the 1200 F units.

4. The Program Plan task includes development of counters and related detectors, such as an array of energy-band ion chambers. Insofar as such devices are needed for the B&W design, they are included in the next task (section 4.2.1.2).

Major Facilities Required — A 1400 F, dry, in-core test location in a fast reactor (EBR-II or Fermi) is required. Other testing can be performed in available conventional facilities.

Cost and Schedule — It is estimated that this effort will require four years at a cost of \$700,000, leading to completion of one year of proof testing late in 1972.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	200,000
1970	200,000
1971	150,000
1972	150,000
Total	700,000

4.2.1.3. Neutron Detector, In-Core Advanced Methods

Objective — This task supplements Task 4.2.1.2 in two ways. First, the installation of detectors in the reference core design would be improved by the development of prompt-response, self-powered fast neutron detectors with a diameter of approximately 1/16 inch. Second, the predecessor plant, and possibly the reference plant, may require energy-selective detectors to supplement the basic broad-response detectors in order to improve the measurement of core performance. Development of suitable self-powered detectors requires a broad R&D effort, and feasibility is not assured. The development of energy-selective detectors comprises a relatively small extension of the in-core ion chamber and self-powered detector efforts, together with a high-risk exploration of other methods.

Summary of Current Technology — Self-powered in-core neutron detectors have been widely used in thermal reactor designs in recent years and will continue to be used because of their low cost, small size, and other advantages over in-core ion chambers. These devices are usually used in quantities of a few hundred per reactor, so operating experience accumulates rapidly. References 3 and 4 indicate the general suitability of delayed-response units for this application, apart from the questions of radiation damage caused by high fast neutron fluxes, signal level, and signal-to-noise ratio. However, efforts to develop a prompt-response self-powered detector have been unsuccessful,⁵ and no known effort has been applied to selecting an emitter material optimized for an LMFBR core neutron energy spectrum.

Ion-chamber energy response has occasionally been modified by using natural or depleted uranium instead of enriched ²³⁵U, thus obtaining a threshold detector. Omission of the neutron-sensitive coating results in a gamma chamber. In both of these cases, the

"new" detector is identical to its parent except for sensitivity. Changing ion-chamber energy response by substituting a new sensitive material is not common. There are no known instances of self-powered detector spectral response being changed significantly. Other approaches to the problem of spectrum measurement have generally employed radioactivants (such as foils), which must be removed from the reactor for reading. Although this approach can be modified to include core thimbles, radioactivants, and counting equipment in a semiautomatic on-line flux wire system, the delay in obtaining a reading is still many minutes. Bennett⁶ has obtained excellent spectral measurements in critical assemblies with a proton recoil counter, but his detector is far from the size and temperature requirements of an LMFBR core. Although some success has been obtained with other methods, such as circulating gas and microwave techniques, these methods generally are unattractive.

Scope of Work — The scope of work for this task is intended to be compatible with, and an extension of, the scope of section 4.2.1.2. The same five phases are required, and insofar as possible, the tests will be performed together.

In the first phase, the specifications of section 4.2.1.2 will apply. In order to bring the two schedules together, it is also necessary during this time to initiate the neutron-sensitive material selection effort, particularly for the self-powered detectors. The background current problem with self-powered detectors will also be investigated, and consideration of other methods will begin.

In the second phase, the preliminary design work will be completed. Ion chambers will be based on the designs developed in the second phase of section 4.2.1.2.

The third phase will involve fabricating preliminary design units which will be tested with the preliminary reference ion chambers. A current assessment of spectral measurement needs will be factored into the selection of the units to be tested. After the tests are completed, the needed improvements will be identified by examining the test units.

In the fourth phase, final designs will be established on the basis of available information, including current specifications.

Final design units will be fabricated and tested in the fifth phase. By this time, the task is expected to lag the 4.2.1.2 task by 6 to 12 months.

Relation to Program Plan — This task includes a portion of the spectrum measurement effort in Task 4-7.1 of Program Plan 4, and includes the self-powered detector effort in Task 4-7.1. Less effort is intended to be devoted to techniques other than ion chambers and self-powered detectors, and perhaps more effort to self-powered detectors, compared with Program Plan 4.

Major Facilities Required — As for section 4.2.1.2, the major requirement is for a 1400 F dry in-core test location in a fast reactor (EBR-II or Fermi). This location should include provision for extensive dosimetry, and either a means of changing the radiation environment or the availability of additional test locations is desirable for calibration testing. Other required test facilities are conventional, although they should include beam sources of controlled-energy neutrons.

Cost and Schedule — This effort is expected to require five years at a cost of \$800,000, with final design of self-powered detectors completed late in 1972 and proof testing completed late in 1973. Ion-chamber design will be completed in mid-1972, and proof testing in mid-1973.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	200,000
1970	200,000
1971	150,000
1972	150,000
1973	100,000
Total	800,000

4.2.1.4. Neutron Detector, Out-of-Core

Objective — The B&W reference design requires out-of-core neutron detectors for basic protection and control of the reactor. The detectors must cover a neutron flux range of approximately

12 decades. Although the design provides alternate detector locations at approximately 200 and 800 F, the detectors must operate at 900 F and must briefly withstand 1300 F in order to monitor a design basis accident. This task includes three improvements over the present proven technology: (1) upgrading the temperature rating from a present value of about 500 F, (2) incorporating advanced cable from section 4.2.3.3 in a form integral with the detector; and (3) proving the use of a single detector for the entire operating range (or, short of this, ensuring that two or three designs, capable of covering the entire range, are developed as needed).

Summary of Current Technology — Current technology has all but proven the suitability of a single fission counter, with a variety of circuits connected to it, to monitor 12 or more decades for reactor protection and control. The fission counter developed in the Naval Reactors Program⁷ has been used extensively in both counting and d-c channels. Although this detector is rated at only 180 F because of polystyrene insulation, ratings of 600 F or more are claimed for its ceramic-insulated descendants and counterparts. References 2 and 8 show, on a laboratory basis, that MSV operation of such detectors bridges the gap between the counting and d-c modes with some overlap. Thus, current technology suggests that temperature upgrading and suitable cabling are the only detector improvements needed for a 900 F single-detector out-of-core system.

Scope of Work — The scope of work will consist principally of evaluating the detectors commercially designed and fabricated for the application. Some information from section 4.2.1.2 will be available for design guidance, particularly in the areas of temperature upgrading and MSV operation optimization. After preliminary tests of early designs, some redesign will probably be necessary, including making advanced cable from section 4.2.3.3 integral with the detector. Final design units will then be fabricated and tested.

Relation to Program Plan — This task is similar to Task 4-7.2 of Program Plan 4, except that only one type of multi-range detector is expected to be included.

Major Facilities Required — Since a relatively low thermal neutron flux (about 10^{10} neut/cm²-sec) is satisfactory, a suitable 900 F test location could be readily prepared in almost any swimming pool test reactor, for example. Thus, there are no major test facilities required that are not quite readily available.

Cost and Schedule — This effort is estimated to require four years at a cost of \$250,000, with proof testing completed in 1972.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	50,000
1970	75,000
1971	75,000
1972	50,000
Total	250,000

4.2.1.5. Fission Product Measurement

Objective — A system is needed to continuously monitor for the existence of failed fuel in the LMFBR. The reference design method is the delayed neutron monitor, which is generally satisfactory in any power reactor and is particularly good with vented fuel. This task will undertake a search for a better method and will then upgrade the selected method to the status of an on-line system.

Summary of Current Technology — EBR-II has been evaluating both delayed neutron and cover gas monitoring systems for detecting failed fuel,⁹ and some other reactors have experimented with these and other types. No system suitable for an LMFBR has been fully proven by operating experience. The extension of the system to include locating the failed fuel has been attempted in a few instances, but the direct location of failed fuel by an on-line instrument appears to be possible only with extremely complex piping systems connected to the core flow channels. The effort devoted to detection (and location) of failed fuel so far is smaller than that given to other areas of reactor instrumentation.

Scope of Work — The delayed neutron monitor and other possible methods will be carefully evaluated in terms of expected plant parameters (including sodium background gamma activity). Considerable input will be received from the Sodium Technology and Fuels and Materials Programs. In the case of the delayed neutron monitor, the successful development, in other tasks, of a 900 F fission counter (suitable for multiple-detector parallel operation) and the necessary cable and circuitry will be assumed for this evaluation. Upon selection of the reference method, a final design will be made for EBR-II or Fermi (or both). A system (or two systems) will then be constructed and installed, in addition to the system(s) already in the plant(s). Operating experience will then be accumulated for normal plant operation and (hopefully) with some deliberately defected fuel. Any differences from the reference plant environment will be considered carefully while evaluating performance.

This task will also take the lead in studying on-line and off-line techniques for locating failed fuel and in implementing any method found desirable. However, this effort is not expected to lead to the development of an on-line failed-fuel location technique with good precision.

Relation to Program Plan — This task is similar to Task 4-3.3 of Program Plan 4 with regard to the delayed neutron monitor effort. The cover gas monitor covered in Task 4-3.3 is omitted here because it would be redundant; however, gross monitoring of cover gas activity as an indication of failed fuel will be explored (but will not require equipment R&D in the I&C area).

Major Facilities Required — Based on testing in EBR-II and/or Fermi, no major test facility is required, but provisions for a strong supplementary gamma source at the detector location are desirable to simulate the higher sodium activity in a 1000-MWe plant.

Cost and Schedule — It is estimated that this effort will require four years at a cost of \$350,000. Testing will begin with an available neutron detector, and the 900 F detector developed in

4.2.1.4 will be substituted in 1971. Proof testing will blend into project takeover and operation in 1972.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	100,000
1970	100,000
1971	100,000
1972	<u>50,000</u>
Total	350,000

4.2.1.6. Circuitry for Radiation Measurement

Objective — This task must provide the electronic circuitry required by the detectors that were developed (in the other tasks of section 4.2.1) for radiation measurement. No fundamentally new circuit types are contemplated. The required development effort consists primarily of upgrading MSV circuitry to reactor protection grade, optimizing the operation of the MSV circuits with the detectors and cables, and, possibly, a small effort to increase the environmental resistance and reliability of preamplifiers.

Summary of Current Technology — Counting and d-c channels are well proven and fully satisfy the requirements of B&W's reference design, with the possible exception of counting circuit preamps. The latter will be located in moderately inaccessible regions subject to fairly low radiation levels, and some upgrading along the lines reported in reference 10 may be necessary.

Mean square voltage (MSV) or Campbell circuitry has been laboratory-tested as reported in references 2 and 8, and some information is available through these references on designing detectors for MSV operation. MSV circuits have not been proven for reactor protection application, and some design effort is presumably necessary for their use with LMFBR detectors and cables (both in- and out-of-core types). The MSV preamp may also require some upgrading in reliability and environmental resistance.

Scope of Work — Initially, this task will establish circuit design requirements for use with the detectors and cables

(to be developed in other tasks) for MSV operation. Subsequently, MSV circuits will be procured for use in detector testing under other tasks, and they may be subjected to some modification as the testing progresses. In parallel, the need for preamplifier upgrading will be identified and, if necessary, satisfied.

Relation to Program Plan — This task is similar to Task 4-2.1 of Program Plan 4, except that its scope is limited to relatively minor improvements of MSV circuitry and counting preamps; a major effort involving packaging standardization and digital readout is not considered necessary.

Major Facilities Required — No major facilities are required.

Cost and Schedule — It is estimated that this effort will require three years at a cost of \$100,000. Final design, construction, and limited testing will be completed in 1971; the circuits will then be operated under the detector development test efforts.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	50,000
1970	20,000
1971	<u>30,000</u>
Total	100,000

4.2.2. Sodium Instrumentation

4.2.2.1. General

The Sodium Instrumentation R&D effort supplements current technology to provide the instrument systems necessary for all sodium-associated measurements and observations in an LMFBFR plant. Such parameters as temperature, pressure, flow rate, and sodium level are used for protection, control, and performance monitoring. In these areas the required development effort centers on improving environmental resistance and/or on-line reliability of existing instrument types, which are generally only of laboratory grade. Sodium purity instrumentation is needed to ensure that impurities remain below acceptable

concentrations; present technology does not provide sufficient basis for establishing instrument requirements, and existing instruments are primarily experimental in nature. The hazards associated with sodium leakage require detection of water-to-sodium and sodium-to-gas leaks; the mechanisms of sodium leakage reactions are not fully understood, and better instrumentation must be identified and developed. The possibilities that components may shift during an accident or that an object may be dropped within the pot require a means of "viewing" through optically opaque sodium; present technology provides little more than ideas for partially fulfilling this requirement.

This effort includes eight tasks. Separate tasks are devoted to these relatively straightforward sodium parameter measurements: temperature, flow, pressure and differential pressure, level, and purity. Less straightforward tasks cover the following important areas of leakage: sodium-water reaction detection and sodium-to-gas leak detection. Finally, the novel problem of viewing through sodium is treated in a task with a high probability of less than complete success. The Sodium Instrumentation effort is diversified and extensive; it accounts for approximately 40% of the total Instrumentation and Control R&D budget.

4.2.2.2. Temperature Measurement

Objective — Detectors must be developed for sodium temperature measurements throughout the LMFBR plant (except inside the core itself). These measurements are used for protection, control, and plant monitoring (including heat balances). The reference design requires enclosed-sheath, ungrounded-junction, 1/16 inch chromel-alumel thermocouples housed in dry wells for most applications and platinum resistance thermometers in wells for some out-of-core locations where temperatures are limited to 1000 F. In general, the thermocouples must operate continuously up to 1100 F, but core outlet sensors must withstand and operate during transients approaching 1600 F, and the neutron shield monitors may be exposed to 1500 F. The response time requirement of 10 seconds for thermocouples is quite reasonable. The major improvement needed over current technology is increased on-line reliability, with emphasis on seating the thermocouple

in its well and on minimizing errors caused by high temperature and radiation levels.

Summary of Current Technology — Numerous articles discuss the performance of chromel-alumel thermocouples in water reactors from Shippingport on.¹¹ Reported performance has ranged from excellent to unsatisfactory and is clearly a strong function of installation and handling conditions. The ability of chromel-alumel to withstand temperatures up to 2000 F or more is known,¹² and transmutation effects are small.¹³ EBR-II is accumulating operating experience in a fast flux environment. Radiation damage by large integrated fast neutron fluxes has not been established.

The behavior of well-type platinum resistance thermometers in temperatures up to 1000 F is well established, and no R&D is needed for the B&W application.

Scope of Work — An early effort is needed to specify the mechanical aspects of the thermocouple installation design: permissible bending and cold-working, details of seating the tip in the well, and the like. In parallel with this effort, a materials and processes selection effort will prepare detailed procurement specifications for test units; the selection will consider the type of insulation and sheath, inerting or evacuating, cold-end seal, and similar factors, and will reflect improvements that become available from the cable, insulation, and body material tasks. A few carefully controlled designs will be subjected to extensive laboratory and oven tests including a mockup of typical in-reactor temperature profiles. They will then be reactor-tested along with the final design in-core ion chambers of section 4.2.1.2. Careful attention must be given to gamma-heating effects during the reactor testing. As for the ion chambers, very long-term irradiation testing is not necessary, because the thermocouples will be replaceable. A major goal of the testing is to determine sensitivity changes, so that corrections can be applied to in-service data.

Relation to Program Plan — This effort is very similar to Task 4-7.3 of Program Plan 4. It includes additional effort on determining mechanical handling effects, but it excludes detector types other than thermocouples and considers only a dry detector

environment. Additional scope reduction results from the exclusion of thermocouples intended for the measurement of fuel and cladding temperatures.

Major Facilities Required — The major facility required is a fast reactor in-core test position, which can be the same as the one used in section 4.2.1.2. A secondary facility is needed to simulate installation geometry and temperature profile; this setup could be made quite readily in a large laboratory, assembly area, or shop.

Cost and Schedule — This effort is expected to require four years at a cost of \$550,000, with proof testing to be completed late in 1972.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	100,000
1970	200,000
1971	150,000
1972	100,000
Total	550,000

4.2.2.3. Flow Measurement

Objective — Permanent magnet flow meters must be developed to measure sodium flow rates in the intermediate system and in selected fuel assembly inlets (primary flow is measured by pump calibration and differential pressure measurements in the B&W reference design). The intermediate flow measurements are used for information purposes, including heat balance; the fuel assembly measurements are for information only. The flow meters may be exposed to sodium temperatures of up to 1100 F (at the IHX outlets under accident conditions) and to core-level radiation (at the fuel assembly inlets); monitored pipe diameters range up to 32 inches at the steam generator outlets. The principal R&D efforts required are (1) the development of a permanent magnet material that will withstand the peak temperature and radiation levels, and (2) the extension of size up to 32 inches.

Summary of Current Technology — References

14, 15, and 16 agree that permanent magnet flow meters are the most promising type for LMFBR use. Their major advantages are the lack of moving parts, safe failure in the sense that meter failure does not cause flow blockage, and the absence of large alternating currents which could interfere with other instruments. Since it is very probable that satisfactory permanent magnet flow meters can be developed, no back-up effort in the form of developing other types appears to be necessary.

As reported in reference 16, Alnico V magnetic alloy has shown negligible change after 1000 hours at 1000 F, and preliminary ANL results indicate its stability up to at least 1200 F. Alnico V has shown negligible change after 5×10^{20} neut/cm² exposure. These results approach the highest temperature and radiation levels in the B&W reference design.

Although permanent magnet flow meters up to 32 inches in diameter are not available now, there appears to be no fundamental limitation in reaching this size. Non-linearities, however, will increase.¹⁷

Good operating experience has been reported in the Fermi reactor, although non-linearities in meters as small as six-inch diameter were observed.^{17, 18}

Scope of Work — The first phase of work will include two parallel efforts. Magnet materials, with emphasis on Alnico V and promising lesser-known alloys, will be tested for temperature and radiation stability. The factors that influence linearity and accuracy in large-diameter flow meters (up to 32 inches) will also be studied.

In the second phase, meters representing the required range of sizes will be designed and constructed. The designs will incorporate the improved cable from Task 4.2.3.3. The meters will then be tested under both prolonged and cycled temperature conditions in sodium, and the fuel assembly meters will also be subjected to combined temperature and radiation testing in a reactor. Since this is a relatively low-risk effort, it is expected that failures can be identified and corrected without the need for a second round of flow meter testing. It will be important to measure the calibration changes that

occur during testing, so that corrections can be applied to in-service data.

Relation to Program Plan — This effort is similar to Tasks 4-7.5 and 4-7.6 of Program Plan 4. The major difference is that this effort concentrates on permanent magnet flow meters, with no attempt to develop alternate or backup types as in the program plan. The test facility covered in Task 34 of Program Plan 4 is omitted, as discussed in the next paragraph.

Major Facilities Required — A dry, 900 F, in-core facility is desirable but not necessary for preliminary testing of magnet materials for fuel assembly flow meters; EBR-II or Fermi would be suitable. Later, a sodium-wetted location in EBR-II or Fermi will be necessary for testing these meters. Most out-of-reactor flow meter testing can be performed in the Sodium Components Test Installation at LMEC, with the large sizes to be tested in the Sodium Pump Test Facility, also at LMEC. Conventional laboratory facilities will be suitable for the out-of-core preliminary magnet tests.

Cost and Schedule — It is estimated that this effort will require four years at a cost of \$1,150,000, with proof testing completed in 1972. The accumulation of test loop operating experience at the LMEC will begin then.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	400,000
1970	350,000
1971	200,000
1972	200,000
Total	1,150,000

4.2.2.4. Pressure and Differential Pressure Measurement

Objective — Reliable pressure and differential pressure (ΔP) detectors are required for several measurements in the LMFBR plant. Primary sodium flow measurements in the B&W reference design are performed entirely by ΔP detectors across inherent

system pressure differentials; these detectors are calibrated in place against known pump characteristics. The pressure and ΔP measurements are used for protection (power-to-flow scram) and for information purposes, including heat balances. The required ranges vary from a few to 150 psi, and sodium temperatures to 1200 F. The design permits all transducers to be located in accessible, low-radiation environments below 200 F. The core plenum ΔP response time must be about 0.2 second, and primary sodium-filled heated pipes are run to these transducers. Longer permissible response time allows the use of filled capillary systems for the other measurements. Little R&D effort, other than testing, is needed to provide the necessary instruments.

Summary of Current Technology — Reference 19 comprehensively documents the status of current sodium pressure and ΔP measurement technology. The art is well developed, and considerable experience is available from several manufacturers. In some cases the experience is somewhat spotty, particularly with units using bellows, but the conclusion is undeniable that current designs are very nearly satisfactory "as is."

Scope of Work — On the basis of a review of existing experience (documented in reference 19 presumably through late 1967) and discussions with manufacturers, several commercial units will be selected for testing. The testing can be performed in existing facilities, such as the Sodium Components Test Installation at LMEC, and should concentrate on obtaining additional operational data. Careful attention should be given to determining criteria for heating sodium-filled pipes and NaK-filled capillaries.

Relation to Program Plan — This effort is nearly identical with Task 4-7.8 of Program Plan 4. Slight additional effort is provided to allow for the piped-sodium detector testing. Task 4-7.8 apparently covers a much longer and broader effort than we have allowed for here.

Major Facilities Required — SCTI or the equivalent is the only major facility required.

Cost and Schedule — This effort is expected to require approximately 4.5 years at a cost of \$200,000, with proof testing

blending into accumulation of operating experience at LMEC in the first half of 1973.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	100,000
1970	25,000
1971	25,000
1972	25,000
1973	25,000
Total	200,000

4.2.2.5. Level Measurement

Objective — Level measurements are required for primary and intermediate sodium and for the NaK decay heat removal system at various places. The NaK measurements are used for protective emergency cooling, and all level measurements are used for information, alarm, and verification. Measuring ranges vary from 3 to 30 feet, at temperatures up to 1200 F; the core vessel level monitor sees moderate radiation levels. For simplicity and replaceability, the reference design specifies inductance detectors in dry wells for all level measurements. The required development effort consists primarily of temperature upgrading.

Summary of Current Technology — Reference 20 describes several existing and planned applications of a number of different inductance level detectors. Although substantial temperature effects exist, they can be compensated. Operation above 800 F is evidently not clearly proven, although the capability for operation at temperatures approaching 2000 F is expected with suitable materials choices.

Differential pressure measurement as a back-up detection method will be provided by section 4.2.2.4, using remote transducers.

Scope of Work — A review of existing experience, consideration of material temperature and radiation resistance, and discussions with manufacturers will result in identification of units that are available commercially for testing. It is expected that at least two or three different kinds of inductance probes will be included in the test samples. Testing can be performed in any facility that is capable of handling 1200 F sodium.

Relation to Program Plan — This effort is similar to Task 4-7.10 of Program Plan 4. It differs mainly in that only inductance probes are considered, since differential pressure measurement is available from section 4.2.2.4 as a backup method.

Major Facilities Required — No major facilities are required.

Cost and Schedule — It is estimated that this effort will require four years at a cost of \$200,000. Proof testing should be completed by mid-1972.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	50,000
1970	50,000
1971	50,000
1972	50,000
Total	200,000

4.2.2.6. Purity Measurement, On-Line

Objective — Reliable on-line instrumentation must be developed to measure the purity of sodium and NaK. Impurities in the coolants are carried throughout the heat transfer systems and can greatly affect corrosion rates and other conditions. The B&W reference design requires plugging meters for primary and intermediate sodium and for NaK. Since the plugging meter determines the temperature at which solid impurities precipitate, it is an effective operating instrument but gives only a crude upper limit of actual impurity content. For this reason, the reference design also calls for on-line carbon, hydrogen, and oxygen meters, in addition to including sampling provisions for off-line analysis. The need for these meters is somewhat questionable, and further work in the Sodium Technology Program may show that they are not necessary.

The major required R&D efforts are to upgrade the plugging meter to a continuous on-line instrument, and to follow the completion of existing developments of specific impurity meters in the Sodium Technology Program and possibly upgrade them to on-line status.

Summary of Current Technology — References

21 and 22 are surveys of the current status of on-line sodium purity instrumentation. Although plugging meters have not developed as instruments supplied by instrument vendors, a number of units have been built and operated satisfactorily as manually or semi-automatically controlled devices; extension of these efforts to provide a continuous on-line instrument is considered straightforward. A British experimental design for such a device is described in reference 23. A sensitivity of 10 ppm is acceptable and is regarded as routinely obtainable. A second type of gross purity device, the electrochemical cell or "rhometer," also exists, but the need for this instrument is doubtful; it does not provide the precipitation temperature as does the plugging meter, and it does not measure specific impurity contents as do other methods.

Single-vendor developments of an on-line electrochemical oxygen meter and carbon and hydrogen diffusion meters are in progress. The performance objectives of these efforts appear to be satisfactory.

Scope of Work — Commencement of this effort will be delayed until the on-line continuous plugging meter is further developed in the Sodium Technology Program. The possible need for other on-line purity instruments will also be determined in that program, which will develop prototype instruments as required. I&C will then develop commercial sources of supply for reliable on-line instruments and perform proof testing.

Relation to Program Plan — This effort is similar to Task 4-3.3 of Program Plan 4.

Major Facilities Required — SCTI or its equivalent is the only major facility required.

Cost and Schedule — This effort will probably include only upgrading of the plugging meter to commercially available on-line status. However, in view of the possibility that it may become necessary to upgrade perhaps two specific impurity meters, the total effort is estimated at four years, beginning in 1970, at a cost of \$700,000. Proof testing of the plugging meter will be completed in 1972, and that of other possible types will be finished in 1973.

<u>Calendar Year</u>	<u>Cost, \$</u>
1970	200,000
1971	200,000
1972	200,000
1973	<u>100,000</u>
Total	700,000

4.2.2.7. Sodium-Water Reaction Detection

Objective — Sodium-water leak detectors must be developed to provide positive detection of this violent chemical reaction, which can occur in a faulty steam generator or reheater and which requires a reactor scram. The B&W reference design calls for pressure switches to monitor this reaction, with the provision that a better method will be used if one is developed; backup is provided in the design by gas chromatographs, which are not considered suitable for protection system use. A major objection to the pressure switch for this application is that it is suitable only for large leaks. The R&D effort in this area depends on obtaining better fundamental knowledge of the sodium-water reaction; it then consists of identifying and developing suitable on-line detectors. Close interfaces exist with the Sodium Technology and Safety programs.

Summary of Current Technology — APDA is conducting an investigation of leak detection methods as part of their Sodium-Water Reactions Program.²⁴ Although the Fermi reactor detects a sodium water-reaction by monitoring for hydrogen in the cover gas, the APDA studies show that as much as 99% of the reaction product hydrogen may remain in the sodium. Additional basic investigation of the sodium-water reaction on various scales and under various conditions is needed before good reaction detection criteria can be formulated. In particular, the prompt detection of small water leaks into the very large mass of sodium in a steam generator or reheater needs additional fundamental effort.

APDA has experimented with both hydrogen (in cover gas) thermal conductivity meters and acoustic detection of the sodium-water reaction. These techniques, in addition to the in-sodium

hydrogen diffusion meter covered in the Sodium Technology Program and some other types, provide a nucleus of detection methods.

Scope of Work — This effort will continue the APDA effort and will complement the Sodium Technology and Safety Programs as necessary to provide a firm basis for detection criteria. In parallel, leak detection methods will be studied and promising detectors will be included in tests conducted in the Sodium Technology and Safety Programs insofar as possible. After the criteria are established, one or more detection methods will be selected for upgrading to an on-line protection instrument. Testing must consider such factors as response time as a function of leak rate and leak detector geometry.

Relation to Program Plan — This effort is similar to Task 4-3.6 of Program Plan 4.

Major Facilities Required — This effort will borrow major facilities in the Components, Sodium Technology, and Safety Programs.

Cost and Schedule — It is estimated that this effort will require seven years at a cost of \$450,000. Proof testing of the most advanced instruments may not be completed until 1975, but it could be completed as early as 1972 for devices available now, such as the gas chromatograph.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	100,000
1970	100,000
1971	50,000
1972	50,000
1973	50,000
1974	50,000
1975	50,000
Total	450,000

4.2.2.8. Leak Detection, Sodium-to-Gas

Objective — Detectors must be developed to warn of primary and intermediate sodium leakage to argon and air in a

variety of geometries. The presence of sodium in unwanted areas, the reduction of the quantity of sodium in heat transfer paths, and the mechanical defects symptomized by the leakage can all have serious consequences. The B&W reference design requires sodium detection in the reactor vessel annulus (an automatic shutdown function), in the shrouds surrounding the intermediate pipes inside the containment, and in the containment atmosphere. Additional detailed design effort would probably lead to additional requirements for the detection of sodium vapor in the steam generator room, in conjunction with the operation of the refueling machine, and possibly in other applications. The required R&D effort mainly comprises upgrading the flame photometer to an on-line general atmospheric monitor and to develop techniques for local leak detection, such as leakage from pipes.

Summary of Current Technology — The reference design calls for a smoke detector as a containment atmosphere monitor; however, unpublished tests at ANL and a NaK leakage incident at the British Dounreay Fast Reactor²⁵ show that the sensitivity of this method is poor, and only quite large fires can be detected. Smoke detection would, of course, be generally useless in an argon atmosphere (such as the refueling machine) because it responds only to combustion. Reference 25 indicates that a flame photometer is considerably more sensitive; it provides an additional advantage in that it is capable of monitoring for sodium vapor in argon (the argon sample must be mixed with air before it can be ignited in the detector). The flame photometer is now available only as a laboratory-grade instrument. The reference also suggests the use of a direct light absorbing detector; however, this approach appears to be sensitive only in a somewhat darkened room of favorable geometry.

The alternative to detecting sodium vapor in the atmosphere is to detect liquid sodium at or near the leak location. "Spark plug"-type detectors are used at DFR²⁵ and Fermi²⁶ for this purpose. There is no question that such detectors will work if wetted by sodium, but leaking sodium may be trapped in (or react with) thermal insulation, or it may solidify on a cool surface after leaking and never reach the detector. This is no small problem in a 1000-MWe LMFBR with its vast surface area for potential sodium leakage. Reference 26

suggests wrapping two-conductor electrical tape around pipes, so that leaking sodium will immediately cause a low-resistance signal. Reference 25 suggests a two-dimensional grid extension of this approach. The pipe- or vessel-wrapping approach may encounter problems with the electrical heaters used for cold startup and during long shutdowns. Other approaches may be possible.

Scope of Work — The flame photometer will be tested to upgrade it to on-line status, and its use for the remote monitoring of argon interstices, such as the reactor vessel annulus, will be studied. The electrical wire tape and grid methods will also be evaluated, and other local-to-the-leak detection methods will be investigated.

Relation to Program Plan — This effort is the same as Task 4-3.7 of Program Plan 4 with regard to the flame photometer; however, concentration on the wire tape and grid methods of local detection is substituted for Task 4-3.7's emphasis on the study of sodium's chemical reaction with insulation. The overall level of effort is slightly greater than for Task 4-3.7.

Major Facilities Required — All necessary testing can be performed in existing facilities, such as those at the LMEC.

Cost and Schedule — This effort is expected to require four years at a cost of \$100,000, with proof testing to be completed in 1972.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	25,000
1970	25,000
1971	25,000
1972	25,000
Total	100,000

4.2.2.9. Viewing Through Sodium

A method of under-sodium surveillance is needed. At present there is no means for locating or identifying objects

or for surveying such operations as fuel transfer. A development program is in progress, but there is little chance of developing a completely successful viewer.

Summary of Current Technology — APDA began a development program by surveying the feasibility of sonic and ultrasonic scanning systems,²⁷ since this approach appears to offer the most promise. This study led to the development of a system, targeted for 600 F operation, which combines in a single unit an ultrasonic transmitter and an acoustic image converter which electrically forms a "picture" of the target. Reference 28 is a recent report on the progress of this effort, which remains in the feasibility study stage. Only small areas can be scanned; to search an entire reactor vessel for a dropped object, for example, would require considerable time. No known technique appears likely to be superior, although an intensive effort might lead to the applicability of laser holography.

Scope of Work — The scope here consists of continuing and broadening the present effort. Quantitative performance measurements will be evaluated as they become available, and other approaches will continue to be studied. Information is required on the effects of sodium density and temperature variations, sodium turbulence, radiation effects, and extension to 900 F operation (1200 F would be a little more useful, but 900 F now appears to be sufficient). The work is directed toward the construction of a practical viewer and the identification of application designs for it. The program should be subject to periodic overall assessment; the cost and schedule summary below is to be regarded as strictly a preliminary estimate.

Relation to Program Plan — This effort is nearly identical to Task 4-3.5 of Program Plan 4.

Major Facilities Required — No special facilities are expected to be needed.

Cost and Schedule — This effort is arbitrarily scheduled for seven years' work at a cost of \$1,600,000, leading to the development of a device with limited usefulness in 1973 and a more

useful device by 1975. Periodic evaluation of the effort should result in year-to-year changes in both schedule and cost.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	250,000
1970	250,000
1971	250,000
1972	250,000
1973	250,000
1974	250,000
1975	<u>100,000</u>
Total	1,600,000

4.2.3. Parts and Materials

4.2.3.1. General

The Parts and Materials R&D effort supplements current technology to provide the temperature- and radiation-resistant "bits and pieces" common to most of the LMFBR instruments. The high temperatures, in-core radiation levels, corrosive sodium liquid and vapor environment, and the need for low-leakage containment penetrations place a burden on cables and connectors (as components) and on sheath metals and electrical insulation (as materials) for in-containment instruments of all types. Although suitable materials are generally available, test data and design theory are lacking and must be developed.

This effort has four tasks. The first task covers the development of the electrical connectors necessary to bring detector signals through the containment. The second task provides the cables necessary to conduct a variety of detector signals through the plant. The last two tasks approach two broad material areas differently. The metals task provides coordination within this program and liaison with other programs to ensure the proper choice of detector and cable metals, with very little testing included in the task. In contrast, the electrical insulation task includes both literature searching and testing to catalog the fundamental behavior of insulation in areas of high temperature and radiation levels.

All four tasks in this effort provide service functions to other efforts. The success of these tasks will be measured by the performance of detectors developed in other efforts (primarily 4.2.1 and 4.2.2).

4.2.3.2. Electrical Connector, Gas

Objective — Electrical connectors are needed to join electrical leads in the cover gas and to facilitate lead penetrations through the containment barrier. Literally hundreds of detector leads penetrate the containment, and in most cases a connector will be required at that joint. In addition, connectors will often be required to join leads to detectors. Although the B&W reference design avoids placing connectors in sodium, many are exposed to the cover gas and the sodium vapor it contains. The R&D effort consists mainly of selecting suitable connector types from the many varieties now available.

Summary of Current Technology — Connectors are supplied by more vendors, and in a wider selection of types, than perhaps any other electrical component. More than one existing type, with minor modifications at most, is probably satisfactory for the LMFBR application. Several manufacturers claim temperature ratings of 500 F or more, adequate radiation resistance, high reliability, and the like. Little clear information exists, however, on the effects of sodium vapor. ANL has been evaluating one type of connector;²⁹ although this connector is expensive, the results are encouraging.

The ANL evaluation effort is believed to be limited to multiple-contact connectors for single-conductor leads. It is also necessary to identify coaxial and possibly triaxial connectors for nuclear instrument use, subject to the alternative of using special shielded junction boxes. There is no known effort in this area at present.

Scope of Work — The required effort will identify requirements, survey available connectors, and test the most promising ones. Faults exposed during testing will be corrected by redesign, and the improved units will be tested again. Testing must evaluate both in-line (completely in gas) and feed-through (bulkhead containment) applications. The connectors that are found to be satisfactory will receive

considerable additional testing by using them in the detector development tests of sections 4.2.1 and 4.2.2. Mechanical and electrical compatibility with cables will be considered.

Relation to Program Plan — This effort is similar to Task 4-5.4 of Program Plan 4. Much less effort is devoted to connector design initially, and more attention is paid to coaxial and triaxial connectors.

Major Facilities Required — No special major facilities are required, although a sealed box containing argon loaded with sodium vapor is needed. Facilities otherwise available under the LMFBR program should be adequate.

Cost and Schedule — It is estimated that this effort will require five years at a cost of \$275,000. Near-final designs should be available early in 1971 for use in detector tests, but proof testing of final designs and final evaluation are not expected to be completed until the middle of 1973.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	50,000
1970	75,000
1971	50,000
1972	50,000
1973	<u>50,000</u>
Total	275,000

4.2.3.3. Electrical Cable

Objective — Electrical cables must be developed to satisfy the needs of the LMFBR plant. These needs include high-temperature, radiation-resistant coaxial or triaxial cable for in-core and out-of-core neutron detectors and single and multi-conductor cables for thermocouples and other primary plant detectors. This R&D effort consists mainly of evaluating existing cables and introducing some improvement.

Summary of Current Technology — Interest in this effort centers primarily on the class of cables known as "MI" (mineral-insulated). These cables, which are available from several manufacturers, generally consist of one or more conducting wires surrounded by one or more coaxial metal tubular sheaths with the interstice(s) filled with ceramic electrical insulation in the form of a densely compacted powder. Reference 2, however, indicates that neutron detectors may not be able to use this type of cable because of its high impedance and low rf shielding capability; they may instead require quartz-insulated cable wrapped with multiple layers of metal braid. The problems associated with MI cable apply also to the alternative for nuclear instrument use: a lack of proven performance in the temperature (up to 1400 F for in-core neutron detectors), radiation (up to 10^{16} neut/cm²-sec), and in some cases sodium-exposed, conditions of the LMFBR plant. Radiation-produced signals in in-core cables have long been a matter of concern, although reference 30 may provide the theory needed for quantitative understanding of this effect. Temperature effects are also understood incompletely.

Scope of Work — To a large extent, the development of suitable electrical cables requires two things: first, selection of materials that will withstand the environment; and second, arrangement of these materials into a design that will satisfy the electrical performance requirements of the various instruments. As a result, the scope of work for this effort consists largely of coordinating the Fuels and Materials Program Plan and the materials efforts of sections 4.2.3.3 and 4.2.4.4 with the instrument development efforts of sections 4.2.1 and 4.2.2.

Cable requirements will first be identified, primarily on the basis of output from sections 4.2.1 and 4.2.2. Some early fundamental studies will also be performed to determine the effects of processing, impurity content, and similar variables. Cables that appear to satisfy the various requirements will then be procured from several manufacturers and will be subjected to limited testing in- and out-of-core. At this stage, considerable attention will also be devoted to connecting the cables to the connector. The majority of the testing required will be accomplished by using these cables for detector

tests in sections 4.2.1 and 4.2.2. Test results will be evaluated and improvements will be initiated as necessary throughout the tests.

Relation to Program Plan — This effort is similar to Task 4-5.3 of Program Plan 4. Although a greater variety of cable types is covered here than in Task 4-5.3, more of the effort is covered in other tasks, and the cost is considerably less.

Major Facilities Required — A 1400 F, dry, in-core test location in a fast reactor (EBR-II or Fermi) is the major facility required.

Cost and Schedule — This effort is expected to require five years' work at a cost of \$250,000. As cables are required in the detector tasks for testing, the types believed best at that time will be used. Proof testing of final design units and final evaluation will be completed in 1973.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	50,000
1970	50,000
1971	50,000
1972	50,000
1973	50,000
Total	250,000

4.2.3.4. Metals - Cladding for Detectors and Cables

Objective — Metals are needed for detector and cable construction, both dry and exposed to sodium and cover gas, in and out of the core. In most cases, materials that were developed and proved for other portions of the plant can be used, but they must be selected carefully. In some cases, such as bellows and diaphragms for pressure transducers, instrumentation presents special requirements that will not be basically satisfied in other portions of the LMFBR program. Thus, the objectives of this effort are to ensure that suitable metals are available and to provide selection guidelines for detector and cable development.

Summary of Current Technology — Reports of experience from EBR-I, EBR-II, Fermi, the LMEC, and many other sources provide considerable information on the behavior of detector and cable material exposed to sodium at temperatures well below 1200 F. Considerable uncertainty exists with respect to what happens at 1200 F; significant efforts to resolve the uncertainty are already underway and will be continued in other portions of the LMFBR program.

Scope of Work — In general, fundamental materials R&D is not considered part of the Instrumentation and Control Program. For most metals needed for detector and cable construction, suitable R&D will be performed in other program plans, such as Fuels and Materials, Sodium Technology, and Components. It is further noted that this metals effort within the I&C program does not serve an end in itself; the results must be reflected in other efforts, such as individual detector developments. For these reasons, this effort is deliberately limited to provide mainly metals information liaison between the various efforts in the I&C program and the other materials-oriented programs.

The first phase will be an overall survey of metals selections in the detector and cable efforts and comparison with the information and plans of other programs. The objectives of this phase are overall coordination and guidance of metals choices in the individual efforts and identification of any basic materials R&D requirements that are unique to the I&C program. A few efforts are expected to be identified, such as the selection of optimum transducer bellows materials (since bellows requirements are much different for use in instruments than for use in seals).

In the second phase, the limited R&D efforts will be pursued to the point where they can be taken over by individual detector efforts. For example, after a few promising bellows candidate metals are identified through literature search and testing, the pressure detector effort will take over.

Finally, this effort will maintain continuing surveillance over metals performance in the detector and cable efforts and over progress in other programs. As problems arise in the I&C efforts, assistance will be furnished.

Relation to Program Plan — This effort is similar in subject matter to Task 4-6.1 of Program Plan 4, but it differs in approach. Whereas Task 4-6.1 includes large efforts devoted to materials testing and handbook compilation, this effort concentrates on limited coordination and liaison. The overall effort is much less extensive than Task 4-6.1.

Major Facilities Required — No major special facilities are believed to be required.

Cost and Schedule — This effort is expected to require five years at a cost of \$175,000. Information will be made available to the other tasks continually during that time.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	50,000
1970	50,000
1971	25,000
1972	25,000
1973	25,000
Total	175,000

4.2.3.5. Electrical Insulation

Objective — Electrical insulators used in detectors and cables are subjected to high temperature and radiation levels in LMFBR plants. Current knowledge of the behavior of insulation under these conditions fails to provide a basis for optimum selection of insulator materials. In contrast to most efforts in this program, this effort is devoted to measuring and cataloging more than to improving; however, in so doing, needed improvements in the application of insulation are expected to result.

Summary of Current Technology — As noted in section 4.2.3.3, the electrical behavior of electrical insulation exposed to high temperature and radiation levels is not well understood. Most available data were obtained with assembled devices, such as cables and insulators, where performance is clouded by geometry and by the presence of other materials. Although the mechanisms of temperature and

radiation effects are known, the application of theory to practical cases is generally not possible. A particular difficulty exists because there are many variations of a few standard materials (such as magnesium oxides of various purity levels and with various impurities or additives) with vastly different behaviors. Finally, there is no comprehensive compilation of existing data. It is believed that a collection of data covering commercially available insulation materials would benefit the detector and cable R&D efforts significantly.

Scope of Work — The electrical behavior of electrical insulation is the largest area of the Instrumentation and Control program in which basic materials information is not expected to be supplied by other programs. Although insulators will be tested in all of the device-oriented efforts of this program, it is desirable to isolate insulation as the subject of a single effort concentrating on fundamental behavior. This effort will comprise two main endeavors: first, cataloging of available data on temperature and radiation effects on commercially available insulators of interest; and second, a test program intended to fill the gaps in this catalog. The specification and standardization of insulating materials for LMFBFR instrumentation are the goals of this effort.

Relation to Program Plan — This effort is similar in subject matter to Task 4-6.2 of Program Plan 4. However, Task 4-6.2 includes considerable advancement in basic insulation technology, such as the development of techniques for growing large crystals and advancement of theory, which are replaced herein by a basically empirical approach at much less cost.

Major Facilities Required — Small in- and out-of-core high-temperature (up to 1600 F) test locations are required.

Cost and Schedule — It is estimated that this effort will require five years' work at a cost of \$350,000. The major output will be completed in mid-1971, and subsequent efforts will be devoted primarily to evaluating detector and cable test results.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	100,000
1970	100,000
1971	50,000
1972	50,000
1973	50,000
Total	350,000

4.2.4. Actuator Instrumentation and Controls

4.2.4.1. General

The actuator instrumentation and controls effort includes only a single task, which supplements current technology to provide sensing and control devices for such plant actuator components as control rod drives, pumps, and valves. Although current technology is generally well developed in this area, the operation of electro-mechanical devices in argon (possibly containing sodium vapor) is not fully understood, and environmental temperatures will be somewhat higher in the LMFBR than in most present applications. This task also provides for the possibility that the development of new devices may become necessary as the actuating component development effort progresses.

4.2.4.2. Sensing and Control Devices for Actuators

Objective — Conventional sensing and control devices must be upgraded to operate in the argon and sodium vapor environment of the LMFBR plant. A major application is the reactivity control system, which requires position indicators, limit switches, and other electrical components which must operate very reliably in the control rod drive housings. General applicability extends to other components, such as valve position detectors. This effort is not so much device-oriented as it is directed toward identifying the problems and their solutions for using the electrical devices in the LMFBR environment.

Summary of Current Technology — In general, satisfactory devices are available for an air environment. Industrial competence has not been extended to the problems of argon (with poorer cooling and electrical insulating capabilities than air) and of sodium

vapor. The LMFBF environmental temperatures are also somewhat higher than the usual application.

Scope of Work — This effort is closely related to the Components Program. Most input requirements will be received from that program, and much of the output can be utilized in it.

The effort will study electrical contact and enclosure problems of the argon and sodium vapor environment at temperature, primarily by testing available devices that are expected to be satisfactory and will produce guidelines for their selection and use. Specific devices will be investigated as much as necessary to complete the guidelines.

Relation to Program Plan — This effort is not provided for in Program Plan 4. Presumably, it can be incorporated in Task 4-3.8 as a subtask.

Major Facilities Required — No special facilities are necessary beyond those available in the Components Program.

Cost and Schedule — This effort is expected to require five years' work at a cost of \$200,000. Guidelines will be prepared by late 1971, and the development of specific devices should be completed by the end of 1973.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	25,000
1970	25,000
1971	50,000
1972	75,000
1973	25,000
Total	200,000

4.2.5. Computer Technology

4.2.5.1. General

The Computer Technology R&D effort supplements current technology to provide the necessary on-line and off-line computer capabilities for LMFBF analysis, control, and protection.

Suitable techniques are well proven for small pressurized water reactor plants, usually based on, at most, a one-dimensional core representation; primarily analog equipment, supplemented by a few digital analysis codes and a relay-logic protection system, has proved satisfactory.

The recent advent of very large pressurized water reactors has strained the capabilities of this traditional approach. The necessity for multi-loop operation, the greater importance of xenon oscillations, the economics-dictated needs for reduced overpower margin, and operation optimized for best fuel management and maximum plant efficiency require better design and operational techniques. Thus, from the design and analysis aspect, current technology is writing spatially dependent digital kinetics codes and improving simulation techniques with better component models and multiplexed analog circuits. From the operational aspect, the operators are being provided with many control rods, which they are to manipulate manually on the basis of calculations from an open-loop, on-line digital computer receiving inputs from hundreds of in-core detectors; hundreds of other instruments throughout the plant are monitored by a digital computer (again open-loop) which provides periodic reports on the plant's condition. In a few cases outside the United States, the loop between the digital computer and the reactor plant has been closed. The water reactor trend is approaching the point of overtaxing the capabilities of both the analog simulation equipment and the operator.

Large LMFBFR plants represent an even greater increase in complexity. Both components and plants are more complicated and delicate. Accidents are more likely to occur and can be more severe. Very large capital costs increase the importance of economics, and great emphasis is placed on on-line reliability. Thus, it becomes necessary to accelerate the present growth in PWR capabilities and to develop capabilities in the areas that are unique to LMFBFRs, such as the presence of an intermediate heat transport system, different core kinetics, and the need for hundred-channel protection functions.

Although only two tasks are included in this effort, both are major efforts of extreme importance to the LMFBFR program. The first task is to develop the capability for on-line computer protection and control, with primary emphasis on the digital mode;

hardware, software, and interface capabilities are all included. The second task is to provide the improved analytical tools necessary for simulation and analysis of LMFBR plants, including both component models and overall plant analysis techniques.

4.2.5.2. Computer, On-Line

Objective — Greatly improved on-line computer capabilities are needed for LMFBR plants. All LMFBRs will require increased capacity for plant control and data logging; the FFTF, and possibly other early plants, will require additional scope for control of experiments; and the B&W reference plant requires a digital computer for protection—a feature that is expected to appear in most other LMFBR designs at least to some extent. The role of on-line analog and hybrid computers in LMFBR plants will be explored, but this effort is primarily devoted to a broad upgrading of digital on-line capabilities, including interfaces with the plant and the operator.

Summary of Current Technology — Digital computers are employed routinely with power reactors for on-site data acquisition and performance calculations, such as rod height predictions and fuel management calculations. (Although this application technically should be classed as "off-line," it is included with the discussion of the on-line computer because it is intended that the latter will take over this function.) Digital computers perform this service very well; perhaps the major improvement desired is in the area of selecting the best way to extract and display meaningful information from the vast quantities of data that are handled.

Direct digital control of industrial processes is a field of rapid growth in the 1960's. Most users are finding some (or considerable) difficulty in starting and operating digitally-controlled processes, but these problems have not stifled growth. Both computer manufacturers and users are acquiring experience on the practical advantages and disadvantages of direct digital control. The following is believed to be a fair assessment of the present situation: In many cases the control system can be made operational only with delays and cost increases, and the final operational system falls short of the original performance objectives; however, the finally achieved performance is

impressive and generally superior to alternate methods. No fundamental problems requiring serious reassessment of the future role of direct digital control are known. The Louisiana Power and Light Company's "Little Gypsy" plant, a digitally-controlled steam-electric station, is a major example supporting this assessment.³¹

Direct digital control of nuclear reactors is in its infancy, particularly in the United States. The French³² and the Canadians³³ are now controlling power reactors digitally (some protection functions are also performed digitally) with very good success; the French even control reactor and turbine plant startup automatically. One domestic manufacturer of boiling water power reactors has come very close to digital control and partial digital protection and intends to come even closer.³⁴ A careful study led to the selection of digital control for the FFTF.³⁵ APDA, considering the Fermi fuel melting incident in retrospect, has recommended the use of a digital computer for Fermi core protection and has identified a substantial increase in protection from its use.³⁶ Thus, although there is a slowly growing trend to perform reactor control and (partially) protection functions digitally, the existing technology provides only a starting point for developing the capabilities needed for LMFBR plants.

Analog control systems and analog/relay protection systems for reactors are quite well developed in the United States and elsewhere. A large amount of accumulated experience facilitates the design of systems that not only perform well, but (perhaps more importantly) are accepted by users and regulatory agencies. The latter claim certainly cannot be made for digital systems in this country. However, for an LMFBR, the complexity of the control system and at least some of the protective functions appears to be sufficient that digital equipment becomes superior. As an extreme example, fuel assembly voiding in the B&W reference core is a 2-of-288 coincidence scram. With conventional protection system techniques, where a simple 2-of-3 coincidence requires six relays per (redundant) scram breaker, the required number of relays would be fantastic; in contrast, this function can be conveniently handled digitally.

Scope of Work — This is a broad effort comprising two major areas: first, improving computer software, interface

equipment, and (to a limited extent) hardware; and second, improving human abilities for efficient use and reasonable acceptance of on-line digital computers. These two areas interact extensively, since progress in either must be accompanied by advances in the other; they cannot be separated into separate tasks. Accordingly, the scope of work must consider both areas in parallel.

The first phase will essentially scope the entire effort. It will consider the protection, control, and monitoring needs of LMFBR plants based on the latest reference, predecessor, and demonstrator plant designs, as well as the projected analog, hybrid, and digital equipment capabilities for meeting these needs. Close coordination with the Plant Design Program will be maintained during this phase. Preliminary consideration will be given to the work load division between the computer and the operator. The remainder of this scope of work is written on the assumption that all or nearly all of the functions will be performed best digitally; if this assumption is incorrect, some revision of the remainder of the scope will be necessary. Several parallel efforts will follow the first phase, as described below.

One of the largest efforts will be devoted to software, beginning with the selection of a compiler. In order to make the results of this effort broadly applicable to a variety of reactors and computers, machine language is undesirable, but it is not clear that any existing compiler language will be able to meet all requirements. Driver³⁷ advocates the development of a new compiler language, an admittedly expensive task. Wolfson³⁷ suggests using linkage editor programs to form a "hybrid" compiler system from parts of existing languages. This area will receive intensive study, with the desired objective of avoiding the generation of a new compiler, since—at least initially—its applicability would be limited to very few machines:

The other important aspect of software is the development of protection and control programs. This effort will be coordinated with section 4.2.5.3 to minimize the amount of work, but it will entail considerable effort in writing protection and control programs. Although much of this work will be closely reactor-oriented, the relative newness of sampled data control systems may also cause some non-reactor problems.

The hardware effort will be smaller than that devoted to software work and will involve more selection than development. Computers to be available in the next few years, including both general- and special-purpose types, will be surveyed, and a few will be selected as most suitable. Capability for multi-programmed operation is essential. Other efforts will be built around the characteristics of the selected machines.

Computer interfaces will be considered in two stages: first, the exact point at which computer peripheral equipment meets the various instruments and actuators will be optimized (e. g., where does the instrument circuitry stop and the computer start?); second, hardware development will be undertaken as necessary to ensure the availability of suitable peripheral equipment.

Operator input/output devices will be treated in the same way as electrical interfaces, with the added provision that considerable effort will be devoted to deciding what information should be presented to the operator. Rigid conclusions are not necessarily expected in the latter area, but as a minimum, a set of guidelines and practical limits should be established. The hardware development effort should be small.

The ability of the computer to test itself and other equipment is extremely important. Self-testing is a potential strong point of a digital protection system, but the matter has not been attacked with the care and depth necessary for reactor protection. This effort must consider such aspects as the types of test signals, their frequency, priority relations to other functions, and action to be taken in case of failure. Fault analyses must examine redundancy and independence requirements. This is a dual-natured effort intended, first, to establish a number of design criteria and guidelines and, second, to contribute to safety and reliability analyses.

The products of the various parallel efforts described so far will be assembled into what could be described as a "preliminary design." This "design" will be complete enough that it can be subjected to a round of revision and evaluation.

The overall demand on the computer and the load split with the operator will be reviewed and again optimized to a

range. The results of specific control system studies in the Plant Design Program will be considered carefully.

The programs will be evaluated and improved by constructing a control and protection system and testing with a plant model. This effort will explore and compare various techniques, but it is not intended to produce final system designs for any specific plant; that task will be performed in the Plant Design Program and under specific project design efforts.

Comprehensive safety and reliability analyses, performed in conjunction with the Safety Program and supplemented by testing as required, will determine the validity of the effort. These analyses will be even more stringent than is customary for a particular plant because of the differences in approach and application.

Finally, the entire effort will again be reviewed and improved as indicated by the performance and safety analyses, and close coordination will be maintained with individual reactor projects to assist with design problems.

Relation to Program Plan — This effort compares closely in scope with Task 4-4.2 of Program Plan 4. However, Task 4-4.2 still retains much of the "Computer Control Center" advocated by Driver,³⁷ in that a considerable amount of plant-related work is to be performed by an organization responsible for supplying neither computers nor reactors. Thus, although this effort includes the major addition of developing digital protection system techniques, the estimated cost is considerably less than that for Task 4-4.2. Part of this difference may also result from the apparent probability that Task 4-4.2 would develop a new compiler.

Major Facilities Required — The availability of suitable computing facilities on a rental basis is assumed. No other major facilities are required.

Cost and Schedule — This effort is expected to require nine years' work at a cost of \$2,800,000.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	250,000
1970	350,000
1971	400,000
1972	400,000
1973	400,000
1974	400,000
1975	300,000
1976	200,000
1977	100,000
Total	2,800,000

4.2.5.3. Simulation and Analysis

Objective — Better methods are needed for analytically predicting the performance of the LMFBR and overall plant. Although light water reactor plant analysis techniques are generally satisfactory, the additional physics and engineering complexities of an LMFBR require a considerable advancement in technology. This effort will develop the improved capabilities necessary for simulation and analysis of individual components (including the core) and the overall LMFBR plant. The effort covers general studies which can be applied to particular LMFBR designs but excludes work on those particular designs.

Summary of Current Technology — Performance analysis studies of light water reactor plants are routinely performed on analog computers, supplemented by a few digital codes such as WIGLE, PARK, and CHIC-KIN. The simulation of an LMFBR plant rapidly overflows the capacity of any practical analog facility. Some of the reasons for this difference are (1) the complex arrangement of multiple series and parallel heat transfer paths coupled in the B&W reference design with a pot; (2) variable flow in all heat transfer paths; and (3) the greater accuracy needed because of the high system temperatures, large temperature variation, and fast core response. Reference 38, describing a model of a single heat transfer path for the FFTF (which does not have a steam generator), illustrates the facility-crowding problem. In turn, the complexity of the LMFBR model increases the difficulty of analyzing plant performance and designing control systems.

Purely analog techniques are not sufficient for LMFBFR plants; hybrid and/or digital methods are required. Existing digital methods are too limited in capacity and scope to satisfy the needs of the LMFBFR. Hybrid techniques are in their infancy, although some work with direct applicability to LMFBFR plants is being performed.³⁹ Hybrid machines on the market to date represent an uneven combination of digital and analog virtues; better-optimized hybrid systems will become available soon.

Major improvements are also needed in the simulation of components. Reference 40 is an example of recent work in the digital simulation of a steam generator. Space-dependent kinetics models for core simulation are quite new and can be very complex; hopefully, additional effort in this area will be fruitful.

A final problem in LMFBFR analysis is the lack of operating plants approximating the design of future LMFBFRs and the resultant inability to measure plant performance directly. The vital interplay between theory and experiment is severely limited; hence, methods development is more difficult and more urgent.

Scope of Work — The scope of work broadly consists of (1) developing models for plant components including the core (in conjunction with other programs); (2) developing overall plant dynamics models, working with the Safety and Plant Design Programs; (3) adapting these models to modern hybrid and digital equipment to develop usable codes; and (4) improving these codes based on maximum interplay with experiments in other programs. Generally, better models will be fitted to better equipment. This effort will interact with section 4.2.5.2 to some extent, but the on-line and off-line applications are sufficiently different that the overlap of interest will be quite small.

The Components Program will develop good analytical descriptions of the major plant components. This effort will be responsible for turning these descriptions into practical models for plant analysis, in cooperation with the Plant Design Program. Heat transfer simulation is of particular importance. This effort will also work with the Physics Program to develop the necessary reactor kinetics models. In these areas, multiple models must be developed to cover

major variations from one plant to another and to permit the derivation of analytical methods of varying depth and complexity.

With some interfacing with the Safety and Plant Design Programs, overall plant dynamics models of varying complexity will be developed. The component descriptions will be fitted into these models, which will be shaped for smooth application to modern hybrid and digital equipment. The treatment of variable flows and time delays will be an important part of this phase. Again, a variety of models will be necessary to cover variations in plant designs.

Usable computer codes and techniques will be prepared, based on the component and plant models. Maximum interplay will be sought with tests and experiments in the Components, Physics, and Safety Programs and with operation of EBR-II, Fermi, and finally the FFTF, to permit improving the techniques. It is also essential to the success of individual plant control studies that the codes and techniques be compatible with control system models based on the on-line computer R&D in section 4.2.5.2 (evaluated in the Plant Design Program).

Finally, the results of these efforts will be reviewed in terms of experience gained in specific plant designs in the Plant Design Program and later with the FFTF. Some methods improvement will be performed on the basis of this experience. It is expected that after several years the individual projects will be able to assume the methods development.

Relation to Program Plan — This effort is very similar to Task 4-4.1 of Program Plan 4. A final transition of work to specific reactor projects is planned, perhaps sooner than in Task 4-4.1.

Major Facilities Required — It is assumed that the hybrid and digital computers coming on the market in the next few years will be available on a rental basis. On the premise that pertinent tests will be performed under other programs, no other major facilities are required.

Cost and Schedule — This effort is expected to require eight years' work at a total cost of \$1,300,000.

<u>Calendar Year</u>	<u>Cost, \$</u>
1969	200,000
1970	200,000
1971	200,000
1972	200,000
1973	200,000
1974	100,000
1975	100,000
1976	100,000
Total	1,300,000

4.3. Cost Benefit

Failure to meet the goals of the Instrumentation and Control R&D plan can affect future LMFBR plants significantly. The permutations of many different R&D tasks, many possible shortcomings per task, and many possible effects per shortcoming, are too numerous for analysis here. However, these effects generally fall into the two main categories discussed below.

4.3.1. In-Service Failures

Random or end-of-life failure of instruments which otherwise satisfactorily perform their intended functions may lead to plant downtime for maintenance and repair. The associated fuel cost penalty, based on a nominal plant capacity factor (PCF) of 0.861, is given by

$$\Delta(\text{cost}) \text{ in mills/kWh} = 2.97 - 3.45 (\text{PCF})'$$

$$(\text{PCF})' = \frac{314 - \text{days/yr down}}{365}$$

In the range of one to ten days per year added downtime, the annual fuel cost penalty is 0.01 mill/kWh per day.

If the plant must be shut down to repair an instrument, or if the instrument causes a false shutdown, the time required for subsequent plant startup must be added to the maintenance time, as must any time required to prepare for unscheduled maintenance. Thus, in many cases the repair of a failed instrument can cause a full day of downtime. It is further noted that simple translation of downtime to fuel cost penalty

in this manner may underestimate the actual penalty to the utility operating the reactor, since unscheduled shutdowns may well occur at inconvenient times.

Unsafe failure of protection instruments is a special case of instrument failure. The possible consequences are so severe that an operating license would not be granted to a plant whose protection system permits a significant possibility of undetected failures. Much of the R&D plan is directed toward the improved quality control and precise performance determination needed in this area.

Instrument failure can also lead to another intangible disadvantage: loss of operator confidence. Since no automatic protection system will ever give complete protection against all possible damage to plant equipment, the operator can prevent or limit incidents in many cases. The Fermi fuel meltdown is just one of many instances where operator action based on existing instruments could have forestalled serious consequences, had those instruments been operable and had they been relied upon by the operator.

4.3.2. Performance Deficiencies

This category relates to the effects of less than complete success in meeting the basic instrument performance objectives in the R&D program, without regard to random or end-of-life failures. In general, the R&D effort must identify these shortcomings, so that plant designs can be changed accordingly.

Some detectors may suffer long-term damage from radiation or corrosion. If these effects are quantitatively determined by R&D, minor effects can be compensated for and major effects can be avoided by substituting scheduled replacements for unscheduled shutdowns.

The effects of temperature on detectors are important. Simple errors can be compensated for if they are known from the R&D work. Gross inability to withstand environmental temperatures should be corrected if possible, but they must be identified so that alternates, such as forced cooling, can be designed into plants if necessary. It may even be necessary to substitute "second-choice" instruments in some cases (e.g., thermocouples instead of resistance temperature detectors).

Such alternates can be invoked at much less cost if they are identified before plant construction begins.

The under-sodium viewer is a special case. Cost benefit must be measured primarily in terms of downtime required for unpredictable circumstances. In extreme cases, a high-capability viewer could conceivably save months of downtime.

Instrument errors directly affect plant economics, since performance is ultimately determined by measurement. Errors in the instruments that are used for power measurement and for the calculation of design overpower margin relate linearly to fuel cost. Errors in protection instruments affect setpoints linearly; thus fuel cost is affected indirectly (if these errors are known; if unknown, they affect protection).

The computer technology R&D area has much the same effect on economics as has instrument error, with the stipulations that the protection system must meet minimum standards set by regulatory agencies, and the control system reliability affects plant capacity factor directly. A lack of knowledge of plant performance and controllability during the design stages requires greater design conservatism, and a poorly performing control system yields poorer performance. Reduced capital and maintenance costs are further expected with optimum digital protection and control systems, which should also reduce downtime by virtue of their better maintainability.

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5. SODIUM TECHNOLOGY

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5. SODIUM TECHNOLOGY

5.1. Introduction

5.1.1. General

The sodium technology needs for the B&W reference design are predicated on the basis of the following design parameters, features, and assumptions.

1. A maximum hot-spot sodium temperature of 1250 F in the reactor core.
2. An integral pot design for the primary system.
3. The use of a low-alloy steel in the intermediate sodium system.
4. A maximum intermediate sodium temperature of 1000 F.
5. The assumption that sodium purity requirements would not dictate the use of a gettering system.
6. A vented fuel design.
7. The presence of carbon in thermal shield cans.
8. Maximum sodium velocities of 25 fps in system pipe-work.
9. Argon blanket gas throughout the plant.
10. A minimum of hard facing materials on components in the reactor vessel.

The effects of these items on the R&D required for sodium technology will be apparent in the following discussion.

5.1.2. Summary Budget and Schedule

Table 5-1 is a summary of the budget and schedule for sodium technology research and development tasks.

5.2. R&D Requirements

5.2.1. Materials Compatibility

5.2.1.1. Objective

The potential materials compatibility problems associated with the operation of the primary and intermediate sodium systems on the B&W 1000-MWe LMFBR power plant will be investigated.

Table 5-1. Summary of Budget and Schedule for Sodium Technology

Task	Cost, \$ × 10 ³								Total
	Calendar year								
	1969	1970	1971	1972	1973	1974	1975	1976	
<u>Materials Compatibility</u>									
<u>Metallic mass transfer</u>									
1. Short-term effect on cladding	--	--	--	--	500	1500	1000	--	3,000
2. Long-term effects on structural SS	500	500	500	500	1000	1000	1000	--	5,000
3. Long-term effects on bimetallic systems	500	500	500	500	1000	1000	1000	--	5,000
<u>Carbon transfer</u>									
1. Short-term effects at high carbon contamination	--	--	--	--	1000	1000	1000	--	3,000
2. Long-term effects on bimetallic systems	500	500	500	500	1000	1000	1000	--	5,000
3. Carbon transfer control	--	--	--	--	--	--	--	1000	1,000
<u>Surface cleaning methods</u>									
	--	--	--	--	500	500	--	--	1,000
									23,000
<u>Sodium Purification</u>									
1. Review of design requirements	100	--	--	--	--	--	--	--	100
2. Design survey	--	100	--	--	--	--	--	--	100
3. Cold trap design	--	--	500	--	--	--	--	--	500
4. Model tests	--	--	--	100	--	--	--	--	100
5. Supplemental development programs	--	--	--	100	--	--	--	--	100
6. Development of test facility requirements	--	--	--	50	--	--	--	--	50
7. Design of test facility adaptors	--	--	--	100	--	--	--	--	100
8. Test program development	--	--	--	200	--	--	--	--	200
9. Development of test data extrapolation methods	--	--	--	--	200	300	--	--	500
10. Data analysis	--	--	--	--	--	--	300	--	300
									2,050
<u>Fission Products in Sodium</u>									
1. Fission product fate study	--	--	--	--	500	1000	1000	500	3,000
2. Transient behavior of short-lived fission products	--	--	--	--	500	1000	1000	500	3,000
3. Fission product effects on maintenance and decontamination	--	--	--	--	--	--	--	500	500
4. Development of noble gas separation system	200	200	1000	600	--	--	--	--	2,000
									8,500
<u>On-Line Monitoring Method</u>									
1. Plugging meter development	--	--	--	--	500	500	--	--	1,000
2. Na-H ₂ O reaction detector development	--	--	--	1000	1000	2000	500	500	5,000
									6,000
<u>Sampling and Analysis</u>									
1. State-of-the-art assessment for O, C, H, N, Cr, Ni, Fe, Co	200	--	--	--	--	--	--	--	200
2. Development of most promising analytical methods	800	1000	1000	1000	--	--	--	--	3,800
3. Development of sampling methods	200	200	400	200	--	--	--	--	1,000
									5,000
<u>Fundamental Properties of System Na</u>									
1. Transport properties (O, C, N, H)	--	--	--	1000	1000	2000	500	500	5,000
2. Kinetics of solution and deposition of Fe, Ni, Co, and Cr	--	--	--	1000	1000	2000	500	500	5,000
3. Kinetics of solution and deposition of fission products	--	--	--	500	500	500	500	--	2,000
									12,000
Grand total									56,550

5.2.1.2. Summary of Current Technology

Metallic Mass Transfer — Metallic mass transfer is a poorly understood phenomenon that is now receiving some attention in studies at General Electric and Atomics International. Their contribution consists mainly of empirical equations relating the solution and deposition rates of type 316 stainless steel. No adequate theoretical explanations of the solution-deposition process have been advanced, and a great deal of work is needed in this area to guide experimental programs and to assist in data correlation.

Carbon Transfer — Characterizing the behavior of carbon impurities in sodium is complicated by the fact that at least five species can exist in a system, and activity cannot be correlated with a total carbon content determined by sampling and analysis. However, some qualitative conclusions have been drawn regarding the behavior of carbon in sodium; some of them are listed below.

1. Carbon transport is limited by its diffusion rate in the alloy.
2. The transport potential is a function of the difference in carbon activity rather than in concentration.
3. Carbon is lost from such low-alloy ferritic steels as Croloy 2-1/4 and is gained by austenitic stainless steels like type 304.
4. Loss of carbon from Croloy 2-1/4 tends to weaken the material.
5. Carbon retention by type 304 stainless steel tends to embrittle the material in a structural member.
6. Carbon can be transferred from nonstabilized to stabilized types of stainless steel.
7. Carbon transfer is a more serious metallurgical problem than metallic mass transfer, because carbon diffusion rates are faster; this results in a greater penetration of the structural member.
8. Carbon activity is significantly affected by other non-metallic impurities; however, interaction mechanisms are not understood.
9. The solubility of elemental carbon in sodium is too small to measure with available chemical or instrumental techniques. Quantitative data on carbon transfer are difficult to obtain and are unavailable at the present time.

Surface Cleaning Method — A number of methods have been used successfully to remove residual sodium from components and fuel assemblies; however, deleterious surface reactions have been noted in some cases. Section 17.9 of the Liquid Metals Handbook gives a detailed discussion of the merits and demerits of the accepted techniques. It appears that most of the development work on cleaning methods has been concentrated on achieving a sodium-free surface by economical means and that relatively little attention has been given to evaluating potential permanent damage to the surface being cleaned.

5.2.1.3. Technical Justification

Metallic Mass Transfer — Current data indicate that the solution attack on austenitic stainless steel by sodium is of little concern at temperatures in the neighborhood of 1000 F. The cladding surface in the reactor coolant system is the only area of the B&W design where short-term solution attacks might prove to be a serious problem and should be thoroughly investigated. This conclusion is supported by the following considerations:

1. Temperatures can reach a hot-spot maximum of 1250 F.
2. Cladding is only 10 mils thick (1.5-mil corrosion allowance).
3. All structural members exposed to sodium above 1000 F can have generous corrosion allowances without significant economic penalty. Only the upper part of the core vessel and the IHX will operate in temperatures that exceed 1000 F ($T_{\max} = 1100$ F).

Long-term mass transfer must be investigated for temperatures up to 1100 F for types 304 and 316 stainless steel to obtain conclusive data to support reliability and safety claims for an expected plant life of 30 years. Similar data must be generated on Croloy 2-1/4, which finds use in the intermediate system up to 1050 F.

Carbon Transfer — All components and structural members exposed to primary sodium will be stainless steel; hence, carbon transfer in this system would not normally be a serious problem. On the other hand, there is always the possibility that a thermal shield can fail, resulting in a large source of carbon contamination, which

would ultimately lead to gross surface carburization of the stainless steel. Again, the area of greatest short-term concern is the thin, highly stressed fuel cladding. The long-term effect of gross carburization on the remainder of the primary system is also of great concern from a reliability and safety standpoint.

Because the intermediate system is constructed of dissimilar metals (type 304 stainless steel in the IHX, Croloy 2-1/4 everywhere else), carbon transfer will be an everpresent problem.

A minimum requirement of the carbon transfer R&D program will be to obtain accurate data on carbon transfer rates to support the intermediate system design. The most desirable result of an R&D program would be a technique to inhibit carbon transfer.

Surface Cleaning Methods — The present refueling procedure in the B&W reference design does not include a step to remove the sodium from a subassembly prior to shipment off-site for reprocessing. If a fuel failure occurs, it is not likely that the exact subassembly having the failed pins can be detected and removed from the reactor without inspecting a number of adjacent subassemblies. For this reason, it is anticipated that a thorough cleaning process will be needed to remove sodium from subassemblies so that the thin cladding (10 mils) will not be attacked. Any deleterious alteration of the surface of the fuel pins and/or subassembly can result in scrapping a good unit, with a resulting cost penalty of up to \$100,000.

5.2.1.4. Scope of Work

Metallic Mass Transfer — The investigation of metallic mass transfer effects for the B&W reference plant is divided into three tasks, as follows:

Task 1: The reference design fuel cycle calls for a subassembly exposure of 1050 effective full power days in a total residence time of 3.33 calendar years. To ascertain the net effect of short-term metallic transfer on the cladding, a test will be run to simulate conditions in the fuel bundle for a period that exceeds the number of full power days. Sodium purity will be provided by high-performance cold traps (8 to 10 ppm), such as those found in the reference plant. The results from this test will indicate either the validity of the present

design assumptions or the need to investigate alternate claddings and/or lower sodium purity limits.

Task 2: Long-term metallic mass transfer tests on types 304 and 316 stainless steels in a sodium environment simulating the reactor coolant system will be conducted in this task. The maximum sodium temperatures will not exceed 1100 F, and the minimum will not be less than 800 F. The test duration will be sufficient to provide data to support extrapolations to 30 years.

Task 3: Long-term metallic mass transfer tests on a system simulating the LMFBR intermediate system, which contains type 304 stainless steel (in the IHX) and Croloy 2-1/4 will be conducted. Temperatures in this system vary from 700 to 1050 F. Again, the test duration will be dictated by extrapolation confidence.

Carbon Transfer — The important R&D tasks related to carbon transfer are as follows:

Task 1: Short-term tests in grossly contaminated sodium will be run on type 304 stainless steel at temperatures ranging from 800 to 1250 F to determine the extent of the deleterious metallurgical changes after an accidental release of large quantities of carbon in the primary system. Similar tests will be run on type 316 stainless steel at temperatures up to 1100 F.

Task 2: Tests designed to obtain long-term carbon transfer data for the bimetallic intermediate system will be pursued in this task. Since there are no potential sources of gross carbon contamination in the intermediate system, emphasis will be placed on the accurate determination of the carbon transfer rates between type 304 stainless steel and Croloy 2-1/4 over a temperature range of 700 to 1050 F.

Task 3: A means to control carbon transfer in the intermediate system will be developed in this task. Inherent control by the addition of stabilizing alloys to the structural materials will be investigated. Alternate techniques, such as additives to the sodium coolant, will also be investigated.

Surface Cleaning Methods — The objective here is to develop an economical surface-cleaning method for the B&W reference design vented fuel subassemblies; a method that definitely will not produce harmful metallurgical changes is needed. Ideally, the solvents used will be compatible with the oxide fuels used in the B&W core.

5.2.2. Sodium Purification

5.2.2.1. Objective

The primary objective of this program is to develop a cold trap for controlling the purity of the sodium coolant that is compatible with a pot-type reactor design. The reference plant, as it now exists, calls for two cold traps in the pot for purification of the reactor sodium and a single cold trap for each loop for purification of the steam generator sodium. After detailed consideration of the problems associated with fabrication, performance, maintenance, and cost, this program will determine whether an in-pot or a separately shielded, piped purification system should be developed for reactor sodium. A design representing the type of cold trap selected will then be developed. The program will also determine the arrangement for the purification cold trap for the intermediate system, based on performance, maintenance, cost, and its interrelationship with the intermediate system.

5.2.2.2. Summary of Current Technology

A number of reactor plants have operated with piped cold trap systems; these include EBR-I and -II, HNPF, SRE, and the Enrico Fermi plant. In most large installations a bypass system is used to contain the cold trap, plugging meter, or other oxide sensor; flow meter; pump; and other associated devices. Generally, the maximum flow in the bypass system is rated at 1% of the full system flow.

The cold trap itself consists of an economizer and a vessel packed with stainless steel mesh or other suitable material. The coolant can be any fluid that is compatible with sodium, such as NaK.

The Fermi plant uses a 500-gallon-capacity cold trap cooled with NaK. The sodium flow rate is 100 gpm, and the primary system sodium inventory is 500,000 pounds.

The cold trap in the HNPF is similar in design to the Fermi cold trap and to the one used to remove fission products from the SRE after the fuel damage occurred.

In general, primary system cold traps remove oxygen (present in the initial charge), carbon, iron, silicon, manganese, and chromium, as well as fission products such as ^{137}Cs . These traps have worked satisfactorily in the past, but they have not been used with vented fuel elements.

Maintenance problems associated with the highly radioactive contents of the trap, which have not been completely worked out, will depend to a large extent on the effects of the radioactivity in the cold trap. In general, cold trap designs for intermediate sodium purification are quite similar to those used for reactor sodium cleanup.

5.2.2.3. Technical Justification

Past experience has shown that removing oxides from sodium by precipitation is a feasible method that can be employed to reduce oxide concentration to approximately 10 ppm. Precipitation is also effective, to varying degrees, for removing other impurities as well. However, sufficient knowledge of the basic mechanisms that will provide the designer with quantitative information is still lacking. This fundamental R&D effort is described in the more basic programs covered in section 5. The program described here addresses itself specifically to the principles of design, fabrication, and maintenance of precipitation purification systems for the reactor coolant as well as the intermediate coolant. The in-pot purification system to be used for the reactor is considerably more complex than that for the steam generator. The complexity of the reactor coolant purification system arises from the following situations:

1. The sodium to be purified is highly radioactive, since it contains the long-lived fission products ^{24}Na and ^{22}Na .
2. The pot configuration of the plant makes it very desirable to retain the sodium purification system in the reactor vessel without external processing.

Item 2 requires the cold trap to function in a high-temperature (about 800 F) sodium environment during operation. This program will evaluate the desirability of such a location; if it is found attractive, we will

the problems associated with the design, operation, and maintenance of such a cold trap will be solved. In addition, the program will provide design information for piped systems, in both the reactor and intermediate coolant systems, tailored to B&W's 1000-MWe LMFBR reference design.

5.2.2.4. Scope of Work

Task 1: The design requirements already formulated for the reactor and intermediate coolant systems must be reviewed and evaluated in light of the desired performance of the purification systems. The requirements must be compatible with what is already known about the basic characteristics of precipitation traps, regardless of whether an in-pot or piped design is adopted. A thorough understanding and justification of the basis for the requirements and a realistic evaluation of the effects of only partial achievement are required.

Task 2: A literature survey should be made to assimilate the more recent experience of others in the areas of design, fabrication, maintenance, and operation of cold traps. Special attention should be given to materials, cooling requirements, capacity, instrumentation, maintenance, and design. The results from this survey, combined with information developed in the basic program and the evaluation of design requirements, will be used to develop modified design requirements.

Task 3: A detailed engineering study will determine the optimum arrangement of cold traps for both the reactor and intermediate coolant systems. Special attention will be given to the probability of solving the in-pot insulation, temperature, maintenance, and instrumentation problems. A study will be conducted to determine the size and number of in-pot cold traps required, based on methods of removing fission products and oxide impurities. Studies will also be conducted to determine other cold trap characteristics, such as the capacity for removing oxygen and other impurities, compatibility with plant downtime, reliability, equipment maintenance, the arrangement's compatibility with other equipment, and the effect of seal leakage on the plant waste gas disposal system. In addition, a piped cold trap purification system for the reactor coolant will be investigated. The results of these

studies will be evaluated to determine which approach to follow, i. e., in-pot or piped cold traps. The intermediate coolant requirements will also be studied to determine whether an integral design (as is now shown for the reference plant) or a number of smaller cold traps best meet the purification needs in light of task 2. After the work described above has been completed and the technical approach has been chosen, a detailed design, adequate for fabrication of the pertinent parts of the purification system, will be made. The design work will form the basis for the hardware portion of this program.

Task 4: In all cases, a testing program will either confirm assumptions or supply information required by the designers. Wherever practical, the tests will utilize full-scale models or, where not practical, models as near as necessary to full scale without undue penalty on the test facility or the time requirement. The models will provide representative information associated with temperatures, thermal duties, sodium impurities, velocities, pressure drops, operational instrumentation, residence time, packing densities, purification performance, and the like. The models will be so constructed that they prove out fabrication techniques and environments where there is any doubt about either. The models will also be capable of demonstrating the method of removing the oxide-laden radioactive sodium before placing the cold trap back in service.

Task 5: Based on the detailed design accomplished under task 3, a separate program may be required to develop special weld and mechanical joints. Also depending on the design, there may be requirements for special remote handling tools and equipment. Should it become necessary to provide this kind of information, a program will be undertaken, using mockups wherever possible to prevent delays in the main program. Problems of this nature will be identified during the early design stages.

Task 6: As a result of model testing, facility requirements will be developed. Because of the relatively large models that will be needed to verify and obtain design and performance information, the use of National Program facilities will be assumed. The 4-inch loop now under construction at ANL appears to be a satisfactory

facility for this purpose. However, detailed scheduling of the many activities may indicate that other approaches have to be considered.

Task 7: A design effort will be required to integrate the component models with ANL's 4-inch loop test facility. Modifications to the loop and adaptations to the scaled cold traps will be made.

Task 8: A complete testing program must be written to cover the short- and long-term testing, the parameters to be held constant during each run, and the variables to be investigated. Provisions should be made for non-destructive testing at various time intervals, as well as chemical and metallurgical analysis—all as part of the testing program. The test program should be based on developing data that will allow the designer to obtain the necessary information for the configuration in question, and it should allow time for evaluating the relative importance of each variable. In addition to performance testing, the program should include periodic maintenance, repair, and regeneration of the cold trap under conditions identical to those found in an operating plant.

Task 9: Since the purpose of the entire program is to provide design information, it will be necessary to take the raw data and transform them into meaningful correlations. Therefore, it will be necessary to develop analytical techniques that will utilize the experimental data to predict performance as well as to direct modification of the model and functional requirements. This effort should be started as soon as the technical approach becomes clear.

Task 10: At the completion of the program, the results will be organized to establish, with a high degree of certainty, those factors that are pertinent in the design of cold traps. The results will include information that will be necessary to establish design standards relating to performance, lifetime, fabrication, materials, maintenance, instrumentation, and the requirements associated with the trapped fission product.

5.2.3. Fission Products in Sodium

5.2.3.1. Objectives

The objectives of this program are as follows:

1. To determine the fate of fission products released in the reactor coolant system of B&W's integral pot LMFBR by cladding failures and by intentional fuel venting.
2. To support the development of sensitive and accurate instrumentation for detecting failed fuel and to develop techniques for assessing the magnitude of the failure.
3. To evaluate the effects of vented fuel and failed fuel on equipment maintenance and decontamination problems.
4. To develop an economical radioactive noble gas separation system for the B&W reference design LMFBR.

5.2.3.2. Summary of Current Technology

A great deal of information has been accumulated on fission product release in water-cooled reactor systems mainly in the process of satisfying safety requirements. To a limited extent, these data are applicable to sodium reactors. Atomics International is conducting experimental and analytical studies to evaluate the release and transport characteristics of fission products in molten sodium. The general objective of their work is to evaluate the ability of the sodium coolant to retain the specific fission products that have been released to the coolant stream. Their specific objectives have been (1) to determine the ability of sodium to retain radioactive iodine as a function of such parameters as sodium temperature, elapsed time, sodium-to-fuel and sodium-to-gas volume ratios, geometric relationships, and conditions of released iodine; and (2) to evaluate the ability of boiling sodium and sodium in the vapor phase to retain fission products. These experiments have demonstrated that sodium tends to retain fission products, but extensive plating and deposition of fission products on various surfaces was observed throughout the experiments. An examination of contaminated reactor sodium piping indicated some correlation of fission product activity with carbonaceous impurities in the system.

At the Aix conference, Saroul of CEA-Cadarache presented the results obtained from experiments on the behavior of fission products in sodium and argon.¹ The results indicated that the only fission product detected in the cover gas, other than the inert gasses, was iodine-131. The iodine was found in the vapor trap and in the asbestos filter. No iodine was detected in the activated carbon filter; thus Saroul was led to conclude that the iodine is present only as sodium iodide.

Some insight into the problems associated with fission product release in sodium systems has been gained in the operating experiences with sodium-cooled reactors in the U.S. and in other countries. EBR-I, EBR-II, SRE, and Fermi have operated with sodium or NaK as coolant, and they have experienced fuel meltdowns in which fission products were released. Considerable information has been gained by examining these incidents.

A number of technically sound methods for separating the noble gasses are available today. Major candidates include absorption, fractional distillation and condensation, liquid absorption, thermal diffusion, and diffusion through selective membranes. Adaptability and economics are the unknown areas that must be resolved before an acceptable system can be formulated for the 1000-MWe LMFBR.

5.2.3.3. Technical Justification

Vented fuel gives a new dimension to the problems of safety and of handling radioactive materials at a power reactor site. Many areas of concern that were previously the domain of the fuel reprocessing plant have become priority problems for the LMFBR designer. The disposal of radioactive waste gas is notable among these inherited problems.

The production of radioactive noble gases is a direct function of the thermal power output of a reactor core. As the installed capacity on a single site increases, the problems of disposing of the gaseous radioactive wastes safely are increased. There is now a practical activity release limit of approximately 5000 $\mu\text{C}/\text{sec}$ for ^{85}Kr in a single area. Exact limits depend upon meteorological conditions and the reactor location. Pollution control trends point toward even more strict limitations than are now imposed. By the time a 1000-MWe LMFBR goes on line, the release of any radioactivity at all may be prohibited.

Sodium-cooled breeder reactors tend to compound the problems of waste disposal, especially gaseous waste, in these ways:

1. They require large-capacity plants in order to be economically competitive with PWR systems.

2. The direct relationship of fuel cost to cladding thickness (10^6 \$/mil) is very sensitive; hence, a vented fuel design is favored.
3. There are numerous unsolved seal problems in the reactor vessel cover structure and other critical areas.
4. They require large volumes of inert gas to blanket the highly reactive sodium. Candidate blanket gasses are either expensive, cause metallurgical problems, or become activated.
5. Activated sodium vapor exists in the blanket gas atmosphere.

The alternatives are quite clear for the B&W reference design. If adequate seals (dynamic and static) can be developed to limit gas leakage into the reactor vessel to less than 0.1 scfm and leakage of the radioactive blanket gas out of the vessel to 4.1 cc/day, then a holdup-for-decay scheme will work satisfactorily (assuming a release rate of 5000 μ c/sec for ^{85}Kr). Otherwise, a continuous purge of sufficient magnitude to preclude blanket gas leakage into the reactor building, followed by a noble gas separation and storage system, will be required. Most of the xenon and krypton will be separated from argon and permanently stored. The remaining gas stream could be recirculated, vented to the atmosphere, or both. Most of the radioactivity in the cover gas would then have been isolated.

Although the reference design 1000-MWe LMFBFBR calls for a holdup-for-decay system, it is the opinion of the B&W Systems Engineering staff that an economical separation and permanent storage system must be developed as a backup for use with large sodium-cooled reactors.

The B&W reference design core comprises 95,328 vented fuel pins clad with 10 mils of stainless steel. Fuel swelling from high burnup (up to 100,000 MWd/T) coupled with hot-spot temperatures approaching 1250 F will no doubt result in some cladding failures. The addition of fission products to the reactor coolant via cladding failure apparently will not jeopardize plant safety or cause increased operational difficulties. However, reactor operation with detected fuel failures will be limited if it cannot be positively guaranteed that a failure will not propagate into an accident condition.

Plans for the development of failed-fuel detectors are found in section 4.2.1.5 of this report. The most promising technique is that of measuring the activity of short-lived neutron emitters produced by fission. Detectors in the reference design plant are located apart from the core, and quantitative estimates of the severity of a cladding failure will be meaningless without a knowledge of fission product behavior in a pot-type system. Without a reliable means of correlating the degree of cladding failure to the indicated fission product activity, there will be no recourse to reactor shutdown for fuel inspection.

Although the reference plant has been designed with cognizance of fission products in the reactor coolant and the argon blanket gas, a great deal of work must be done to substantiate the assumptions made. In particular, an accurate means for predicting the fate of fission products must be developed before the final design of reactor components and systems is established. As a minimum, techniques must be developed for predicting the character and quantity of radioactive deposits on components that must be removed from the reactor for maintenance.

5.2.3.4. Scope of Work

The study of fission products in sodium will be handled in four tasks:

Task 1: A comprehensive analytical program, supported by experimental verification when necessary, will develop reliable techniques for predicting the fate of all fission products released to the reactor coolant or blanket gas in a 1000-MWe integral pot reactor design.

Task 2: The transient behavior of short-lived fission products released to the reactor coolant will be investigated to provide a basis for correlating the extent of fuel failures with detection indications for a pot reactor concept.

Task 3: Following task 1, a study will be conducted to ascertain the effects of fission products on component decontamination and maintenance procedures. In addition, positive

recommendations will be made for the design of handling equipment and other facilities exposed to contaminated components.

Task 4: A noble gas separation system for the B&W 1000-MWe LMFBR will be developed in this task. System size will be a function of the inefficiency of the reactor vessel seals. Since this is an unknown factor at this time, the R&D program will be based on developing systems capable of handling a waste gas purge rate from 0.1 to 100 scfm.

Candidate systems will include but not be limited to the following basic operating principles:

1. Cryogenic distillation.
2. Absorption on activated charcoal.
3. Solution in liquids.
4. Permaselective membrane separation.
5. Absorption on silica gel.
6. Trapping in molecular sieves.
7. Trapping as clathrates.
8. Thermal diffusion.

If, after a preliminary examination, it is not clear which separation principle should be developed, alternate designs will be studied in greater depth.

5.2.4. On-Line Monitoring Methods

5.2.4.1. Objectives

The objectives of this program are (1) to develop an accurate and reliable plugging meter for the B&W reference design LMFBR, and (2) to develop a quick-response instrumentation system for detecting the small sodium-water reactions that are likely to occur in the B&W helical-coil steam generator and reheater.

5.2.4.2. Summary of Current Technology

References 2 and 3 survey the current status of on-line sodium purity instrumentation. Although plugging meters have not yet been developed as marketable instruments supplied by instrument vendors, a number of units have been built and operated satisfactorily as manually or semi-automatically controlled devices; the extension of these efforts to provide a continuous on-line instrument is considered straightforward. Reference 4 describes a British experimental

design for such a device. A sensitivity of 10 ppm is regarded as routinely obtainable. Greater sensitivity, reliability, and accuracy will be the goals of R&D work on these instruments.

APDA is conducting an investigation of leak-detection methods as part of their Sodium-Water Reactions Program.⁵ Although sodium-water reactions are detected in the Fermi reactor by monitoring for hydrogen in the cover gas, the APDA studies show that as much as 99% of the reaction-produced hydrogen may remain in the sodium. Additional fundamental investigation of the sodium-water reaction, at different leak rates and under various conditions, is needed before good reaction detection criteria can be formulated. In particular, the prompt detection of small water leaks into the very large mass of sodium present in a steam generator or reheater requires additional basic effort.

APDA has used both hydrogen (in cover gas) thermal conductivity meters and acoustic detection for experiments on the sodium-water reaction. These techniques, in addition to the in-sodium hydrogen diffusion meter and some other types, provide a nucleus of detection methods.

5.2.4.3. Technical Justification

Objective 1 — One of the basic assumptions of the B&W reference design is that, to prevent excessive corrosion, sodium impurity concentrations will not have to be lower than can be provided by high-performance cold traps. If corrosion studies substantiate this assumption, then the only on-line instrument needed will be a plugging meter that monitors gross changes in the concentration of impurities. Since R&D to develop more sensitive instruments which are practical for on-line use will be very expensive, and since success before the reference plant is scheduled for construction is unlikely, it seems prudent to await conclusive metallurgical data before making firm recommendations in this area.

Objective 2 — A possible sodium-water reaction in the steam-raising equipment cannot be discounted for the B&W helical-coil design (a total of 3792 tube-to-tube sheet welds), regardless of the manufacturing excellence and quality control used. Indeed, this

statement applies to any candidate design now being proposed for LMFBR service.

B&W's reference design calls for pressure transducers to monitor the blanket gas pressure in the reheaters and steam generators and to initiate emergency procedures when a predetermined overpressure condition exists. Unfortunately, a sodium-water reaction would be in an advanced stage before a significant rise in pressure was noted. A hydrogen-sensitive gas chromatograph is also included in the instrumentation as a backup to the pressure transducers. A sodium-water reaction may or may not be in an advanced stage when the chromatograph sounds the alarm. A chromatograph is sensitive, but it cannot qualify as an early warning instrument because of the long lag time between the start of a sodium-water reaction and the diffusion of hydrogen into the blanket gas space.

Because a large-scale sodium-water reaction could adversely affect plant safety and would have serious economic repercussions, it is mandatory to develop a way to detect small leaks in time to take precautionary action.

5.2.4.4. Scope of Work

Efforts directed toward on-line monitoring methods will be pursued in two tasks—plugging meter development and small sodium-water leak detection.

Task 1: Utilizing all previous experience with such devices, an acceptable plugging meter will be developed for use in the primary and intermediate coolant systems. When necessary, additional data on the precipitation properties of the elements and species of concern will be made available through experimentation carried out under 5.2.6, Fundamental Properties of System Sodium.

Task 2: A reliable, quick-response system for detecting small sodium-water reactions in the B&W helical-coil steam-raising equipment will be developed in this task. Since the characteristics of sodium-water reactions of any size are now ill-defined, a great deal of work on fundamentals is expected before a suitable system can be designed.

5.2.5. Sampling and Analysis

5.2.5.1. Objectives

The objectives of this program are as follows:

1. To develop methods applicable to the B&W reference design LMFBR for obtaining representative samples of radioactive and non-radioactive sodium, and to formulate procedures for preventing the contamination of these samples.

2. To develop laboratory techniques for identifying and accurately measuring small concentrations of the deleterious forms of various impurities, such as oxygen, carbon, hydrogen, and nitrogen. In addition, equally precise techniques must be developed for measuring the corrosion products containing chromium, nickel, iron, and cobalt that are expected to be present in the sodium heat transport systems of the LMFBR.

5.2.5.2. Summary of Current Technology

The accuracy, precision, and speed of current laboratory methods for sodium impurity analyses are not considered acceptable, despite extensive R&D work in this area over the past 20 years. The results of the "round robin" analyses for oxygen and carbon that have been conducted thus far point out the need for standardizing the analytical techniques. Another round robin exercise is now in progress; it is aimed at achieving this objective for oxygen in sodium. The results of this effort have not yet been reported.

Although refined procedures and techniques have been developed for other impurities in sodium, none have been standardized or universally accepted. Round robin exercises are planned for some of the critical species, and the results will be valuable in assessing the actual current state of the art.

5.2.5.3. Technical Justification

The difficulties in the analysis of impurities in sodium in the range of LMFBR interest (usually below 10 ppm) can be attributed largely to the high chemical reactivity of the sodium. Some chemists express optimism that the accuracy of sodium impurity analysis

can approach that now available with water; however, this statement is always qualified by saying that R&D for comparable accuracy in sodium will be much more expensive than that for water. On the other hand, comparable accuracy may not be a prerequisite to successful LMFBR operation.

The philosophy of sodium impurity surveillance for the B&W reference design is characterized by a minimum number of on-line instruments. These devices will be augmented by laboratory analyses accurate enough to discern any variations in impurity concentration that could endanger plant safety or longevity. Impurities of current interest are listed in objective 2. No doubt other elements or compounds will be added to this list before the reference plant is constructed.

5.2.5.4. Scope of Work

The scope of work for this task encompasses varying degrees of effort in the following areas:

Task 1: Recent literature will be reviewed to assess the state of the art for methods of sampling, preserving, and analyzing sodium containing less than 10 ppm of individual impurities that are either known or expected to be important to the design and operation of the B&W reference design LMFBR. At the present time, these impurities include various compounds of oxygen, carbon, hydrogen, nitrogen, chromium, nickel, iron, and cobalt. Other compounds or elements will be added as the need develops before the reference plant is constructed.

Task 2: From the review in task 1, we will select the methods and procedures that appear to be the most promising for achieving the objectives of this task for further development to overcome any deficiencies.

Task 3: Concurrent with task 2, efforts will be made to develop new or advanced sampling and analytical methods for both laboratory and plant operator applications.

5.2.6. Fundamental Properties of System Sodium

5.2.6.1. Objectives

The objectives of this program are listed below.

1. To determine the reaction products, solubility, and distribution coefficients of the important impurities found in reactor and intermediate sodium. Oxygen, hydrogen, carbon, nitrogen, and their compounds have been identified as the important impurities that require extensive study for the reference design LMFBR.

2. To determine the solubility and the kinetics of the solution process for chromium, nickel, iron, and cobalt in sodium at temperatures of 200 to 1250 F.

3. To determine the solubility and the kinetics of solution for fission products that enter the reactor coolant system sodium from failed fuel or from the intentional venting of fission gasses to the sodium.

5.2.6.2. Summary of Current Technology

The transport properties of impurities and corrosion products in sodium are ill-defined in the area of interest to our reference design LMFBR, largely because techniques for the identification and measurement of the various species are relatively unrefined. Some properties can be determined by adding known quantities of an impurity to ultrapure sodium (prepared by multiple distillation) and observing the mixture produced. For example, the solubility of some species in sodium can be determined by the controlled addition of small quantities until saturation occurs. Since sodium is so chemically reactive, a portion of the impurity added may react and form other compounds, thus obscuring the true solubility of the impurity being investigated. This tendency, however, may be partially overcome by analyzing the mixture for other species or elements that might be liberated or formed in a reaction between the original added impurity and the pure sodium. Although a great deal of useful data can be obtained by such indirect measurements, the more direct approach via accurate laboratory analysis is preferred. As indicated in section 5.2.5, accurate methods for determining most impurities in sodium in the area of interest (usually below 10 ppm) are not available. A corollary to this fact would be that the accuracy of transport properties parallels analytical accuracy.

5.2.6.3. Technical Justification

An understanding of the chemical behavior and transport properties of corrosion products, fission products, and other impurities in sodium is essential to the success of the B&W 1000-MWe LMFBFR. There are a number of specific reasons for requiring this knowledge:

1. To permit an accurate correlation of the extent of fuel failure and to assist in failure location. Without a knowledge of diffusion coefficients and other transport properties, it would be difficult to estimate the quantity of fission products released to the coolant on the basis of the measured activity of short-lived radioisotopes.
2. To support studies of the fate of fission products released to the reactor coolant system. Although the B&W LMFBFR is designed with full cognizance of fission products in the reactor coolant system, sinks, sources, and transport phenomena must be known to adequately estimate the consequences of a DBA.
3. To permit accurate estimates of metal waste due to solution attack in both primary and intermediate sodium systems. The degree of solution attack on a metal surface depends on the relative concentration of metallic impurities in the sodium. If a system could be permitted to reach static equilibrium, then metal waste by this mechanism would be suppressed. However, cold traps will act as sinks for dissolved metallic impurities. Some degree of solution suppression may be possible in a dynamic system if accurate transport properties are known.
4. To support experimental corrosion programs. Accurate corrosion data are essential to the success of the LMFBFR, mainly because of the cost penalties encountered in the degradation of breeding ratio by increased cladding thickness in the core.
5. To support the development of high-performance cold traps. Current data indicate that oxide and other precipitable impurities must be kept in the low-ppm range (about 10 ppm). Cold traps are capable of reducing many of the important impurities to this level, but a better understanding of the basic mechanisms is required to prolong

the useful life of a unit and to reduce its physical size. Although size is not important in nonradioactive systems, it is very important in radioactive systems because of the handling and waste disposal problems.

5.2.6.4. Scope of Work

Fundamental properties of system sodium will be pursued in three major tasks.

Task 1: The transport properties of low concentrations (in sodium) of the important oxygen, hydrogen, carbon, and nitrogen impurities will be determined. Two temperature ranges will be emphasized: the high-temperature range will be concerned with properties from 700 to 1250 F, and the low-temperature investigation will be concerned with properties from 208 to 300 F.

Task 2: The kinetics of solution and deposition of chromium, iron, nickel, and cobalt will be investigated for the range of temperatures indicated in task 1. Corrosion suppression methods will also be investigated.

Task 3: The kinetics of solution and deposition of fission products will be investigated for a pot configuration. Temperature ranges will, of necessity, cover the entire range expected in the reactor coolant system (200 to 1250 F, i. e., cold-trap to core-hot-spot temperature).

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6. CORE DESIGN

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6. CORE DESIGN

6.1. Introduction and Summary

6.1.1. General

The objectives of the core design R&D program are to develop technology for prototype components for both predecessor and reference cores, and to provide for the design and testing of those components. The predecessor core will contain sealed fuel pins with solid pellet, mixed-oxide fuel; the pin will be conservatively rated with respect to linear power, inpile time, maximum cladding stress and surface temperature. It is anticipated that most of the research and development required for designing the predecessor fuel pin will be supplied by present and planned national programs. The predecessor plant will be capable of operating with vented fuel so that the reference fuel pins can be statistically tested for reliability. The reference fuel pin will be a vented-to-coolant pin containing vibratory compacted mixed oxide.

The fuel assemblies, radial blanket assemblies, and control rods for the predecessor and the reference reactors will be similar in concept; the only anticipated major differences are in the assemblies' dimensions. However, it is possible that different types of fuel pin spacers will be used in the predecessor and the reference fuel assemblies. This depends on whether the spacer selection program demonstrates the acceptability of advanced spacer concepts before hardware commitments are made for the predecessor.

The overall core configuration concept, which affects the selection of assembly holddown and assembly can design, should not differ significantly for the predecessor and the reference cores.

The primary objective of the R&D program for the fuel pin (section 6.2.1) is to provide guidance for the materials and integrated pin tests to be conducted under the fuels and materials program. Two exceptions to this objective are (1) investigation of the failure

propagation phenomenon for sealed fuel pins, and (2) outpile thermal shock tests of fuel pins, which will be conducted under the core design program. The fuel pin program is relatively independent of other core design programs with regard to possible redirection of efforts.

The fuel assembly R&D program (section 6.2.2) is set up to develop and test a prototype assembly before hardware commitments are made for the predecessor plant. Work thereafter will concern the upgrading of the prototype design for reference plant application. Also, operation of the predecessor plant after 1980 will afford an opportunity to conduct tests on the reliability of reference fuel pins in a prototypic environment. The reference fuel assembly will be used in these tests since it is realized that spacer and assembly designs have a strong effect on pin reliability. The fuel assembly spacer and wrapper can programs as presently planned are subject to change pending the results of an investigation into the fixed can concept considered in section 6.2.5.3. A fixed can core design (and possibly the perforated wrapper can concept) would change some spacer and can requirements. The Assembly Lower Fitting and Support Grid Junction Development, part E of section 6.2.2.2, would be eliminated if a fixed can concept were chosen.

Steady-State Thermal-Hydraulics, section 6.2.2.3, has as its objective the methods development required to calculate assembly fluid velocity and temperature distributions, cladding surface temperature distributions, and the development of orifices for core flow distribution (the orifices required for the radial blanket assemblies are also included here). Of course, the spacer type has some effect on fuel assembly fluid temperature distributions by virtue of the effects on inter-channel fluid mixing. Wherever these mixing effects strongly influence spacer selection, tests for mixing coefficients will be conducted, as will tests to improve data concerning spacer and pin bundle coolant pressure drop. (Mixing coefficient tests are funded under section 6.2.2.2, Fuel Pin Support System Selection.)

Preliminary data required for fuel assembly shaker tests are for the sake of program continuity, included under the proof test program (part C, section 6.2.2.5, Assembly Mechanics and Shaker Tests). These data will likely be obtained when the prototype spacer

evaluations are made during the spacer selection program. The fuel assembly thermal bow facility is funded in section 6.2.2.5; however, it is planned to use this facility to proof-test the control rod, control rod guide tube, and radial blanket assembly. Although thermal bow of an assembly has some bearing on the spacer design, it is anticipated that this aspect of spacer selection will be investigated by analytical means or by bench tests funded under the spacer selection program, section 6.2.2.2. Negative results from proof tests should be reflected in minor design modifications.

Development programs for the control rod and the radial blanket assembly will draw heavily upon fuel assembly development experience. Analytical model development funded under section 6.2.2, Fuel Assembly, should include applicability to large pin geometries if possible. Separate methods development for control rods and radial blanket geometries in the areas of flow-induced vibration, spacer pressure loss, fluid mixing, and the like are not funded in section 6.2.3, Control Rod, and section 6.2.4, Radial Blanket Assembly Development.

A number of investigations are planned under Core System, section 6.2.5, concerning factors that may effect changes in the overall core configuration. If the concepts of a fixed can, perforated can, or hydraulic holddown prove feasible and desirable from an operating or energy cost standpoint, then the fuel assembly and radial blanket assembly R&D programs will be re-evaluated and possibly redirected if the schedule allows.

6.1.2. Budget and Schedule Summary

The investigation into propagation of fuel pin failure (6.2.1.2) primarily concerns the predecessor plant design. These studies are expected to be completed by late 1973. The difficulties involved in developing a reference fuel pin stem primarily from the vented concept. A nonvented concept for the predecessor plant should be qualified by late 1977. Funding for fuel data development (6.2.1.3) and cladding development (6.2.1.4) is not planned to extend beyond that date. Steady-state and transient tests of fuel pins beyond late 1977 will primarily involve the proofing of vented fuel pins. The vent development program

Table 6-1. Summary of Schedule and Budget for Core Design R&D

		Budget, \$ × 10 ³																				
	FY	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988		
Task	CY	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	Total
Milestones:						1		2					3		4					5		
6.2.1. Fuel Pin																						
Failure propagation		60	105	45	45	45	--	--	--	--	--	--	--	--	--	--	--	--	--	--	300	
Fuel data development		30	30	30	10	10	10	10	10	10	--	--	--	--	--	--	--	--	--	--	150	
Cladding development		24	24	21	21	21	21	21	21	21	--	--	--	--	--	--	--	--	--	--	195	
Irradiations of mixed-oxide fuel pins		100	100	100	100	100	80	60	60	60	60	60	60	60	60	--	--	--	--	--	1,060	
Transient power tests		40	30	30	10	10	10	10	10	10	10	10	10	10	10	--	--	--	--	--	210	
Vent development		40	30	30	10	10	10	10	10	10	10	10	10	10	--	--	--	--	--	--	200	
Total		294	319	256	196	196	131	111	111	111	80	80	80	80	70	--	--	--	--	--	2,115	
6.2.2. Fuel Assembly																						
Fuel pin support system selection		150	582	570	320	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1,622	
Steady-state thermal/hydraulics		1300	1700	1700	600	540	15	15	15	15	15	15	15	15	--	--	--	--	--	--	5,945	
Transient thermal/hydraulics		60	90	70	30	25	15	15	15	15	15	15	15	--	--	--	--	--	--	--	380	
Fuel assembly prototype design and proof tests		--	--	551	482	362	210	--	--	--	--	--	--	--	--	--	--	--	--	--	1,605	
Long-range upgrading		--	--	--	--	--	--	22	22	22	22	22	20	20	--	--	--	--	--	--	150	
Pin reliability and assembly proof tests in the predecessor		--	--	--	--	--	--	--	--	--	--	--	50	50	50	50	--	--	--	--	200	
Total		1510	2372	2891	1432	927	240	52	52	52	52	52	100	70	50	50	--	--	--	--	9,902	
6.2.3. Control Rod																						
Absorber pin		40	40	20	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	100	
Rod design data and criteria		80	70	40	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	190	
Analytical model development		60	80	40	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	180	
Prototype design and testing		--	--	120	120	200	200	100	60	--	--	--	--	--	--	--	--	--	--	--	800	
Total		180	190	220	120	200	200	100	60	--	--	--	--	--	--	--	--	--	--	--	1,270	
6.2.4. Radial Blanket Assembly Development																						
Radial blanket pin		--	--	15	15	--	--	--	--	--	--	--	--	15	15	--	--	--	--	--	60	
Support system development		--	--	40	30	30	--	--	--	--	--	--	--	--	--	--	--	--	--	--	100	
Prototype assembly design and out-pile proof tests		--	--	--	60	200	40	--	--	--	--	--	--	--	--	--	--	--	--	--	300	
Total		--	--	55	105	230	40	--	--	--	--	--	--	15	15	--	--	--	--	--	460	
6.2.5. Core System R&D																						
Hydraulic holddown		50	50	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	100	
Wrapper can concepts		70	70	60	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	200	
Hydraulically operated control rod		100	400	400	100	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1,000	
Total		220	520	460	100	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1,000	
Overall Total		2204	3401	3882	1953	1553	611	263	223	163	132	132	180	165	135	50	--	--	--	--	15,047	

Milestones: △ FTF operation begins. △ Reference plant construction begins.
 △ Predecessor construction begins. △ Reference plant operation begins.
 △ Predecessor operation begins.

will continue to about the end of 1981, and funding after 1971 will hold at a relatively low level (from a core design standpoint).

Selection of the fuel pin support system (6.2.2.2) should be made by late 1972, and prototype fuel assembly proof tests (6.2.2.5) should end in 1974. The experimentation for steady-state and transient flow and heat transfer methods verification (6.2.2.3 and 6.2.2.4) should end by late 1973. Work extending beyond that date is planned for methods upgrading. The results of methods improvement will be utilized in the reference fuel assembly upgrading (6.2.2.6), which is expected to be carried out between 1974 and 1982. Fuel pin reliability tests (6.2.2.7) are funded from 1980 through 1983, but many uncertainties affect this scheduling. If by 1980 the fuel pin design has progressed satisfactorily to the point where reliability tests can be initiated, reliability tests will probably be carried on for eight years instead of four.

Development of the control rod and the radial blanket assembly is not expected to extend beyond 1976. Funding beyond that date is not provided since predecessor control rod and radial blanket technologies should be sufficient for the reference design effort. The control rod absorber pin tests (6.2.3.2) are planned for 1969 through 1971 but could be delayed by three to four years. Control rod prototype design and testing (6.2.3.5) is scheduled through 1976, but control rod life tests (3.2.7.3) should begin before then. Development of methods for the control rod and the radial blanket assembly will draw heavily upon those developed under the fuel pin support system program (6.2.2.2).

Core system R&D (6.2.5) investigations should be completed very early in the overall program. Results of these programs may affect the emphasis and direction of other programs. Programs for hydraulic holddown (6.2.5.2) and the wrapper can concept (6.2.5.3) will be completed by 1971 or sooner if possible.

Table 6-1 summarizes the core design budget and schedule.

6.2. R&D Requirements

6.2.1. Fuel Pin

6.2.1.1. General

Objective — The goal of the fuel pin program is to develop a concept that is technologically feasible for LMFBR operation in the late 1980's. An intermediate goal is the development of an interim concept for use in a predecessor plant. Fuel pin development is primarily a cooperative effort between core design and fuels and materials programs; core design will supply the system engineering effort which involves the following:

1. Translate core performance and safety requirements into fuel pin requirements.
2. Guide the selection of fuel pin concepts and parameters with consideration of the effects of pin performance on the core and the plant.
3. Develop technology concerning uncertainties in areas where the fuel pin strongly affects or limits core design (viz., effects of pressurized fuel pin failures).
4. Generate design information or oversee programs in other areas that generate information required to specify the fuel pin configuration resulting in minimum power generation cost.
5. Set integral fuel pin testing environments and objectives; this necessarily involves capabilities for fulfilling objectives and providing valid or usable information.

Summary of Current Technology —

1. Design Criteria: Theoretically, cladding failure should be precluded for normal operation, normal transients, and certain specified abnormal operating conditions for the average removal burnup (in-core time). A cladding failure is defined as a cladding breach, or 1% deformation ($\Delta D/D$).

An in-reactor cladding failure should not result in the failure of adjacent fuel pins. Propagation of a failure might be caused by vapor blanketing, flow blockage, or molten fuel and cladding contact.

2. Design Data: No data are available for the probable types of in-reactor fuel pin failures (extent and location) and the probability of failure propagation. Fuel property data for mixed oxides are generally lacking. The designer is presently forced to use data for UO_2 (obtained under thermal flux conditions) where the behavior of mixed oxides is unknown.

Mechanical properties of unirradiated cladding candidates are generally documented well enough for design purposes up to and slightly beyond the normal operating temperatures expected in the LMFBR. However, short-term, high-temperature properties (required for safety analyses) are generally lacking, as are data on the effects of high fluences.

Performance data for the low-burnup exposure of mixed-oxide fuel pins are available, but uncertainties in the data must be resolved. Fuel pins containing mixed-oxide fuel have been run up to 76,000 MWd/T burnup without cladding failure. Low-burnup, fast-flux tests have been performed on pellet and powder fuels with peak power ratings from 14.8 to 16.8 kW/ft²; thermal flux tests have demonstrated the capability of pellet fuels to operate at up to 23 kW/ft², whereas powder fuels of equal smeared density and weight percent plutonium are expected to be capable of 90% of the pellet fuel rating.

Some information is available on the performance of fuel pins subjected to power transients after steady-state irradiations up to 70,000 MWd/T. Data for fuel pins at higher steady-state burnups are lacking.

Performance data for the diving-bell concept for venting fission gasses to the coolant have been obtained from in-reactor tests. No information, however, is available for inpile performance of porous-plug vent devices.

3. Analytical Techniques: The lack of analytical techniques for fuel pin failure propagation makes it impossible to predict successfully the mechanics of fission gas and coolant interaction at, and downstream from, the cladding rupture. Analytical tools for the two-phase flow of liquid sodium are generally lacking. Since the mechanics of cladding failure are probably not entirely amenable to

analysis, the desired information concerning types and extents of cladding failure must be provided in part by tests.

Analytical methods for predicting the steady-state thermal performance of fuel pins are considered adequate; the accuracy of design calculations is limited primarily by fuel property and fuel-cladding gap heat transfer coefficient data. Methods for predicting the mechanical interaction between fuel and cladding require some improvement. No models are available to predict the possibility of fuel movement as a result of a sudden core shock or vibration.

The analytical models for the thermal and mechanical performance of the fuel pin under transient conditions require further development.

No sophisticated models have been developed for predicting the performance of vent devices with regard to the holdup time of fission gasses in a fuel pin.

Scope of Work — The fuel pin concepts will be limited to a mixed-oxide, pellet, sealed-plenum configuration for the predecessor plant, and to a mixed oxide, vibratory compacted, vented-to-coolant type for the reference plant. Emphasis on the predecessor fuel will be directed toward developing a configuration that just provides initial competitive power generation costs. The predecessor will be designed to accommodate the advanced reference fuel pin when it becomes available. Existing and planned test programs should provide sufficient information to permit the design of a conservative predecessor fuel pin with little testing other than proof-testing. Since the number of fast flux test facilities is limited, it should not be assumed that many fast flux tests could be performed before the FFTF fuel development program is completed.

Of primary importance in the reference fuel pin program are the development of a vent device and the demonstration of an average fuel burnup of 100,000 MWd/T. The reference design has included a porous plug as the venting device. This concept has an economic advantage over the diving bell concept, but the latter will undergo some examination and development since it is the logical backup to a porous plug device. Demonstration of extremely high burnup capability is the single most important economic factor for the reference fuel pin.

Most of the developmental effort will be directed toward obtaining high burnup capability while maintaining assurance of high reliability.

Neither the reference pin nor the predecessor fuel pin is expected to have any fuel-cladding bond other than helium gas. Fuel pin shelves to limit fuel movement or to allow operation with molten fuel are not being considered. Since the core design responsibility in the fuel area is primarily that of a systems engineering nature, no experimental work will be performed except in areas dealing with failure propagation of pressurized fuel pins and fuel pin thermal-mechanics performance; both of these problem areas will involve outpile tests which are funded under core design.

6.2.1.2. Failure Propagation With Pressurized Fuel Pins

Objective — Some effort must be directed toward defining the relationship between the allowable fission gas volume, gas pressure, fuel pin failure types and locations, and fuel pin lattice geometry to avoid possible failure propagation due to cladding failure with a pressurized fuel pin. This is obviously a consideration for the predecessor fuel pin, since it will be a sealed-plenum type. Also, consideration of this failure propagation mechanism may be applicable to the vented-to-coolant concept since the pin pressure may be appreciable (see 6.2.1.7, Vent Development).

Emphasis will be directed toward developing analytical methods to determine fuel pin design requirements in relation to stored fission gas pressure, volume, and the like based on the fission gas choking effects on coolant flow. Outpile tests will be conducted to support the analytical effort. Considerable effort will be devoted to design methods to circumvent the propagation problem (e.g., a gas throttling device, internal to the pin, between plenum and fuel or axial blanket) instead of defining the exact phenomenological nature of failure propagation due to fission gas release.

It should be noted that in a program of this nature, very large expenditures of time and money are required to produce results with a high level of confidence. A great deal of basic groundwork must be laid in order to arrive at meaningful and truly representative results. Very little is known of the probability of cladding failure as a

function of axial pin position or in-core time. Also, the types of failures that fuel pins subjected to fast fluence may undergo are not documented at present, and probably will not be in the near future. Hundreds of in-pile failure tests would have to be conducted to demonstrate that failure propagation will not occur in an operating reactor. A small number (such as from one to ten) inpile failure tests are likely to add very little to the level of confidence of outpile test results. Here, we are faced with the fact that an extensive analytical effort backed by outpile tests may provide results at a given confidence level, and that an extensive inpile test program may increase the confidence level only marginally. Consequently, at this time we do not feel that inpile failure propagation tests are justified.

Summary of Current Technology — As far as is known, no model exists that can predict coolant flow and heat transfer under the transient condition of fission gas flowing in a coolant channel. Some experimental work on the hydrodynamics of two-phase liquid-metal flows has been conducted,³ but it is difficult to measure the void fraction. No acceptable flow correlation has been developed for liquid metals, including sodium.

Flow boiling (or nonboiling, two-phase flow) heat transfer data for liquid sodium (in bundle flow) is virtually nonexistent. Some experimental data are available for mercury, potassium, rubidium, and some amalgams over scattered ranges of pressures, velocities, qualities, and heat fluxes. Some empirical correlations for boiling heat transfer coefficients have been obtained, but apparently not for liquid sodium (for the flow regimes of interest).

Methods for calculating fission gas release rates from a failed pin can be developed with existing theory. Analytical method accuracy would require experimental evaluation. Very little information on types of fuel pin failures is in evidence.

Scope of Work — This program will consist of three major efforts:

1. Development of the most sophisticated fuel pin-coolant interaction model consistent with present technology.

2. Design efforts to circumvent situations leading to possible failure propagation, as indicated by the interaction model.
3. Outpile testing to substantiate the model or fix empirical constants (to the extent possible) and evaluate problem solutions from item 2.

Most of the analytical effort will be directed toward developing a computer program to simulate the gas dynamics (internal to the fuel pin) and the two-phase flow (fission gas and sodium coolant) phenomena. The extent of coolant flow throttling and the extent of downstream or adjacent pin gas blanketing will provide a basis for evaluating the likelihood of added failures. The analytical results desired include the pin area and/or coolant channels affected and the cladding temperature-time envelope for a given set of stored gas volume, stored gas pressure, cladding breach flow area, and breach location conditions.

Proposed design solutions may be accounted for in the failure simulation. Possible design solutions may include an internal throttling orifice plate between the fuel chamber and the fission gas plenum. The analytical results may indicate safety factors that must be applied to the predecessor fuel element, such as added plenum volume, burnup limitations, or changes in fuel pin pitch-to-diameter ratios.

Outpile tests should be conducted in low temperature water (<250 F). The test specimens will be mock fuel pins whose diameter and pitch-to-diameter ratios closely approximate those of a proposed predecessor configuration. The tests will utilize flow visualization techniques to confirm predicted flow regimes (annular, bubbly, etc.) and the extent of adjacent pin gas blanketing opposite the simulated fissure or hole. Gas escape rate measurements will be of interest. Test variables will parallel those examined in the analysis above.

Major Facilities and/or Equipment Required —

The test facility will be an existing conventional hydraulic flow loop with modifications.

Relationship to Current or Planned Programs —

The LMFBR Program Plan sets forth a program of this nature (LMFBR Program Plan, Core Design subtask 6-2.2.3, WASH-1106). The plan, however, allows for the possibility of in-core failure propagation tests. We have discounted the worth of such tests here. Also, we have discounted any near-term hopes of incorporating into the analytical work any quantitative evaluation of the probability of cladding failure type, location, or time (related to burnup). We have approached the problem mainly from the standpoint of evaluating the effects of design approaches for circumvention.

Budget and Schedule — These tests and analytical results are required in order to design the predecessor fuel pin properly. As mentioned previously, some of the results may be applicable to the reference vented-to-coolant concept. Budgeting for manpower includes pre-test engineering and support engineering for tests. The costs in thousands of dollars are shown below.

	Calendar year					Total
	1969	1970	1971	1972	1973	
Manpower	60	60	30	30	30	210
Facilities	--	30	15	15	15	75
Equipment	--	15	--	--	--	15
Total	60	105	45	45	45	300

6.2.1.3. Fuel Data Development

Objective — The goal of this effort is to develop pertinent fuel, fuel-sodium, and fuel-cladding data required for the confident design of predecessor and reference core fuel pins. We will emphasize obtaining data for vibro-pak fuel, helium bonded with a low internal pin hydrostatic pressure. Existing or planned programs will be utilized for the design of the mixed-oxide, pellet, sealed fuel pin anticipated for use in the predecessor plant; data from these programs should be generally applicable with little or no additional development. The predecessor fuel pin will be designed conservatively with respect to the indicated allowable burnup, heat rating, and maximum cladding surface

temperature. After completing two or three predecessor fuel cycles with pellet, sealed pins, we will use the advanced pin design (vibro-pak, vented) in this plant.

This program will be a coordinated effort between the core design and the fuels and materials sections. The experimental work is funded under the fuels and materials area. The core design responsibility includes:

1. Delineation of the types of data required, the test variable ranges, and the desired accuracy of results.
2. Aid in planning experiments with regard to test conditions and the type of facilities used.
3. Evaluation of the usefulness and relevancy of results.
4. Re-evaluation of the feasibility of concepts and firming-up of pin design criteria.

Summary of Current Technology — Fuel data state of the art is discussed in section 7.2.1.

Scope of Work — The core design section will set forth the fuel properties of interest for vibro-pac, mixed-oxide fuel, as well as the range of variables affecting these properties. Tentative data requirements are as follows:

1. Effective fuel thermal conductivity as a function of burnup, stoichiometry, fabrication methods, and smeared fuel density.
2. Melting temperature as a function of burnup, composition, stoichiometry, and fuel impurities.
3. Fuel specific heat.
4. Coefficient of thermal expansion.
5. Volumetric expansion upon melting.
6. Molten fuel viscosity.
7. Fuel modulus of compression and creep strain rate as a function of temperature, applied stress, and grain structure (as-fabricated or restructured).

8. Unconstrained swelling rate and fission gas release as a function of temperature, fission gas pressure, and fissioning rate.
9. Fuel-sodium compatibility as a function of burnup and stoichiometry.
10. Fuel-cladding compatibility.
11. Plutonium migration rate and extent.

Any data affected by internal pin pressure will have to be re-evaluated as results from the vent device program are received. There is some uncertainty concerning the pin pressures required to make the porous plug device operable from the standpoint of fission gas holdup time. Pin internal hydrostatic pressures can affect the effective fuel thermal conductivity and fission gas release.

Major Facilities and/or Equipment Required —

The experimental work will be conducted under the fuels and materials program. See section 7.2.1 for a discussion of test facilities.

Relationship to Current or Planned Programs —

The emphasis of the LMFBR Program Plan Task for fuel properties (6-2.4) of mixed oxides is on the properties of sintered pellets under conditions experienced in sealed pins. Very little of the effort is directed toward investigation of vibro-pak fuel. Since some of the fuel properties are expected to pertain to both pellet and vibro-pak, there is some overlap. The extent of overlap is difficult to assess since the difference in the amounts of fission gas retention may significantly affect the melt temperature, volumetric expansion coefficient upon melting, fuel mechanical properties, and diffusion rates of plutonium.

Budget and Schedule — The manpower costs, in thousands of dollars, are as follows:

	<u>Calendar year</u>									
	<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	<u>Total</u>
Manpower	30	30	30	10	10	10	10	10	10	150

6.2.1.4. Cladding Development

Objective — The core design area shares the developmental responsibility for cladding with the fuels and materials area. Core design is responsible for establishing design and test requirements, screening cladding candidates for advanced tests (preliminary screening and pre-irradiation tests to be the sole responsibility of fuels and materials), and evaluating test results for the selection of prime candidates for prototype tests. We will emphasize the development of a cladding consistent with the requirements of a vented pin utilizing a vibro-pak, mixed-oxide fuel, although the predecessor plant will initially use a sealed pin containing fuel pellets. It is hoped that the cladding for this pin will require little evaluation beyond present and planned programs—the pins for the early loadings of the predecessor will be conservatively designed.

One of the primary considerations in selecting the cladding material is the cladding ductility degradation with fast flux exposure. Also, any serious cladding contenders must be thoroughly tested to determine their ultimate strength, stress at 0.2% offset (or modulus of elasticity), creep strength, and ductility in order to properly evaluate the effect on fuel burnup, which is the main operating parameter through which cladding affects plant economics. These data will be used in a fuel-cladding interaction model to be developed by the fuels and materials program.

Before inpile tests are conducted on a prototype pin (see section 6.2.1.5) employing a cladding candidate, outpile thermal shock tests will be conducted. The cladding integrity must be demonstrated under situations simulating sudden changes from normal to low power-to-flow ratios.

Summary of Current Technology — The cladding candidates for fast reactor service in the United States⁴ are primarily 304 and 316 stainless steels, Hastelloy-X, Inconel-625, and vanadium alloys. The British are considering 316L stainless steel, FV548 (niobium-stabilized type 316), 20 Cr-25 Ni-Cb alloy, and Nimonic PE 16. Some minor considerations of 347 stainless steel, Incoloy-800, 19-9DL, and 20 Cr-25 Ni have been undertaken.

No information is available concerning the minimum cladding thickness that is practical from a production, handling and fabrication standpoint. General Electric⁵ has produced Vipak test specimens with cladding thicknesses of 0.015 inch for irradiations in EBR-II. Westinghouse has produced claddings for pin irradiation tests as low as 5 mils in thickness⁶.

Insufficient information is available to define the allowable failure rate for pin claddings. And the definition of clad failure other than rupture has not been set; some question exists as to whether 1, 2, or 3% clad creep deformation should be considered as a design limit. The mechanical and chemical properties considered of interest for cladding use with vibratory compacted fuel are as follows:

1. Ductility.
2. Sodium compatibility.
3. Ultimate tensile strength.
4. Creep strain rate and creep rupture strength.
5. Properties for determining thermal stress.
6. Fatigue properties.

Scope of Work — The core design responsibility in the cladding material screening process involves the establishment of material requirements. For cladding use in a vented, vibro-pak fuel concept, the following minimum cladding requirements must be determined:

1. Ductility.
2. Sodium corrosion rate.
3. Ultimate tensile strength.
4. Creep strain rate (a function of strain and temperature).
5. Properties for determining thermal stress.

The temperatures, fluence, sodium purity, and the like at which these requirements are to be met must be supplied. Using these requirements, fuels and materials personnel will perform preliminary screening, including test evaluations.

Prime cladding candidates selected in the fuels and materials program will be evaluated to determine the effects of each on core performance. The most promising cladding materials will then be subjected to combined fuel-cladding tests, both outpile and inpile. This program assumes responsibility for outpile tests to demonstrate (by simulation) the cladding ability to withstand the stresses induced by cladding "shrinking" on the fuel as an operating reactor goes rapidly from normal to low power-to-flow ratios. Again, the inpile test data required, as well as the test parameters to be varied, will be specified in the core design program. Inpile tests are funded under section 7.3.

Outpile tests for cladding integrity under conditions of thermal shock (mentioned above) will be performed in a sodium pot-type facility. Since the actual high fuel temperatures cannot be obtained, we propose a test whereby a cladding specimen would be fitted over a UO_2 sintered pellet, heated to an appropriate temperature in an inert or nitrogen atmosphere, and then immersed in a sodium bath held at a reduced temperature. The pellet and cladding dimensions will be fixed so as to simulate as nearly as possible the incore fuel-cladding mechanical interaction.

Major Facilities and/or Equipment Required —

Materials test facilities are discussed in section 7.3 (inpile and outpile). Cladding thermal shock tests will be conducted in a conventional sodium pot-type facility.

Relationship to Current or Planned Programs —

The LMFBR Program Plan (subtask 6-2.4.2) deals with the core design effort regarding cladding development. We have emphasized claddings consistent with the goal of developing a vibro-pak, vented fuel pin. A predecessor fuel pin cladding can be selected on the basis of present knowledge of candidates.

Budget and Schedule — The costs in thousands of dollars are shown below.

Calendar year

1969 - 1978

Manpower	150
Facility	30
Equipment	15
<hr/>	
Total	195

6.2.1.5. Irradiation of Mixed-Oxide Fuel Pins

Objective — These steady-state tests will demonstrate the feasibility of, provide design data for, and demonstrate the operability of the fuel-cladding vent combination in a prototype stage of development. The fuel pin development is a combined core design and fuels and materials effort. Core design responsibility is as follows:

1. Statement of design data and performance demonstration requirements.
2. Test goals, environment, and range of test parameters delineation.
3. Assist in evaluating the capabilities of available test reactors, including thermal reactors.
4. Determine the extent of the applicability of test data to the conditions expected in the LMFBFR.
5. Evaluate test results (with respect to the effects of variations in projected performance) on core design and economics.

These irradiations will deal primarily with design data and performance for vibro-pak fuel. The existing and planned programs for the pellet fuel configuration in combination with sealed pins—the predecessor concept—should be adequate for pin design with a minimum amount of additional testing. This philosophy results from the assumption that a conservative pellet, sealed pin design will be competitive or nearly so in 1981 (the year of proposed predecessor plant startup); the startup date of the predecessor allows adequate time for testing in the FFTF if required.

Summary of Current Technology — Required design information that is obtainable from steady-state tests of fuel pins is as follows:

1. The effects of fuel form—powder or pellet—and smeared fuel density on the maximum linear power rating with no fuel melting.
2. The effects of in-core time and burnup on the linear power rating.
3. Fuel temperatures at which the core can operate without danger of fuel slumping due to a sudden core mechanical shock or vibration.
4. The fuel and cladding temperatures and the fuel form effects on attainable burnup without cladding rupture.
5. The fuel and cladding effects on fission gas holdup time.
6. The vent device performance.

Some experimenters report that the maximum power rating of powder fuels is approximately 10% less than that of a pellet fuel configuration of the same smeared density. The performance of the two fuel forms would reasonably be expected to be very close, since the powder fuel sinters in the reactor, leaving only a small circumferential band of as-fabricated fuel.

Use of the B&W fuel model code TAMPAs indicates that a decreasing smeared density for powder fuel may allow an increased linear power rating. The code has indicated an increase in the linear power rating of from 17.0 to 18.7 kW/ft while reducing the smeared density from 85 to 80%. This change in the maximum linear power rating has not been confirmed experimentally.

The effect of burnup on the melting temperature of mixed oxides has not been adequately investigated. The data obtained for UO_2 by Christensen⁷ are considered the most accurate; the mixed-oxide melting temperature is then obtained approximately by correcting for PuO_2 concentration.

The thermal conductivity of mixed-oxides is known to change considerably with burnup. It has been found that the thermal conductivity of UO_2 decreases with irradiation for fuel temperatures below 600 C; irradiation has an apparently small effect on the thermal conductivity of fuel above this temperature. Changes in the fuel stoichiometry due to temperature gradients and irradiation have been

postulated as the cause for the conductivity changes.⁸ No known published data are available for the effects of burnup on the thermal conductivity of mixed oxides.

Vipac fuel pins have been irradiated in the Dounreay Fast Reactor. The United Kingdom reported that pin failures occurred in the burnup range of from 61,000 to 76,000 MWd/T. Short-term irradiations of pellet and powder fuel have been reported by General Electric,⁹ irradiations in EBR-II were short term with a peak burnup of 1800 MWd/T. Results of burnup tests to 50,000 and 100,000 MWd/T, again in EBR-II, have not been published. (The latest published results are for burnups to 27,000 MWd/t.)

No information is available on the operating fuel temperatures necessary to prevent fuel slumping during sudden core mechanical shock or vibration.

Scope of Work — Early testing must be directed toward obtaining necessary design data and demonstrating the feasibility of the concept. Most information available on mixed-oxide, vibro-pak fuels has been obtained in thermal fluxes. Fast flux irradiations are required to reduce the considerable uncertainties in present data; however, because of the limited test space available, few fast flux tests can be conducted until the FFTF fuels development program is concluded.

The prime feasibility questions affecting the vented, vibro-pak concept are as follows:

1. Does the unsintered, annular ring of vibro-pak fuel offer, in fact, a measurably more elastic structure at or near 140,000 MWd/T burnup than does a sintered pellet of originally identical smeared density?
2. What effect does the thin reference fuel pin cladding have on restraining the outward radial fuel growth by inducing inward radial and axial fuel creep?
3. What are the effects of long-term sodium logging of vibro-pak fuel, and are those effects different in quality from those on pellet fuel?

4. What are the inherent fuel-cladding combination effects on fission gas holdup time, excluding the vent device?

Design data acquired in early testing will provide a basis for determining design criteria, especially the effects limiting the maximum fuel temperature criterion, that is, fuel slumping due to a sudden core vibration or shock and fertile-fissile segregation due to plutonium migration. The accuracy of thermal flux tests in supplying information of the nature given above will be evaluated. If the accuracy is insufficient, then tests to supply this information cannot be conducted until space is available in a fast flux test facility. The slumping problem will be investigated to determine the effects of center void size, and to explore the possibility of thermal ratcheting creating axial gaps that afford the possibility of fuel movement during core vibration.

Other design data of interest include:

1. The allowable linear power rating of mixed-oxide, vibropak fuel as it is affected by as-fabricated smeared fuel density, incore time, and burnup (of course, the allowable linear power rating as set by the fuel temperature is determined by the power - time envelope that the reactor is required to survive; see section 6.2.1.6, Transient Power Tests of Mixed-Oxide Fuel Pins).

2. Irradiation-induced changes in fuel stoichiometry.

3. Irradiated cladding creep ductility and the allowable burnup. Demonstration of allowable burnup requires in-reactor testing to failure. Fuel pins irradiated in transient power tests, without failure, will be re-irradiated in steady-state tests to demonstrate the effects of transient power-time envelope on long-term pin integrity. Reliability tests are discussed in section 6.2.2.7.

Major Facilities and/or Equipment Required —

No difficulties are expected in obtaining thermal flux test space. Possible facilities include the BWTR, MTR, ETR, CP-5, GETR, LTR, NRX, and BR-2. Fast flux tests which involve an appreciable probability for pin failures will be conducted in the FFTF. Some testing will be performed in EBR-II and Fermi, but the extent of testing will be limited to sealed pins.

Relationship to Current or Planned Programs —

Irradiations are being conducted in EBR-II, but data for pellet type fuels are being emphasized. The EBR-II capability limits the tests to sealed pins.

This program is similar to subtask 6-2.4.3 of the LMFBR Program Plan, WASH-1106, except that it is considerably reduced in scope; it covers almost entirely the irradiations of a mixed-oxide, vibro-pak, vented-to-coolant concept. The predecessor plant fuel design will draw from the present and planned programs and should require minimum developmental testing.

Schedule and Budget — The experimental effort is funded under Fuels and Materials, section 7.4.3. The budgeting shown below includes pre- and post-test engineering. Costs in thousands of dollars are as follows:

	Calendar year														Total
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	
Manpower	100	100	100	100	100	80	60	60	60	60	60	60	60	60	1060

6.2.1.6. Transient Power Tests of Mixed-Oxide Fuel Pins

Objective — This program will establish design criteria for the maximum allowable operating fuel temperature with respect to the limit set by the power transients that the core should withstand without sustaining damage. Experimental work to obtain data required to determine the credible power-time envelope will be carried out in other programs; proof tests of prototype pins to demonstrate cladding, fuel, and vent integrity under the power-time envelope will be the responsibility of the fuels and materials program. Other objectives of this program are as follows:

1. Determine design criteria regarding the transient performance of the fuel pin.
2. Cooperate in determining test facilities that will adequately meet test goals.
3. Set test environments and parameters to be varied.

4. Interpret design data and determine the effects on core design and economics.

The transient power tests will primarily concern the demonstration of the acceptability of a vibro-pak, vented prototype design. A minimum amount of testing should be required to demonstrate or proof-test a predecessor fuel pin of conservative linear power rating and burnup. (As mentioned previously, the predecessor design will probably be a pellet type, sealed pin.)

Effects of transient power on the cladding temperature-time envelope will be determined analytically and backed by experiment. Core design personnel will develop analytical models to predict fuel-side and coolant-side performance. The development of transient heat transfer models is discussed in section 6.2.2.4.

Summary of Current Technology — Tests on unirradiated and irradiated fuels (to 70,000 MWd/T) indicate that the integrity of fuel pin cladding is not jeopardized by reactor transients in which no fuel melting occurs.¹⁰ In these tests, sample fuel pins have survived 22% melting (% of volume) with no cladding deformation, and the cladding deformation resulting from large amounts of fuel melting is attributed to the increased pin pressure accompanying fission gas release occurring upon fuel melting.

No information on analytical models or test results is available for the slumping characteristics of high-temperature fuels subjected to sudden mechanical shock or vibration. This problem area has some bearing on the design overpower value since the probability of simultaneous reactor overpower and episodes of core vibration exists.

Data are not available on the results of fuel pin thermal shock analyses or tests where the thermal shock is due to a sudden coolant temperature reduction in the neighborhood of several hundred degrees Fahrenheit.

Published data show that short-term effects of sodium logging on fuel performance under severe transient overpower conditions are not detrimental to fuel pin integrity. The tests referred to were performed on non-vented pins.

Scope of Work — A number of uncertainties must be resolved to set the power-time envelope that the core should survive. The core design program will investigate this problem area, determine a criterion, and delineate fuel pin tests to demonstrate fuel, cladding, and vent integrity. The proof tests of prototype pins for transient tests will carry pin power to various levels—below, up to, and beyond melting. Post-test examinations will indicate any fuel slumping or melting.

Tests carried beyond the fuel melting point will demonstrate the ability of the cladding to survive the strains induced by the fuel volume change upon melting. Some overpower tests will be conducted to produce expected operating maximum hot pin temperatures and a subsequent simulated scram to demonstrate the absence of shrinkage slumping.

Criteria must be set for the number of thermal cycles that the fuel pin must withstand. Tests will be conducted to demonstrate the integrity of claddings subjected to stress cycles in appropriate excess of the established design requirement. These tests should demonstrate the presence, if any, of fuel thermal ratcheting problems. Also, proof tests of cladding integrity under the stresses induced by cladding "shrinkage" on the fuel during sudden drops in the core bulk outlet temperature must be performed (these outpile tests are discussed in section 6.2.1.4).

The acceptability of long-term sodium logging in vibro-pak fuel pins must be determined; then, the porous plug candidates must be shown to perform adequately. It must be demonstrated that plugging of the vent device, which would jeopardize the integrity of the pin because of fission gas pressure buildup during overpower transients, is not possible.

Some pins surviving the transient power tests must be re-irradiated (steady-state) to the maximum burnup to ascertain the long-term effects of transient overpower, thermal cycling, and the like.

Major Facilities and/or Equipment Required — The TREAT facility might be used for transient testing, but it would probably limit investigation of the power-time envelopes of interest.

If possible, the FFTF will be used for sustained overpower tests in an effort to compensate for the shortcomings of TREAT tests.

Relationship to Current or Planned Programs —

Subtask 6-2.4.4 of the LMFBR Program Plan concerns the design limit and transient survival testing of fuel pins. This program is reduced in scope somewhat from that of the program plan; tests will be conducted primarily on vibro-pak, vented pins.

Schedule and Budget — Manpower costs, in

thousands of dollars, are shown below.

Calendar year														
<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	<u>1978</u>	<u>1979</u>	<u>1980</u>	<u>1981</u>	<u>1982</u>	<u>Total</u>
40	30	30	10	10	10	10	10	10	10	10	10	10	10	210

6.2.1.7. Vent Development

Objective — The fuels and materials program

is responsible for the test portion of this program, both outpile and in-pile. The core design effort will be directed toward the following:

1. Set forth the candidate vent concepts.
2. Determine vent performance criteria.
3. Assist in the material screening program for the porous plug.
4. Supply to the fuels and materials program the test data required for design, including setting outpile and inpile test requirements concerning the specification of temperature, fluence, and sodium impurity level variations.
5. Select the vent device configuration to be employed in prototype fuel pin tests; once the vent device is incorporated into a prototype pin, the vent development considerations will be undertaken in other portions of the fuel pin development program (section 6.2.1.5).

The vent developmental effort will be concentrated on the porous plug device; the diving bell concept will be carried along as a backup in the event that we fail to produce a successful plug material.

Summary of Current Technology — The only significant amount of in-reactor vent device testing is reported in GEAP-4770.¹² A fuel pin employing a diving bell device for venting fission gases directly to the coolant was tested. The results showed that the fission gas holdup time was in the range of from 5 to 10 days and that the only fission gas daughter present outside of the fuel pin in significant quantities was ^{133}Cs . A very small amount of sodium backflow into the fuel pin was apparently due to back diffusion of sodium vapor and subsequent condensation in the pin.

No known results have been published for in-reactor tests of porous plug venting devices. Some outpile tests of porous plugs have been reported by Smidt and Sommer¹³, who tested porous plugs made of Ni, Cr-Ni 18/8, Cr-Ni 18/10, and Fe with pore sizes between 1 and 5 microns. The pressure differential across the plugs at which sodium leakage was detected was at approximately 1.5 atmospheres for a sodium temperature of 500 C and 0.5 atmosphere at 600 C.

Vent device design criteria for the allowable sodium ingress have not been determined. Tests on fuel pins defected with sodium have shown that the presence of sodium poses no threat to the pin integrity for severe overpower transients, at least for short-term sodium logging.¹⁴

Steady-state tests of defected (with sodium) pins have shown that operation up to 47,500 MWd/T peak burnup is possible with no apparent fuel or cladding damage.¹⁵

Scope of Work — A study must be conducted to determine the allowable cover gas activity and the cost penalties associated with the system necessary to maintain a value of the activity equal to or less than the allowable value. This study, to be conducted by plant design, will set the minimum holdup time for fission gases in the pin and provide an evaluation of cost incentives for maintaining the minimum practicable holdup time.

A reasonable holdup time may not be attainable with a porous plug without incurring appreciable internal pin hydrostatic pressures. (Gas transit or holdup time is inversely proportional to gas velocity, which in turn is inversely proportional to gas pressure, making holdup time proportional to pin pressure; this is based on a simplified

slug flow model.) This raises the possibility of eliminating the main safety advantage of vented fuel, namely, no danger of fuel pin failure propagation due to a release of quantities of fission gases in the coolant channels. If, indeed, the fuel pin hydrostatic pressure must be measurably higher than that outside the pin, then the question of failure propagation must be answered. Thus, the investigation of failure propagation, which is considered in section 6.2.1.2, must determine the threshold value of stored fission gas pressure or a combination of gas pressure and volume which poses a threat of possible failure propagation. This threshold value will represent one vent device design criterion.

The design criterion for the allowable sodium ingress will be set by the information gained in section 6.2.1.5, Irradiation of Mixed-Oxide Fuel Pins.

The mechanical design data of interest include fabricability (machinability, weldability, etc.) and the coefficients of thermal expansion. Chemical compatibility with sodium, sodium coolant impurities, cladding and fuel must be demonstrated at the high in-core temperatures expected. It must be shown that the effects of irradiation do not result in compromising the integrity of the porous plug by degrading its mechanical properties.

Bench tests of porous plugs will provide a basis for material screening and will provide design data for determining prototype plug characteristics, such as pore size, quantity, and shape. These tests will evaluate the diffusion or flow rates of both sodium and fission product gases through plugs of various materials and pore sizes, etc., at elevated temperatures.

Plugging tests of different types of vents will be conducted before the prototype fuel pin is selected. The plugging tests must be incore tests with the pin immersed in sodium. The plugging characteristics must be determined with respect to the formation of sodium chemical reaction products on the sodium or the fuel side of the plug; also, the effect of plugging due to solid and volatile fission product accumulation must be determined.

Major Facilities and/or Equipment Required —
Outpile tests will require relatively conventional bench tests and

materials test facilities. Inpile tests for vent plugging determination will be carried out in either EBR-II or Fermi.

Relationship to Current or Planned Programs —

Subtask 6-2.4.4 of the LMFBFR Program Plan, Wash-1106, concerns the development of vented fuel elements. Whereas the program plan at this stage is apparently concerned with the feasibility and performance benefits of a vented concept, we here are concerned with feasibility, performance benefits, development, and qualification for in-reactor use of a vent device.

Budget and Schedule — Costs in thousands of dollars are as follows:

	Calendar year													Total
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	
Manpower	40	30	30	10	10	10	10	10	10	10	10	10	10	200

6.2.2. Fuel Assembly

6.2.2.1. General

Objective — The objective of the research and development program for the fuel assembly is to develop fuel assemblies for a predecessor plant and the reference plant; the reference fuel assembly is expected to be an upgraded or refined version of that for the predecessor plant. It is the goal of the developmental effort to produce prototype assemblies that contribute to the safe, economical operation of the LMFBFR plant. The intermediate goals are as follows:

1. Evaluate fuel pin spacer concepts and select a spacer type.
2. Refine the analytical tools for determining pseudo-steady-state thermal and hydraulic performance and conduct the required testing.
3. Develop methods for calculating the transient thermal and hydraulic fuel assembly performance and conduct the required testing.

4. Perform proof tests on the prototype fuel assembly at the predecessor stage of development.

5. Upgrade the fuel assembly during the time between prototype qualification and reference qualification.

6. Perform reliability tests on a number of complete fuel assemblies before reference plant startup; operation will be performed in the proposed predecessor reactor.

Summary of Current Technology — The fuel pin spacer types in use or planned for use in operating reactors or reactors under construction include the grid type, wire wrap, and spiral fin. Little is known about the spacer pitches required to guarantee acceptable performance with respect to fuel pin vibration, fretting corrosion, and autocatalytic bow. Also, the effects of fuel assembly thermal bow on spacer structural integrity, hot channel factors, and fretting corrosion of fuel pin cladding are not well known. Very little work has been done in the area of spacer effects on the local flow maldistributions causing "hot spots" in the cladding. Some published results pertain to the effects of spacers on intra- and inter-channel fluid mixing, but the data are experimental measurements and difficult to apply to designs except in a qualitative sense.

There are considerable uncertainties in the empirical formulations available for the liquid sodium heat transfer coefficient; the film temperature drops are small enough, however, to make the large uncertainties in the heat transfer coefficient of little concern except where the effects of the pitch-to-diameter ratios are concerned. Pitch-to-diameter ratios of less than 1.1 can result in a considerable reduction in the heat transfer coefficient. Analytical determination of local coolant velocity and temperature profiles depend on the setting of empirical constants with experimental measurements; further experimental confirmation of analytical results is required for these low pin pitch-to-diameter ratios.

Pressure drop correlations for fuel pin bundles with and without spacers are not considered accurate enough, for design purposes, to reduce "built-in" flow maldistribution to the desired low level. Also, these pressure drop correlations are not applicable to

very low flow rates, namely, those occurring in the natural convection range.

The transient flow and heat transfer methods are generally considered adequate for engineering purposes as long as the reactor transients considered are moderate. Analyses for fast reactor transients require further development.

The methods and design data now used to predict the structural adequacy of a fuel assembly are not accurate enough to allow elimination of mechanical and thermal-shock proof tests. Tests must be conducted to certify the performance of the assembly in cases involving fluid pressure forces, thermal shock, thermally induced assembly stresses, and vibration of earthquake intensity.

Scope of Work — The fuel assembly configuration of interest for the predecessor and reference plants is a design that is similar to the present reference design with the possible exception of the spacer. A solid wrapper can is anticipated for both the predecessor and the reference assembly. The direction of the research and development program is subject to change, however, depending on the outcome of further studies (and consideration of design innovations), which could demonstrate the feasibility of such concepts as hydraulic holddown, perforated wrapper cans, or the fixed can (discussed in section 6.2.5, Core System). The direction of the fuel assembly program will be reassessed in the event that other practical configurations show promise of reducing operating difficulties, energy costs, and the like.

The type of spacer employed in the LMFBR affects core performance in many ways, so that its importance in contributing to the overall plant success cannot be ignored. Even though the reference design now uses the helical ribbon spacer, many questions concerning its performance and feasibility still must be answered. Therefore, the fuel assembly development program includes a provision for evaluating a number of different types of spacers; this number should be reduced to the most promising two or three when hardware commitments are made for the predecessor.

All the fuel assembly testing is funded under this program. Ex-reactor prototype assembly tests will be conducted

before the hardware commitment for the predecessor is made. In-reactor tests of reference fuel assemblies will be conducted in the predecessor.

6.2.2.2. Fuel Pin Support System Selection

Objective — The objective of this program is to evaluate, select, and qualify a fuel pin spacer that contributes to the safe operation of the LMFBR yet allows the maintenance of a low power generation cost. This objective also includes the development of a fuel assembly wrapper can and end fitting geometry. The spacer selection process involves the following:

1. Methods development and design data acquisition in the areas of flow-induced vibrations, fuel pin bow (thermal and hydraulic induced), spacer stress analysis, fretting phenomena, spacer pressure drop, assembly bow induced loads on pin spacer and interchannel fluid mixing.
2. Materials screening depending on the spacer operation (such as compliant or non-compliant).
3. Determine design criteria for the allowable pin bow, vibration amplitude (cladding fatigue or fretting corrosion), "hot spot" temperature increase or thermal gradient, and the required amount of fractional pin diameter increase to be accommodated.
4. Establish the feasibility of candidate spacer concepts based on their satisfying the design criteria, their fabricability, and their structural integrity over the fast fluence and temperature ranges expected especially the compliant-type spacers).
5. Design a prototype and an alternate or alternates.
6. Design outpile tests to measure the level of performance of various types of spacers.
7. Evaluate the significance of test results in predicting inpile performance, and fix a design for the predecessor plant; recommend a candidate or candidates for the reference core.

Difficulties are anticipated in developing analytical models of sufficient accuracy, and in obtaining usable design data in view of the very complex

phenomena of flow-induced vibration, pin bowing, and fretting corrosion, among others. Problems of spacer evaluations in these performance areas may have to be attacked by the trial-and-error developmental approach or circumvented by eliminating spacer concepts with extremely dubious qualities in these areas. Thus, we realize that qualitative evaluations will play some part in selecting a spacer, at least in the foreseeable future.

The fuel can development will be a matter of analytical model development, material selection, fabrication methods, development prototype construction, and proof testing. Development of the fuel assembly end fitting and grid support juncture will involve determining a geometry that minimizes assembly bypass flow but allows no opportunity for fluid cavitation and the consequent fitting and/or grid material erosion.

Summary of Current Technology — The spacers that are currently in use or planned for use in sodium-cooled fast reactors include the grid, wire-wound, and spiral-fin types. Reactors using or planning to use the grid type are Fermi and PFR. The wire-wound spacer is used in EBR-II and has been planned for use in the FFTF, although indications now point to a switch to the spring spacer. The Rapsodie reactor in France employs a spiral-fin spacer. The geometries of spacers of the same type, may vary considerably; for example, the helical pitch of the wire-wound spacer in the EBR-II is 6.0 inches and in the FFTF is 12.0 inches. This indicates that very little is known about the relationships between spacer geometry and performance.

Although the helical ribbon spacer (which is the reference spacer concept) has not been employed in any operating reactor, it has undergone extensive testing for possible use in boiling water reactors. Most of the published results describe the tests carried out by the Atomic Division of SNECMA in France. Researchers in Germany (Allgemeine Elektrizitäts Gesellschaft; Frankfurt am Main, West Germany) and Italy (Italian Atomic Research Center; Ispra, Italy) have been testing this type of spacer.

The feasibility of employing grid-type spacers is evidently not limited by the ability to fabricate grids of very low

thickness, since grids have been satisfactorily produced with thicknesses of 7 mils.¹⁶ The wire-wound and spiral-fin concepts apparently pose no fabricability problems. Helical ribbon spacers have been fabricated for experimental purposes. Twist pitch-to-width ratios of 12.0 have been reported¹⁷ for ribbons constructed from stainless steel and Zircaloy-2. Brass ribbons with pitch-to-width ratios of 3.62 have also been made. Twisted tapes (ribbons) of Inconel have been made with pitch-to-width ratios of 4.16, but in no case is information available on the dimensional variations (from the nominal) of the finished product.

The structural integrity of spacer candidates is a very important consideration since the spacers will be subjected to unprecedented temperatures and fluences. The wire-wound and spiral-fin types are not expected to be limited by structural integrity considerations. The use of compliant spacers (grid and helical ribbon), however, may be limited because it is known the fast fluence severely embrittles the candidate materials. Very little work in this area of spacer loadings and structural integrity has been done for LMFBR application.

Pressure-drop correlations are available for the grid, wire-wound, spiral-fin and helical-ribbon spacers. de Stordeur¹⁸ reports methods of calculating spacer pressure losses for both the grid and wire-wound spacers. His correlation for the drag coefficient for wire-wound spacers may suffer somewhat in accuracy, but his correlation for grid spacers appears to be satisfactory. The French¹⁹ have reported a pressure-drop correlation for in-line fluid flow through a tube bundle employing helical ribbon spacers. This correlation has a rather limited range of spacer twist pitches for good pressure-drop accuracy.

One consideration in selecting the spacer and pitch (or distance between fuel pin support locations) is the minimization of flow-induced pin vibration amplitudes. Although empirical relationships for predicting vibration amplitudes of pins (or beams) for in-line coolant flow have met with some success, these relationships fall far short of enabling the designer to determine acceptable support locations or intervals for fuel pins. Empirical formulations by Burgreen and Paidoussis have been compared with experimental results for pins supported at intervals by spacers.²⁰ Burgreen's prediction of vibration

amplitudes ranged up to approximately 15 times too high; Paidoussis's results indicated amplitudes of up to twice those observed experimentally, depending on the fluid temperature. Based on an approach proposed by E. P. Quinn,²¹ General Electric has presented the results of an analysis to predict the required distance between fuel pin supports for grid spacers and wire-wrap spacers. It has been asserted that their results need no experimental confirmation unless the value of a vibrational stability factor falls below a certain fixed value.

The pin vibration aspect of spacer selection has some effect on the cladding fretting corrosion problem. In the interest of minimizing fretting corrosion, the vibrations of the fuel pins must be minimized or eliminated. The possibility of eliminating pin vibrations entirely is remote. Other factors enter into the fretting corrosion phenomenon, such as cladding and spacer contact loads and the spacer shape at the contact point. Little or no data exists for predicting the corrosion fretting performance of a spacer system.²² Each spacer type and fuel assembly design must be evaluated in flow-loop lifetime tests.

Another consideration in selecting a spacer and pitch is the possibility of autocatalytic bow of fuel pins. The buckling-type of fuel pin instability has been shown to occur under certain conditions for isothermal flows.²³ The extent to which thermal bow in fuel pins contributes to this phenomenon is not known, but it is felt that the instability would occur at lower fluid velocities in the case of heat generation in the pin as opposed to the isothermal case. A deflected or buckled pin should produce a flow distribution which in turn produces a peripheral clad temperature variation tending to deflect the pin further.

Some spacer types affect the liquid sodium heat transfer because of the effects on local velocity distributions. A spacer characteristic that results in increased inter-channel mixing is very desirable since it permits a reduction in the surface temperatures of the hot channel cladding. The effects of a wire-wrap spacer on inter-channel fluid mixing have been measured.²⁴ Appreciable fluid mixing occurs with wire-wrap pitches of from 10 to 18 inches without excessive pressure drop. Some types of spacers may distort the local velocity profiles so as to decrease the amount of coolant "service" to the cladding and thereby produce "hot spots". At this time, there is not enough

knowledge of the turbulent flow mechanisms to predict accurately the effects of spacers on local flow distributions; the three-dimensional flow patterns produced by some types of spacers result in added complexity and increase the errors in applying approximate methods for the solution of flow distributions. The effects of the type of spacer on hot channel factors (both hot spot formation and inter- and intra-channel fluid mixing) will likely be adequately determined only by combined analysis and experiment in the foreseeable future.

The effects of fuel assembly bow on spacer selection must be considered. Present techniques are considered adequate for determining the effects of fuel assembly bow on hot channel factors for compliant spacers (assembly bow deflections will tend to decrease the flow area for some channels), and the added contact loadings at spacer-cladding support points can be determined for non-compliant spacers. What cannot be ascertained at this point, however, is the effect on cladding fretting corrosion due to the fuel assembly deflection-induced cladding-spacer contact loads.

Scope of Work — The program for selecting the fuel pin support system will emphasize developing an assembly with a solid can (nonperforated); however, if the results of a study on the fixed-can concept (section 6.2.5.4) or a similar concept demonstrate the feasibility of an alternate assembly concept allowing safer operation and lower power generation costs, then this program will be re-evaluated and redirected if the schedule allows. The program is now directed toward evaluating the most promising spacer concepts and narrowing the field to as few as two by the time that hardware commitments are made for the predecessor (one predecessor type and an alternate, probably an advanced concept for the reference plant).

Figure 6-1 is a schematic diagram of the spacer selection process; it is not intended to represent completely the entire design process, but outlines the main steps to be taken in the selection. The figure shows that methods development feeds into the selection process at various steps or stages. Methods development, both analytical and supporting experimental work, will be carried on during the entire time allotted for spacer selection, as will the materials evaluations. During the spacer development program, a spacer may be

Figure 6-1. Spacer Selection Schematic Diagram

at stage 1, 2, or 3 up to the time the predecessor design is fixed (stage 4); therefore, we must allow for incorporating promising design innovations during the developmental program. Economic and schedule considerations, however, will seriously limit the possibility of developing any types not examined at the outset of the program. Some types may not be considered because of shortcomings in analytical techniques, test techniques, or design and fabrication data.

Some overlap exists between this program and sections 6.2.2.3 (Steady-State Thermal Hydraulics) and 6.2.2.5 (Fuel Assembly Proof Tests). The spacer affects the flow distribution in the pin bundle primarily through the effect of hydraulic resistance. Since this has some bearing on spacer selection, in this program we will generate empirical formulations for spacer pressure loss and inter-channel mixing and supply these to the Steady-State Thermal/Hydraulics program. Likewise, some information generated in that section will be used in selecting a spacer; information such as the minimum fuel pin pitch-to-diameter ratio as determined by thermal considerations will be supplied to this section of the overall program. The shaker proof tests of section 6.2.2.5 will require some preliminary information concerning the mass, natural frequency, stiffness, etc., for a prototype fuel assembly. Although this information could conceivably be a responsibility of the spacer selection program, for the sake of continuity it will be generated under the shaker test program. The structural limitations of fuel assembly interaction during earthquake-induced vibration will enter into the spacer selection problem in a relatively minor way; it is felt that the fuel assembly wrapper can must be designed for minimum deformation under loads resulting from mechanical interaction of the assembly during earthquake phenomena.

Considerable work is required in the area of engineering methods development. The phenomenon of flow-induced vibration must be investigated to obtain (1) a thorough understanding of the mechanisms involved, and (2) an accurate and flexible calculational method supported by experimental measurements. Experiments conducted in water will be sufficient to confirm any proposed theoretical model. No liquid sodium tests are planned. Again, no tests in flowing

sodium are proposed for a model to predict fuel pin autocatalytic bow; we have taken this approach because:

1. The transverse hydrodynamic forces which occur with flow-induced pin vibrations or pin buckling instability (zero-frequency vibration) should be independent of the presence of a pin heat flux, neglecting the fluid's physical properties variations with fluid temperature.

2. A successful flow-induced vibration model should lend itself to a determination of zero-frequency amplitudes.

3. The calculation of zero-frequency amplitudes will then allow superposition of calculated cladding temperatures and thermal-induced bow deflections on the isothermal case where velocity maldistributions as a function of pin deflection can be determined experimentally.

Item 1 above pertains to the case of a fuel pin exactly centered in its coolant channel with no flow maldistribution for the pin in its normal position. Of course, situations arise in a practical case where transverse coolant thermal gradients exist as a result of power gradients or differing flow areas on opposite sides of a pin; in such cases, the autocatalytic bow phenomenon can be considered independent of the pin-fluid interaction as far as the initiating mechanism is concerned. Tests in sodium, however, should still not be required with the proper use of the results from experiments in water in combination with analytical model calculations. Velocity profile measurements in water can provide means for calculating fluid and cladding temperatures. Tests in water can also provide transverse fluid forces on a fuel pin.

An analytical model to predict fretting corrosion rates is very desirable. This is admittedly a very complex problem, which from present indications will probably not yield even to empirical approaches in the near future. Therefore, the effects of fuel pin and support contact geometries (very simple geometries only), contact loads, and pin vibration amplitudes should be determined by tests. If an empirical correlation is not forthcoming, perhaps some qualitative indications may result for methods of minimizing fretting corrosion.

Tests will be conducted in water to determine the pressure-drop characteristics of various spacers. Empirical correlations will be developed to give the pressure drop of different spacer types (the main candidates, primarily) and different configurations of these types over a range of pin diameters and pitch-to-diameter ratios. Information on flow distributions for eccentric and bowed fuel pins is also required.

Models must be developed that can be used to determine accurately the fuel assembly bow-induced pin-cladding contact loads for compliant and non-compliant spacer types. This involves the development of a spacer deflection-load model, which will be particularly difficult for the helical ribbon spacer. From the results of these methods developments we will gain information on the effects of fuel assembly bow on flow maldistribution (due to channel area changes) and the resultant hot channel cladding temperature addition.

It is felt that the effects of the type of spacer on intra-channel fluid mixing and local hot spot formation are not amenable to analytical treatment without experimental data and will not be in the foreseeable future. The performance of the various types of spacers in these problem areas will be rated qualitatively or tested in an experimental mockup to ascertain their effects on cladding temperature (no analytical or empirical formulations will be sought).

Some tests may be conducted to formulate empirical relationships to predict inter-channel mixing (due to transverse quasi-steady velocity components) on spacers such as the wire-wound type for which mixing is a decided advantage. Water tests would be sufficient in this case.

Stress analysis methods must be developed to calculate thermally and mechanically induced stresses in spacers. Both safety and energy cost considerations make methods improvement imperative.

The materials evaluations and screening for spacer candidates is included in the spacer selection process and is illustrated in Figure 6-1. Since the spacer is subjected to the same environment as the cladding, except for the absence of contact with the fuel, and has similar performance requirements regarding ductility,

creep strength, and the like, the spacer material problem area will draw heavily on the core materials development programs (see sections 6.2.1.4 and 7.3). Fuels and materials personnel are responsible for test evaluations of materials for spacers.

Stage 2 of the spacer selection process as illustrated in Figure 6-1 involves the delineation of spacer criteria. The calculational methods developed above will aid in setting the following criteria:

1. Fuel pin maximum bow deflection as a function of fuel pin diameter and pin pitch-to-diameter ratio.
2. Vibration maximum amplitude (determined by cladding fatigue or fretting corrosion).
3. Fractional cladding diameter increases ($\Delta D/D$) to be accommodated without spacer-induced cladding failure.
4. Spacer-induced "hot spot" temperature increase and/or thermal gradient in the pin cladding due to local flow maldistributions. (There is a low probability that the designer could translate these criteria into the allowable extent of flow maldistributions resulting from various spacer geometries; however, approximate results from flow visualization mockups may suffice for most spacer configurations.)

When methods or design data are lacking, it may not be possible to judge whether some spacer configurations satisfy the design criteria. Figure 6-1 reflects this possibility and indicates the resulting options under these circumstances. Further consideration of such a spacer configuration may be delayed until methods or test data become available, or evaluation of the spacer may proceed until it is confirmed that the design criteria are satisfied (step 5 of Figure 6-1, performance evaluation).

Satisfaction of the design criteria directly affects the question of feasibility of spacer geometries. Satisfaction of the criteria may require impractical spacer geometries from the standpoint of energy cost. Other factors affecting feasibility are considered in step 3 of Figure 6-1 (steps 2 and 3 are arbitrarily sequenced to some extent, since feasibility factors other than the satisfaction of the design

criteria may be investigated as the first step in considering a type of spacer). These factors affecting feasibility are as follows:

1. Can the spacer be fabricated at reasonable cost and does it result in an acceptable dimensional tolerance buildup (as it affects hot channel factors and pin loading uniformity)?

2. Does the spacer configuration allow assembly fabrication techniques that do not jeopardize the cladding or support integrity?

3. Will a compliant-type spacer perform properly over the levels of irradiation exposure expected?

Steps 4 and 5 in the spacer selection process involve the design of a predecessor and an alternate(s) spacer configuration, the design of tests for performance evaluation, and the performance measurement, both analytical and experimental. The selection of predecessor and alternate concepts will be based on the following performance considerations, assuming that the design criteria are met:

1. Dimensional tolerance buildup effects on hot channel factors (or maximum cladding surface temperatures).

2. Fretting corrosion rates as they affect required cladding thickness or maximum allowable pin inpile time.

3. Spacer manufacturing costs as affected by fabrication problems.

4. Spacer component of coolant pressure drop.

5. Spacer steel volume required for structural integrity under forces induced by quasi-steady and dynamic fluid forces, thermal gradients, fuel assembly bow, pin thermal bow, and pin growth ($\Delta D/D$).

The mixing of coolant induced by a spacer configuration is not a main consideration except where the spacer is known to appreciably affect inter-channel mixing, as in the wire-wound case.

Major Facilities and/or Equipment Required —

Assuming that the development of methods is successful, only the corrosion fretting tests will require a flowing sodium loop. Since experimentation in water is much less involved and less expensive, we have

elected to pursue this path. Admittedly, some degree of error will be inherent in the results when the thermal aspects of a flow problem are simulated in water. If the errors appear to be appreciable, the approach will be re-evaluated and the course of investigation altered accordingly.

Existing hydraulic facilities are available for water tests and would require only slight modifications. Sodium test loops to measure the rate of fretting corrosion include facilities at APDA and PNL.

Relationship to Current or Planned Programs —

This program for selecting a fuel pin support system is similar to that of Task 6-2.3 of the LMFBR Program Plan, Wash-1106, with the exception of subtask 6-2.3.4 as described in the same document (subtask 6-2.3.4 concerns Thermal Gradient and Thermal-Transient Proof-Testing of Fuel Assemblies). This program provides for the thermal gradient and thermal shock tests mentioned in section 6.2.2.5. The costs in thousands of dollars are as follows:

Schedule and Budget —

	<u>Calendar year</u>				
	<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>Total</u>
Manpower	150	266	320	166	902
Facilities		182	160	115	457
Equipment		134	90	39	263
Total	150	582	570	320	1622

Detailed Tests and Development Programs —

The following sections A through E are detailed test and development program descriptions pertaining specifically to the helical ribbon spacer. These descriptions delineate typical types of tests to be conducted and the costs incurred in evaluating a single type of spacer. These costs are not to be added to the budget figures quoted in the foregoing Schedule and Budget subsection; the costs have already been included with an appropriate multiplier to account for the additional similar investigations of other spacer types that will be performed.

A. Helical Ribbon Spacer and Apron Development

Information is desired on the fabrication methods required and on projected fabrication costs to produce helical spiral-ribbon spacers and aprons to appropriate tolerances. Several promising materials will be evaluated for fabricability. Spacer and apron properties, such as crush strengths and spring constants (or deflection versus load), will be determined to provide feedback on the design analytical models. Effects of temperature, irradiation, and corrosion on performance will not be necessary since the information generated in the materials research program is considered sufficient.

Prototype aprons and spacers that meet dimensional requirements will be fabricated. The manufacturing processes will be documented (forming methods, die dimensions, heat treatment, etc., where applicable) so that cost figures can be projected.

The spacer and apron performance test specimens will be unirradiated. Several spacers and aprons of various cross-sectional dimensions and pitches will be tested in air at room temperature.

The measurements desired include spring constants (or deflection versus load) and crush strength (in a transverse direction).

The costs in thousands of dollars are as follows:

	<u>Calendar year</u>
	<u>1969 - 1971</u>
Manpower	110
Facilities	20
Equipment	20
	<hr/>
Total	150

B. Spacer, Apron, and Cladding Erosion, Fretting Corrosion and Wear

Economic studies have shown that the thickness of the fuel pin cladding has a significant influence on reactor economics and should be minimized to improve nuclear performance. It should be remembered, however, that the cladding performs a vital function in ensuring against coolant-fuel interaction. For this reason, the cladding thickness must remain sufficiently thick throughout core lifetime to prevent failure. A

10-mil cladding thickness has been selected for the reference design. This thickness will be continually reduced during lifetime by the effects of corrosion, erosion, and fretting. This test will (1) provide design information on the individual problems of corrosion, erosion, and fretting to allow a more exact design basis, and (2) determine means of reducing material losses by corrosion, erosion, and fretting.

The tests will be designed to include the following:

1. Cladding erosion due to flow — Current data indicate that materials in a sodium environment will be subjected to increased corrosion rates with increasing sodium velocities.
2. Corrosion due to sodium impurities — Corrosion has been shown to be increasingly significant with increasing sodium temperature or impurity level.²⁵ Impurities in the sodium tend to react with the cladding material, resulting in a loss of material and/or strength. The cost and engineering problems associated with a highly efficient cleanup system make the effects of sodium impurities of particular interest since some optimum cleanup capacity may be found.
3. Fretting — Fretting and vibration-induced wear between the clad, twisted ribbon spacers, apron, and assembly can is a major area of uncertainty because of the novel spacer design. Thermal cycling, bowing, and coolant-induced vibration all tend to produce relative movement between components which results in material wear.

Tests for corrosion, erosion, and fretting should be performed in sodium using a full-sized assembly or a rod cluster. The coolant velocity should be varied between 20 and 30 ft/sec at temperatures ranging from 1200 to 1300 F. Impurity levels should be varied from 5 to 20 ppm oxygen to cover the range of interest sufficiently.

The costs in thousands of dollars are as follows:

	<u>Calendar year</u>
	<u>1970 - 1973</u>
Manpower	70
Facilities	60
Equipment	<u>30</u>
Total	160

C. Flow-Induced Vibration

The timely development of analytical methods to predict the vibrations characteristics of a very complex geometry such as the helical ribbon spacer and fuel pin combination cannot be presumed. Thus, the assurance of acceptable fuel pin vibrations under reference core conditions rests on a combined analytical and experimental development program. An experimental approach, however, is not without difficulties for two prime reasons: first, the vibration characteristics of the elements (pins and spacers) are strongly temperature dependent since the rigidities drop off sharply at the temperatures of interest; second, no completely satisfactory methods of instrumentation for measuring vibration amplitudes exist for use in the very tight LMFBR fuel pin lattices. Thus, a supposedly meaningful outpile experimental program set forth at this time would be based on conjecture—further studies are required to delineate a test program other than an inpile assembly lifetime test for proper pin characteristics, which is quite impractical. A tentative program must be put forth at this time, however, as an immediate goal. This program will be predicated on two assumptions: (1) that dummy fuel pins and spacers at ambient or moderate temperatures can be constructed to closely simulate the vibration characteristics of the actual pins and spacers at operating temperatures; (2) that acceptable instrumentation can be developed.

Vibration tests will be performed with a rod cluster in water. These tests will be directed toward measuring vibration amplitudes as a function of spacer pitch, thickness, support grid, and entrance and exit nozzle geometry. Vibration amplitude readings will be taken for velocities in the range of 15 to 30 ft/sec.

The anticipated costs in thousands of dollars are as follows:

	<u>Calendar year</u>
	<u>1970 - 1972</u>
Manpower	214
Facilities	48
Equipment	<u>48</u>
Total	310

D. Assembly Wrapper Can Development

An analytical model will be developed to provide information on bending, burst strength, and flexure versus load. The integrity of the can wall must be shown under the expected differential pressures, misalignment due to the differential thermal expansion of the upper and lower support grids, and flats deflection versus load, so that fuel pins are properly constrained.

A bending analysis considering the wrapper can as a beam under creep will be conducted. The ends will be misaligned and a plot of deflection versus time will be developed. The pressure analyses will be performed using finite element techniques. This will show flat deflection versus pressure and also provide some estimate of burst strength.

Tests will provide information on bend proof testing, burst strength, and flats deflection versus load. Proof tests demonstrating the integrity of the wrapper can under loads induced by lateral misalignment of the upper and lower end fittings must be conducted.

Bend tests will be conducted with the fuel assembly wrapper can attached to simulated upper and lower end fittings. The thickness of the wrapper can will be reduced to the value expected at end-of-life conditions (accounting for erosion and corrosion). The wrapper can will be unpressurized, and the tests will be conducted at temperatures ranging from 1100 to 1300 F in an inert gas environment. The two end fittings will be laterally displaced, and the required displacement sufficient to cause rupture will be noted. Since the performance of the wrapper can in the embrittled condition (due to fluence) is of prime interest, strain measurements will be taken by which performance can be projected.

Tests will be conducted in which the direction of lateral displacement is (1) parallel to a diagonal of the hexagonal cross section, (2) perpendicular to the wrapper can flats, and (3) at various intermediate directions. The test data required include the amount and direction of the end fitting misalignment required for failure, the mode of failure, and full pre- and post-test characterization.

The burst strength test will be conducted on a wrapper can of thickness equal to nominal, less manufacturing tolerance and less allowance for erosion and corrosion. The internal pressure will range from atmospheric to that at which failure occurs. The pressurizing fluid will

be an inert gas or nitrogen, at temperatures ranging from 700 to 1300 F. This test may conceivably be combined with the flats deflection versus load test where the data of interest are flats displacement as a function of internal fluid pressure (from which the load may be deduced). Data of interest include the internal pressure resulting in failure of the can wrapper as a function of temperature.

The anticipated costs in thousands of dollars are as follows:

	<u>Calendar year</u>
	<u>1969 - 1970</u>
Manpower	60
Facilities	60
Equipment	40
	<hr/>
Total	160

E. Assembly Lower Fitting and Support Grid Juncture Development

A fuel assembly lower fitting and support grid mating geometry resulting in minimal leakage or bypass flow must be developed. A sealing geometry or technique that results in relatively low leakage velocities is also desired in order to minimize erosion of the lower grid support structure. The absence of flow cavitation in the leakage path must be demonstrated.

Test specimens will be full-scale mockups of the actual assembly configurations. Tests may be performed in water to determine the pressure-flow characteristics of the leakage path geometry. Also, water may be used as the test fluid for determining the required flows and pressures resulting in cavitation.

The functional parameters to be tested include the pressure-flow characteristics of the leakage path, including the effects of dimensional variations in the mockup geometry (to account for effects of manufacturing tolerances). The flow and system pressure effects on fluid cavitation inception should be investigated with leakage flows and pressure drops ranging up to approximately 3000 lb/hr and 70 psig, respectively. System pressures will range up to 130 psig.

Anticipated costs in thousands of dollars are as follows:

	<u>Calendar year</u>
	<u>1973</u>
Manpower	30
Facilities	15
Equipment	<u>15</u>
Total	60

6.2.2.3. Steady-State Thermal-Hydraulics

Objective — The steady-state thermal-hydraulics program objectives are as follows:

1. Develop methods to calculate accurately the local coolant velocity and temperature distributions to allow determination of peripheral cladding temperature variations for normally and eccentrically placed fuel pins.
2. Related to that in 1 above is the development of methods to calculate the outlet coolant flow and enthalpy distribution taking into account the inter-channel mixing under ideal conditions of no spacer effects.
3. Investigate the effects of low pin pitch-to-diameter ratios on the average heat transfer coefficient if the incentive is shown to reduce the P/D to less than 1.1 based on economic or safety (sodium voiding Δk) considerations.
4. Investigate and determine the effects of low Peclet numbers and coolant impurities on the average heat transfer coefficient of in-line-flow through rod bundles.
5. Develop the bundle orifices employed for core flow distribution.

These objectives involve both methods development and testing, which will be the responsibilities of this program.

Summary of Current Technology — A number of investigators have presented empirical expressions for the heat transfer coefficient for liquid sodium flowing on the inside of round tubes. Considerable discrepancy results from using the various expressions for calculating the Nusselt number as a function of the Peclet number.

The disagreement in values of the Nussult number is large; e. g., differences of up to 40%, depending on the value of the Peclet number, have been obtained. However, since the film temperature drops at the heat fluxes of interest are small, a discrepancy of 40% in the Nussult number or the heat transfer coefficient is of little significance for LMFBR design purposes. Additional error can be introduced by applying results obtained for flow inside a tube to in-line flow through closely packed pin bundles; this error also can be considered of little importance.

Dwyer²⁶ has shown that the average heat transfer coefficient is reduced for pitch-to-diameter (P/D) ratios below about 1.3. Since the heat transfer coefficient might be reduced by as much as an order of magnitude in reducing the pitch-to-diameter ratio to 1.0, this effect deserves further investigation (low effective P/D's may possibly arise as a result of pin bow or spacer fabrication tolerances). Pitch-to-diameter ratios of 1.0 result in very large peripheral cladding temperature variations, but these low P/D ratios may offer some benefit with regard to the safety aspects of sodium void propagation; according to Dwyer²⁶, the Russians have shown some interest in these low P/D ratios.

The determination of maximum cladding surface temperatures for a fuel pin is dependent upon the calculation of coolant velocity and temperature profiles. Velocity profiles can be calculated at a given axial location for fully developed flow (absence of intermittent or radially non-symmetrical spacer effects) with probably what is acceptable accuracy for engineering purposes at P/D ratios greater than 1.3. Bender²⁷ has calculated local velocity distributions in pin bundles using a mixing length theory for calculating turbulent momentum transport. The authors state that more detailed experimental data are necessary to demonstrate the general validity of the method.

Although results have been published for the calculated velocity distribution in a single channel, as mentioned above, no published results are available for the calculation of coolant velocity and temperature profiles for a number of adjacent channels. It is important for the LMFBR designer to be able to predict the effect of

inter-channel turbulent mixing on the local coolant temperature profiles, since mixing is expected to have a measurable effect on hot channel maximum cladding surface temperatures. Experimental measurements are available which provide mixing coefficients²⁸ for PWR application, but these results, of course, cannot be applied to liquid metal situations.

The fuel assembly flow distribution may have a strong effect on the hot channel cladding surface temperature. Since the type of fuel pin spacer affects the channel pressure drop and consequently the flow distribution, there are means to calculate the flow distributions, but the accuracies are such that the channel flows require experimental confirmation. Experimental measurements for fuel assemblies employing water (at an appropriate temperature up to 250 F) as the flow medium can provide sufficient accuracy for determining sodium flow distributions.²⁹ Accurate measurements of a fuel assembly's coolant outlet temperature profiles can be obtained only by tests with liquid sodium as the fluid because the heat transfer characteristics of sodium are very different from those of conventional fluids like water.

The uncertainties involved in determining heat transfer coefficients and flow distributions at very low flow rates (corresponding to those required for decay heat removal) will require additional experimental data. The accuracy of available spacer pressure-drop correlations does not allow their use at very low Reynolds numbers.

Scope of Work — A model must be developed to predict accurately the thermal performance of the fuel assembly. In particular, this includes a complete temperature map of the coolant at the axial location of the peak cladding surface temperature; inter-channel mixing and fuel pin eccentricity, off the normal channel axis, must be accounted for. For the foreseeable future, it is anticipated that analytical methods to predict fluid velocity and temperature profiles will be semi-empirical. Thus, experimentally determined constants will be required. Experimental measurements of velocity and temperature profiles will be conducted in flowing sodium.

Should the incentives be demonstrated in energy cost savings and Δk voiding coefficients for operating at low P/D ratios, then the effect on the average heat transfer coefficient will require

investigation. Again, experimental apparatus employing flowing sodium will be necessary.

The flow Peclet number and cladding surface condition are known to influence the average heat transfer coefficient. The average heat transfer coefficient at Peclet numbers corresponding to free convection flow rates may be of interest if methods are devised to remove the fuel decay heat by natural circulation. Of course, the Peclet number, which is based on the Reynolds number, should not be expected to correlate heat transfer measurements where the Grashof number becomes the pertinent dimensionless variable. The effect of coolant impurities on the average heat transfer coefficient is of little concern in the LMFBR core except where it may aggravate peripheral cladding temperature variations. An investigation should be entered into to determine whether the effect of coolant impurities on cladding "fouling" is dependent upon circumferential location around the pin, and if so, to what degree.

Since corrections can be made for the fluid property effects on the heat transfer coefficient, experimental measurements to validate the methods or to obtain the empirical formulations discussed above can be made in relatively low-temperature sodium. This is important from an economic standpoint since the development of high-powered rod heaters and high-temperature instrumentation is costly. "Fouling" effects, however, may be so dependent on the coolant temperature that we must admit the possible necessity of operating tests up to 1300 F. Measurements of interest in these heat transfer studies generally include flow rates, pin local heat generation rates, coolant temperatures, and cladding surface temperatures. Coolant impurity levels should be held to a minimum in most studies except the case where the specific effects of impurities are desired. Characterization of the cladding surface condition and finish of the heater (simulated fuel pin) is extremely important in these measurements. Since mean fluid velocities, depending upon the test objectives, will range from less than 1 ft/sec to more than 30 ft/sec, multiple flow loops will be required for accurate flow control. Experimental loop flow capacities have not been set at this time, since the size requirements for test specimens have not been set.

Various sizes of orifices must be tested to determine the pressure-flow and cavitation characteristics. These characteristics are dependent upon upstream and downstream velocity profiles as well as the upstream level of turbulence and system pressure. Therefore, accurate prediction of operating orifice characteristics will be dependent upon experimental results obtained in a simulated inlet plenum and fuel assembly mockup. Orifices for both the fuel assemblies and the radial blanket assemblies will be tested. No transient flow measurements are anticipated; quasi-steady flow measurements should suffice for projected fuel assembly transient performance (with regard to orifice effects).

The test apparatus will consist of an inlet plenum and fuel assembly mockup. Orifice plates, or some other throttling device, with a range of hole sizes will be tested. Water will be used as the flow medium. The test data required are the pressure-flow characteristics of the orifices and the flowrates resulting in cavitation as a function of system pressure.

Major Facilities and/or Equipment Required —

A sodium test loop will be required for this program. The same facility is expected to be employed by other programs in the core design R&D plan. The entire capital cost of the facility, however, is funded under the Steady-State Thermal-Hydraulics program.

The exact capacity, instrumentation, temperature limitations, etc., of the facility have not been determined. The projected cost of \$3,000,000 involves considerable uncertainty.

Relationship to Current or Planned Programs —

The LMFBR Program Plan, Wash-1106, states that there is little incentive for further investigation into values for the liquid sodium heat transfer coefficient for rod bundle flow. We have admitted the possible necessity of investigating values of heat transfer coefficients for pin pitch-to-diameter ratios of less than 1.10. The LMFBR Program Plan describes a program for determining the inter-channel conduction and turbulent heat transport.

Schedule and Budget — Budgeting beyond 1974

provides for possible additional methods development to maintain the engineering tools for the reference fuel assembly design. The costs in thousands of dollars are shown below.

	Calendar year												Total
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	
Manpower	100	330	655	430	390	15	15	15	15	15	15	15	2010
Sodium Loop	1200	1000	800										3000
Facilities (excluding sodium loop)		150	135	100	85								470
Equipment		220	110	70	65								465
Total	1300	1700	1700	600	540	15	15	15	15	15	15	15	5945

6.2.2.4. Transient Thermal and Hydraulic Performance

Objectives — The following goals are part of this program:

1. Improve methods for calculating the effects of plant transients on individual channel flow rates, fluid velocities, and temperatures, and cladding temperatures for single-phase liquid flow.
2. Determine the minimum P/D (pin pitch-to-diameter ratio) acceptable from a transient flow standpoint in the event that energy cost and safety (Δk voiding) incentives demonstrate the desirability of a P/D less than 1.1. This determination must be made in concert with a heat transfer coefficient investigation at $P/D < 1.1$ (section 6.2.2.3).
3. Parallel the effort described in 6.2.1.2. in developing methods to determine the effects of the small amounts of fission gas release expected upon failure of vented-type fuel pins.
4. Develop methods to calculate the thermal transient feedback on core reactivity due to fuel assembly displacements (bowing and support structure movements).

Fluid and cladding temperature predictions for reactor transients are important, since boiling and sodium vapor formation must be avoided. In this program techniques will be developed to accurately predict conditions in which sodium boiling does not occur.

The safety program will investigate the two-phase flow phenomena occurring during sodium boiling. We are, however, interested in the two-phase flow case with sodium as the liquid phase and fission gas as the vapor phase. The effects of the presence of the small amounts of fission gases expected to occur in the flow channel after a cladding failure of a vented pin (in particular with a porous plug vent) are of interest.

Summary of Current Technology — Transient heat transfer models presently available for single-phase flow of a liquid are generally considered sufficient for moderate transients occurring in the LMFBR. Accuracies may be improved by making fewer simplifications in the governing differential equations for fluid flow. No known computer codes have been written which determine the local velocity profiles or temperature profiles resulting during flow or power transients.

There are no known published results on analytical methods or test results concerning the effects of fission gas and coolant interaction occurring as a result of pin failures. Some quasi-steady two-phase flow methods may be applied to the study of resulting flow disruption and cladding temperature excursions, but the accuracies would not be sufficient for design purposes.

Scope of Work — Improved methods must be developed for calculating the effects of plant transients on flow rates, coolant velocity, and temperature profiles as well as cladding temperatures. Reactor transients involving rapid power changes but moderate flow transients should be handled adequately with one-dimensional flow models. Two- and three-dimensional models for predicting coolant velocity and temperature profiles in tight lattice cores will be required for high-rate flow transients. It is anticipated that quasi-steady measurements of the heat transfer coefficient should suffice for predicting cladding temperatures under transient conditions. The importance of accurately predicting coolant velocities and temperatures as well as cladding temperatures will depend on the safety program investigation of required superheat to initiate sodium boiling. The superheat is known to vary with local fluid velocity as well as numerous other variables. Experimental confirmation of the accuracy of methods will likely be

limited because of instrumentation problems; however, confirmation of velocity profiles and cladding temperature measurements is definitely desirable. It is hoped that the accuracy of methods can be checked at rather low fluid bulk temperature levels with extrapolation to higher temperatures in the range of 1600 F+. It is not the purpose of this program to demonstrate in a proof-type test, the flows, flow transient rates, and power levels at which boiling is initiated. Predicted velocity profiles can be confirmed in isothermal tests with water as the fluid. Cladding temperature verification, however, will require electrical heater pin simulation in flowing sodium. All variables of interest will be measured in flowing sodium if practicable, accurate instrumentation can be developed.

As mentioned previously, the lower P/D ratios of interest depend on a demonstration of the desirability of $P/D < 1.1$ based on energy cost and Δk voiding considerations. Transient flow effects will likely set a lower limit (design criterion) on the P/D and should be investigated both analytically and experimentally.

Work is required in the area of two-phase flow of sodium liquid and fission gas. Methods must be developed to calculate the coolant velocity profiles, temperature profiles, and cladding temperature excursions resulting from the release of fission gases from failed cladding. Section 6.2.1.2 covers the effects of a release of relatively large amounts of fission gases accompanying nonvented pin failure. This program is an extension of the investigation into the effects of a release of relatively small amounts of fission gases (expected to be released upon failure of porous plug-vented fuel pins). Tests in flowing water may be sufficient for methods verification. Data on fission gas escape rates through cladding fissures will be forthcoming from the above-mentioned study (section 6.2.1.2). Water tests will provide local coolant maldistribution data (velocities and flow regimes), but measurements to verify fluid temperatures and cladding temperatures will not be made because they require sodium as the coolant for a heated pin bundle (with simulated pins).

Design calculational tools must be developed to determine the thermal transient feedback effects on core reactivity due to changes in thermally induced assembly displacements. Scale-model

mockups to verify the methods will probably not be required although this is undetermined now.

Major Facilities and/or Equipment Required —

Conventional water flow loops are available. The sodium flow loop discussed in section 6.2.2.3 will be used for flow transient tests where cladding temperature measurements are of interest.

Relationship to Current or Planned Programs —

Subtask 6-2.2.1 of the LMFBR Program Plan, Wash-1106, involves transient heat transfer investigations that are very similar to those discussed above for the two-phase flow considerations for sodium liquid and fission gas. The effects of fuel pin failure in the LMFBR Program Plan are considered under Subtask 6-2.5.1.

Schedule and Budget — Funds for research and

development beyond 1973 are specifically for methods upgrading after initial experimental work. The costs in thousands of dollars are as follows:

	Calendar year												Total
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	
Manpower	60	60	40	30	25	15	15	15	15	15	15	15	320
Facilities													
Equipment		30	30										60
Total	60	90	70	30	25	15	15	15	15	15	15	15	380

6.2.2.5. Fuel Assembly Prototype Design and Proof Tests

Objective — After a spacer concept is selected

for the predecessor fuel assembly, a prototype assembly must be designed, constructed, and proof-tested. In proof-testing, we are primarily concerned with demonstrating the structural integrity; design methods and data should be sufficient to guarantee acceptable fretting corrosion and vibration behavior at this point in the developmental program. See section 6.2.2 for R&D pertaining to fuel pin support selection.

The objectives of this test program are as follows:

1. Design a prototype fuel assembly based on the methods developed and the design data generated in programs 6.2.2.2, 6.2.2.3, and 6.2.2.4.
2. Conduct a proof test in an outpile isothermal sodium flow loop to demonstrate the spacer integrity under the quasi-steady hydraulic forces.
3. Conduct thermal shock and thermal bow tests on prototype fuel assemblies.
4. Conduct shaker and impact tests to simulate earthquake-induced internal (inertial) and external loadings.

Tests for reference core fuel assemblies, which will likely be similar to the predecessor type, are discussed under 6.2.2.7, Fuel Pin Reliability and Assembly Proof Tests in the Predecessor Reactor.

Summary of Current Technology — The methods and design data now used to predict the structural adequacy of a fuel assembly are not accurate enough to allow elimination of mechanical and thermal shock proof tests. Simplifications are made in analytical models to reduce the mathematical complexities to a practical engineering level. Design data such as those used in determining fluid pressure forces, heat transfer coefficients, fuel/cladding friction forces and cladding/spacer friction forces are not available, are unreliable or difficult to obtain. It must be shown that the uncertainties involved in the designer's use of methods and data do not compromise the integrity of the fuel assembly within the accuracy of "proof" tests that reasonably simulate the operating environment.

Scope of Work — Sections A through D describe the prototype proof tests.

A. Dynamic Proof Test and Pressure Drop Test

This test will attempt to "proof" the prototype fuel assembly to the extent possible. Such a proof test is beset by numerous problems:

1. Simulating the end-of-life materials properties at temperature.
2. A complete "proofing" would require the simultaneous thermal stresses, hydraulic loadings, and external loadings (simulated supports and adjacent fuel assemblies).
3. At-temperature tests in sodium present numerous instrumentation problems.

A separate test for assembly thermal bow is discussed in section C; the reason for separating these tests is that two different fluids will be used for tests of the effects of hydraulically induced and thermal-gradient-induced stresses.

There is a very low possibility of developing a material that can duplicate the irradiated mechanical properties of the wrapper can and spacers, and even less of a possibility that a structure could be fabricated of such a brittle material for a reasonable cost. Consequently, the best alternative is to instrument an assembly with strain measurement gages, subject the assembly to the loadings expected, and infer stresses from strain measurements. This approach should give acceptable results if the material modulus of elasticity and Poisson's ratio are not greatly affected by irradiation. If irradiation does induce appreciable changes in the modulus of elasticity, perhaps the test temperature or the type of material can be adjusted to compensate. This method of strain measurement to infer stresses is simplified by considering an isothermal test so that temperature measurement at various locations is not required. Strain measurements can also be used to indicate the expected fatigue life of an assembly.

The identical problem of simulating a brittle structure pertains to the case of investigating the effects of large thermal gradients in the assembly. A proof test without structural failure proves little, since plastic deformations can occur undetected. Therefore, an identical approach of strain measurement to infer stresses is considered. Two separate tests can provide the total stress "map" if no part of an assembly undergoes plastic strain (so that superposition of stress is possible). Stresses set up by thermal gradients in a constrained assembly are considered in section C, Thermal Bow Tests.

The test of spacer integrity under the quasi-steady hydraulic forces will provide an opportunity to measure assembly flow distributions and pressure drop. The flowing fluid will be sodium because the need for a facility with sodium has been demonstrated (see 6.2.2.3 and 6.2.2.4). Test temperatures will range up to 1300 F if instrumentation can be developed for these temperatures. The test assembly will be identical to a prototype assembly design except that dummy fuel pins will be used and material thicknesses will not include allowances for corrosion and erosion. It is considered impractical to employ pre-irradiated assemblies, but the feasibility of doing so has not been closely examined.

Flows for this test should range above expected operating flow rates by some appropriate margin. Typical external loads will be applied to the assembly under test. Upper and lower support misalignment will be applied, as well as the maximum loads expected to arise as a result of assembly interactions due to bow induced by thermal gradients.

B. Thermal Shock Tests

After a reactor scram, the core may be subjected to rapid temperature change. Structures must be designed to withstand the thermal shocks encountered without permanent damage or failure of individual components. This test is designed to evaluate the fuel assembly integrity under such conditions.

The problem of thermal shock on the fuel assembly is too complex to evaluate analytically. Consequently, it appears necessary to proof-test individual designs. The problem is difficult in that it is not possible to fully simulate core conditions. Temperature gradients and power profiles cannot be accurately duplicated in such a test. Therefore, results obtained from this test, which simulates the worst case, must be evaluated in the light of these conditions.

The use of an alternate coolant in the thermal shock tests has been considered. However, the use of sodium coolant appears essential to duplicate fully a severe transient condition because sodium, owing to its heat transfer properties, would present a more severe shock than that of either water or gas.

The test facility as presently conceived consists of an apparatus to test a full-scale reference core bundle containing dummy fuel. After

placing the assembly in a test tank, the temperature of the assembly and the surrounding sodium will be raised to 1200 F. A manual cycle start will then dump the 1200 F sodium into the sump and dump tank, and 800 F sodium will be introduced from the fill tank. The transient time (defined as the time required to dump the 1200 F sodium and fill with 800 F sodium) is 0.3 second.

A can failure will be detected remotely as the flow characteristics of the sodium change from the test vessel to the dump tank. The fuel elements will be sealed with inert gas inside containing a known amount of oxygen. Failure in any element will be detected in the oxide indicating and control system. If a failure occurs in the can which does not affect the sodium flow and an element failure that is outside the oxygen-containing portion, then visual inspection will be required after a predetermined number of cycles.

Spacer performance under transient conditions will be of major importance. Any loosening or tightening of the fuel assembly may prove sufficient to destroy the integrity of the fuel assembly.

After the transient tests are completed, the assembly will be destructively tested to determine whether any permanent damage has occurred to individual components. Specifically, tests must be made to determine whether changes have occurred in the material ductility or whether surface cracking has occurred indicating severe thermal stresses. Fuel pin integrity when subjected to thermal shock will be demonstrated in the program discussed under section 6.2.1.4.

C. Fuel Assembly Thermal Bow

The LMFBR Program Plan, WASH-1106, considers the inpile or outpile thermal gradient testing of a prototype fuel assembly to be a necessity. The Program Plan, Subtask 6-2.3.4, Thermal Gradient and Thermal-Transient Proof Testing of Fuel Assemblies, states the following:

"This subtask will definitely require a new [sodium]* loop facility. . . . There is agreement that it is desirable, although opinions differ as to the practicality of the ex-reactor simulation vs. an in-reactor test where prototypal thermal gradients are inherent. . . . Ex-reactor.

* Author's insertions in brackets.

testing, however successful, will not be taken as a complete substitute for in-reactor (FFTF) testing of assemblies or structurally representative portions of them. . . . [For feasibility] the facility must meet the following requirements:

(1) A test section large enough to receive a full-sized fuel assembly (e.g. 217 fuel elements) or, failing that, a model of full assembly length and 1/6 the cross section (i.e., a full-sized triangular module of a hexagonal fuel assembly comprised of six such triangles).

(5) Power supply—electrically heated rods of 0.2 - 0.35 in. diam. simulate fuel elements. The elements must produce a controllable power to a heat flux of at least 1×10^6 Btu/(hr) (sq. ft.).

Assuming. . . [feasibility]. . . , the follow-on work would proceed. . . .

The purpose of the testing program (Phase 4) will be demonstration of: long-term structural endurance; resistance to fretting, galling, and seizing;. . . ."

Our plan for testing the reference plant fuel assembly in the predecessor reactor fortunately eliminates the necessity of our performing the above-mentioned outpile simulation for the reference case (see section 6.2.2.7, Fuel Pin Reliability and Assembly Proof Tests in the Predecessor Reactor). The question of such a test is, of course, pertinent to the case of the prototype assembly. We feel that such a test in sodium should not be considered for the following reasons:

1. The inability to simulate fuel pin character will negate any supposed benefits in indicating fretting corrosion behavior.

2. The prime effect on structural performance is absent, i.e., material property changes due to fast fluence (in particular, material embrittlement).

3. Such a test, regardless of the level of success (as quoted above), is no substitute for in-reactor tests.

4. In the light of 1, 2 and 3, above, the benefits would certainly not justify the cost of the tests.

The predecessor plant will involve a conservative fuel pin and fuel assembly design. Therefore, we have two alternatives regarding thermal gradient tests of fuel assemblies: (1) We can forego this type of test and

design the assembly ultraconservatively, or (2) we can utilize an ex-reactor thermal gradient test without sodium coolant. We have elected to pursue the second alternative since the test benefits should justify the expenditure in this case.

A test facility is envisioned which will utilize:

1. An entire bundle (up to 331 pins) of electrical heaters operating at very low power.
2. A very low flow rate of nitrogen or some inert gas.
3. Strain and local temperature instrumentation.

The low pin powers and low flow rates can combine to give representative axial and transverse thermal gradients. Stresses inferred from strain measurements will circumvent the problem of simulating the brittle characteristics of the actual irradiated assembly as long as no plastic strains are incurred (discussed in section A, Dynamic Proof Test and Pressure Drop).

The test specimens will be unirradiated prototype assemblies instrumented with strain gages and temperature sensors. Pins will be individually electrically heated to provide representative transverse thermal gradients, while the power levels and fluid flow rate will control axial thermal gradients. Power levels to provide the required axial temperature gradients have not yet been determined. Some provision must be made for fluid temperature control external to the can wall. External restraints simulating upper and lower supports and adjacent fuel assemblies will be provided. Transverse thermal gradients of up to 200 F will be imposed with axial gradients up to 400 F.

Information of interest includes strain and temperature measurements in strategic locations (on the supports and the can wall in particular). Strain measurements will be used for determining the expected assembly fatigue life. No transient tests are intended, nor are measurements of corrosion fretting expected. Destructive tests will allow examinations for evidence of failure at points of stress concentration.

D. Assembly Mechanics and Shaker Tests

The integrity of the fuel assembly must be proof-tested under conditions simulating earthquake-induced vibrations. Possible modes of

failure include internal inertial effects and loads induced by external fuel assembly interactions. The types of information required before simulated proof tests can be conducted include:

1. Fuel assembly static and dynamic mechanics information, such as transverse deflection versus load, assembly mass (actual and virtual), natural frequency, and log decrement.

2. From the data in 1 above, an analog simulation of the dynamic behavior of a group of assemblies will be conducted from which interaction loads and accelerations can be calculated for proof testing. The method of analog simulation has yet to be developed.

The proof test will be conducted by applying the accelerations, as calculated above, and the impact loads to a prototype assembly.

The foregoing information concerning data gathering and the proof test itself has some bearing on spacer selection as discussed in section 6.2.2.2. The data on static and dynamic properties of an assembly will, for the sake of continuity, be a responsibility of this program (Assembly Mechanics and Shaker Tests). Some iteration in assembly development may result because of the relationships of this program with those of Fuel Pin Support System Selection, since the support system design is somewhat dependent upon the results of the analog study mentioned in item 2 above. The fuel pin spacers should not fail under the inertial loads due to vibration and impact, and some design criteria pertaining to dynamic loadings should be supplied to the spacer selection program.

Tests on the static and dynamic assembly behavior (step 1 above) will be performed on full-sized assemblies containing dummy fuel (UO_2 only). The test temperatures should range up to approximately 1050 F. Effects of axial thermal gradients will be neglected for test purposes. Static tests can be conducted in nitrogen or some inert gas. Dynamic tests should be conducted in a fluid (perhaps NaK) whose vapor pressure and density approach that of sodium. Dynamic tests will not be conducted with a flowing fluid; effects of axial fluid pressure gradients and fluid Coriolis forces should be negligible.

Proof tests of assemblies have the same drawback as discussed in sections A and C; that is, the problem of dissimilar material properties of unirradiated and irradiated assemblies. This problem may be

circumvented by instrumenting assemblies for dynamic strain measurement; however, high-temperature strain gages have yet to be developed. Strains will be measured at strategic locations on spacers and on the can wall. Stresses will be inferred from these measurements, and post shaker test destructive examination will indicate possible failure in areas of stress concentration. Dynamic proof tests at elevated temperatures will be conducted, perhaps with NaK as the fluid, to determine the dynamic behavior of the assembly. The results of shock load tests conducted on ductile material structures so as to infer acceptability of a brittle structure are not valid if plastic deformations occur during loading.

Major Facilities and/or Equipment Required —

The sodium flow loop described in section 6.2.2.3, Steady-State Thermal/Hydraulics, will be utilized for the proof tests of spacer structural integrity described in section A, Dynamic Proof Test and Pressure Drop.

The thermal shock tests described in section B will require the construction of a special facility, which has been discussed previously.

The test facility for thermal bow effects and proofing is discussed in section C, Thermal Bow Tests. This facility will not be a sodium facility for the reasons previously enumerated.

The mechanics and shaker tests are fairly conventional as planned. Assembly immersion in sodium is not considered necessary.

Relationship to Current or Planned Programs —

A program to investigate the feasibility of utilizing an outpile test facility combining flowing sodium, thermal gradient, and thermal transient effects is considered in Subtask 6-2.3.4. of the LMFBR Program Plan, WASH-1106. An operating facility of this type might provide results of the type desired in sections A, B, and C as discussed above. No section of the Program Plan covers earthquake-induced vibrations and loading as discussed in Section D.

Schedule and Budget — The following budget figures are in thousands of dollars.

	Calendar year				Total
	1971	1972	1973	1974	
Manpower	282	350	227	129	988
Facilities	167	60	66	81	374
Equipment	102	72	69		243
Total	551	482	362	210	1605

6.2.2.6. Reference Fuel Assembly Long-Range Upgrading

Objective — The purpose of this program is the long-range upgrading of the reference fuel assembly between the time that commitments are made for predecessor hardware and reference hardware. This involves:

1. Methods upgrading.
2. Additional prototype design and proof testing if radical changes are instituted.

Major modifications are not anticipated at this stage of development, but it is possible that desirable changes other than improved materials or upgraded operating conditions may be made.

Summary of Current Technology — N. A.

Scope of Work — We cannot expect completely adequate methods development for the fuel assembly design by the date that commitments for predecessor hardware are made. This is especially true in the case of the pin bowing, pin vibration, and corrosion fretting phenomena. Also, some inroads may be forthcoming on methods for calculating spacer effects on fluid mixing and "hot spot" formations. Methods improvements in each of these areas will allow performance upgrading of fuel assemblies.

Utilization of improved materials for spacers or wrapper cans may require additional proof testing, since new materials may have different coefficients of thermal expansion, moduli of elasticity, Poisson ratios, etc.

Major Facilities and/or Equipment Required —

Any required testing in this program should be performed in the facilities described in preceding sections.

Relationship to Current or Planned Programs —

N. A.

Schedule and Budget — The budget values are in thousands of dollars.

	<u>Calendar year</u>
	<u>1975 - 1981</u>
Manpower	100
Facilities	25
Equipment	<u>25</u>
Total	150

6.2.2.7. Reference Fuel Pin Reliability and Assembly Proof Tests in the Predecessor Reactor

Objective — The predecessor reactor will be capable of operating with vented fuel although the initial fuel and several succeeding fuel batches will not contain vented fuel. The plant capability offers an excellent opportunity for reliability testing of vented reference fuel pins. Pin testing in regular fuel assemblies (intended for application in the reference plant) will constitute an in-reactor proof of structural integrity under simultaneous hydraulic-force- and thermal-gradient-induced stresses.

Summary of Current Technology — The British have conducted tests on more than 500 fuel pins using the Dounreay Fast Reactor. As of early 1967, there were no reported failures in the tests. Burnups as high as 7% were achieved. A U.S. program of fast flux testing in EBR-II has been initiated. FFTF development plans call for the testing of more than 550 fuel pins.

Scope of Work — Approximately 2000 reference fuel pins will be tested in the predecessor reactor to demonstrate pin reliability. Since the reliability is strongly affected by spacer and wrapper can design, meaningful reliability data can be obtained only by fuel pin and fuel assembly tests in concert.

Major Facilities and/or Equipment Required —
Fuel pin reliability tests will be conducted in the predecessor reactor.

Relationship to Current or Planned Programs — N. A.

Schedule and Budget — Budget figures are in thousands of dollars.

	<u>Calendar year</u>
	<u>1980 - 1983</u>
Manpower	50
Equipment	<u>150</u>
Total	200

6.2.3. Control Rod

6.2.3.1. General

Objective — The core design R&D program is responsible for developing a control rod (less the drive) and guide tube configuration that contributes to reliable reactivity control. The development of a rod drive and lifetime tests of the complete mechanical assembly are discussed in section 3.2.7.3.

The work required in developing a control rod involves some establishment of design criteria, the generation of design data, and design and testing of prototype components. The design criteria that are not settled at this time are as follows:

1. Must the control rod structure be required to survive, with no damage, one or more incidences of scram snubber failure?
2. Are rollers or wear pads required between the control rod sheath and the guide tube?

The design data of interest include:

1. Behavior of tantalum and/or B₄C under irradiation (swelling, gas generation, heat generation rates).
2. Structural loadings due to scrams, core vibration (earthquake), thermal transients, and quasi-steady thermal stresses.

3. Expected maximum fluence as a function of rod axial location (for determining mechanical properties).

4. Mechanical properties at high fluence and high temperatures for structural materials.

The prototype component tests will primarily be drop tests (simulated scrams) and structural integrity tests. Considerations of control pin vibration and fretting corrosion will be of concern only for the case of B_4C used as the absorber material.

Summary of Current Technology — Candidate absorber materials are B_4C , tantalum, and hafnium. Considerable experience has been obtained in the use of boron carbide, but it has a high gas generation rate and is not compatible with stainless steel at high temperatures. Existing data indicate that tantalum does not generate gases, swell, or have an incompatibility problem with stainless steel or sodium.

For large cores, control assemblies usually are spread over the entire core to avoid local reactivity anomalies. Most designs are intended for both regulating and shim control. The problem of wear and galling of the guide tube and control rod sheath requires investigation. Types of bearing surfaces, wear pads, and the problems incurred by loose fits require evaluation.

Present methods should be adequate for predicting guide tube deformation under thermal gradients and adjacent fuel assembly bow. The effects of guide tube and sheath friction on scram displacement versus time, however, are not amenable to analytical treatment other than empirical. Structural loadings due to most scram or core vibration situations can be determined adequately by present methods.

The number of uncertainties involved in developing a new mechanical configuration of control rod for the LMFBR environment makes imperative the exhaustive proof and lifetime testing of a prototype assembly before design acceptance.

Scope of Work — Two absorber materials, tantalum and B_4C , will be evaluated. Most of the work on B_4C will be to confirm present data on gas generation rates. All test data accumulation will be the responsibility of the fuels and materials program.

Information concerning the effects of temperature and fluence on the candidate structural materials will also be furnished by fuels and materials personnel.

The design criteria concerning the necessity of a control rod to withstand one or more incidences of snubber failure will be forthcoming from investigations of the possibility of detecting imminent snubber failure (as determined in the drive assembly development program). The criteria for the need for control rod sheath bear pads or bearings will result from analytical investigations of guide tube and sheath deformations due to thermal gradients and interactions (static and dynamic) with adjacent fuel assemblies.

Hardware development is likely to be an iterative process if the structural requirements to guarantee rod integrity sacrifice scram performance (due to excess fluid drag caused by support or spacer components). Model development for scram simulation is anticipated, but model accuracies will suffer where guide tube and sheath interference produces significant friction. Thus, scram tests on a full-scale model must be performed. After suitable guide tube and control rod hardware is developed, the combined control rod and drive assembly lifetime tests will be conducted under the components program (section 3.2.7.3).

Some uncertainties pertain to the use of grid spacers for the absorber pins. Other spacer concepts will be considered for use in the control rod. Types to be considered depend on whether tantalum or boron carbide is selected as the absorber material. The overriding spacer consideration for tantalum pins is the structural integrity of the spacers themselves; as to B_4C , the effects of the type of spacer on pin vibration and fretting corrosion must be considered. Pin vibration may cause cladding failure for cladding subjected to a large static stress, as is the case with B_4C sealed pins.

6.2.3.2. Absorber Pin

Objective — The two prime candidates for absorber material in the control rod are boron carbide and tantalum. Studies must be conducted to furnish design data on which a material

selection can be made. After material selection, prototype absorber pins will be irradiated to determine the adequacy of the design.

Summary of Current Technology — The problems involved in the design of absorber pins where the absorber material generates gases and is subject to irradiation damage are very similar to those of nonvented fuel pins. With a knowledge of gas generation rates, swelling rates, heat generation rates, and compatibility of absorber materials with cladding and sodium, pins can be designed and qualified for reactor use with little developmental effort.

Scope of Work — The fuels and materials program will obtain the pertinent design information for boron carbide and tantalum. General information desired includes:

1. Cladding compatibility.
2. Sodium Compatibility.
3. Amount of gas release (B_4C only).
4. Extent of irradiation damage (B_4C only).
5. Heat generation rates.

Prototype absorber pins will be designed, built, and proof-tested in pile. If the B_4C material candidate is included in the pin tests, the tests will be conducted in sodium at temperatures slightly in excess of 1300 F. Inpile times will be sufficient to demonstrate proof of the design. Outpile tests will be conducted to demonstrate the cladding integrity with pins subjected to thermal shock.

Major Facilities and/or Equipment Required — Irradiations may be conducted in EBR-II, Fermi, or FFTF. Since the absorber material affects the choice of absorber pin spacer selection, early irradiations in EBR-II or Fermi will probably be required.

Relationship to Current or Planned Programs — Task 6-3.2 of the LMFBR Program Plan deals with the evaluation and selection of absorber materials for LMFBR application.

Schedule and Budget — The following figures are in thousands of dollars.

	<u>Calendar year</u>			<u>Total</u>
	<u>1969</u>	<u>1970</u>	<u>1971</u>	
Manpower	30	30	10	70
Facilities		10	10	20
Equipment	<u>10</u>	<u> </u>	<u> </u>	<u>10</u>
Total	40	40	20	100

6.2.3.3. Control Rod Design Data and Criteria

Objective — Certain design data are required for an efficient control rod design and development process. Some of the data may be obtained by analysis requiring analytical model development; this is the situation anticipated for load determination and guide tube and control rod deformation due to thermal gradients. Tests will furnish design data on fluid drag coefficients, material mechanical properties and the like. Information concerning guide tube and shroud frictional forces and wear are of interest, but meaningful data of this type are very difficult to obtain; the design effort will be aimed at circumventing or alleviating the effects of friction and wear.

Design data for maximum snubber accelerations and the number of load cycles likely to be applied by the control rod drive will be forthcoming from sections 3.2.7 and 2.2.3. The fuels and materials program will supply design data on irradiated and non-irradiated mechanical properties of structural materials.

Summary of Current Technology — Analytical methods for determining control rod loadings are adequate except for flow and coulomb friction forces occurring during scrams. Fluid friction drag coefficients can be determined empirically, but difficulties arise in attacking the problem of coulomb friction since all pertinent variables are not always known or measurable.

Methods for determining guide tube and control rod deformation (bow) due to thermal gradients are sufficient for engineering purposes. Interactions of numerous fuel assemblies and control rod guide tubes are possible to predict but comprise a very complex numerical analysis problem.

Little is known of the phenomenon of crud buildup in cold zones of the core (such as the inlet) and consequent effects on control rod operation.

Scope of Work — An analytical model will be used to determine guide tube deformation (short-term and creep) under the influence of thermal gradients and the actions of adjacent assemblies. Calculated control rod bow due to thermal gradients will then enable a basis for determining the guide tube and sheath gap or the requirement for bearing surfaces or devices. Analytical model developments for the fuel assembly wrapper can and the complete fuel assembly should be applicable to the deformations of interest here.

Full-scale tests on control rod components will be made in flowing water to determine drag coefficients. Nonsupport-type grids, upper fittings, and pin partial bundles will be immersed in flowing water and tested under quasi-steady conditions. Pin partial bundles will be tested in flow channels with walls simulating (geometrically) the missing, next peripheral row of pins. Inaccuracies arise in attempting to obtain the sodium drag coefficients in water tests because dynamic similitude cannot be maintained in turbulent (pin channel) and laminar (boundary layer and wake) regions simultaneously. The inaccuracies, however, will be overshadowed by inaccuracies in a transient flow model for predicting control rod scram displacement versus time. Prototype design scram tests in sodium will be conducted in the final development stages (see section 3.2.7.3).

Dynamic loadings, such as those experienced under earthquake conditions, will be obtained through an analog simulation of fuel assemblies and guide tubes. Loadings can be transformed into guide tube deflections, which may have some bearing on the minimum guide tube and sheath clearances.

Major Facilities and/or Equipment Required — The water loop tests for experimental drag coefficients will be conducted in a conventional test facility, many of which are available.

Relationship to Current or Planned Programs — Task 6-3.2, Control and Safety Assembly Design Technology, of the

LMFBR Program Plan, WASH-1106, deals with design criteria and design data (in particular, Subtask 6-3.2.1).

Schedule and Budget — The following figures are in thousands of dollars.

	<u>Calendar year</u>			
	<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>Total</u>
Manpower	60	60	30	150
Facilities	10	10	10	30
Equipment	10	—	—	10
Total	80	70	40	190

6.2.3.4. Control Rod Analytical Model Development

Objective — The control rod and fuel assembly have similar requirements for model development. The models developed for fuel pin vibration, assembly bow, thermal stresses, and the like should apply to the control rod design with little modification. An effort will be made to ensure that models developed for the fuel assembly will be applicable to the control rod if practical.

Models to predict control rod local stresses under the transverse and longitudinal accelerations likely to occur as a result of scrams must be developed. A model to predict scram displacement versus time is desirable.

Summary of Current Technology — The present technology for pin vibrations induced by in-line fluid flow, fretting corrosion, and other spacer-associated phenomena is discussed in section 6.2.2.2, Fuel Pin Support System Selection.

Technology is available for determining guide tube and control rod deformations under the action of thermal gradients and external loads except where friction forces are involved. Models for determining stresses and deformations of complex (such as hexagonal) geometries usually require the development of separate models because of complexities in providing model versatility.

Scope of Work — Analytical models must be developed for determining:

1. Guide tube deformation (short-term and creep) under the action of thermal gradients and transverse loadings (due to interactions with adjacent fuel assemblies).
2. Control rod deformations due to thermal gradients (short-term and creep).
3. Control rod and guide tube dynamic loadings during scrams or earthquake-induced core vibration.
4. Control rod scram displacement versus time.

The methods and models developed for the fuel assembly will be utilized for control rod application where practical. The different absorber pin requirements concerning the method of support, however, will very likely severely limit application of the fuel assembly model to control rod problems. Experimental verification of models to predict control rod bow under the driving forces due to thermal gradients may be required if frictional forces internal to the sheath play a significant part in determining rod stiffness (static). Full-sized mockups will be required in these tests. Of course, the need for such tests cannot be determined at this time; further studies and evaluations of model results, taking a conservative approach to handling frictional uncertainties, will determine the need for tests.

Analytical model development for scram displacement versus time should require no additional testing other than the drag coefficient measurements discussed in section 6.2.3.3.

Major Facilities and/or Equipment Required — Experimental verification of model predictions for guide tube and control rod deformation due to thermal gradients will require use of the facility described in section 6.2.2.5, Fuel Assembly Prototype Design and Proof Tests, part C. Capital costs for the construction of that facility are not covered in this section.

Schedule and Budget — Facility and equipment costs are contingent upon needs as outlined below. Costs are in thousands of dollars.

	<u>Calendar year</u>			<u>Total</u>
	<u>1969</u>	<u>1970</u>	<u>1971</u>	
Manpower	60	60	30	150
Facilities		10	10	20
Equipment		10		10
Total	60	80	40	180

6.2.3.5. Prototype Control Rod Design and Testing

Objective — This program is concerned with the design of control rod components and the assembly for construction and test evaluations. Tests will be conducted to determine:

1. Structural integrity of individual components.
2. Structural integrity of assembled control rod and guide tube under static and dynamic loadings.
3. Amount of guide tube and control rod deformation under action of thermal gradients and simulated adjacent fuel assembly loads.
4. Displacement versus time measurements in water for simulated scrams.

Qualification of components and the assembled control rod prototype will be completed before combined control rod and drive tests are carried out in the components program (see section 3.2.7.3). The type and extent of testing will vary somewhat depending on the selection of an absorber material; the tests described herein are for tantalum absorber material. Consequently, considerations of pin vibration and cladding fretting corrosion are minor.

Summary of Current Technology — The methods and the design data now used to predict the structural adequacy of a fuel assembly are not accurate enough to allow elimination of mechanical and thermal shock proof tests. Simplifications are made in analytical models to reduce the mathematical complexities to a practical engineering level, thereby introducing a measure of inaccuracy. Design data for accelerations and frictional forces are not available, are unreliable or difficult to obtain. It must be shown that uncertainties involved in the designer's

use of methods and data do not compromise the integrity of the fuel assembly within the accuracy of "proof" tests that reasonably simulate the operating environment.

Scope of Work — The guide tube, sheath, spacers and fittings will be constructed and tested separately for structural integrity. Long-term creep tests are not anticipated. Thermal shock tests for components will be conducted in a sodium facility. The possibility of obtaining meaningful results from thermal shock tests on a representative short length of a control rod will be evaluated. If such a test is not acceptable, then thermal shock testing of a complete assembly will be considered. Control rod improvements will be tested for rigidity and load carrying capacity for comparison with calculated values. Here we are confronted with the same problem as discussed in the section on fuel assembly prototype proof tests, namely, how do we account for irradiated structural material properties in tests on unirradiated specimens. The logical answer is to equip test specimens with strain measurement instrumentation and infer stress levels from the values of strain. Areas of stress concentration can be examined after tests for evidence of failure. This approach may require development of special, high-temperature strain instrumentation. Specimens in component and assembled control rod tests for structural integrity will be manufactured to nominal dimensions less corrosion and erosion tolerances where applicable.

As mentioned in section 6.2.2.5, part A, it is considered impractical to test pre-irradiated structures as large as a fuel assembly or control rod, although this approach might merit further consideration. The method that we have proposed for structural tests is not without serious drawbacks, however. Plastic strains occurring under static or dynamic loadings in any part of an unirradiated test assembly will introduce errors in predicting the performance of embrittled assemblies by strain measurement. This problem may be partially alleviated by applying reduced loads, but this introduces further uncertainties in the validity of the "proof" tests.

Assembled control rods will be subjected to simulated scram loads. This loading may include that occurring under snubber failure if the rod must be designed to withstand an incidence

of snubber failure (this design criterion will require investigation as mentioned previously in section 6.3.2.1). Other dynamic loadings of interest include:

1. Cyclic loads imposed by drive motor positioning, and the number of cycles likely to induce fatigue failure.
2. Accelerations and external loads simulating earthquake incidences.

An assembled control rod will be tested in an outpile thermal gradient facility as described in section 6.2.2.5, Fuel Assembly Prototype Design and Proof Tests, part C. Likewise, the guide tube should be tested for deformation under simulated thermal gradients. Guide tube tests could be conducted in a separate bench test facility where loads simulating the action of adjacent assemblies can be imposed. Information of interest includes local strains and total deformations to indicate possible control rod and guide tube interference.

Dynamic load tests of a guide tube should be conducted to indicate its performance while subjected to fuel assembly impact loads caused by earthquake incidence. The primary information of interest is the resulting local strains sufficient to indicate failure of a brittle material. Transverse displacement (bow) measurements are also of interest.

Drop tests of an entire control rod in water will be performed. The measurement of interest is the rod displacement versus time. Exact simulation of sodium effects are not possible. Water tests, however, should provide adequate information.

Major Facilities and/or Equipment Required —

Static loading of control rod structures will require conventional testing facilities, of which there are many. Thermal gradient test facilities, thermal shock facilities, and assembly shaker facilities are described in the fuel assembly R&D program; these facilities should be capable of accommodating control rod or guide tube prototypes. The facility for drop tests of a control rod in water should require a conventional water flow loop with minor modifications.

Relationship to Current or Planned Programs —

Task 6-3.2 of the LMFBR Program Plan, WASH-1106, is concerned with

the development of control rod concepts for LMFBR application. This program primarily concerns the development of the reference design; major changes in the concept are not anticipated.

Schedule and Budget — The costs in thousands of dollars are as follows:

	Calendar year						Total
	1971	1972	1973	1974	1975	1976	
Manpower	90	60	120	120	60	30	480
Facilities		30	80	80	40	30	260
Equipment	30	30					60
Total	120	120	200	200	100	60	800

6.2.4. Radial Blanket Assembly Development

6.2.4.1. General

Objective — The objective of this program is to develop a radial blanket pin and assembly for use in both the predecessor and the reference reactor. The predecessor blanket pin will be nonvented and, as such, will merely require an extension of thermal reactor fuel pin technology to account for differences in environment. The vented radial blanket pin design for the reference plant use will draw heavily on the vented fuel pin development program for design data and methods. This is also the case for the blanket assemblies, although the pins and assemblies face worse thermal gradient conditions than do the fuel pins and fuel assemblies. This program will (1) set blanket pin and assembly design criteria, (2) guide the fuels and materials program in materials evaluations and inpile pin tests, and (3) set the prototype assembly design and "proof" test the design to the extent possible. After operation of the predecessor plant is begun, reference radial blanket assembly inpile tests will be conducted.

Two special problems requiring attention in the blanket pin and assembly program are (1) pin bow resulting from the very steep thermal gradient experienced by the pin in closest proximity to the core, and (2) assembly bow induced by the very large thermal

gradients experienced by some assemblies (spacer or pin damage would be of prime concern here rather than reactivity insertion due to assembly bow).

Summary of Current Technology — Many of the radial blanket pins will contain appreciable plutonium at some time near the end of the assembly's service lifetime. In effect, then, some pins become low power fuel pins and must be considered as such. The design criteria concerning fuel pin failure and failure effects on sodium voiding propagation apply to some extent to the blanket pin design (see section 6.2.1.1.). Fuel properties of UO_2 can be successfully applied to the design of blanket pins, which can be designed and operated conservatively with little economic penalty.

No information is available for inpile tests of porous plug vent devices. Performance data for the diving bell concept, however, have been obtained from in-reactor tests. Effects of long-term sodium logging in UO_2 have not been determined.

Analytical techniques for predicting steady state thermal performance of blanket pins are considered sufficient. Design data, however, are lacking in some areas. Models for determining the behavior of blanket pins subjected to thermal transients require development.

The fuel assembly state-of-the-art discussed in section 6.2.2.1. applies to the blanket assemblies as well.

Scope of Work — Radial blanket pin configurations of interest include nonvented, solid-pellet UO_2 for the predecessor plant, and vented, vibratory compacted UO_2 for the reference plant.

A blanket assembly employing a solid wrapper can is of interest for both the predecessor and the reference design. The type of spacer for the blanket pin is open.

The orifice development program under section 6.2.2.2 includes the investigations required to produce an acceptable blanket throttling device.

No radial blanket material or pin experimental work is funded under this program except for the outpile tests of pin thermal shock. This program is responsible for, and provides funds for all required blanket assembly tests.

No inpile tests of the reference radial blanket assembly in the predecessor reactor are anticipated. The structural and operating condition differences between the predecessor and reference assemblies should not be significant. Predecessor blanket assembly operation should provide any information desired concerning the structural integrity of the assembly.

6.2.4.2. Radial Blanket Pin

Objective — The development program for the radial blanket pin has as its objective the design and qualification of prototype predecessor and reference blanket pins. Blanket pins, in which an appreciable amount of plutonium is built up and a corresponding amount of power is generated during core operation, involve a slightly different problem. The melting temperature of a mixed oxide decreases with plutonium concentration (for the concentrations of interest) and therefore sets a limit on the power rating of the pin. Existing data on the melting point are believed to be accurate enough for design purposes. Information on the fast flux swelling rates of blanket material is of interest, but again, existing data should suffice for design purposes.

Analytical models have been developed for predicting the bow of UO_2 pins in pellet fuel form.

Scope of Work — Inpile tests of prototype blanket pins must be designed by the core design and fuels and materials programs. Environment and exposure times will be designated. Information of interest includes:

1. Cladding deformations due to swelling.
2. Evidence of fuel melting or slumping.
3. Evidence of sodium ingress and its consequences (for vented pins only).

Tests will be conducted in a fast flux environment. Steady-state power tests are of interest, and the need for transient tests will be determined later. Short-term tests may be performed with initial concentrations of plutonium set at predetermined values.

Outpile tests will be conducted to verify blanket pin bow values predicted by analysis or to set empirical constants.

Methods to produce the cladding temperature gradients of interest have not been determined. It may be possible to use a regular blanket pin with a portion of fertile material removed so that an eccentrically positioned electrical heater may be inserted. Measurements of interest include the transverse pin displacement (or bow) as a function of transverse thermal gradient.

An outpile test to "proof" the cladding material under conditions of rapid changes in coolant temperature is anticipated. Specimens will be alternately subjected to liquid sodium at temperatures as high as 1400 F and as low as 700 F (approximately).

Major Facilities and/or Equipment Required —

Inpile pin tests will be conducted in EBR-II.

Blanket pin bow tests could be conducted in a bench test facility. A facility for thermal shock tests is described in section 6.2.2.5).

Relationship to Current or Planned Programs —

Task 6-3.1 of the LMFBR Program Plan, WASH-1106, deals with the development of blanket pins and assemblies. The pin design being considered in that task, however, is a sealed pin concept.

Schedule and Budget — The costs in thousands of dollars are given below.

	<u>Calendar year</u>		<u>Total</u>
	<u>1971-1972</u>	<u>1981-1982</u>	
Manpower	30	30	60
Facilities	5	5	10
Equipment	<u>5</u>	<u>5</u>	<u>10</u>
Total	40	40	80

6.2.4.3. Blanket Pin Support System Development

Objective — This program is responsible for evaluating the types and geometries of spacers and selecting a type for application to the blanket assembly. Design data generated in the program set up to evaluate types of fuel assembly spacers will be utilized

to the fullest extent, as will the methods developed in that program. This program will follow the same plan as outlined in the section 6.2.2.2, Fuel Pin Support System Selection; rather than to repeat the objectives here, we will point out the major differences expected in this program. The major problem areas in selecting a blanket spacer are as follows:

1. The structural integrity of the spacer under the combined action of thermal stresses, pin bow, and assembly bow (bow due to thermal gradients).
2. Fabrication tolerances due to the very tight pin lattice desired.
3. Fretting corrosion of the pins subjected to the largest thermal gradients.

The overriding problem in developing the wrapper can is expected to be the thermal stresses set up by combined external constraints and thermal gradients.

Summary of Current Technology — Section 6.2.2.2, Fuel Pin Support System Selection, gives the state-of-the-art for criteria, design data, and methods pertaining to the design of the rod bundle assembly.

Scope of Work — Figure 6-1 indicates steps in the spacer selection process. As mentioned previously in section 6.2.2.2, it is not intended to represent the design process in its entirety, nor is it intended to be interpreted in a strict chronological sense. Of course, the step 4 is not applicable here, since one type of spacer is expected to be sufficient for both the predecessor and the reference assembly.

Step 2 in Figure 6-1 has to do with establishment of assembly design criteria. The criteria for the blanket assembly are slightly different from those of the fuel assembly. In particular, the blanket assembly will have different criteria regarding:

1. Cladding "hot spot" formation.
2. Minimum pin pitch-to-diameter ratio.
3. Pin vibration.
4. Pin bow.

The very small blanket pin ligaments (pin pitch less the pin diameter) will accentuate the difficulties involved in guaranteeing proper individual fuel pin constraint due to tolerance buildup in the assembly. Evaluation of the spacer in this problem area will be accomplished by model fabrication and development (step 3).

Structural requirements (load capacity) of the blanket assembly will likely be greater than those of the fuel assembly since the thermal gradients are larger in the blanket. Pin sizes and pin bow may combine to produce large forces on parts of a spacer structure or large forces between pins where pins and spacers are combined (wire wound and helical fin). Large loads between pins or between pins and spacers, especially in the case of non-compliant spacers, can result in serious fretting corrosion problems.

The types of spacers will be evaluated when possible from the analyses provided by methods development; this includes satisfaction of the criteria, as well as the performance level. Fabrication characteristics will be evaluated by engineering judgement, model construction, and fabrication methods development. The test effort will be primarily directed toward analytical methods verification, if possible. Some "brute force" cut and try approaches are required where phenomena do not lend themselves (with present technology) to empirical or analytical approaches, such as the corrosion fretting phenomenon.

Major Facilities and/or Equipment Required —

Test facilities required for this program are identical to those discussed under section 6.2.2.2, Fuel Pin Support System Selection.

Relationship to Current or Planned Programs —

Task 6-3.1 of the LMFBR Program Plan, WASH-1106, pertains to the technology development and design of blanket assemblies.

Schedule and Budget — The costs in thousands of dollars are as follows:

	<u>Calendar year</u>			<u>Total</u>
	<u>1971</u>	<u>1972</u>	<u>1973</u>	
Manpower	20	20	20	60
Facilities		10	10	20
Equipment	<u>20</u>	<u>—</u>	<u>—</u>	<u>20</u>
Total	40	30	30	100

6.2.4.4. Radial Blanket Assembly Prototype Design and Outpile Proof Tests

Objective — The objective of this program is to design, construct, and "proof" test prototype blanket assemblies to the extent possible. These tests will follow closely those planned for the prototype fuel assembly; see section 6.2.2.5, Fuel Assembly Prototype Design and Proof Test. However, for the blanket assembly, the emphasis will differ slightly. For example, in a dynamic proof test and pressure drop test, less concern is placed on structural integrity under the influence of quasi-steady hydraulic forces than on pressure drop and assembly flow distribution. Also, the shaker tests for blanket assemblies will not be as detailed as those for the fuel assembly. Thermal shock and thermal bow tests will closely parallel those tests for the fuel assembly.

Summary of Current Technology — The methods and design data used now and anticipated for the near future in predicting the structural adequacy of a complex structure such as the radial blanket assembly are not accurate enough to allow elimination of thermal shock and mechanical proof tests. Simplifications are made in the analytical models to reduce the mathematical complexities to a practical engineering level. Also, design data of interest in many cases are not available, are not accurate or are difficult to obtain. The uncertainties involved, then, in the designer's use of methods and design data must be shown to give results that do not result in compromising the fuel assembly's integrity within the accuracy of "proof" tests that reasonably simulate the operating environment.

Scope of Work — Proof tests will be conducted as follows:

1. Flow tests for assembly flow distribution and pressure drop.
2. Thermal shock tests.
3. Thermal bow tests.
4. Assembly mechanics and shaker tests

These tests are similar to the "proof" tests planned for the fuel assembly. Therefore, we shall point out here any major changes in test objectives rather than repeat those discussed in section 6.2.2.5 with the changes made for blanket assembly tests.

A dynamic proof test is planned for the fuel assembly and is discussed in part A, section 6.2.2.5. There is little concern over the structural integrity of the radial blanket spacer under the action of quasi-steady hydraulic forces. Rather, this test affords an opportunity to measure flow distributions and coolant temperature distributions at an assembly outlet. Electrical heaters will be used to simulate peripheral assembly pins, and peripheral channel flows and temperatures will be measured. These measurements are important because they set test conditions for thermal bow measurements of the assembly. It is anticipated that a spacer will be selected with primary emphasis on interchannel fluid mixing effects to alleviate pin bow and assembly bow due to thermal gradients. This test will measure spacer performance in this area. Tests will be conducted in a sodium loop which is described in section 6.2.2.3, Steady-State Thermal/Hydraulics.

Thermal shock tests and thermal bow tests will be very similar to those described for the fuel assembly, parts B and C of section 6.2.2.5. The large transverse thermal gradients experienced in a blanket assembly may be difficult to duplicate in a simulated test. The addition of devices in the assembly may have to be considered to limit transverse heat and fluid flow for simulation purposes. These devices must not appreciably affect the assembly's mechanical characteristics, such as rigidity, internal friction, and the like.

The mechanics and the shaker tests of the blanket assembly may be considerably reduced in scope from those of the corresponding test for the fuel assembly (part D, section 6.2.2.5). With identical wrapper can designs, it may be possible to eliminate

these tests entirely and to infer performance of the blanket assembly from tests of fuel assemblies.

Major Facilities and/or Equipment Required —

Test facilities for these tests are discussed in section 6.2.2.5.

Relationship to Current or Planned Programs —

Task 6-3.1 of the LMFBR Program Plan, WASH-1106, pertains to the technology development and design of blanket assemblies. No specific proof tests are described, however.

Schedule and Budget — Costs in thousands of

dollars are given below.

	<u>Calendar year</u>			
	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>Total</u>
Manpower	30	60	10	100
Facilities		80	30	110
Equipment	<u>30</u>	<u>60</u>	<u>—</u>	<u>90</u>
Total	60	200	40	300

6.2.5. Core System R&D

6.2.5.1. General

Objective — The objective of this program is to conduct investigations into alternate core and core component configurations not presently considered or allowed for in the programs set up in sections 6.2.1 through 6.2.4.

Summary of Current Technology — Present

design practices in providing fuel assembly holddown include mechanical connections on the lower fitting and utilization of a holddown structure at the upper fitting. Hydraulic holddown has been considered in some designs, along with an additional positive mechanical restraint.

Fuel assembly and support displacements due to thermal expansion must be provided for without producing an increase in core reactivity with temperature. The Fermi design utilizes spacer pads between assemblies which are placed so that an increase in reactor

temperature produces a net outward fuel movement that contributes to a negative reactivity temperature coefficient.

The fuel assemblies have wrapper cans which limit the extent or the rate of sodium void propagation in the assembly. Information is not now available from which to determine the entire set of wrapper can criteria, including the required pressure it must withstand and the extent to which it may be perforated (to reduce wrapper can loadings during normal operation).

The primary control rod assembly arrangement presently used in reactor design involves hanging the assembly from the drive unit at the top of the reactor vessel. Misalignment of the core with the drive locations might result in binding a control rod. Development of a safety rod not requiring a core-to-reactor vessel mechanical connection would alleviate this problem.

Scope of Work — Evaluations will be conducted to determine the feasibility and the performance improvements of alternate core support, holddown, and assembly designs. Promising design innovations will be examined and steps will be taken to redirect individual programs accordingly, depending on schedule and budget considerations. The designs or concepts now planned for further study include hydraulic holddown, the fixed can concept, the perforated fuel assembly wrapper can, and the independent hydraulically operated safety control rod.

6.2.5.2. Hydraulic Holddown

Objective — The feasibility of using a hydraulic holddown technique in the fuel assembly will be investigated to determine primarily the following:

1. What possible situations arise during normal or abnormal plant operation that could result in a direction reversal of the fluid forces on an assembly?
2. Evaluate the feasibility with regard to the safety consideration in 1 above.
3. Once feasibility is shown based on safety considerations, evaluate the results in terms of reactor design, operation, and energy cost by employing hydraulic holddown.

In the hydraulic holddown technique referred to here, the fuel assembly lower fitting is extended through the high-pressure inlet plenum to a plenum of low fluid pressure. The geometry of the inlet nozzle then, may be arranged so that the net hydraulic force on the assembly is downward. The change in direction of the fluid through the fuel assembly can produce a downward fluid impulse on the assembly.

Summary of Current Technology — The hydraulic holddown technique described above can result in a net downward fluid force on an assembly during steady-state operation. All types of normal and abnormal reactor operation incidences tending to compromise the holddown have apparently not been well defined. The prime advantage of the hydraulic holddown is the elimination of the upper hold-down structure and its associated positioning mechanism with some attendant reduction in the diameter of the core vessel.

Scope of Work — Application of the hydraulic holddown concept to the LMFBR requires an investigation of several problem areas, including safety considerations where a holddown malfunction may occur. The holddown technique may be compromised under conditions where:

1. A sudden high-pressure pulse occurs in the low-pressure plenum.
2. Core vibrations of earthquake intensity lift an assembly and cause removal of the lower fitting from the low-pressure plenum.
3. Fuel assembly interaction causes enough ratcheting of an entire assembly to remove the lower fitting from the low-pressure plenum after a number of reactor power cycles.
4. Improper positioning of a fuel assembly occurs.

The foregoing situations and others not now apparent will be examined to determine the probability of occurrence; methods to minimize the opportunity for compromise of the holddown will be examined.

Other areas requiring investigation include:

1. Possible improvement of the problem of reactivity insertion accompanying thermal bow of fuel assemblies, since the upper hold-down will be eliminated.

2. Provision for fuel meltdown in an assembly without jeopardizing any part of adjacent assembly fittings.

3. Inlet plenum flow distribution.

4. Effects of seal failure.

Regular operating condition requirements, as well as abnormal and safety considerations, demand investigation to establish design criteria. The ability to meet these criteria will determine the feasibility of hydraulic holddown. Accompanying the feasibility investigation should be a thorough examination of the operating and economic incentives and penalties. It is not anticipated that any hardware development and testing will be required in evaluating this concept.

Major Facilities and/or Equipment Required —

None.

Relationship to Current or Planned Programs —

No other known evaluations of hydraulic holddown are planned for LMFBR application.

Schedule and Budget — Costs in thousands of

dollars are given below.

	<u>Calendar year</u>		
	<u>1969</u>	<u>1970</u>	<u>Total</u>
Manpower	50	50	100

6.2.5.3. Alternative Fuel Assembly Wrapper
Can Concepts

Objective — Alternative fuel assembly wrapper can concepts should be evaluated for reference LMFBR application. The concept of a perforated wrapper can should be further investigated as should the fixed can concept. Successful application of either concept will result in a reduction of core steel volume fraction since thinner cans may result. A perforated can allows increased fuel assembly bypass flow thereby reducing the pressure differential across the can wall (this reduces the can wall stresses neglecting effects of thermal

stresses). A fixed can concept involves a fixed honeycomb structure into which the canless fuel assemblies are placed (one assembly per honeycomb cell). This eliminates the fluid pressure differential across the can wall for all practical purposes, thereby possibly allowing the use of a very thin can wall.

The objective of this study is to evaluate these design alternatives. Evaluations of the fixed can concept will require:

1. Analytical model development to determine the thermal stresses in the honeycomb structure.
2. Establishment of design criteria concerning sodium void formation (pressures) and propagation.
3. Proof of the concept's practicality by fabricating and testing hardware.

Evaluation of a perforated can concept will involve similar requirements, but methods to alleviate the severe thermal gradients across the can wall are needed. Analytical methods should be adequate for design purposes in this case except for determining fuel assembly bypass flow areas and flow distributions.

Summary of Current Technology — As is the case with the reference fuel assembly's solid wrapper can, the criteria involving sodium void propagation must be determined for the fixed can and the perforated can. Questions must be answered concerning internal transient can pressure requirements, extent or types of allowable can failure with the presence of sodium vapor in an assembly, allowable rates of sodium void propagation, and other matters.

Analytical models for complex structures such as the honeycomb-type fixed can must be individually developed. Analytical methods for thermoelastic stress analysis are sufficient.

Scope of Work — Much of the effort for determining wrapper can criteria regarding void propagation will be performed under the safety program. These criteria are important for wrapper can design whether or not the present fuel assembly's solid can wall is used in the reference plant.

A model will be developed for determining thermal stresses in a honeycomb structure subjected to transverse and longitudinal thermal gradients. This model will then be used to design a fixed can arrangement. This concept will be evaluated in the light of the design criteria, operating considerations (especially fuel handling), and the effects on plant economics. Some hardware development and testing will be required if the concept progresses beyond the feasibility investigation stage. Testing may be conducted on full- or partial-scale mockups. Complete test program details are not set at this time.

The feasibility of using a perforated wrapper can for the reference assembly depends on wrapper-can requirements concerning the rate of sodium void propagation in the assembly and the development of mechanical methods to reduce thermal stresses in the can wall due to bypass flow. Considerable effort must be expended also in determining bypass flow areas, which are likely to vary greatly owing to thermal gradient-induced bow of assemblies.

Major Facilities and/or Equipment Required —

At this time, test facility requirements are open, but it is anticipated that conventional bench test apparatus should be sufficient.

Relationship to Current or Planned Programs —

No known development efforts are in progress or planned for the fixed core concept. Task 6-2.7 of the LMFBF Program Plan, WASH-1106, apparently plans for evaluations of perforated wrapper cans.

Schedule and Budget — The costs in thousands of dollars are as follows:

	<u>Calendar year</u>	
	<u>1969 - 1971</u>	<u>Total</u>
Manpower, Facilities, and Equipment	200	200

6.2.5.4. Hydraulically Operated Control Rod

Objective — The large sodium-cooled fast breeder reactor (LMFBR) presents new safety shutdown problems to the

reactor designer. The safety system may be limited in its effect by the rapidity of nuclear events or by minor incidents within the core immobilizing some or all of the rods in the conventional safety system. A shutdown system that has no operating connection to the top reactor closure plug might offer some safety improvements over the conventional concept.

A study showing the need for such an independent backup safety system has been performed, and the independent safety rod concept is offered as a possible solution. This system is hydraulically (sodium) operated and is integral with the fuel element assembly.

As a first step in a systematic program, the shutdown problem must be thoroughly investigated. The prime objective will be to establish the operating criteria and the functional requirements for the shutdown system.

The next step in the program is the conceptual study and design phase in which a system description is prepared, along with an analysis, to develop the functional adequacy and performance characteristics. The detailed design of new components will follow, and a development and testing program will be formulated. Preliminary component and prototype testing will be performed in a water environment, followed by full assembly testing in hot sodium.

Testing requirements will be established, and the necessary facilities for testing will be selected. Fabrication and performance specifications and standards will follow successful development, along with the general criteria necessary to incorporate the integral safety rod system into the LMFBR design.

Summary of Current Technology — Hydraulically operated control rods have not been utilized previously in fast reactor designs. All fast reactor systems, including FARET, EBR-I and II, and Enrico Fermi, use conventional control rod systems with individual drives mounted on the top structure. Some of the small test reactors, including SEFOR, LAMPRE-I, and the zero-power reactors, use various means of moving the reflector, but none uses an integral hydraulic system.

Scope of Work — A four-step program is proposed to develop an integral safety rod system for a reliable auxiliary shutdown system for use in sodium-cooled fast breeder reactors:

1. Criteria Investigation: This step will begin with a thorough investigation of the shutdown problem. The prime objective will be to establish the operating criteria for the auxiliary safety shutdown system and its functional requirements.

2. Study and Design: This step will begin with a thorough study of a preliminary design. The study will size the valves and orifices and establish flow rates. The sodium pump will be rated, along with the actuating air cylinders and solenoid valves. The snubber systems will be studied thoroughly in combination with the latch design and its operation. The problems of sodium chemistry, flow corrosion, and plugging will be studied. This step will produce complete layout drawings of the entire system and detailed drawings of all the components that require development and testing in the second step of the program. In addition to the analytical effort to support this design, step 1 will establish materials and designate the most probable components for purchase. A planned procedure for step 2 will result, as will estimates for component fabrication costs and fabrication lead times. Preliminary component testing schedules and costs will also be given.

3. Fabrication and Testing of Components: This step will begin with the fabrication of full-sized test components. Concurrently, existing test facilities will be studied thoroughly, and a complete development program will be established. Specific tests will be planned and scheduled. Scheduling of the design and fabrication of testing equipment will be started. As now foreseen, the development procedure will first require testing in water to develop corrections to clearances, orifices, flow rates, or other details. This will be followed by testing in sodium facilities that duplicate reactor conditions as nearly as possible. Long-term lifetime, wear, or compatibility tests will be planned and started. This step will be completed with preliminary planning, cost, and schedule estimates for full system testing in the next step. Drawings will be revised and updated concurrently with the development program.

4. Fabrication and Testing of System: The first effort in this step is completion of the scheduling and estimating begun in phase 3. Facilities to accomplish the testing will be designed, built, or scheduled as required. The fabrication of a complete system is part of this step; it will be a full-sized system consisting of the actuating sodium pump, the valve and actuating mechanism, the command line, a simulated absorber column mounted in an actual guide tube and complete with the latches and snubbers developed in the previous steps. This complete system will be mounted in a facility having sodium environment that simulates the reactor conditions as nearly as possible.

In addition to this complete system testing, a displacement check is planned for after completion of all the system tests. The dummy fuel element will be displaced mechanically to simulate minor core rearrangement and thus establish the degree of immobilization of the integral safety rod system.

Fabrication and performance specifications and standards will be started during the later stages of full-sized system testing. Successful completion of the tests and specification documents will establish the criteria necessary to incorporate the integral safety rod system into the large sodium-cooled fast reactor system.

Major Facilities and/or Equipment Required —

Testing in a water environment will require a conventional bench test setup. Although sodium test facilities for control rods are available, the construction of a separate facility should be considered for these tests. A decision to use existing facilities will rest on an examination of the required modifications to accommodate the sodium pump and hydraulic lines necessary for the tests.

Relationship to Current or Planned Programs —

Development of the integral safety rod system would result in no duplication of effort with current or past fast reactor programs.

Schedule and Budget — The costs in thousands of dollars are given below.

	Calendar year				
	1969	1970	1971	1972	Total
Manpower	100	200	200	50	550
Facilities		100	150	50	300
Equipment		100	50		150
Total	100	400	400	100	1000

6.3. Cost Benefit Studies

The following sections define the penalties for nonattainment and/or cost benefits for performing the research and development described in the preceding sections. Quantitative evaluations of penalties for nonattainment draw heavily from the results of the 1000-MWe LMFBR Follow-On Study, Parametric Studies.³⁰ Power generation costs are consequently quoted for the reference design; these values are low, since the parametric studies were initiated before the estimate of plant capital costs was completed. Evaluations of the penalties (cost differences) of nonattainment are not affected by the absolute accuracies of the power generation cost, but only by the relative accuracies. Of course, the percentage of change due to nonattainment is not valid and consequently is not quoted.

6.3.1. Fuel Pin

The fuel pin for the reference design incorporates a mixed-oxide, vibratory compacted fuel with fission gas venting-to-coolant through a porous plug vent. The vibratory compacted fuel is expected to offer two advantages over pellet fuels: (1) the fabrication costs should be lower by an estimated 10%, and (2) higher burnups should be attainable. In addition, the attainable burnup is expected to increase because of the presence of a low-density, unsintered annular ring of fuel adjacent to the cladding; a pellet having the same initial smeared density as a powder fuel configuration will contact the cladding with a less elastic mass. Vibratory compacted fuel should distribute the cladding loads more uniformly. Two problems arise from the use of powdered fuel: 1. Some upper limit on the practically attainable smeared density may limit the density to less than the design goal; 2. Powdered fuels are

apparently limited to approximately 90% of the pellet fuel's linear power rating to produce fuel melting.

Other advantages of the vented fuel pin are that claddings of low thickness can be used and the pin length required for the fission gas plenum can be eliminated and thus improve the core breeding performance and the fuel's specific power rating. The porous plug is an ideal vent because it allows the use of shorter pins than the diving bell and check valve schemes do, and it employs no moving parts. The possible problem areas include (1) fuel burnup (incore time) limitations due to long-term sodium logging, and (2) attainment of an allowable fission gas product buildup time as set by the activity level of the cover gas.

6.3.1.1. Fuel Pin Fabrication

Fabrication difficulties could prevent attainment of the reference fuel pin configuration. The possible shortcomings and their consequences are as follows:

1. Nonattainment of 85% smeared density resulting primarily in a reduction of core breeding performance. Lower densities involve an increase in enrichment (fertile loading decreases more rapidly than the fissile loading with decreasing smeared fuel density).

At the reference core's linear power rating of 17 kW/ft, a reduction in smeared fuel density of from 85 to 80% results in a fuel cost increase of 0.045 mill/kWh.³⁰

The effects of lowered smeared fuel density on the restructured fuel geometry would indicate that a linear heat rate increase to 18.7 kW/ft is possible. A reduction, then, of 85 to 80% density accompanied by a linear power rating change from 17.0 to 18.7 would result in a net fuel cost increase of 0.005 mill/kWh.³¹

An alternative in the design that would circumvent the density problem is the use of pellet fuel, but, some increase in fabrication cost is expected to accompany this design choice. The unit fabrication cost of the vipac fuel is estimated to be 10% less than for pellet fuel. The resulting power generation cost penalty for employing pellet fuels at reference conditions is $3.349 - 3.315 = 0.034$ mill/kWh as obtained from BAW-1328³² (accounting for differences in fabrication costs only).

2. Nonattainment of the 10-mil cladding thickness is conceivable because of fabrication and handling limitations. Fifteen-mil cladding thicknesses have been demonstrated on a laboratory scale. Using this figure as a minimum limit for a production pin, the penalty for nonattainment of the reference design thickness becomes $3.385 - 3.314 = 0.071$ mill/kWh.³³ Reference 33 accounts for a reoptimization of the cladding's inside and outside diameters as well as fuel pin pitch. The relationship between power generation cost and cladding thickness over the thickness range from 10 to 20 mils is very nearly linear.

3. Problems arising from the fabrication of a vent device are discussed in the section below. Fuel density and enrichment variations (tolerances) are covered in section 7.6.

6.3.1.2. Vent Concept

Nonattainment of the vented fuel pin concept for the reference design may arise from difficulties in several areas. These problem areas, the consequences and/or alternatives are as follows:

1. Difficulties in fabricating the porous plug may preclude its use. Substitution of a diving bell, requiring considerable research and development in itself, would result in a penalty of $3.465 - 3.432 = 0.033$ mill/kWh.³⁴ Reference 34 does not include reoptimization of pin pitch and diameter, which would accompany the added fuel pin length required for the diving bell.

2. The combined effects of the porous plug, fuel, and cladding on the fission gas holdup time may not meet that time required without maintenance of an appreciable internal gas pressure. Assuming slug flow of fission gases within the pin (no mixing and no concentration gradients in the direction of flow), the holdup time of a particle is proportional to the distance traveled and inversely proportional to the velocity of the particle. At a given mass flow rate of fission gases, the local gas velocity decreases with increasing fission gas density accompanying an increase in pin pressure; this increases the holdup time.

An approximate holdup time has been calculated for the reference fuel pin using the following assumptions:

1. The effect on the holdup time of gas passing through the sintered portion of fuel was neglected.
2. A fission gas release rate per pin of $10.8 \times 10^{-9} \text{ ft}^3/\text{min}$.
3. A flow area over the active fuel length equal to the as-fabricated fuel annular ring area multiplied by the quantity 1.0 minus the fraction of theoretical density.
4. A flow area over the axial blanket length of the cladding's inside cross-sectional area multiplied by the quantity 1.0 minus the fraction of theoretical density.
5. Internal gas pressure of 20 psia.

The result is an average holdup time of 6.4 days over the active fuel length and a holdup time of 4.7 days over the blanket length, making a total of 9.3 days*. If the internal gas pressure were raised to 200 psia, the holdup time could conceivably approach 93 days (for full power steady state operation). This fuel pin internal pressure increase, per se, would not limit the useful life of fuel pin cladding for the reference thickness of 10 mils.

3. It is conceivable that a porous plug guaranteeing exclusion of sodium from the interior of the pin cannot be developed. The long-term effect of sodium logging has not been adequately determined; it is possible that some sodium ingress might be acceptable if the in-core fuel pin time is not severely limited due to fuel swelling induced by sodium logging. Even with the successful testing of G. E., it should be assumed at this point that any sodium ingress is unacceptable and that failure to develop a successful porous plug would lead to the diving bell

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* This is probably a high estimate; a calculation for a combination of pellet fuel with a diving bell vent similar to that reported in GEAP-4770 gives a holdup time of approximately 6.2 days, whereas the experimental measurement indicated a holdup time of 5 days.

concept or to sealed fuel pins. Penalties associated with the diving bell concept are discussed in 1 above. Penalties in power generation cost associated with sealed fuel pins, vibratory compacted fuel, 15-mil cladding, and 90% fission gas release amount to $3.577 - 3.432 = 0.145$ mill/kWh.³⁴ Again, no account is taken of any reoptimization of fuel pin diameter or pitch. Penalties in power generation cost associated with sealed fuel pins, pellet fuel, 15-mil cladding, and 50% fission gas release would be approximately $3.536 - 3.432 = 0.045$ mill/kWh, neglecting the difference in fabrication cost between pellet and powdered fuels. Considerable research and development would be required to permit the use of pellet fuels (sealed) with a 10-mil cladding (the corresponding penalty would reduce to $3.477 - 3.432 = 0.045$ mill/kWh, neglecting any burnup limitations). The foregoing penalties for resorting to sealed fuel pins do not include any savings for the resulting relaxation of requirements for the reactor seals. Other factors in comparing sealed and vented concepts have been ignored in the discussion above, since they cannot be quantitatively supported at this time. They should be accounted for, however. First, it is felt that internally pressurized fuel pins may present a serious safety problem since cladding failure and subsequent fission gas release at high rates may promote further pin failures. Thus, a sealed concept may require a sizable reduction in design burnup to minimize the probability of failure and the ensuing failure propagation. Another approach might be to increase the cladding thickness; however, both methods of minimizing the probability of failure will considerably increase the power generation cost. Also, the sealed-pin, solid-pellet concept may involve an added penalty other than that discussed above, since the lower fission gas release rate promotes increased fuel swelling, thereby possibly limiting the fuel burnup.

6.3.1.3. Fuel Burnup

Penalties for nonachievement of the 100,000 MWd/T burnup goal may be assessed from BAW-1328.³⁵ Burnup, as referred to here, may alternatively be termed inpile time. The predominant limitation on inpile time is a cladding failure due to fuel swelling; other considerations are fertile-fissile segregation due to plutonium

migration and fuel or cladding failure due to the effects of sodium ingress through cladding or weld defects. Other problem areas such as the fuel melting temperature and fuel-cladding chemistry problems are more properly classified as linear power rating, coolant temperature, cladding thickness, or material considerations.

Cladding failure due to fuel swelling is expected to be the prime limitation on inpile time. The effects of fuel swelling due to long-term sodium logging cannot be quantitatively assessed at this time. Sodium ingress through the vent device may be eliminated by selecting a sealed pin design; the economic penalties for this alternative are discussed under the heading Sealed Versus Vented Fuel Pin. Sodium ingress through cladding or weld defects may be minimized by improved production methods which involve increased fabrication costs; some trade-off between fabrication costs and burnup would result. In this case we do not have enough data on which to base a prediction of the cost penalty.

It is anticipated that irradiation-induced fuel swelling should not limit the average fuel burnup to less than 50,000 MWd/T. The penalty for nonattainment, then, would be $1.296 - 0.816 = 0.480$ mill/kWh in fuel cost.³⁵ The figure in reference 35 is based on the assumption that absolutely no outward radial swelling will be tolerated by the cladding; the penalty would be lessened somewhat by increasing the smeared fuel density to 95% while maintaining 50,000 mWd/t. The smeared density of the reference fuel was set at 85% to provide void volume in which swelling might be accommodated up to approximately 80,000 MWd/t (based on $0.7\% \Delta v/v/10^{20}$ fissions).

6.3.1.4. Linear Power Rating

The present design philosophy for establishing a maximum allowable linear power rating is this: The hottest pin in the core must operate with its central fuel temperature at or slightly below the melting point with the reactor operating at 111% power. The power rating is a strong function of the fuel's effective thermal conductivity, the fuel's melt temperature, and the fuel temperature at the cladding-fuel interface. The uncertainties in determining the fuel's effective thermal conductivity are considerable at this time; also, some questions

arise as to the allowable maximum fuel temperature at which it is safe to operate. Other considerations of lesser importance are the factors affecting outside fuel temperatures. These are of less importance since it is felt that more confidence can be placed on the presently available data and/or analytical techniques concerning these. The following two paragraphs discuss possible limitations on the maximum allowable linear power rating.

The effective thermal conductivity is dependent upon the restructured fuel geometry, the distribution of the restructured fuel density, and the distribution of the temperature. These, in turn, are dependent upon the fuel-cladding gap's unit surface thermal conductance, equiaxed and columnar grain densities, and the as-fabricated fuel density. A consideration of the effects of uncertainties in the effective fuel thermal conductivity would indicate that a confident design limit for the linear power rating is 14 kW/ft. From BAW-1328³⁶ we obtain a power generation cost penalty of $3.370 - 3.310 = 0.060$ mills/kWh, where the penalty is incurred by adding fuel pins to the reference core. A cost penalty of $3.350 - 3.310 = 0.040$ mill/kWh results if the active fuel height of the core is increased.

A decrease in the maximum linear heat rate may be required because of uncertainties in the fuel melt temperature; also, the extent of long- or short-term fuel slumping may require a reduction in the operating fuel temperatures. By short-term slumping, we mean slumping fuel movements due to fuel pin or core vibration and shock. Report BAW-1328³⁷ shows that the maximum linear heat rate must be decreased by 0.5 kW/ft per 100 F reduction in the maximum operating temperature. This report³⁶ also shows that the power generation cost increase is approximately 0.007 mill/kWh for a 0.5 kW/ft decrease in the maximum allowable heat rating.

The total length of active fuel in the core is set by the thermal power requirements of the core and the average linear heat rating of the pins. The average heat rating of the fuel pins is the maximum allowable linear heat rating (discussed above) with corrections for power profiles, core overpower, and uncertainties. Some of the uncertainties are related to the fuel pin; the power profile and core overpower considerations are discussed in section 6.3.5., Core System. The effect of an increase in uncertainties can be evaluated

from reference 36; an increase of 10% in the uncertainties would have the same effect on the total length of fuel as would a decrease of 10% in the maximum allowable linear heat rating. A decrease from 17.0 to 15.3 kW/ft (equivalent to a 10% increase in the aforementioned uncertainties) increases the power generation cost by approximately $3.334 - 3.310 = 0.024$ mill/kWh. For this change in heat rating, no appreciable difference is evidenced by increasing either the number of fuel pins or the height of the core.

6.3.1.5. Fuel Pin Cladding

The cladding thickness and the type of cladding material used for the pin affect the fuel burnup, the core breeding ratio, and the reactor mixed mean outlet temperature. The reference design goal of obtaining an average fuel burnup of 100,000 MWd/T is predicated on the assumption that the cladding will limit the rate of outward radial fuel swelling. Thus, the cladding thickness and the mechanical properties become extremely important. The mechanical properties of the cladding may be degraded by the cladding temperature (effects on allowable stress, creep strength, etc.), fluence, fuel-cladding chemical incompatibility, and sodium-cladding incompatibility. Any addition to the cladding thickness to counteract the deleterious effects of temperature, fluence, and the like on the mechanical properties has an adverse effect on the breeding performance. From reference 33 the power generation cost penalty per mil of cladding thickness is $(3.385 - 3.315) / (15.0 - 10.0) = 0.014$ mill/kWh, which is primarily the result of a decrease in breeding performance and the plutonium credit component of fuel cost; this figure is quoted, of course, for constant burnup. No fuel-cladding model is available for a realistic evaluation of the burnup equivalent of 1 mil of cladding. The deleterious effects of fluence, mass transfer, and the like on the cladding's mechanical properties can be countered by lowering the cladding temperatures. As mentioned in the section on fuel burnup, the cost penalty associated with a reduction of coolant temperature is 0.04 mill/kWh per 50 F.

If, as proposed, the cladding can exert enough force on the fuel to retard outward radial growth of the fuel, then the cladding temperature will significantly influence the allowable inpile

time (the cladding creep strength and the modulus of elasticity vary inversely with temperature). No analytical model is available to relate the clad temperature to the allowable burnup. If a 50 F reduction in the core outlet temperature is required, however, to maintain the goal of 100,000 MWd/T burnup, then the penalty in power generation cost would amount to $3.360 - 3.320 = 0.04$ mill/kWh. This figure reflects the effects of maintaining the 300 degree coolant temperature rise for the reference core (and the reference steam cycle) in order to maintain the breeding performance. The cost penalty arises from the increased size of the heat transport system equipment.³⁸ One mil of cladding thickness added to the reference design results in a penalty of $(3.385 - 3.315)$ mills/kWh/ $(15.0 - 10.0)$ mils = 0.014 mill/kWh per mil in power generation cost. Therefore, the addition of $0.040/0.014 = 2.85$ mils of cladding that would result in the same penalty as would a reduction of 50 F in the mixed mean core outlet temperature. This in no way implies, however, that 2.85 mils of cladding, and a 50 F temperature change have an equivalent effect on the allowable burnup. (For cladding thickness effects on power generation cost, see reference 33.)

The primary problem anticipated in the area of cladding limitations on the allowable burnup is the reduction of ductility occurring at high values of fast fluence. All stainless steel candidate materials are expected to become severely embrittled under the anticipated irradiation exposures. Consequently, we will have to take advantage of the high strengths of candidate materials to limit the tendency of the fuel to swell in an outward radial direction; high-strength claddings may supply sufficient restraining force to force the plastic deformation of the fuel into the center void and in the axial direction without appreciable cladding deformations. With the absence of an acceptable plastic fuel model, it is not possible to accurately predict the penalties associated with the loss of cladding ductility as it affects the allowable burnup or as it affects the required cladding thickness to maintain the reference burnup of 100,000 MWd/T.

6.3.1.6. Plutonium Migration

The possibility of plutonium migration in mixed-oxide fuels may pose a safety problem (a reduction of the Doppler

coefficient) at a given time during the incore lifetime of the fuel and thus limit the inpile time. Little is known of the mechanism for, or the extent of, the migration and consequent fissile-fertile segregation. As a result, no quantitative measurements of possible burnup curtailment are available. Two design alternatives would be to reduce the central fuel temperatures by (1) reducing the linear power rating of the fuel pin, or (2) maintaining the linear power rating but employing a fuel-cladding sodium bond. These approaches may prove feasible if the migration phenomenon is primarily temperature dependent. If the phenomenon is strongly thermal-gradient induced, then other methods must be found to alleviate the difficulty (if it materializes).

6.3.1.7. Thermal-Mechanics

A potential problem regarding cladding integrity lies in the possibility of failure due to sudden changes in the power-to-flow ratio; these changes may occur as a result of scrams from operating power levels. The cladding may "shrink" on the fuel and fail because of the stresses induced by the differences in fuel and cladding temperatures and coefficients of thermal expansion. It is felt that this problem is minimized by the use of vipac fuel since the fuel next to the cladding is not a hard, sintered mass as is the case with pellet fuel. Two possibilities exist for alleviating the problem should it arise. First, the cladding thickness could be increased to reduce the cladding stress. As discussed previously, an increase in cladding thickness of 1 mil will increase power generation costs by 0.014 mill/kWh (at constant burnup). An alternative would be to reduce the coolant temperature rise through the core by increasing the coolant flow rate. Maintaining the reference core mixed mean outlet temperature at 1100 F and increasing the inlet temperature to 900 F results in a power generation cost penalty of $3.395 - 3.320 = 0.075$ mill/kWh. This 100 F increase in inlet temperature lowers the hot channel (at 103%) temperature rise from $1210 - 800 = 410$ F to $1176 - 900 = 276$ F, so that the cost penalty is $0.075/134 = 0.00056$ mill/kWh per degree Fahrenheit (hot channel).³⁸

6.3.2. Fuel Assembly

6.3.2.1. Spacer Selection

The LMFBR designer is faced with a multitude of problems concerning the selection of a fuel pin spacer. The problem areas are discussed in detail in section 6.2.2.2, Fuel Pin Support System Selection, but we shall repeat them here for convenience:

1. Fuel pin vibration.
2. Fuel pin cladding and spacer fretting corrosion.
3. Fuel pin bow.
4. Assembly bow.
5. Intra- and inter-channel coolant mixing.
6. Fabricability (spacer and fuel assembly).
7. Effects of spacer geometry and material on plant economics and fuel conservation.
8. Structural integrity when subjected to high temperature, high fluence, vibration, and hydraulic forces.

Owing to the state of technology for spacer selection, the absence of analytical methods and general knowledge (e.g., in fabricability) precludes quantitative spacer comparisons of the required accuracy. We can, however, obtain some qualitative results by examining the effects of using different types of spacers in the reference core which are now in use or planned for use in other sodium-cooled fast reactors. The types of spacers and their geometries are as follows:

<u>Reactor</u>	<u>Existing</u>	<u>Planned</u>	<u>Spacer type</u>	<u>Spacer pitch, m.</u>	<u>Fuel pin lattice</u>
EBR-II	x		Wire-wound	6.00	Hex.
FFTF		x	Wire-wound	12.00	Hex.
Fermi	x		Grid	3.75	Square
PFR		x	Grid	4.00	Hex.

Figure 6-2. Effect of Various Spacer Designs on Power Generation Cost

Figure 6-3. Effect of Fuel Pin Pitch and Type of Spacer on Fuel Cost

Figure 6-4. Effect of Various Spacer Designs on Doubling Time

Figure 6-5. Effect of Various Spacer Designs on Pin Bundle Pressure Drop

Figure 6-6. Effect of Various Spacer Designs on Core Steel Volume Fraction

Figure 6-7. Effect of Spacer Types and Pitches on Power Generation Cost for Reference Pin Pitch and Diameter

Figure 6-8. Effect of Spacer Types and Pitches on Doubling Time

Figure 6-9. Effect of Spacer Types and Pitches on Bundle Pressure Drop for Reference Pin Pitch and Diameter

Figure 6-10. Effect of Spacer Types and Pitches on Can Thickness and Core Steel Volume for Reference Pin Pitch and Diameter

For this study, we shall quote numerical differences in power generation cost arising from the use of the helical ribbon in place of the grid and wire-wrapped types used or planned for use in other reactors. The numerical differences should be considered as qualitative only, since the analytical tools are absent for determining adequate spacer geometries (from a performance standpoint) for any type of spacer.

For types of spacers that perform equally well with respect to the aforementioned problem areas, the differences arising in power generation cost and doubling time result from each type's peculiar characteristics in adding to the core's steel volume and coolant pressure drop (in comparison to no spacer at all). Any spacer adds to the core's steel volume; the spacer's steel volume and the added can thickness (due to the increased coolant pressure drop) increase the core's steel volume. Also, the spacer's pressure drop component of the total pressure drop through the core increases the primary system's pumping power. The spacer's steel volume and component of coolant pressure drop vary with spacer thickness (cross-sectional), axial pitch (axial distance between leading edges for the grid type), and spacer height (applies only to the grid type). Before examining the effects of particular types and geometries on the power generation cost and the doubling time for the reference fuel pin's diameter and pitch (0.280 and 0.337 inch, respectively), we will consider the effects of spacer types on the optimum fuel pin pitch.

Figure 6-2 illustrates the effects of spacer types and geometries on the optimum pin pitch for minimum power generation cost at a constant pin diameter of 0.280 inch. An important limitation of Figure 6-2, as well as Figures 6-3 through 6-10, is that the performance of certain or all geometries shown may not be acceptable. The discontinuity occurring in the curves between pin pitches of 0.340 and 0.350 inch is a result of a step change in the radial blanket pin size, the number of pins, and consequently the fabrication component of fuel cost. It can be seen from Figure 6-2 that the helical ribbon of 2-inch pitch and 20-mil thickness and the grid of 11.4-inch pitch and 15-mil thickness result in minimum power generation costs occurring very close to the reference pitch of 0.337 inch. The wire-wound case and the grid cases shown for a 3.8-inch pitch all result in a minimum

occurring at pitches somewhat more removed from the reference pin pitch value. Note that in Figure 6-2 the power generation cost (PGC) for the wire-wound spacer at a pin pitch of 0.337 inch is slightly higher than the PGC for the same spacer at this spacer's optimum pin pitch (approximately 0.327 inch); therefore, little inaccuracy will be incurred by neglecting this difference in PGC for the figures to follow (6-7 and 6-8). We will, however, incur a somewhat greater inaccuracy if we consider the 0.337 inch pitch instead of the optimum pitch for some grid geometries. We shall neglect this inaccuracy in Figures 6-7 through 6-10, since the inaccuracy is not sufficient to invalidate the trends intended to be shown. Figure 6-3 shows the effects of spacer types and geometries on the optimum pin pitch for minimum fuel cost. Note that the minimum fuel cost for the wire-wound spacer occurs at a lower pin pitch than does the minimum PGC: the reverse is true for the grid type, i. e., the fuel cost minimum occurs at a larger pin pitch than does the PGC minimum. This is because the high pumping power required in the wire-wound case overrides the usual effects of the reactor vessel capital costs which tend to lower the optimum pin pitch. The differences in total core pressure drop for the two cases can be seen in Figure 6-5.

The arguments presented above for the justification of examining the effects of spacer types and geometries on PGC while maintaining constant fuel pin diameter and pitch (0.280 and 0.337 inch respectively) do not pertain to determining the effects of spacer types and geometries on doubling time. From Figure 6-4, it is seen that large differences can occur between the doubling time at a pin pitch of 0.337 inch and the actual minimum doubling time for some types of spacers. Consequently, Figure 6-8 shows the effect of spacer types on doubling time for pin pitches of both 0.337 inch and the optimum pin pitch for minimum PGC (the optimum pitch depends on the type and geometry as shown in Figure 6-2).

An explanation for the relatively small pin pitches for minimum PGC and doubling time for the wire-wound type and the relatively large pitches for the grid type is found in Figures 6-2 and 6-6. With the absence of any spacer, the optimum pitch for minimum PGC is approximately 0.342 inch for a pin diameter of 0.280 inch, giving, in this case, an optimum ligament of 0.062. The addition of a

grid-type spacer changes the slope of the core's steel volume fraction curve as shown in Figure 6-6, thereby driving the optimum ligament upward. The addition of a wire-wound spacer reverses the slope of the core's steel volume fraction curve, thereby driving the optimum ligament downward. The opposite slope for the wire-wound case is primarily a result of the unique dependence of the spacer's cross-sectional area on the fuel pin pitch.

Figure 6-7 illustrates the effect of spacer types and geometries on the power generation cost. The spacer pitch for the helical ribbon is the axial distance required to make one complete 360 degree twist. In the wire-wound spacer, the pitch is the axial distance for one complete traverse around the pin's periphery. The pitch of the grid spacer is the axial distance between leading edges of adjacent spacers. The thickness of the helical ribbon is the small cross-sectional dimension, whereas the thickness of the grid is the web thickness measured along a line connecting fuel pin centers. Of course, the diameter of the wire in the wire-wound spacer is equal to the pin ligament (pin pitch less the pin diameter). The height of the grid is 1.0 inch in Figures 6-7 through 6-10. The spacer types and pitches used in the EBR-II and planned for use in the PFR and FFTF reactors are marked with the symbol $\oplus$, as is the reference type and pitch. There is little difference between the PFR and the Fermi grid pitches as indicated in the foregoing tabulation, so that we have chosen to show only the PFR grid pitch; also, the Fermi pin pitch is square, which would add additional uncertainties to the validity of comparisons of spacer geometries.

It may be seen from Figure 6-7 that at constant inpile time the power generation cost is relatively insensitive to changes in the helical ribbon thickness; appreciable effects, however, can result from decreasing the pitch from 2.0 inches (the reference value) to 1.0 inch or less. Design changes in the helical ribbon's thickness or pitch may result from future disclosures of problems with fuel pin vibration, bowing, fretting corrosion, and the like with the present spacer geometry. If the spacer requirements dictate a change in the type of spacer for the reference design, the cost penalty (based on the designated spacer types and geometries) will range from 0.03 to 0.196

mill/kWh depending on the spacer alternative employed. Cost penalties may be evaluated using the left-most ordinate.

Figure 6-8 shows the effects of spacer types, geometries and pitches on the doubling time for a fuel pin diameter of 0.280 inch. The solid lines are for a pitch of 0.337 inch and the dashed lines are for the optimum pitch for minimum doubling time depending on the type of spacer. For a helical ribbon pitch of from 2.0 to 6.0 inches, the doubling time is relatively insensitive to the geometry; however, the doubling time becomes quite sensitive to pitch at pitch values less than 2.0 inches. If the spacer requirements dictate a change in the type of spacer, the doubling time penalties might range up to 5.0 years. The left-most ordinate may be used to evaluate the doubling time penalties or doubling time reductions.

Figures 6-9 and 6-10 illustrate the causes for the very high doubling times for the wire-wound types. In Figure 6-9, which is a plot of the total bundle pressure drop as a function of spacer pitch, the wire-wound spacer results in a pressure drop comparable to that obtained with the 15-mil-thick grid. Figure 6-10 shows that the resulting fuel assembly can thicknesses for the wire-wound type and 15-mil-thick grid are comparable; a result of nearly identical coolant pressure drops. Figure 6-10 also shows the relationships between the core's steel volume fraction and the spacer pitch for various spacers and spacer geometries. The wire-wound spacer results in a high core steel volume fraction over the entire range of wire pitches examined; in fact, the core's steel volume for the wire-wound case exceeds that for any other type of spacer considered. This very high steel volume fraction is a direct result of the very high steel volume of the wire spacers themselves, which results in the very high doubling times shown. The doubling time increases with the core steel volume fraction because the increasing steel results in both higher fissile loadings and lower breeding ratios (because of higher enrichments and greater neutron parasitic absorption by the steel).

Table 6-2 lists the advantages, disadvantages, and performance uncertainties of various types of spacers. The integral-fin spacer is included here even though no power generation costs or doubling times are presented for it in Figures 6-2, 6-4, 6-7, and 6-8.

Table 6-2. Performance Characteristics and Uncertainties for Various Types of Spacers

Spacer type	Comments	Possible spacer geometry or reactor changes to meet performance requirements	Cost penalties incurred by geometry changes
1. Helical Ribbon	A. Advantages		
	Low spacer steel		
	Enhances inter- and intra-channel fluid mixing		
	Minimizes "Hot Spot" formation at spacer and cladding contact points		
	Spacer compliance accommodates fuel pin swelling		
	B. Disadvantages		
	Difficult assembly fabrication		
	C. Performance Uncertainties		
	Fuel pin vibration	Decrease pitch Increase spring constant (by increasing thickness)	0.024 mill/kWh per inch 0.0002 mill/kWh per mil
	Fretting corrosion	Decrease pitch (decrease load per contact) Increase thickness	0.024 mill/kWh per inch 0.0002 mill/kWh per mil
	Spring constant variation with temperature and nvt	Change spacer concept	
	Tolerance buildup effect on hot channel factors (reduction of channel flow area)	Change spacer concept	
	Fuel pin autocatalytic bow	Decrease average coolant temperature Increase spring constant (by increasing thickness)	0.0008 mill/kWh per F 0.0002 mill/kWh per mil
	Structural integrity	Increase thickness	0.0002 mill/kWh per mil
	Spacer pressure drop component	Change fuel can thickness and pumping power	Approximately 0.0028 mill/kWh per psi
	Assembly thermal bow effect on hot channel factors (reduction of channel flow area)	Decrease average coolant temperature Increase spring constant (by increasing thickness)	0.0008 mill/kWh per F 0.0002 mill/kWh per mil
2. Grid	A. Advantages		
	Spacer compliance accommodates fuel pin swelling		
	B. Disadvantages		
	High drag coefficient		
	Spring design and development problems incurred for small ligaments		

Table 6-2. (Cont'd)

<u>Spacer type</u>	<u>Comments</u>	<u>Possible spacer geometry or reactor changes to meet performance requirements</u>	<u>Cost penalties incurred by geometry changes</u>
	C. Performance Uncertainties		
	Fuel pin vibration		
	Fretting corrosion		
	Spring constant variation with temperature and nvt		
	Tolerance buildup effect on hot channel factors		
	Fuel pin autocatalytic bow		
	Structural integrity		
3. Wire-Wound	A. Advantages		
	High spacer structural integrity		
	Noncompliance eliminates concern over fuel assembly bow effects on hot channel factors		
	Enhances interchannel fluid mixing		
	B. Disadvantages		
	No inherent provision for accommodation of fuel pin swelling		
	Noncompliance can increase corrosion fretting due to high contact load per unit area		
	High spacer steel volume at the pin pitches of interest (in the particular case of the reference core)		
	C. Performance Uncertainties		
	Fuel pin vibration		
	Fretting corrosion		
	Tolerance buildup effect on hot channel factors		
	Fuel pin autocatalytic bow		
	Fuel pin contact loads		
4. Integral fin (helical, with fin-to-fin contact)	A. Advantages		
	No cladding fretting corrosion		
	High structural integrity		
	Noncompliance eliminates concern over fuel assembly bow effects on hot channel factors		
	B. Disadvantages		
	No inherent provision for accommodation of fuel pin swelling		
	C. Performance Uncertainties		
	Tolerance buildup effect on hot channel factors		
	Fuel pin contact loads		

In the integral-fin concept considered here, fin-to-fin contact between adjacent fuel pins is obtained. The table lists the spacer geometries or reactor changes that may be required in case the helical ribbon spacer falls short of performance goals or criteria in the problem areas listed. Also, cost penalties are associated with each possible change, although it is conceivable that some changes may be made in a direction that lowers power generation costs.

6.3.2.2. Steady-State Thermal-Hydraulics

A number of uncertainties exist in the projected thermal-hydraulics performance of the reference fuel assembly. Most of these uncertainties affect the maximum cladding surface temperature in the hot channel, but all of them affect, to some degree, the allowable linear power rating (due to the cladding temperature at the core midplane being in excess of those calculated for the hot channel). The uncertainties are as follows:

1. Factors affecting the maximum cladding surface temperature include uncertainties in the hot channel bulk temperature (channel heat rate, flow rate, and extent of interchannel mixing), the local velocity and temperature profiles, and the heat transfer coefficient. The present B&W hot channel factor on the coolant bulk temperature (enthalpy rise) includes physics calculational uncertainties, fuel loading, pin dimension and reactor power level uncertainties plus factors to account for coolant maldistribution (inlet plenum effects) and coolant mixing. We wish to consider here only the effects of uncertainties in the hydraulics. Coolant maldistributions within an assembly that are caused by fuel pin bow, transient flow phenomena, or pressure drop calculational uncertainties may increase the uncertainty for the coolant enthalpy rise from the present 1.132 to 1.19, for an additional maldistribution factor of 1.05. In order to maintain the reference design's maximum cladding surface temperature of 1209 F in this event, the bulk outlet temperature would have to be reduced to 1086 F; this would result in an increase of approximately 0.010 mill/kWh at constant inlet temperatures.³⁸

2. A factor of 0.95 has been used to take advantage of interchannel coolant mixing and has been applied to the bulk coolant

temperature rise. In the unlikely event that mixing does not appreciably affect interchannel thermal energy transport, the effect on the bulk outlet temperature would be the same as that quoted above; that is, a reduction of 14 F resulting in a penalty of 0.010 mill/kWh.

3. The local coolant velocity and temperature profiles may result in appreciable variations in the peripheral cladding temperature. The temperature variations cannot be predicted without a three-dimensional flow model.

4. The adverse effects of non-wetting, gas entrainment and/or corrosion films on the heat-transfer coefficient might conceivably increase the coolant film temperature drops by as much as a factor of two (an upper limit). Some small reduction in the mixed mean outlet temperature may have to be made as a result of unexpectedly low heat transfer coefficients. The maximum cladding surface temperature is a rather weak function of the heat transfer coefficient, since the heat flux is relatively low at the upper end of the active fuel length and the film temperature drop is calculated to be less than 5 F based on present data. An increase of 5 F in the maximum cladding surface temperature would require a decrease of approximately 3.5 F in the mixed mean outlet temperature. This appears to be a relatively insignificant effect, but no judgment can be passed on its importance until the fuel burnup equivalent of 1.0 F in cladding temperature is determined. The cladding creep rate increases with temperature and thus results in cladding stress relaxation and less restraint on fuel outward radial growth (postulated).

All the foregoing factors affect the cladding surface temperatures at the axial location where the peak fuel temperatures are experienced (near the core midplane). Consequently, these factors will affect the maximum linear power rating, but not significantly since a 100 F rise in temperature would decrease the maximum allowable linear power rating by approximately 3% (0.51 kW/ft), which results in a power generation cost penalty of 0.007 mill/kWh.

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