949

CEND-346

MASTER

1000 MWe

L M F B R

Liquid Metal Fast Breeder Reactor

FOLLOW-ON STUDY

**COMBUSTION ENGINEERING
TASK IV REPORT**

**A Research and Development
Program Needed for the
C-E Reference Concept**

Prepared for ARGONNE NATIONAL LABORATORY

RESTRICTION OF THIS DOCUMENT IS INDICATED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

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

Printed in USA. Price \$3.00. Available from the Clearinghouse for Federal Scientific and Technical Information, National Bureau of Standards, U. S. Department of Commerce, Springfield, Virginia 22151.

1000 Mwe LIQUID METAL FAST BREEDER REACTOR FOLLOW-ON STUDY:
TASK IV REPORT

A RESEARCH AND DEVELOPMENT PROGRAM
NEEDED FOR THE C-E REFERENCE CONCEPT

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Contract 31-109-38-1999
Argonne National Laboratory

Issued: February, 1969

Unclassified

Classification

Approved by:

William H. Lamm 2/69

Authorized Classifier

Date

W. P. Staker

W. P. Staker,
Project Manager

NUCLEAR POWER DEPARTMENT
COMBUSTION ENGINEERING, INC.
WINDSOR, CONNECTICUT

leg

The material in this report represents the work of numerous personnel on the Liquid Metal Fast Breeder Reactor Project. Those involved and their areas of responsibility are:

W. P. Chernock - Fuels and Materials
J. C. Gilbertson - System Studies
W. J. Mordarski - Fuels and Materials
H. O. Muenchow - Reactor Support Systems
R. C. Noyes - System Studies and Technical Guidance
J. J. Prabulos, Jr. - System Studies
R. S. Stankiewicz - Mechanical Design Studies
G. Zuromsky - Fuels and Materials

W. P. Staker,
Project Manager

TABLE OF CONTENTS

<u>Section No.</u>	<u>Page No.</u>
1.0 INTRODUCTION	1
References	3
2.0 SUMMARY AND DISCUSSION OF PROBABILITY OF ACHIEVING R&D GOALS AND PENALTIES ASSOCIATED WITH NONACHIEVEMENT	4
References	10
3.0 PLANT DESIGN	11
3.1 Introduction	11
3.2 Description of Current Technology	11
3.3 New Information Needed	12
3.4 Program Description	12
4.0 FUELS AND MATERIALS	14
4.1 Introduction	14
4.2 General Development Plan	20
4.3 Program Definition -- New Information Needed	22
4.4 Program Description	27
References	44
5.0 SAFETY	54
5.1 Introduction	54
5.2 Core Configuration Safety Studies (Subtask 10-1.1.0)	57
5.3 Fuel Failure and Melting (Task 10-2.4)	63
5.4 Research Related to Hypothetical Accidents (Task Area 10-2)	72
5.5 Protective System (Task 10-3.1)	73
References	92

TABLE OF CONTENTS (Cont.)

<u>Section No.</u>		<u>Page No.</u>
6.0	CORE DESIGN	98
	6.1 Program Definition	98
	6.2 Description of Current Technology	99
	6.3 New Information Needed	108
	6.4 Program Description	116
	References	126
7.0	COMPONENTS DESIGN: REACTOR VESSEL AND REACTIVITY CONTROL SYSTEM	127
	7.1 Definition	127
	7.2 Description of Current Technology	128
	7.3 New Information Needed	132
	7.4 Program Description	136
8.0	COMPONENTS DESIGN: FUEL HANDLING SYSTEM	146
	8.1 Program Definition	146
	8.2 Description of Current Technology	148
	8.3 New Information Needed	153
	8.4 Program Description	156
	8.5 Probability of Accomplishment	176
	8.6 Effect of Nonattainment of R&D	176
	References	182
9.0	PHYSICS	183
	9.1 Introduction	183
	9.2 Acquisition of Nuclear Data	183
	9.3 Performance of Critical Experiments	190
	9.4 Improvement and Development of Calculational Methods	193

LIST OF ILLUSTRATIONS

<u>Figure No.</u>		<u>Page No.</u>
2-1	Economic Incentive for Development of Higher Performance Sodium-Bonded Carbide Fuel	7
2-2	Estimated Probability of Achieving Fuel Development Goals within Time and Cost Estimates	9
4-1	Fuels and Materials R and D Schedule and Milestone Chart	45
5-1	Safety R and D Schedule and Milestone Chart	94
6-1	Core Design R and D Schedule and Milestone Chart	124
7-1	Component Design R and D Schedule and Milestone Chart	144
8-1	Pin Drive and Seal Testing	158
8-2	SATM Decay Heat Cooling R and D	162
8-3	Automatic Welding Machine	166
8-4	Subassembly Canning System	170
8-5	Mechanism Storage Tests	174
8-6	Fuel Handling System R and D Schedule and Milestone Chart	179

LIST OF TABLES

<u>Table No.</u>		<u>Page No.</u>
2-1	Summary of Estimated Fuel Cycle Cost Penalties	5
4-1	Summary of Estimated Costs (1970-1975)	47
4-2	Detailed Program	48
4-3	Yearly Costs by Subtask	53
5-1	Safety R&D Manpower and Cost Estimate	97
6-1	Materials Information Required for Fuel Subassembly Design	114
6-2	Core Design R&D Manpower and Cost Estimate	123
7-1	Components Design R&D Manpower and Cost Estimate	143
8-1	Pin Drive and Seal Cost Estimate	159
8-2	Cost Estimate SATM Decay Heat Cooling R&D	163
8-3	Automatic Welding Cost Estimate	167
8-4	Subassembly Canning System Cost Estimate	171
8-5	Mechanism Storage Tests Cost Estimate	175
8-6	Fuel Handling System R&D Manpower and Cost Estimate	181
9-1	Effects of Possible Higher Alpha Values for Pu-239 on Performance of C-E Reference Design	185
9-2	Schedule of Highest Prior Cross Section Measurements from LMFBR Program Plan Applicable to C-E Design Effort	187
9-3	Other Cross Section Measurements from LMFBR Program Plan Applicable to C-E Design Effort	189
9-4	Recommendations Concerning Research and Development in the Area of Nuclear Data Acquisition	190
9-5	List of Basic Physics Experiments in Support of LMFBR Program	192
9-6	Research and Development Required in Computational Methods	194

1.0 INTRODUCTION

The 1000 Mwe Liquid Metal Fast Breeder Reactor Follow-on Study is an extension of 1000 Mwe design studies performed previously for the Atomic Energy Commission. The objective of the Follow-on program is to provide information which will be of assistance to the AEC in the furtherance of the Liquid Metal Fast Breeder Reactor (LMFBR) Program Plan. The scope of the work for Combustion Engineering is in the area of the heat generation system of the nuclear steam supply system. The objective of this study of the heat generation system has been to develop a reference design⁽¹⁾ and to identify the research and development programs which are required to permit commitment of the selected design by approximately 1980. The Task IV work described in this report is concerned with these research and development programs.

To the extent applicable, the description of R&D programs includes: (1) a definition of each R&D item, (2) a description of current technology, (3) a statement of new information needed, (4) a program description for each item, (5) an estimate of the probability of accomplishing objectives within reasonable funding and time estimates, and (6) an estimate of the cost penalty which would result from nonattainment or partial attainment of R&D objectives. It is difficult, however, to define items (4), (5), and (6) unambiguously without reference to the environment in which the proposed R&D is to be conducted. For example, the program description for fuels development must consider the future availability of irradiation space and, in order to evaluate item (6), consideration must be given to the future availability of alternate fuels. It was neither practical nor consistent with the objectives of this study to attempt to predict the most probable future conditions in this respect. Rather, we have formulated a set of working assumptions which define future conditions in a logical manner which attempts as much as possible to be consistent with the results of the present C-E Follow-on Study, the 1968 version of the AEC LMFBR Program Plan and the broad objectives of the U. S. power reactor development program. In some respects, these assumptions are, of necessity, arbitrary; however,

generally they reflect real possibilities which together represent one practical way to achieve the overall goals of the LMFBR program. The basic assumptions thus developed are as follows:

- a) The national LMFBR program will follow a two-concept approach to the development in appropriate areas. First emphasis will be on a conservative design which has a very great probability of operating, although it may not meet all performance goals. A strong additional program will be pursued toward a higher performance system which, if it succeeds, will be much more likely to have the competitive economic performance needed by our economy;
- b) An LMFBR basic technology research and development program which is not concept oriented will be vigorously pursued in the areas of physics, fuels and materials, safety, instruments and sodium technology;
- c) For the immediate near term, in cases where available facilities are not adequate to meet needs, the R&D requirements of the Fast Flux Test Facility (FFTF) will be given first priority and the preferred high performance concepts will be given second priority except for certain very important experiments. During this period, development of FFTF driver fuel will be adequate to satisfy the development requirements of a conservative oxide LMFBR fuel;
- d) During the development period, at least one carbide-fueled demonstration plant will be built. This plant will be planned and operated on a semicommercial basis but will be available for certain experiments and tests which are most efficiently accomplished in such a plant.

In accordance with the contract for this study, the R&D programs described in this report are generally limited to those particularly associated with the C-E reference design. The main exception to this rule is in the area of general LMFBR base technology research where certain important items are discussed. Since, however, this R&D is applicable and essential to the LMFBR concept in general and not to a particular concept, its costs should not be associated with the C-E concept alone. In much of this area, the economic penalties associated with nonattainment of R&D goals for individual studies are not definable in a meaningful way because choices between alternates are not directly involved (in measurement of neutron cross sections, for instance). The most that can be said is that

if this research is not done or is not successful, there is some likelihood that serious mistakes will be made either in the design of demonstration plants or perhaps even in the decision to build a particular demonstration plant. The possibility of such a mistake does constitute a risk with which economic penalties can be associated; however, there is no apparent reasonable way to subdivide this "contingency cost" among the numerous individual R&D programs which contribute to its reduction. Specific identification of base technology items is made in the text.

For items such as fuel handling components and control rod drives for which several alternate components and systems are expected to be developed, but for which specific details of alternates are not available, it has not been practical to attempt to cost the alternates and accurately identify penalties associated with nonattainment of development objectives. The results of the fuel handling trade study have been used, however, to obtain a general indication of the penalties associated with what could be considered a typical alternate system.

REFERENCES

for Section 1.0

1. "1000 Mwe Liquid Metal Fast Breeder Follow-on Study: Report of Tasks II and III, A Reference Conceptual Design of a 1000 Mwe LMFBF", Vols. I-III, CEND-337 (1968).

2.0 SUMMARY AND DISCUSSION OF PROBABILITY OF ACHIEVING R&D GOALS AND PENALTIES ASSOCIATED WITH NONACHIEVEMENT

From the point of view of broad research and development objectives, the core of the C-E reference design has the following characteristics:

- a) High performance sodium-bonded mixed carbide fuel in a vented fuel element;
- b) Cylindrical core geometry with a positive void effect, and if unmoderated, a voided Doppler effect which is smaller (less negative) than for comparable oxide cores;
- c) Redundant and highly reliable core cooling and protective systems and an extensive local temperature monitoring system;
- d) A core mechanical arrangement characterized by a relatively low steel fraction and close spacing of fuel rods.

Comparative fuel cycle cost calculations have been made to determine cost penalties of various alternatives to the reference design which have characteristics less desirable than those for a reactor with those indicated above but which may be dictated by nonattainment or only partial attainment of R&D objectives. Table 2-1 is a listing of the principal alternates that fall into this category together with fuel cycle cost penalties associated with each alternate. The cost penalties shown in table 2-1 represent perturbations on the reference design assuming that other characteristics are unchanged. The penalty listed for moderator addition, however, is not a perturbation since the reference design contains moderator. The indicated penalty has already been "paid" in the reference design. Research in the area of safety, however, may show that it will not be necessary to pay this penalty for a safe design; therefore, from the point of view of R&D planning, this item is equivalent to the other items listed.

Table 2-1. Summary of Estimated Fuel Cycle Cost Penalties

<u>Item</u>	<u>Penalty (mills/kwh)</u>	<u>Remarks</u>
<u>Fuel System</u>		
1. Sodium-bonded carbide, performance reduced to:		
peak burnup = 70,000 Mwd/T	0.27	see Fig. 2-1
peak power = 32 kw/ft	0.05	
combination of above	0.35	
2. Change to gas-bonded carbide (100,000 Mwd/T avg)	0.25-0.35	Ref (1)
3. Change to gas-bonded oxide (100,000 Mwd/T avg)	0.35-0.50	Ref (1)
4. No practical LMFBR fuel. Change to LWR	~0.8	Ref (2)
<u>Safety</u>		
1. Change from regular to spoiled geometry to obtain a negative void effect (modular or pancake)	0.30	
2. Add moderator to increase Doppler effect	0.04-0.14	Depends on subassembly development
3. Large LMFBR unsafe. Change to LWR	~0.8	Ref (2)
<u>Core Design</u>		
1. Change to hex subassembly with moderator pins	0.08	
2. Reduce peak sodium velocity to 20 ft/sec (fuel rod P/D increase to 1.25)	0.12	

The cost penalty or benefit of a given alternative design characteristic usually cannot be unambiguously associated with a particular R&D task or subtask, although there are exceptions to this rule. The rather general association of alternates with R&D areas (Fuel, Safety and Core Design) indicated in table 2-1 is the best that can be done in this respect; however, even this general relationship must be qualified. For example, it is reasonable to assume that the fuel development program will establish the important performance limitations on the fuel (burnup, linear heat rating, etc.). Therefore, reduced allowable fuel performance can be said to be the penalty associated with partial attainment of the goals of the fuels R&D program. The probabilities and penalties for this program, indicated below, have been established on this basis. However, it should be kept in mind that reduced fuel performance might be dictated by results of research in other areas such as safety. An unlikely, but not impossible, result of safety studies could be the elimination of the sodium bond. It is also not impossible that the sodium bond could be ruled out because of nonattainment of the goals of fuels R&D subtask on carbon transport from fuel to clad. In either case, the alternative is the same (gas-bonded carbide)* and the penalty is identifiable (0.25-0.35 mills/kwh) but the alternative is associated both with safety and fuels R&D.

Figure 2-1 is a plot of fuel cycle costs as a function of peak fuel heat rating and peak fuel burnup. The reference design point is indicated and a lower limit performance point (70,000 Mwd/MT at 32 kw/ft) is shown which represents the approximate lower limit of the break-even range for competition with high performance gas-bonded oxide. The incentive for development of a fuel of lesser performance is minimal; therefore, the R&D program is designed to have a high probability (98 percent) of reaching this point. Estimated probabilities of achieving higher fuel performance within the limits of the R&D program are

* Actually, the alternatives might not turn out to be the same in this case if, for instance, the problem is carbon transport, not safety. Then, sodium-bonded nitride might be the logical alternative, provided of course that nitride is developed successfully. Because there are so many such possibilities, consideration of alternatives has been restricted to a few obvious examples.

Economic Incentive for Development of
Higher Performance Sodium-Bonded
Carbide Fuel

COMBUSTION ENGINEERING, INC.
WINDSOR, CONNECTICUT

Figure
2-1

shown in figure 2-2. The reduced probabilities corresponding to higher performance levels represent a combination of the estimated uncertainty of the intrinsic ultimate potential of the fuel system and the effect of the limited nature of proposed R&D program. This program is deliberately designed in this manner because it is felt to be the most efficient approach. Rather than propose a much larger and broader program with a very high probability of achieving or exceeding reference design performance, it is more efficient to concentrate the effort on a sequential approach which has a high probability of assuring a truly viable LMFBR fuel system in the near term and a reasonably good chance of producing a truly advanced fuel system at minimum cost. As with all fuel systems, it is expected that further development work will be done which will achieve higher performance levels. The ultimate probability of achieving the reference design performance or better is estimated to be at least 90 percent.

Necessary R&D has been placed in the following sections more or less in accordance with the organization of the LMFBR Program Plan. In several instances, however, it has seemed more natural to cover an item together with a related item in a section not identical with that of the Program Plan. For example, various aspects which might be covered under Fuel Recycle, Fuels and Materials or Core Design are all located in the Fuels and Materials Section.

Probability
of Achievement

Estimated Probability of Achieving
Fuel Development Goals
Within Time and Cost Estimates

COMBUSTION ENGINEERING, INC.
WINDSOR, CONNECTICUT

Figure
2-2

REFERENCES

for Section 2.0

1. "Development and Evaluation of C-E Advanced 1000 Mwe LMFBR Design", R. C. Noyes et al, International Conference on Sodium Technology and Large Fast Reactor Design, Argonne National Laboratory, ANL-7520, Nov. 7-9, 1968.
2. "Atomics International 1000 Mwe LMFBR Follow-on Study Progress Report", R. E. Aronstein et al, International Conference on Sodium Technology and Large Fast Reactor Design, Argonne National Laboratory, ANL-7520, Nov. 7-9, 1968.

3.0 PLANT DESIGN

3.1 INTRODUCTION

The entire 1000 Mwe Follow-On Study is within the compass of the Plant Design section of the AEC LMFBR Program Plan. Most research and development items, however, have been assigned to the specific elements of the Program Plan to which they are most closely related.

Among the items required before commercial sale of the C-E reference design will be feasible is proof testing of the carbide fuel system. However, in addition, it is desirable that an entire demonstration plant based upon the vented carbide fuel system be operating.

For demonstrating the C-E LMFBR plant concept, the demonstration plant should be of the loop-type with a cylindrical core using sodium-bonded, mixed carbide fuel in thin, vented stainless steel tubes. Such a plant would test licensibility, extended reactor operation, and refueling concepts. It is assumed that a previous demonstration plant, or plants, will utilize oxide fuel and a loop or pool primary system, or both. If such is the case, it is assumed that this demonstration plant will utilize technological advances made after the designing of the first demonstration plant.

3.2 DESCRIPTION OF CURRENT TECHNOLOGY

In the U.S.A., EBR-II and FERMI are liquid metal cooled fast breeder reactors. Their operation has been at relatively low power with metal fuel and has been rather variable in reliability. FERMI, especially, has had an unfortunate long-term shutdown. It is for this reason that the demonstration plants will be built to the most reliable standards and will utilize well-engineered and tested components.

In Europe, Dounreay and the PFR being built in England will precede us. Similarly, Rapsodie and Phenix in France will provide prior information. Each of these is different in one or more major concept, but on the other hand will provide considerable information of value.

As mentioned earlier, it is assumed that the demonstration plant to test the C-E concept will not be the first demonstration plant, but will follow at least one oxide plant. Therefore, it can utilize technology developed later in addition to using test data on the carbide fuel system which will be developed in forthcoming years.

3.3 NEW INFORMATION NEEDED

This item of study, the demonstration plant, provides the testing of the total synthesized system concept. It should test all of the system design features of the reference concept, including numerous features of the complete plant design which are in areas not studied under the Follow-on Study contract. It would also test the logic of the design through the licensing process. That is, if the demonstration plant is properly designed in relation to the large reactor plant, it will not skirt the safety issue but will face up to the safety problems similar to those which are present in a 1000 Mwe plant. A major result from the demonstration plant would be the experience gained by the reactor vendor, architect-engineer (A-E), Utilities and numerous subcontractors in designing, building and testing such a sodium-cooled fast breeder reactor and the components included therein. In addition, the demonstration plant will prove out the many aspects of the design which cannot be tested in any other way. For example, the correctness of the total core loading, the control element worth in the reactor, the reactivity change with core life, and many other items will all be verified. All components will be tested in the proper reactor environment rather than in simulated test conditions. By this means, the proper functioning of such components will be demonstrated.

3.4 PROGRAM DESCRIPTION

The R&D program for the demonstration plant will not be presented in detail here since it is considered better to emphasize and spend time on those key items, such as the fuel and cladding system, which will lead up to the demonstration plant of the reference concept. The demonstration plant is

mentioned here, however, in order that this major item will be considered in long range planning.

The costs for a demonstration plant should be those additional costs above the regularly expected costs for routinely built light water reactors. Included in additional expenses will be first-of-a-kind engineering and fabrication, and pre- and postoperational testing beyond normal resulting from the identification of the plant as a demonstration plant.

4.0 FUELS AND MATERIALS

4.1 INTRODUCTION

The vented fuel rod selected as the basis for the reference design for this follow-on study contains sodium-bonded, hyperstoichiometric, mixed carbide fuel which is clad with stainless steel. The bases for selecting this fuel concept center upon its high potential for meeting the major fuel goals of the LMFBR Program which are the production of economical nuclear power with a fuel cycle capable of high fissile atom utilization. Thus, low doubling times and concomitant high burnups are necessary to meet these objectives.

It is not the objective of this section of the report to compare all candidate fuels and establish the basis for the selection which has been made for this study. A comparison of doubling times and fuel cycle costs for helium-bonded mixed oxide, helium-bonded mixed carbide and sodium-bonded mixed carbide has been presented elsewhere⁽¹⁾ and this evaluation forms the basis for selection of the vented, sodium-bonded, hyperstoichiometric mixed carbide fuel for this design study. Although an evaluation of mixed nitride fuels was not performed, its major characteristics are sufficiently analogous to those of the mixed carbide to permit general inferences to be drawn concerning its potential value as an LMFBR fuel.

In the 1000 Mwe follow-on studies, the attainment of average burnups of 100,000 Mwd/MTM is presented as one of the objectives. This burnup forms the basis for estimates of fuel cycle costs and doubling times. There is little doubt that all of the fuels being considered for LMFBR application will be capable of attaining average burnups of 100,000 Mwd/MTM. However, the capability of fuel rods to attain this burnup level is dependent upon a number of factors, the most important of which is the ability of the cladding to accommodate the strains imposed by swelling of the fuel, fission gas release, ΔT across the cladding thickness and structural accommodation to the fuel bundling scheme. It seems that the major limitation with respect to fuel rod performance will be provided by the properties of the cladding after irradiation to fast fluences in excess of 10^{23} nvt. In this respect, continued

and vigorous Commission programs are aimed at identifying the characteristics of stainless steel cladding exposed to fluences in excess of 10^{23} nvt as well as aimed at possible solutions to the clad swelling and ductility problems.

The cladding properties will control the achievable burnup for a given fuel rod concept. Therefore, a fuel rod concept which minimizes cladding strain offers the best potential for achieving the target burnups indicated above. In this respect, the stainless steel clad, vented, sodium-bonded, hyperstoichiometric mixed carbide fuel offers considerable potential for fulfilling LMFBR targets. Evaluations pertaining to doubling times have been included in other sections of this report. A few of the parameters controlling high burnup potential are discussed below.

- 1) Fission Gas Release - Although fission gas release at burnups below about 30,000 Mwd/MTM are quite low, there is good evidence indicating substantially higher release rates at burnups approaching and exceeding 50,000 Mwd/MTM. The use of a vented fuel rod concept eliminates internal fission gas pressure as a factor with respect to cladding strain. Furthermore, the use of a vented fuel rod concept eliminates the driving force necessary to force sodium from the fuel-cladding interface in the event a fuel cladding defect is present. Thus, the design, development and statistical testing of a reliable vent is a vital factor in development of this reference design fuel rod.
- 2) Fuel Swelling - Cladding strains resulting from fuel swelling can be eliminated through proper design of a fuel rod in which the sodium bond annulus between the fuel and cladding is sized adequately to accommodate maximum fuel swelling rates. Although the concept of void deployment within the fuel may permit a reduction of what might otherwise be considered as a "hard" fuel-cladding contact strain, such a concept requires use of lower density carbide and inflicts a penalty of inducing a cladding strain of a magnitude required to cause flow of the fuel into its voids during swelling. Therefore, the concept requiring swelling accommodation through void deployment is less desirable, even if the concept can be demonstrated as feasible, since the cladding strain, a limiting parameter, will be substantially greater than that encountered in a sodium-bonded rod which is adequately designed. Nevertheless, irradiation test programs currently underway to evaluate and develop gas-bonded carbide with Commission support should be continued on the following bases:
 - a) Important additional information on fission gas release from these mixed carbides will be obtained. Although this information may seem to be, at first glance, of

only academic interest to a vented fuel rod design, such information will provide important inputs to the following considerations:

- 1) The quantity of gas which must be accommodated by venting as a function of linear heat rating and burnup;
 - 2) An assessment of the consequences of vent plugging. The resulting cladding strain, burnup limits, and safety related factors for a plugged-vent fuel rod can be more accurately evaluated.
- b) Determination of whether void deployment is a viable concept for mixed carbide. These evaluations will permit an assessment of fuel swelling accommodation in voids and a determination of failure lifetimes for rods with different void deployment capability. This information will be important input to the sodium-bonded concept in that it will permit a better assessment of risks encountered as a result of possible improper total assessment of free swelling rates. If void deployment is possible, the margin to rod failure, even in a high density rod (>95% TD) will be substantially greater than that anticipated if void deployment is not demonstrated as a means for swelling accommodation.
- c) Substantial additional information on swelling of mixed carbide fuels will be obtained. Although "free swelling" rates are needed for the sodium-bonded concept, the swelling data generated in the UNC program will be quite useful in establishing parameters and designs for irradiation testing of the sodium-bonded concept.
- d) Additional information on cladding failure strains can be obtained. Although the fast flux in EBR-II is substantially below that anticipated in LMFBR systems, important additional information on cladding failure strains and fuel swelling accommodation strains can be obtained. However, these must be properly assessed since the fuel burnup-cladding strain capability will not be demonstrated by these irradiations. There is a tendency to presume that a burnup capability for a fuel rod in EBR-II provides demonstration of LMFBR performance to an equivalent burnup. This is not completely true since the cladding ductility in EBR-II is substantially greater than the cladding ductility attained in an irradiation in an LMFBR to an equivalent fuel burnup. Therefore, the ability of the cladding to strain without failure at a cladding fluence is all that can be presumed from EBR-II irradiations. This information is important in assessing cladding strain limits.

- e) Finally, the information on helium-bonded, carbide fuel rods will be important in assessing the performance capability of possible "fall-back" fuel rod concept in the unlikely event that formidable problems are encountered in the use of a sodium-bonded concept.

We recommend continuation and expansion of the current limited UNC effort on sodium-bonded mixed carbide fuels. Although the target linear heat ratings and burnups are low (about 25 kw/ft and 50,000 Mwd/MTM), information gained from this program will be important in providing data points for free swelling of mixed carbides. In addition, performance of the sodium bond can be assessed with respect to fission gas entrapment. Data from this limited effort will provide a baseline for required high performance irradiations of the sodium-bonded concept indicated as part of the required development effort outlined below.

In addition, the Los Alamos program on the development and testing of single-phase mixed carbide should continue to receive AEC support. As part of this program, seven sodium-bonded capsules will be irradiated in EBR-II at peak linear heat ratings of about 30 kw/ft and a peak burnup of about 60,000 Mwd/MTM. This program will provide additional data on free-swelling of mixed carbide as well as performance of the sodium bond. In addition, the potential for fabricating single-phase material will be assessed.

A helium-bonded, stainless steel clad, mixed oxide fuel rod will undoubtedly be selected for the first LMFBR demonstration plant on the basis of the considerable amount of AEC research, development and testing of this system which will have been performed in support of the FFTF effort. The second or third demonstration plants should contain mixed carbide. The choice of mixed oxide for the first plant is based upon the concerted FFTF effort. We must concur with the premise that the mixed carbide fuel will not be ready for commitment until the second or third demonstration plant because the Commission will require that there are sufficient performance data available to indicate that a properly designed fuel rod will be capable of achieving the target performance levels conducive to low doubling times, high burnups and low fuel cycle costs. A partial assessment of the fuel rod concept used in this study was provided earlier in this section and indicated that the one major limitation to successful fuel rod performance was reduced to a minimum through incorporation

of the concept used as a design basis; namely, minimizing cladding strain. One of the other possible deterrents to achieving highly successful high performance from these fuels is the possibility of additional cladding embrittlement through the transfer of carbon from the fuel to the cladding via the sodium bond. In CEND-200 (January 1964), the possible problem of carbon transport in sodium-bonded hyperstoichiometric mixed carbide systems was identified. One of the solutions proposed at that time was the simple heat treatment of the original monocarbide-dicarbide phases to produce the monocarbide-sesquicarbide phases. It was suggested that a discontinuous sesquicarbide phase would not transport carbon through the sodium bond to the cladding. Recent British, French and UNC ex-reactor results have verified this conjecture for hyperstoichiometric mixed carbide fuel containing the sesquicarbide as the nonstoichiometric carbide phase. Therefore, the probability that carbon transport from the fuel to the cladding will limit performance of the sodium-bonded concept has been substantially reduced. In-reactor verification will be necessary. Evaluation of the limited number of UNC sodium-bonded fuel rods being irradiated will provide important information with respect to carbon transport. Additional irradiation data will be required.

Another uncertainty which must be resolved is associated with the performance of the sodium bond under conditions in which fission gas bubbles could accumulate in the bond and reduce the effectiveness of heat transfer from the fuel to the cladding. Although this condition may exist, it has not been observed to date in any operating fuel rod which has been sodium-bonded. It is likely that in-reactor vibrations would move gas bubbles, if any would form, through the sodium to the vent. Nevertheless, this area needs further investigation to establish, on a statistical basis, whether this potential problem needs further consideration. In-reactor testing will be required. Again, the limited in-reactor tests on sodium-bonded mixed carbides now being performed by United Nuclear and Los Alamos will provide some initial information. However, in-reactor testing of sodium-bonded carbide fuel rods at peak linear heat ratings approaching 45 kw/ft and for burnups in excess of 50,000 Mwd/MTM (the burnup above which fission gas release in these carbides increases by about one order of magnitude) will be required before the concern over degradation of the sodium heat transfer bond by fission gas bubbles is finally resolved.

A fuel rod system which is to be developed with Commission support must have considerable potential for achieving the long range objectives of the LMFBR program. We believe that the mixed carbide fuel in a properly designed fuel rod will be capable of achieving burnups in excess of 100,000 Mwd/MTM at peak linear heat ratings approaching 45 kw/ft. The design which will achieve these conditions must provide adequate diametral clearance to accommodate swelling, without fuel cladding mechanical interaction, and must be capable of accommodating the high fission gas release rates associated with these high burnups. Therefore, a fuel rod with sufficient swelling accommodation to prevent fuel cladding mechanical interaction and a vent mechanism to prevent fission gas buildup is required. It is our belief that the rod design concept used as a basis for this study meets the above requirements and that this design can be committed, after additional irradiation testing, to statistical performance demonstrations in early LMFBR demonstration plants. Although at least one of these plants will be fueled with mixed oxide, at least one of the two remaining plants should be designed to accommodate the full potential of the mixed carbide system. Therefore, all components should be sized on the basis of a fuel core loading operating at conditions which are associated with peak linear heat ratings of up to 45 kw/ft. These demonstration plants will play a role in fuel development since they will provide a basis for full scale demonstration of performance of fuel rod and assembly performance. However, before the Commission will commit a fuel system for use in a demonstration plant, adequate performance capability must have been demonstrated via irradiation test programs.

In addition to the performance demonstration of the reference fuel rods via irradiation testing, the development program must be capable of demonstrating fabrication feasibility with respect to the production of a well characterized fuel. This program must also be capable of demonstrating a fuel cycle cost fabrication component consistent with total fuel cycle costs in the neighborhood of 0.4 mill per kwh.

The anticipated high performance capability and low power generating costs which are believed to be achievable for the stainless steel clad, sodium-bonded, hyperstoichiometric mixed carbide justify an intensive development program to

facilitate the inclusion of this fuel rod concept in a demonstration plant at an early date.

In considering the objectives for a research and development program, it is not possible, within the framework of a reasonable allocation of funds, to proceed with all promising concepts at the same level of effort. Therefore, the philosophy with which the outline of this program was approached centered upon maximum use of existing information and use of technology forming the bases of current irradiation efforts. For example, the approach which is utilized in the fuel fabrication area is one which concentrates on the UNC process of carbothermic reduction of coprecipitated mixed oxide followed by pressing and sintering with a 0.1 w/o Ni sintering aid. It is entirely possible that alternate fabrication schemes will lead to lower fabrication costs. Nevertheless, the fuel cycle costs associated with this method are those indicated earlier in this report and are in the range leading to a fuel cycle cost of about 0.4 mill/kwh. Undoubtedly, at a later time, other processing methods may be evaluated for cost reduction. However, the program outlined below is aimed at a concentrated effort in those areas requiring such effort. Very little planning for interesting sidelights or alternate approaches has been included.

4.2 GENERAL DEVELOPMENT PLAN

The general plan for the development effort is summarized below:

- 1) We believe that mixed carbide fuel in a properly designed fuel element will achieve average burnups in excess of 100,000 Mwd/MTM at peak linear heat ratings approaching 45 kw/ft.
- 2) A production process for mixed carbide fuel must be developed and this contribution to fuel cycle costs must be established.
- 3) A fuel element with sufficient swelling accommodation to prevent fuel-cladding interaction and a vent mechanism to prevent fission gas pressure buildup are required.
- 4) Two of the major safety problems which must be considered are:
 - a) The consequences of cladding rupture;
 - b) The consequences of fuel melting.

Cladding rupture, in a properly designed fuel rod, would result if the vent become clogged or if estimates of swelling rates were erroneous. One model which has been proposed as a means for inducing fuel melting is the loss of the sodium bond resulting from leakage through a hole in the cladding. Two conditions must exist simultaneously to account for this possibility. First, there must be a hole in the cladding. Second, the vent in this rod must cease to perform properly, resulting in internal fission gas pressures which can then act to force the sodium from the rod through the hole in the cladding. The probability of occurrence of both conditions in the same rod is very low.

- 5) A program for failure mode analysis is already being initiated by the AEC at ANL. Experimental work in reactors is also being included for verification of proposed models. This program, together with the programs outlined in the Safety section of this report, will provide the necessary safety data.
- 6) The fuel rod design used as the basis for this study is aimed at avoiding any of the possible failure modes discussed above.
- 7) At least one of the LMFBR demonstration plants should be designed on the basis of the mixed carbide system. Therefore, all components should be sized on the basis of a full core loading operating at conditions which are associated with peak linear heat ratings of up to 45 kw/ft.
- 8) The fuel element design for the demonstration plant should be the design which is capable of attaining 100,000 Mwd/MTM burnup at a linear heat rating of up to 45 kw/ft. However, the fuel cycle economics should be based upon 30 kw/ft at 50,000 Mwd/MTM average burnup.
- 9) Performance of the fuel element described in item 8 at 50,000 Mwd/MTM average burnup and at 30 kw/ft will provide fuel cycle costs (about 0.75 mill/kwh) which are lower than those in current PWR systems.
- 10) In the demonstration plant, the early aim for fuel performance is to demonstrate that the fuel element designed for 100,000 Mwd/MTM average burnup and a peak linear heat rating of up to 45 kw/ft is capable of achieving at least 50,000 Mwd/MTM average burnup at 30 kw/ft, parameters which form the bases for the fuel cycle economics for the demonstration plant. Burnups beyond 50,000 Mwd/MTM should be demonstrated and the core should remain in service until an average burnup of 100,000 Mwd/MTM is achieved or until some predetermined level of fuel failure incidence occurs.

4.3 PROGRAM DEFINITION - NEW INFORMATION NEEDED

The development plan is based upon completion of the following prior to the commitment of this mixed carbide fuel in an LMFBR plant.

4.3.1 Development of Fuel Fabrication Processes

It must be demonstrated that fuel fabrication processes are developed sufficiently to provide a well characterized fuel with predetermined properties at costs consistent with those associated with an ultimate fuel cycle cost of about 0.4 mill per kwh.

4.3.2 Design and Testing of a Vent Concept

At the time of commitment, a reliable vent concept needs to be "in hand". This will require a major emphasis early in the program. It is our belief that a reliable vent concept can be developed and that a sufficient number of ex-reactor performance tests can be completed to provide the high performance reliability necessary. Some confirmatory in-reactor testing will be necessary.

4.3.3 Demonstration of the Behavior of Gas Bubbles in the Sodium Bond

Although problems with respect to impaired heat transfer as a result of fission gas bubble accumulation in the sodium bond have not been observed to date in fuel rods which have been irradiated, work needs to be performed in this area to demonstrate acceptable sodium-bond performance under conditions of high linear heat ratings and high burnups. The latter condition is necessary to insure that the concept has been tested under fission gas release conditions which are representative of high burnup fuel. Furthermore, tests are needed to determine whether momentary power excursions can result in fission gas bursts which may induce conditions which are not representative of steady state operation. Some of the irradiation tests already underway by UNC, Westinghouse, and Los Alamos will provide initial assessment of the probability of fission gas bubble problems in the sodium bond. Although some ex-reactor

evaluations will be performed, these may be of questionable value. Final proof testing is needed under in-reactor conditions.

4.3.4 Carbon Transport through the Sodium Bond

UNC, the French and the British ex-reactor tests have indicated that carbon transport is not a problem in sodium-bonded mixed carbide when the phases are monocarbide and sesquicarbide. Westinghouse is pursuing an interesting scheme with respect to chromium carbide additions to hypostoichiometric mixed carbide to reduce carbon transport. The effort at Los Alamos with respect to development of single-phase mixed monocarbide should also provide a solution to this problem. On the bases of ex-reactor tests, it appears likely that carbon transport from the mixed carbide* to the cladding need no longer be of major concern. Certainly, in-reactor verification will be necessary. Initial verification will be obtained from the following programs:

- a) UNC irradiation of sodium-bonded hyperstoichiometric mixed carbide containing the mono-plus sesquicarbide phases;
- b) Westinghouse irradiation of sodium-bonded hypostoichiometric mixed carbide with chromium carbide additions;
- c) Los Alamos irradiation of sodium-bonded mixed monocarbide.

We do not believe that all three approaches need to be pursued in the R&D effort, outlined below, at the same level of importance. We believe that structure control (monocarbide plus sesquicarbide) provides the simplest and most economical method for elimination of the carbon transport problem. The other two approaches might constitute good backup efforts in the event unforeseen surprises are encountered. Therefore, major programmatic effort will be centered upon irradiation testing of sodium-bonded fuel containing the monocarbide and sesquicarbide structures. Within this framework, the maximum concentration of sesquicarbide will be equivalent to that generated by considering the maximum carbon content in the fuel believed probable on the basis of high production yields and low fabrication costs.

* Containing monocarbide and sesquicarbide; structures formed on sintering with a 0.1 w/o Ni sintering aid.

4.3.5 Swelling and Fission Gas Release

Programs now underway by UNC, Westinghouse and Los Alamos should provide some initial information on free swelling of mixed carbide. However, considerably more information than is currently available is required to establish reliable free swelling values for this fuel so that designs can be evolved which provide zero fuel cladding mechanical interaction and a resultant zero cladding strain from this component. Additional free swelling data will be obtained as a result of irradiations described in the program below. The probable approach for the demonstration plant design will be the use of the most conservative swelling rates which have been generated. Fission gas release data will be a less critical requirement in view of the vent concept being employed.

4.3.6 Efficacy of Bundling Method

Testing of fuel assembly bundling schemes can be performed in ex-reactor loops and these programs can provide important information concerning pressure drop, flow distribution, and bundle component integrity. Although in-reactor testing of full size fuel bundles may be desirable, it is not essential prior to commitment of these fuel assemblies in a demonstration plant. Ex-reactor testing programs are described in another section of this report and will not be included in the program descriptions provided below.

4.3.7 Irradiation Testing

Extensive irradiation testing of stainless steel clad rods containing sodium-bonded hyperstoichiometric fuels will be required to:

- a) Demonstrate fuel rod performance capability at peak burnups in excess of 100,000 Mwd/MTM and with linear heat ratings up to 45 kw/ft;
- b) Demonstrate efficacy of the fission gas vent;
- c) Obtain free swelling rates as a function of burnup;
- d) Demonstrate adequacy of the sodium heat transfer bond;
- e) Verify that ex-reactor solutions to the carbon transport problem are applicable to in-reactor cases at high burnup.

Major irradiation programs will have to be performed in EBR-II with full realization that the fast fluence for the cladding at a given fuel burnup will be substantially below that anticipated in an LMFBR system. However, with a design concept centered upon zero fuel cladding mechanical interaction, the consequences of these differences will be minimized. Nevertheless, the fact that there will be differences cannot be ignored; extrapolations will be necessary, especially with respect to clad swelling. These EBR-II irradiations probably will be performed in capsules initially. However, we anticipate that the capsule requirement will be removed, as it has been for mixed oxide, after additional experience has been gained through successful irradiation of the mixed carbide.

Concurrent irradiations will be performed in thermal reactors and will be aimed at demonstrating, to the extent that this is possible, the following:*

- a) Solution to the carbon transport problem;
- b) Ability of the vent to function under irradiation conditions;
- c) Performance of the sodium bond.

4.3.8 Cladding

Stainless steel cladding will be utilized throughout the development program and will be the cladding utilized for fuel for the LMFBR demonstration plant. Stainless steel cladding is being extensively irradiated as part of the FFTF program, in support of mixed oxide LMFBR systems, and as part of base line cladding irradiation efforts. Stainless steel incorporating the ORNL modifications are also being extensively irradiated. We believe that fuel rod performance will be limited in fuel rod designs which incorporate fuel cladding mechanical interactions as a result of drastic reductions in rupture life and reductions in elongation to very low values at fracture, particularly under biaxial conditions. The embrittlement is attributed to vacancy clusters or microvoids, helium formation by (n,α) reactions with Fe, Cr, Ni and N alloying

* In addition, information on free swelling and fission gas release rates will be obtained. However, fast flux irradiations will be required to provide design values.

elements in the cladding material and (n,α) reaction with boron as an impurity. Examination of an EBR-II control rod thimble exposed to a maximum fast neutron fluence of 4.8×10^{22} n/cm² and temperatures of 400 to 450°C showed a decrease in uniform elongation to 2 percent at a test temperature of 450°C. Extrapolation of limited available swelling data suggests a possible volume increase on the order of 10 percent for exposures of 10^{23} n/cm². The full effectiveness of this large amount of clad swelling on fuel rod performance has not yet been assessed. Current information indicates that a cladding strain limit of under 1 percent is likely to be imposed for fluences on the order of 10^{23} nvt.

Considerable effort is now being expended as part of AEC programs in the area of LMFBR cladding development, characterization and testing. Furthermore, extensive oxide irradiations will provide much needed information with respect to sodium corrosion and corrosion allowances as well as cladding strain limits. On these bases, the development program which is summarized below does not include an effort specifically addressed to cladding. It is our intent to follow in great detail all the Commission programs and to select that stainless steel cladding alloy which offers the highest probability of success, on the bases of irradiations already performed, and which is commercially available. With the exception of postirradiation evaluation of cladding properties for irradiated fuel rods, no specific cladding development effort is identified in the development program.

The vanadium-base alloys offer some possible relief from low ductility and excessive swelling at LMFBR temperatures and fluences. However, the development and testing of these alloys, especially under oxygen conditions which can be economically controlled in large LMFBR systems, has not yet produced a firm contender. Nevertheless, this important effort is and should be continued by the Commission. Successful completion of such a program is not necessary for the commitment of mixed carbide in a demonstration plant. Therefore, work on vanadium-base alloys has not been included in this development effort. The Commission programs will be followed and, at the appropriate time, vanadium-base alloys may be incorporated in the program if problems are encountered with stainless steel cladding.

4.4 PROGRAM DESCRIPTION

4.4.1 Subtask I: Development of Fuel Fabrication Processes

4.4.1.1 Objectives

- 1) To develop a method for the economical production of mixed carbide pellets.
- 2) To characterize the product at various stages of processing.
- 3) To investigate alternate approaches which may result in cost reductions or which may be required as a result of performance requirements.
- 4) To establish specifications for mixed carbide and uranium carbide pellets and intermediate products.

4.4.1.2 Discussion

The major effort on this subtask is directed toward the development of a commercial process for the fabrication of hyperstoichiometric mixed-carbide fuel for the core of LMFBR plants. Additional effort will be directed toward the fabrication development of uranium carbide blanket fuel.

At this time, in this country, it is not possible to purchase either production or experimental quantities of uranium carbide pellets without paying for new setup charges and plant startup costs. Therefore, this commodity must be now classified in the "commercially unavailable" category. There is considerable research and development effort on the fabrication development of mixed carbide on a laboratory scale. For the most part, the approach which is being utilized to fabricate mixed carbide is associated with the particular scheme being considered for solving the carbon transport problem in a sodium-bonded rod. Although this was not the initial impetus for the UNC approach to pellet fabrication, the 0.1 w/o Ni addition provides a sintered product which contains the desired monocarbide plus sesquicarbide phases.

Process evaluation will initiate with reprocessing plant output and will consider the impact on processing equipment and procedures introduced by realistic variations in plutonium isotope ratios and time between reprocessing fuel fabrication.

This program should be directed toward the development of a large scale commercial production process for the fabrication of hyperstiochiometric mixed carbide fuels. Because the carbothermic reduction process is in a more advanced stage of technological development and appears to be economically attractive, primary program emphasis should be placed on this process. However, the continued development of the fluidized bed technique at ANL should be encouraged because it has substantial potential for development into an economically attractive continuous commercial process. This process was not selected for primary emphasis in this program because it is not sufficiently advanced in development to merit consideration at this time. Among the various techniques developed for mixed carbide powder fabrication, the process developed on a laboratory scale at UNC⁽²⁾ for the production of hyperstoichiometric mixed carbide fuel by the carbothermic reduction of mechanically mixed oxide appears most attractive for immediate development into a large scale process because of its more advanced state of technological development. This process involves the addition of 0.1 w/o nickel^(2,3) to the powder prior to pelletizing as a means of improving powder sinterability and stabilizing the sesquicarbide phase. Using this process, high density (98 percent TD) pellets have been obtained at sintering temperature below 1,600°C. Because of the low sintering temperature required, plutonium losses from the compacts due to volatilization are minimized. Extensive ex-reactor testing⁽²⁾ has indicated that plutonium content can be routinely controlled to 0.13 w/o while the carbon content can be controlled to better than 0.05 w/o. These composition controls are based upon laboratory efforts and will undoubtedly be subject to some change as a result of scale-up of the fabrication operations. On this basis, we have set targets of ± 0.1 w/o for control of carbon content. Therefore, to be assured that free metal will not occur, it is likely that the lower limit for the effective* carbon content will have to be set at about 4.85 w/o. Thus, target compositions of 4.95 ± 0.1 w/o effective carbon may be attained. Nevertheless, initial programmatic effort on carbon transport should center on the maximum conceivable carbon content** which could be set now at about 5.2 w/o.

* Oxygen plus nitrogen plus carbon.

** We may have guessed wrong with respect to the practical carbon control in a commercial process with high yields.

4.4.1.3 Program

Current carbide fuel fabrication technology based on the carbothermic reduction of mechanically blended mixed oxides should be scaled up to facilitate the production of commercial size batches (about 100 kg per day). Pellets close to theoretical density will be prepared using nickel as a sintering aid. Fuel process development will include the following areas of effort in an attempt to optimize the production process which will result in the fabrication of high performance carbide fuel.

4.4.1.3.1 Preparation of Fuel Powder Feed Material

- a) The mechanically blended mixed oxide starting materials will be characterized for physical and chemical properties. Characterization will include particle size determination, impurity analysis, surface area analysis, bulk density, tap density, chemical analysis and plutonium homogeneity.
- b) Coprecipitated mixed oxide starting materials will also be characterized in order to determine whether improved plutonium homogeneity and lower impurity content are achievable so that physics assessment can be performed.

4.4.1.3.2 Oxide to Carbide Conversion

Techniques will be devised to facilitate the conversion of large batches (10 to 25 kg) of oxide materials to carbide. Large capacity furnaces will be designed and built which are capable of operating at temperatures approaching 2,000°C under inert atmosphere or vacuum conditions.

4.4.1.3.3 Carbide Fuel Powder Preparation Characterization

The conversion of the mixed carbide reaction product into a finely divided sinterable powder will present major processing problems because of the excessive reactivity of the powder toward atmospheric impurities. Therefore, special attention will have to be directed toward the identification, analysis and control of substitutional impurity elements such as oxygen and nitrogen. The characterization scheme which was described earlier will also be used for the characterization of the carbide fuel powder. The analysis of the fuel will be complicated by a differentiation between the actual carbon content (%C) and the "equivalent" carbon content ($\%[C + 12/16 O + 12/14 N]$).

4.4.1.3.4 Pellet Fabrication

The commercial pelletizing process for large quantities of material must be developed. Such variables as ram pressure versus green density, binder and/or sintering aid additions, etc., must be identified and assessed so that product quality can be controlled.

4.4.1.3.5 Sintering Process Development

Production of pellets of high density (about 98 percent TD) will be the aim of the sintering process. The effect of initial green density on final pellet density will be investigated. The role of binders and sintering aids on pellet densification and phase stabilization will also be investigated. Prime emphasis will be placed on the Ni sintering aid since it provides also the desired monocarbide plus sesquicarbide structure. Processing variables such as temperature, atmospheric composition and time will have to be defined.

4.4.1.3.6 Pellet Characterization

The following analyses will be run on a statistically significant number of pellets from each batch. This statistically significant number will be defined as required.

- a) Chemical analysis for actinide elements;
- b) Carbon analysis;
- c) Total metallic impurities;
- d) Oxygen and nitrogen analysis;
- e) Open and closed porosity;
- f) C/M and (C + O + N) M ratios;
- g) Autoradiography for plutonium homogeneity;
- h) X-ray diffraction for solid solution formation;
- i) Metallography for microstructure.

4.4.1.3.7 Process Control

The following effects will be investigated as required for overall process control:

- a) The effect of various metallic impurities;
- b) The effect of various nonmetallic impurities (e.g., O_2 , N_2 and H_2);
- c) Selection of techniques for geometry and density control;
- d) Selection of methods for stoichiometry control;
- e) Methods of impurity control.

4.4.1.3.8 Thermal Property Evaluation

Ex-reactor thermal tests will be performed on the mixed carbide pellets merely to confirm already available data. These tests are not intended to be extensive. These will include:

- a) Thermal conductivity determination;
- b) Melting point analysis.

4.4.1.3.9 Physical Property Evaluation

Ex-reactor mechanical properties such as compressive creep will be determined to establish the slumping characteristics of the fuel column which may occur in-reactor. In addition, tests must be run to determine the compatibility of the fuel with cladding materials and bonding material (i.e., liquid sodium).

4.4.1.3.10 Irradiation Testing

To be discussed in a later subtask.

4.4.1.3.11 Additional Efforts

Major programmatic emphasis above has been placed upon the development of the fuel which provides a solution to the carbon transport problem. In the unlikely event that surprises are encountered during in-reactor testing,

emphasis will be placed upon other processing routes. The most likely second contender is the hydrogen reduction process to provide stoichiometric material.

Considerable effort will also be expended in providing a stoichiometry-yield-processing cost relationship which will be helpful in scoping additional fabrication development effort.

Effort will also be expended on the development of large scale fabrication of uranium carbide for both the axial and radial blankets.

Fluidized bed products will be characterized and pellets made from powder prepared by this process will be evaluated after feasibility of the ANL process has been demonstrated.

4.4.2 Subtask II: Fuel Element Vent Development

4.4.2.1 Objectives

- 1) To develop a vent mechanism which will eliminate fission gas pressure buildup within the rod as a result of release of fission gases after adequate holdup times to permit effluent activity to be no greater than that of Na-22.
- 2) To demonstrate via statistical testing programs that the vent concept is highly reliable.
- 3) To develop specifications and fabrication procedures for successful vent concepts.

4.4.2.2 Discussion

The vent mechanism must be capable of:

- 1) Elimination of fission gas pressure buildup and resultant elimination of this contribution to cladding stress;
- 2) Prevention of sodium-bond expulsion and subsequent gas blanketing of the fuel in the event of a cladding rupture at a point below the bottom of the fuel column. Out-of-pile tests⁽⁴⁾ have incorporated porous ceramic plugs containing dispersions of silver and silver halides at the top of the fuel elements. The fission product halides (iodine and bromine) combined with the silver and were retained in the ceramic plug. The fission produced alkali elements (cesium and rubidium) reacted with the silver iodide to form alkali iodide and silver. Such a plug concept will eliminate the possibility of plugging of the vent mechanism or release to the sodium coolant by deposition of the fission products.

Although the fulfillment of all of the economic and fuel conservation advantages offered by the mixed carbides is not totally dependent upon the successful development of a vent concept, the demonstration of a successful concept will permit designs and costs consistent with those used in this study. The alternative, of course, is to utilize a fission gas plenum which we consider as undesirable and unrequired since we are convinced that a highly reliable and functional vent mechanism can be developed and demonstrated if adequate attention is addressed to this area. Therefore, we believe that an adequate vent mechanism can be demonstrated at the time that a commitment is made for an LMFBR demonstration plant which is fueled with mixed carbide.

4.4.2.3 Program

The development program should be initiated with a vigorous design and ex-reactor testing effort to establish a viable vent design. This effort should address itself to the development of appropriate vent designs meeting the following performance requirements:

- 1) Release of fission gases with reduced activity into the coolant following a controlled in-vent holdup time;
- 2) Prevention of volatile fission products (Cs and I) from sealing the gas outflow system;
- 3) Accommodation of sudden large gas releases from the fuel during transient conditions without straining the cladding;
- 4) Provision for reliable vent design with effective in-reactor operating life of 25,000 hours.

An initial concept of a vent mechanism worthy of further development consists of two components in series; a porous plug for the entrapment of volatile alkali and halide fission products (primarily Cs and I), and a capillary or diving bell component for controlled time retention of fission gases prior to release into the sodium coolant.

The porous plug evaluation will consist of the determination of the efficiency of volatile product trapping, its chemical and physical stability under isothermal and thermal cycling conditions and compatibility with the cladding, fuel and sodium.

The capillary and diving bell components will be evaluated under isothermal and thermal gradient conditions. Emphasis will be made on establishing an adequate size for these components to retain fission gases for adequate time prior to release and for provisions to accommodate large volumes of fission gases released during transient conditions. The capability of these devices to perform without plugging of the vent system will be established. Additional ex-reactor tests will be performed in sodium loops in which the sodium inventory will be adjusted to provide oxygen and other impurity levels consistent with the maximum expected in a large operating LMFBR system.

All of the above effort will be on the bases of ex-reactor tests with fission gas release systems incorporating release rates and forms typical of those expected under in-reactor conditions. A statistically significant number of vent mechanisms will be tested under simulated in-reactor steady state and anticipated transient conditions.

It is important that a viable vent design be developed very early in the program. A design which will work is needed for inclusion in early irradiation efforts. It is imperative that a "reference design" be evolved early so that the irradiation testing of fuel rods can be used also to demonstrate the function of the "reference" vent.

Early testing will be performed in loops already in existence at the Liquid Metal Engineering Center. Later testing will be performed in loops which are especially constructed for the carbide fuel element engineering effort.

Irradiation testing of these vent concepts is described in a later subtask.

4.4.3 Subtask III: Sodium Bond Development and Testing

4.4.3.1 Objectives

- 1) To develop a method for economically producing a sodium bond between mixed carbide fuel and the stainless steel cladding.
- 2) To develop nondestructive testing methods for determining the efficacy of the bond.
- 3) To demonstrate that the heat transfer capability of the bond will be retained under anticipated steady state and transient operating conditions.

4.4.3.2 Discussion

The UNC method⁽⁵⁾ of sodium bonding consists of loading a sodium pellet at the bottom of the fuel pellet column, heating the loaded fuel pin to approximately 800°C and vibrating until eddy current scanning indicates the presence of a continuous bond over the entire surface of the fuel pin. Westinghouse⁽⁶⁾ has experienced limited success using vibratory methods to remove voids from the sodium bond during the assembly of fuel pins and is evaluating the ANL method of impact bonding developed for EBR-II sodium-bonded fuel pins. LASL⁽⁷⁾ is centrifuging to remove bubbles from the sodium bond. Nondestructive bond evaluations are generally being performed by eddy current testing. Any void of approximately 0.075 inch in diameter and thickness greater than about 0.001 inch can be detected. This is adequate since the calculated bubble size that can cause serious heat transfer problems is on the order of 0.2 inch in diameter. Eddy current tests are not adequate for the identification of nonwetted conditions or very thin voids (less than 0.001 inch thick). For these conditions, ultrasonic inspection with a lithium sulfide crystal may be adequate. No major problems in fabricating fuel with an integral sodium bond are anticipated.

Current technology, however, is very meager on bond performance evaluations. The EBR-II fuel is sodium-bonded, but it is not prototypical of LMFBR fuel. Considerable useful information will be obtained from the limited irradiations of sodium-bonded mixed carbide rods being performed by UNC, Los Alamos and Westinghouse. However, emphasis must be placed upon burnups in excess of 50,000 Mwd/MTM since it is in these burnup regions that an increase in the amount of fission gas released from carbide is encountered. Sudden transients will increase the rate of fission gas release. The influence of these transients on bond performance needs to be assessed.

4.4.3.3 Program

Although there is a great urge to perform ex-reactor tests on the ability of fission gas bubbles to migrate through the sodium as a function of release rate, temperature, total gas content, sodium annulus thickness (beginning, middle, and end-of-life), and pellet stack configuration, there is considerable question concerning the validity of these ex-reactor tests. Nevertheless, a

limited number of tests will be performed ex-reactor to establish, in a cursory manner, some of the conditions which might be conducive to fission gas bubble formation and retention in the sodium bond.

The limited ex-reactor program will provide for the introduction of gases at the bottom of a fuel column by degassing a porous ceramic plug or by direct bleeding of gas into the bottom of a fuel rod. These tests will be performed with varying pellet to cladding gaps. Tests will be performed with various surface conditions of the pellet and cladding and with various traces of impurities in the sodium, such as oxygen, nitrogen, carbon, cesium and iodine. The object of this test is to determine whether a gas bubble can develop and stagnate in a localized area to produce a cladding hot spot which is sufficient to cause cladding damage. The effect of pressure buildup followed by sudden depressurization will also be investigated. This condition could occur by means of the plugging of a vent followed by sudden clad failure or unplugging.

Major programmatic effort on bond efficiency will be centered upon the results of irradiation tests and will be discussed in a later subtask.

A second effort on this subtask will address itself to the development and testing of economical and reliable methods for providing a sodium bond during rod fabrication as well as the development of reliable nondestructive methods for determining bond integrity.

4.4.4 Subtask IV: Carbon Transport

4.4.4.1 Objectives

- 1) To verify ex-reactor results indicating a solution to the carbon transport problem through use of hyperstoichiometric mixed carbides containing monocarbide and discontinuous second phase sesquicarbide.
- 2) To evaluate alternate solutions to the carbon transport problem in the unlikely event that problems are encountered with the use of simple structure control of monocarbide plus sesquicarbide in hyperstoichiometric compositions.
- 3) To establish the maximum equivalent carbon content limit in the fuel which is compatible with acceptable carbon transport and with economical fuel fabrication methods.
- 4) To verify that uranium carbide for the radial and axial blankets is amenable to a similar solution to the carbon transport problem.

4.4.4.2 Discussion

The motivation for this portion of the program is based upon the premise that sodium bonding is essential if mixed carbides are to achieve their maximum thermal potential. It is our belief that carbon transport from the fuel to the cladding, via sodium represents one of the major barriers to the utilization of the sodium-bonded, hyperstoichiometric mixed carbide fuel concept. We believe that a solution to the carbon transport problem in sodium-bonded mixed carbide fuels can be achieved. Of the concepts discussed earlier, we believe that major programmatic efforts should be expended on the development and testing of mixed carbides containing controlled sesquicarbide-monocarbide phases on the bases that doubling times are not altered and fabrication cost penalties are essentially zero since these structures are readily produced in a sintered product which contains 0.1 w/o Ni as a sintering aid. Hydrogen reduction represents another promising solution to the problem and has been relegated to the role of a backup effort. In this latter respect, the Los Alamos program will be followed with great interest.

The approach which has been selected as the basis for this design with respect to minimizing carbon transport to the cladding is one which relies on structure control within the hyperstoichiometric mixed carbide. The higher carbon phase is the sesquicarbide which is discontinuous within the monocarbide matrix. Carbon transport from this type of structure at anticipated LMFBR temperatures has been found to be quite small and acceptable.

4.4.4.3 Program

A series of isothermal and thermal gradient ex-reactor tests will be performed to verify results already obtained at other installations. The aim will be to demonstrate that the fuel fabricated as part of this development effort exhibits carbon transport characteristics typical of those encountered in current programs. Furthermore, thermal gradient tests will provide indications of whether this parameter is a factor. As part of Subtask I, limits with respect to economical control of carbon contents will have been established. It is imperative that these upper limits be evaluated with respect to carbon transport potential. An iterative effort may be required if borderline conditions are

encountered. The objective here is to establish an upper limit of carbon content amenable to control of carbon transport.

In the unlikely event alternate solutions to the problem become necessary as a result of either ex-reactor or in-reactor test results, the development of these alternate solutions will be given high priority. Therefore, considerable emphasis will be placed upon the results of irradiation tests of sodium-bonded elements being performed by UNC, Los Alamos and Westinghouse. Although the number of in-reactor tests are less than could be considered as statistically significant, the results will establish trends which will be recognized in the pursuit of this subtask. Obviously, if major deviations from contemplated processing schemes are dictated by these early irradiation results, major changes in the process development subtask will also be required. Conversely, as advanced or more economically attractive fuel processing schemes are evolved as part of Subtask I, these will be evaluated with respect to their potential impact on carbon transport and testing will be initiated on fuel concepts involving major deviations from those already tested.

There is some indication that uranium carbide exhibits a greater tendency to transfer carbon than does the mixed carbide even if the same monocarbide-sesquicarbide structures are developed. Tests will be performed on UC and, if differences in performance are found, the variation in performance as a function of Pu content will be assessed.

Final proof testing will be required via extensive in-reactor programs described as part of Subtask V.

4.4.5 Subtask V: Irradiation Testing

4.4.5.1 Objectives

- 1) To determine free swelling rates for mixed carbide fuels and to determine whether there is a burnup dependency.
- 2) To demonstrate the adequacy of the sodium heat transfer bond.
- 3) To evaluate carbon transport from the fuel to the cladding at high burnups and heat ratings.

- 4) To demonstrate the adequacy of the vent design.
- 5) To demonstrate fuel rod and bundle performance at peak burnups in excess of 100,000 Mwd/MTM and with linear heat ratings up to 45 kw/ft.

4.4.5.2 Discussion

Significant information on the performance of the sodium-bond, carbon transport from the fuel to the cladding and performance of the vent can be obtained from properly designed irradiations in thermal fluxes. Information will also be obtained with respect to free swelling and fission gas release. However, more applicable data concerning fuel rod and fuel assembly performance will be obtained as a result of irradiations in EBR-II even in light of limitations discussed in earlier parts of this section. Therefore, the irradiation program addresses itself to parallel efforts in both thermal and fast fluxes.

Although the program description below emphasizes burnup and linear heat ratings as target parameters, most rods will contain vents and sodium bonds and will be evaluated with respect to the ability of the vent to function as well as the efficacy of the sodium bond to provide adequate heat transfer under fission gas release conditions associated with steady state and transient conditions. In addition, carbon transport from the fuel to the cladding will be assessed.

The confirmation of adequate vent performance during in-reactor tests will require special capsule designs and, in the case of EBR-II, may require the design of fission gas containers to prevent release within EBR-II.

It is anticipated that EBR-II irradiations can be performed without a capsule requirement after some confidence in the mixed carbide performance in EBR-II has been established as a result of successful irradiations.

4.4.5.3 Program

4.4.5.3.1 Thermal Irradiations

The target burnup for most of the thermal irradiations will be 75,000 Mwd/MTM on the basis that this burnup is sufficiently beyond the threshold burnup for high gas release and that the functions of the vent and the sodium heat transfer bond will have been adequately tested under high fission gas

release conditons. In addition, free swelling data will be obtained and will require minor corrections for differences in radial burnup distribution within the fuel. An earlier assessment of the performance of these carbide fuels will be obtained via the irradiation of a limited number of rods to peak burnups of 25,000 and 50,000 Mwd/MTM.

Approximately 95 rods will be irradiated under the following target conditions:

Thermal Irradiations

<u>Series No.</u>	<u>No. of Rods</u>	Peak Heat Rating****	Peak Burnup****
		<u>(kw/ft)</u>	<u>(Mwd/MTM)</u>
1*	5	30	25,000
	5	40	25,000
	5	30	50,000
	5	40	50,000
	15***	30	75,000
	15***	45	75,000
2**	15***	30	75,000
	15***	40	75,000
	15***	45	75,000

* Fabricated on basis of current UNC technology.

** Fabricated by pilot plant technology. However, not fabricated in pilot plant.

*** Five of these rods are to be purposely overpowered at 1/3, 2/3 and at end of life. The amount of overpower will be equivalent to that anticipated in the plant.

**** There will be a range within the rod as a result of axial peaking.

4.4.5.3.2 Fast Flux Irradiations

Irradiations in EBR-II will be aimed at demonstrating burnup and heat rating performance capability of fuel rods and fuel assemblies under fast flux conditions. Furthermore, free swelling rates for the fuel will be determined and the efficacy of the vent and sodium heat transfer bond will be demonstrated under steady state and transient conditions. Determinations will also be made of carbon transport from the fuel to the cladding as well as fission gas release.*

The fundamental parameters which will be used as a basis for the design of these irradiation samples will be obtained from results of programs already under way at Los Alamos, UNC and Westinghouse as well as results obtained in England and France. Therefore, an early commitment in EBR-II can be made with little undo risk.

The program, as currently envisioned, is summarized below. Some of the test conditions may be changed as the program evolves. However, it is likely that the number of rods irradiated will remain the same. In the summary provided below, identification has not been made of fuel assemblies. A number of these will be irradiated and the rods to form these fuel assemblies have been included in the summary.

* For later use in safety analyses

Fast Flux Irradiations

<u>Series No.</u>	<u>No. of Rods*</u>	<u>Peak Linear Heat Rating**</u>	<u>Peak Burnup**</u>
		<u>(kw/ft)</u>	<u>(Mwd/MTM)</u>
1***	25	30	75,000
	25	30	100,000
	25	30	125,000
	25	40	75,000
2****	25	40	100,000
	25	40	125,000
	25	45	100,000
	25	45	125,000
3	Approximately 50 rods under conditions which are not yet definable.		
4	Approximately 50 rods under conditions which are not yet definable.		

* Some of these will be in the form of fuel assemblies. Some rods will be overpowered at 1/3, 2/3 and at end of life.

** There will be a range within the rod as a result of axial peaking.

*** Fabricated on the basis of current UNC technology.

**** Fabricated by pilot plant technology. However, not fabricated in pilot plant.

4.4.6 Subtask VI: Program Management

4.4.6.1 Objectives

- 1) To provide overall technical and administrative coordination and management for the Fuels and Materials Programs.
- 2) To provide necessary liaison between design efforts and research and development programs.
- 3) To remain abreast of all pertinent activities at other installations and to ensure proper liaison between these activities.
- 4) To continuously review program objectives, results and required emphasis with the aim of redirection into high priority areas.
- 5) To update the program plan and budget on a yearly basis.

4.4.6.2 Program

We believe that a Fuels and Materials Project Office should contain three engineers charged with the responsibility of meeting the above objectives. The senior individual responsible for all of the Fuels and Materials Program efforts will manage this office and function from this office.

REFERENCES

for Section 4.0

1. R. C. Noyes, et al, ANL-7520, Proceedings of International Conference on Sodium Technology and Large Fast Reactor Design, "Development and Evaluation of the C-E Advanced 1000 Mwe LMFBR Design", Nov. 1968
2. D. Stahl, J. Anderson, "Irradiation of High Density (U Pu)C in EBR-II - Design and Fabrication Report of Rods 81 Through 90", UNC-5170, March 24, 1967
3. K. M. Taylor, C. H. McMurtry, J. C. Anderson, "Sintering Characteristics of U-C and (U Pu)C With and Without Small Additions of Nickel" in Carbides in Nuclear Energy, Vol. 2, pp. 668-679, Macmillan and Company, Ltd., London, 1964
4. H. Beisswenger, S. Dorner, "Retention of Volatile Fission Products in a Vented Fuel Element", EURFNR-437, October 23, 1967
5. Quarterly Report, UNC 6201, February 10, 1968
6. Quarterly Report, WARD-3791-21, June, 1968
7. Quarterly Status Report on the Advanced Plutonium Fuels Program, LA-3933-MS, January 1 to March 31, 1968

Figure 4-1
Fuels and Materials R and D Schedule and Milestone Chart

Figure 4-1: Fuels and Materials Development Milestone Legend

1. Complete Fuel Fabrication Process Evaluation and Development.
2. Complete construction of Fuel Fabrication Pilot Plant.
3. Pilot plant shakedown completed. Ready to fabricate.
4. Complete Development of the most promising alternate Fuel Fabrication Process.
5. Complete the design, the fabrication and the testing of four possible vent concepts and choose reference design.
6. Testing of reference design is completed.
7. Complete reference design refinements.
8. First stage of fabrication development completed.
9. Sodium-bond fabrication development and testing is completed.
10. Complete bonding process improvement and cost reduction program.
11. Complete isothermal and thermal gradient ex-reactor tests.
12. Complete tests of the backup carbon transport preventive technique.
13. Complete Series 1 thermal irradiations to burnups of 25,000 Mwd/MT.
14. Complete Series 1 thermal irradiations to burnups of 50,000 Mwd/MT.
15. Complete the Series 1 thermal irradiations to burnups of 75,000 Mwd/MT.
16. Complete the Series 2 thermal irradiations to burnups of 75,000 Mwd/MT.
17. Complete Series 1 fast irradiations to 75,000 Mwd/MT burnup.
18. Complete Series 1 fast irradiations to 100,000 Mwd/MT burnup.
19. Complete Series 1 fast irradiations to 125,000 Mwd/MT burnup.
20. Complete Series 2 fast irradiations to 100,000 Mwd/MT burnup.
21. Complete Series 2 fast irradiations to 125,000 Mwd/MT burnup.
22. Complete the Series 3 irradiations.
23. Complete the Series 4 irradiations.

Table 4-1. Summary of Estimated Costs
Calendar Years 1970 through 1975

	Total A Man-years	Total B Man-years	Eng. & Scient. Lab. + O/H	Tech, Draftsmen, Others Lab. + O/H	\$ ODC	Total (\$) Based on 1969 (\$) No escalation (\$ x 10 ⁻⁶)
Subtask I: Dev. of Fuel Fab. Proc.	42	90	1,680,000	1,800,000	3,350,000	6.83
Subtask II: Fuel Element Vent Dev.	13	26	520,000	560,000	240,000	1.32
Subtask III: Sodium Bond Dev. & Test.	6	16	240,000	320,000	300,000	0.86
Subtask IV: Carbon Transport	4	6	160,000	120,000	130,000	0.41
Subtask V: Irradiation Testing*	49	92	1,960,000	1,840,000	3,230,000	7.03
Subtask VI: Program Management	12	6	480,000	120,000	60,000	0.66
TOTAL						17.11

* Will require \$100,000 in Calendar Year 1969

Table 4-2. Detailed Program

(Continued)

Table 4-2. Detailed Program (Cont.)

	Calendar Year						
	1970	1971	1972	1973	1974	1975	1976
Subtask II: Fuel Element Vent Development (Ex-Reactor Effort only)							
Detailed mechanical design	→	→	→				
Fabrication of Mark I	→						
Testing of Mark I		→					
Fabrication of Mark II		→					
Testing of Mark II		→					
Fabrication of Mark III		→					
Testing of Mark III		→					
Fabrication of Mark IV		→					
Testing of Mark IV		→					
Reference Design		→					
Fabrication of Reference Design			→				
Testing of Reference Design			→				
Design Refinements*				→			
Testing of new designs*				→			

(Continued)

* only if needed

Table 4-2. Detailed Program (Cont.)

	Calendar Year					
	1970	1971	1972	1973	1974	1975
Subtask III: Sodium Bond Development and Testing (Ex-Reactor Effort)						
Evaluation of sodium bonding methods	→					
Development of nondestructive test apparatus for bond voids		→				
Fabrication development and testing	Devel. →		Test →			
Ex-Reactor testing for fission gas release effects		→				
Process improvement-cost reduction				→		
Subtask IV: Carbon Transport						
Isothermal tests (M & S)*		→				
Thermal gradient tests (M & S)*		→				
Tests of alternate fuels**				→		
Tests of possible backups			→			

(Continued)

* Mono plus sesquicarbide

** Based on results of Subtask I effort

Table 4-2. Detailed Program (Cont.)

	Calendar Year								
	1969	1970	1971	1972	1973	1974	1975	1976	1977
Subtask V: Irradiation Testing									
<u>THERMAL</u>									
Evaluate facilities	→								
Design capsules and samples	→								
Safeguards		→							
<u>Series 1:</u>									
Fabricate samples and capsules		→							
<u>Irradiate plus Post irradiation exam and test</u>									
25,000 Mwd/MT			Irrad. Test.	→					
50,000 Mwd/MT				→	→				
75,000 Mwd/MT						→	→		
<u>Series 2:</u>									
Fabricate samples and capsules				→					
<u>Irradiate plus Post irradiation exam and test</u>								→	→
<u>FAST</u>									
Evaluate facilities	→								
Design capsules and samples	→								
Safeguards	→								

(Continued)

Table 4-2. Detailed Program (Cont.)

		Calendar Year								
		1969	1970	1971	1972	1973	1974	1975	1976	1977
<u>Series 1:</u>										
Fabricate samples and capsules			→							
<u>Irradiate plus Post irradiation exam. and test</u>										
75,000	Mwd/MT					→	→			
100,000	Mwd/MT						→	→		
125,000	Mwd/MT						→	→		
<u>Series 2:</u>										
Fabricate samples and capsules					→					
<u>Irradiate plus Post irradiation exam and test</u>										
100,000	Mwd/MT							→	→	
125,000	Mwd/MT								→	→
<u>Series 3:</u>										
Fabricate samples and capsules						→				
<u>Irradiate plus Post irradiation exam and test</u>										
								→ To 1979 →		
<u>Series 4:</u>										
Fabricate samples and capsules							→			
<u>Irradiate plus Post irradiation exam and test</u>										
								→ To 1980 →		

Table 4-3. Yearly Costs by Subtask
A and B in Man-years

Calendar Year	1970			1971			1972			1973			1974			1975		
	A	B	ODC (\$)	A	B	ODC (\$)	A	B	ODC (\$)	A	B	ODC (\$)	A	B	ODC (\$)	A	B	ODC (\$)
	(A)			(A)			(A)			(A)			**	(A)		**	(A)	
Subtask I	5	10	250,000	8	15	500,000	8	15	300,000	5	10	1,700,000*	8	20	\$300,000	8	20	\$300,000
Subtask II	4	8	100,000	2	6	50,000	2	6	25,000	2	3	25,000	2	3	25,000	1	2	15,000
Subtask III	2	6	100,000	2	5	100,000	1	3	50,000	1	3	50,000	--			--		
Subtask IV	1	2	50,000	1	2	30,000	1	1	25,000	1	1	25,000	--			--		
Subtask V ^(B)	13	28	520,000	2	2	100,000	12	32	650,000	5	5	500,000	8	13	940,000	8	11	480,000
Subtask VI	2	1	10,000	2	1	10,000	2	1	10,000	2	1	10,000	2	1	10,000	2	1	10,000

(A) It is assumed that Pu will be supplied by the AEC with no direct cost to the program. Therefore, costs for Pu are not included.

* Includes \$1,500,000 for pilot plant facility

** Assumes use of pilot plant for production of LMFBR demonstration plant core loadings

(B) Will need 2A and 2B and \$40,000 ODC during the second half of Calendar Year 1969. This is equivalent to 1A man-year + 1B man-year. (Total \$ = \$100,000)

5.0 SAFETY

5.1 INTRODUCTION

The AEC LMFBR Program Plan for Safety⁽¹⁾ has been reviewed in light of safety studies and requirements of the C-E reference LMFBR design. With the exception of its almost exclusive implementation with gas-bonded oxide fuels, the plan was found to be generally appropriate and to the point in terms of identification of problems and plans for their solution. It would not be unreasonable, therefore, to simply endorse the Program Plan as it stands, with the qualification that implementation of the investigations involving sodium-bonded carbide fuel ought to be given a substantially higher priority. This is, in fact, C-E's general and most important conclusion and recommendation with respect to LMFBR Safety research and development.

In view of this general conclusion, it would not be useful to repeat the development of detailed programs for all areas covered in the AEC LMFBR Program Plan. Rather, particular areas have been selected, either because they are closely associated with and affected by the particular characteristics of the C-E design or because they are judged to be particularly important.

The identification and definition of Potential Safety Issues in the LMFBR Program Plan (page 10-43) is especially good and provides an excellent reference point for relating the particular items selected to the overall LMFBR Program Plan. The following tabulation of potential safety issues presents essentially the same issues identified in the Program Plan except for the issue of the Intrinsic Safety of the Reactor and its subsidiary issues. The reasons for this addition to the list are discussed below. Although essentially the same issues are involved, some reorganization and renaming has been done to reflect C-E's point of view towards these issues. Entrained Gas and the Behavior of Sodium-bonded Fuel have been pulled out as separate issues because the former is thought to be more generally significant and the latter is particularly significant to the C-E concept. Sodium Boiling, on the other hand is,

especially for sodium-bonded fuel, too closely connected with fuel and clad failure phenomena to be identified as a separate and distinct issue.

The issues are placed in four groups in which the primary issue is listed first and subsidiary and related issues follow.

Potential Safety Issues

I. Adequacy of Core Protective Systems

- Accident Definition
- Safety Instrumentation
- Probabilistic Evaluation

II. Fuel Assembly Integrity

- Propagation of Fuel Element Failure
- Sodium Boiling, Pressure Pulses and Associated Effects
- Behavior of Sodium-bonded Fuel
- Entrained Gas

III. Definition of Hypothetical Accidents

- Core Slumping
- Energy Partition and Absorption
- Sodium Boiling, Pressure Pulses and Associated Effects
- Equation of State and Fuel Properties
- Sodium Fires
- Fission Product Release
- Behavior of Molten Fuel
- Meteorological Distribution

IV. Intrinsic Safety of the Reactor

- Core Reactivity Coefficients
- Core Mechanical Stability
- Quality and Design Standards

Group IV lists issues related to the first level of safety concern described in the Program Plan (WASH-1110, page 10-34) and these issues are considered in various parts of the plan. Therefore, it was not the

omission from the Program Plan of consideration of these issues that prompted their addition to the list, but rather the desire to complete this list of issues as seen from the reactor engineering point of view. These four groups are seen as the four primary issues for which the reactor engineer needs resolution at an appropriate level depending on the reactor in question, e.g., experimental plant, demonstration plant or commercial plant. The R&D areas which were selected are directly related to these four primary issues.

The requirements of the commercial plant are the concern of this discussion; however, this does not mean that the demonstration plant is not relevant. Quite the contrary; one of the key purposes of the demonstration plant is to resolve some of these safety issues for the commercial plants that follow. This is particularly true with respect to issues involving the intrinsic safety of the reactor since judgement with respect to such factors as the adequacy of the Doppler effect cannot be made in the abstract, i.e., without consideration of all the characteristics of a particular reactor. Furthermore, the issue of Quality and Design Standards will be initially resolved only when standards which are adequate with respect to safety are also shown to be achievable in practice by the building and licensing of the demonstration plant. The demonstration plant is also the most practical and in some cases the only place for the final resolution of short term and long term system and subsystem stability and predictability questions. In a more general sense, it will be the designing, building, licensing and operating of an appropriate demonstration plant which, for that type of plant, will resolve the issue of overall safety at a confidence level which will allow similar commercial plants to follow.

Of course this is not meant to imply that the demonstration plant will resolve all safety issues; certainly the majority of the specific questions (core slumping, sodium fires, etc.) must be attacked individually with the sorts of investigations described in the Program Plan. These investigations will provide essential input to the Preliminary Safety Analysis Report (PSAR) for the demonstration plant and the licensing procedures for the plant will provide feedback to the investigators as

to the adequacy and appropriateness of their work. Continued investigation through the demonstration plant phase and concurrent with the construction of commercial plants will be essential in many areas. Few issues will be finally resolved in one step or by one self-contained program. Several steps will be involved, each of which will increase the confidence in and efficiency of the solutions. The demonstration plant is one of these steps.

The programs which are outlined in the following discussions concentrate more on those elements which must precede the licensing of the demonstration plant and/or which are proposed to be done concurrently with this phase of development. Specific contributions of the demonstration plant to safety development are also indicated in appropriate areas.

The R&D tasks and subtasks which have been selected as particularly important and relevant to the C-E design are described in the following sections. Where feasible, these have been coordinated with the Program Plan in terms of task titles and task numbers. The approach has been to identify the most appropriate task (e.g., 10-2.4, Fuel Failure and Melting) and subtask (e.g., 10-2.4.4, In-Pile Test--Steady-State Reactors) and put the main body of the proposed work under that task. In this context, it is intended that other subtasks are implied to be appropriate and adequately described. For instance, subtask 10-2.4.1, Catalog of Fuel Failure Incidents, is an appropriate part of task 10-2.4, is adequately described and need not be repeated.

5.2 CORE CONFIGURATION SAFETY STUDIES (Subtask 10-1.1.0)

5.2.1 Program Definition

All safety considerations related to the reference core configuration are to be gathered and analyzed under this task. Particular attention is to be paid to the inherent characteristics of this configuration with regard to Doppler effect, void effect, fuel behavior during accidents, structural effects, etc.; however, protective systems will also be integrated into the study. This is not an experimental task, rather it

will use results of experiments and analyses obtained from other Safety tasks and Physics, Core Design and other research and development areas.

Of necessity, studies in various specialized areas (e.g., Physics) must be generalized to a certain extent with respect to core configuration. Also, it is seldom possible to answer important safety questions without reference to a specific reactor. Thus, for example, Physics studies will provide the tools for calculating Doppler effects. In this task, these tools will be used to calculate Doppler effects for the reference core configuration (and related configurations) for various core conditions, and the required transient analysis will be made to determine whether or not the Doppler provides adequate safety margins considering all the other characteristics of this specific configuration.

By application of results of studies of specific problems to the reference configuration, this task will determine the significance of these results and will evaluate the progress of these studies and thus provide essential feedback to maximize the efficiency of research and development. This will be a continuing effort which will develop accident and transient analyses of the reference concept in terms of all available experimental and theoretical information in order that an informed judgement on the inherent safety of the concept can be made and so that determinations can be made of the effectiveness and appropriateness of various design modifications and compromises for safety. An important part of this effort will be the development of general and specific safety related design criteria for the core and protective systems.

This task also includes the study of safety questions related to the demonstration plant. Particular consideration will be given to differences in the state of knowledge on safety questions at the time of the design of the demonstration plant and the commercial plant. Design criteria appropriate for the demonstration plant will be developed. One important function of the demonstration plant will be to contribute to LMFBFR development in the safety area. Appropriate studies will be made to insure that this contribution will be maximized. This will include the definition of safety related tests to be done in the demonstration plant.

5.2.2 Description of Current Technology

The state of the art can best be summarized by reference to the safety studies reported in Volume III of the Tasks II and III Report⁽²⁾, to other safety studies, to discussions for proposed LMFBRs, and to the licensing documents for Enrico Fermi, EBR-II and SEFOR. A great deal of material has been published on LMFBR safety, most of which has some relevance to the reference concept. The bulk of this material, however, is more extensive than intensive, more general than specific. Studies of the type reported in Ref (2) should be extended in more depth. An initial set of general design criteria has been formulated but these need testing in terms of practicality, adequacy and relevance.

5.2.3 New Information Needed

The new information needed is an organized synthesis and interpretation of information gained from experience, analysis and experiments relevant to the reference concept. This information should focus on the inherent characteristics of the reactor, especially transient characteristics, considered within the context of practical protective systems and engineered safeguards. Based on the state of knowledge at a given time, and the type of plant under consideration (demonstration plant or commercial), a practical set of general and specific design criteria which assure safety is needed. This would include a definition of Design Basis Accidents. Also needed are definitions of appropriate system compromises for safety and the effects of these on reactor performance. Whether these compromises are motivated by a need to be conservative because of inadequate knowledge or whether they are essentially permanent characteristics of the system needs to be established. This sort of information will help direct research and development programs. Documentation of this synthesis of concept oriented information is needed in a form appropriate for critical evaluation by authoritative and independent safety groups (e.g., ACRS).

5.2.4 Program Description

The proposed program would be a continuing effort with two principal phases: the Demonstration Plant Phase and the Commercial Plant Phase.

Each phase will include transient studies, development of design criteria, documentation for independent review and identification of research and development.

5.2.4.1 Demonstration Plant Phase

The purposes of the Demonstration Plant Phase of the Core Configuration Safety Studies are as follows:

- 1) Identification of the role and purpose of the demonstration plant relative to development of safety of the concept;
- 2) Development of specific design criteria for the demonstration plant;
- 3) Specification of tests and observations to be made on the demonstration plant;
- 4) Documentation of the safety characteristics of the concept demonstration plant.

Transient studies will be made on various sized plants, including commercial sizes. Variations on the basic core configuration will be studied including various degrees of pancaking and various amounts of moderator. These will include sensitivity analyses relative to uncertainties in input parameters. Those characteristics which are sensitive to plant size and core configuration will be identified. These studies will aim at the general question of the role of the demonstration plant by comparative studies of early and later plants and evaluation of alternate paths leading to the commercial plant.

In the light of the long term objectives of LMFBR development and the state of knowledge at various points in time, these studies will consider the question of conservatism in the demonstration plant. Attention will be focused on the balance of conservatism between inherent characteristics (Doppler, power density, void effect, etc.) and protective systems, engineered safeguards and siting. The contributions of other LMFBR demonstration plants will be considered in this evaluation. Since it is assumed that the sodium-bonded carbide concept will not be used in the first LMFBR demonstration plant, it will be appropriate to

consider the complementary roles of prior plants. For instance, the first plant may have a soft spectrum core in order to be conservative with respect to the Doppler effect prior to in-service demonstration of the response and reliability of protective systems. A later demonstration plant may consider that the response and reliability of protective systems will have been demonstrated sufficiently to allow a hard spectrum core. On the other hand, it may be determined that the first hard spectrum core requires extra redundancy of protective systems because its inherent stability has not been sufficiently proven in service. Implicit in these questions is one of the possible roles of the demonstration plant, i.e., in this example, demonstration of the safety and practicality of a hard spectrum LMFBR. By examination of the safety and other implications of this and other possible plant characteristics (positive void effect, high power density, etc.), the appropriate balance can be struck which is consistent with safety based on the state of the art at the time of the demonstration plant and which makes maximum contribution to advancement toward the goals of the LMFBR development program.

Specific design criteria will be developed partly as a result of the studies outlined above and partly in parallel to these studies. Starting from the general design criteria now being developed for LMFBRs, specific criteria will be formulated and examined for practicality, adequacy and consistency with the features and purposes of the plant. These criteria will constitute a major section of the documentation of the safety of the demonstration plant. Other major sections of the safety document will be design descriptions of safety-related features and the results of transient analyses.

Safety-related tests and observations to be made on the demonstration plant must be carefully planned in advance in order that the necessary facilities and instruments are provided, so that qualified and authorized persons will be in the right place at the right time and so that interference with orderly startup and operation of the plant is minimized. Tests must be justified and alternate methods of acquiring the data evaluated.

Studies in this area will concentrate on the following items which are most important and most appropriate.

1. Statistical reliability data
 - a) Components
 - b) Protective system
 - c) Fuel system
2. Core stability data
 - a) Power, temperature and flow reactivity coefficients
 - b) Mechanical stability
 - c) Fuel burnup effects on stability (fuel redistribution, subassembly bowing, regional power shifts, etc.)
3. Operational data
 - a) Verification of predictions - reactivity, radiation levels, temperatures, etc.
 - b) Operational flow, temperature and reactivity transients - magnitude and frequency
 - c) Cover gas activity and cleanup system efficiency, activity releases, etc.
4. Other special tests
 - a) Tests of local flow blockage detection systems and other in-core instruments
 - b) Emergency cooling systems tests

Test observations will be categorized as ordinary (usual for any reactor) or extraordinary (special because the reactor is first-of-a-kind). The test program will be documented including schedules, costs and justifications.

5.2.4.2 Commercial Plant Phase

The purposes of the commercial plant phase of the core configuration studies are as follows:

- 1) Development of specific design criteria for the commercial plant;

- 2) Identification of safety research and development required to support licensing of commercial plant;
- 3) Documentation of the safety characteristics of the commercial plant.

The Commercial Plant Phase will start at a low level parallel to the Demonstration Plant Phase and will serve as a point of reference for demonstration plant work. The later commercial plant work will depend to a considerable extent on what sort of demonstration plant is eventually evolved. An unlikely but certainly desirable state of affairs might be that the demonstration plant would not be significantly compromised for safety. In this case the commercial plant studies would be limited to investigations of the effects of size, factoring in results of experience with demonstration plants and latest theoretical and experimental developments, and perhaps some inquiring into siting questions. It is more likely, however, that the commercial plant will represent an advancement over the demonstration plant in several areas related to safety (e.g. higher power density fuel, more regular core geometry, less moderator, new in-core instruments and protective systems, etc.). Each such change will be evaluated from the safety point of view in the light of the current state of the art. An important product of studies of this nature will be a carefully reasoned and documented projection of which advancements should be permissible. This will reduce uncertainties in establishing the inherent characteristics of the commercial plant prior to licensing. These advances would be reflected in up-dated specific design criteria for the concept.

5.3 FUEL FAILURE AND MELTING (Task 10-2.4)

5.3.1 Program Definition

Transient and steady state fuel failure thresholds and mechanisms are to be established for sodium-bonded carbide fuel. Normal failure* mechanisms are to be established as part of Fuels and Materials Research and Development with close coordination with this subtask. The effect

* Failures that occur under normal operating conditions.

of various operational and accidental transients on integrity of fuel is to be established. Mechanisms and thresholds of fuel failure under severe accident conditions will be established. Conditions under which single pin failures will tend to propagate to adjacent pins will be determined.

5.3.2 Description of Current Technology

An excellent general review of the state-of-the-art in this area is contained in the LMFBR Program Plan⁽¹⁾ and need not be repeated here. The significant points for sodium-bonded carbide can be summarized as follows.

1. No irradiation testing of sodium-bonded carbide elements which have experienced power or temperature transients has been done. No theoretical mechanisms for fuel failure due to transients short of fuel melting or bond boiling have been identified for this fuel; therefore, it is not now possible to set meaningful limits on operation of fuel which has experienced transients in this range. Since quite severe transients are possible without the immediate damaging effects of fuel melting or bond boiling, there is a wide "gray" range which needs investigation to determine if failure mechanisms exist.
2. Only one TREAT experiment involving sodium-bonded carbide has been performed.⁽³⁾ This experiment suggests that, for certain power transients, the failure mechanism for this fuel is different than for gas-bonded ceramic fuels. In this case, vaporization of the bond apparently caused the fuel to become thermally insulated from the clad prior to fuel melting. Then when fuel melting progressed to the point where contact with the clad occurred, rapid, general and rather violent dispersion of fuel and clad occurred. This was attributed to high localized sodium vapor pressure caused by very hot fuel and clad coming into contact with the sodium coolant. Similar experiments with sodium-bonded metal fuels have not shown similar failure patterns.⁽⁴⁾
3. The problem of failure propagation has not received much experimental attention for any fuel system and none for the sodium-bonded carbide system. In one related experiment,⁽⁵⁾ loss of the sodium bond occurred in a uranium carbide specimen during an irradiation test. Increased temperatures caused accelerated swelling to the point where fuel contacted clad and reduced fuel temperatures. Although clad failure did not occur before the termination of the experiment, it was

predicted that it would occur after an additional exposure of 5,000 to 10,000 Mwd/MT. Fuel power density during this test was sufficiently low, however, that fuel melting apparently did not occur after loss of the bond. Theoretical predictions of fuel temperatures after loss of the bond in the reference LMFBR design indicate that melting would occur except near the end of life when the fuel-clad gap is small.

5.3.3 New Information Needed

Failures and failure mechanisms associated with normal operation will be studied under the Fuels and Materials research area. However, in order to prevent failures under normal operation which could be caused by previous off-normal operation, there needs to be a study of the effects of operational and accidental transients on subsequent fuel behavior. Part of the studies in this area will be done under this subtask. This is one of the three topics listed below on which new information is needed.

<u>Topic</u>	<u>Information Needed</u>
1. Effect of accidental transients on fuel integrity. (Series I Tests)	Failure history and condition of fuel pins which have experienced a short term coolant temperature transient approaching bond boiling and a short term power transient approaching bond boiling and/or fuel melting.
2. Propagation of single pin failures. (Series II Tests)	a) Behavior of a fuel pin cluster in which bond loss has occurred in a single pin. b) Behavior of a fuel pin cluster in which clad rupture has occurred in a single pin due to excessive fuel swelling.
3. Mechanisms of fuel failure in unprotected transients and propagation from plugging of a single subassembly. (Series III Tests)	a) Behavior of a single fuel pin in flowing sodium during a flow transient and a power transient severe enough to cause bond boiling and fuel melting. b) Behavior of a cluster of fuel pins under above circumstances.

The investigations listed under topic 3 have two objectives which are combined because essentially the same experiments are proposed to obtain the needed information. The plugging of a single subassembly is not an unprotected accident; however, it leads to the same local conditions as a general pump failure without protective system action. In the case of the latter accident, the information is needed in order to allow fairly realistic prediction of the ultimate outcome of such an accident. For the case of subassembly plugging, there must be assurance that unmanageable propagation to adjacent subassemblies will not occur before the event is detected and the reactor is shut down.

5.3.4 Program Description

5.3.4.1 In-Pile Tests - Steady State Reactors (Subtask 10-2.4.4)

The objectives of the in-pile tests in steady state reactors will be to provide the bulk of the new information needed which is listed under topics (1) and (2) above. For tests related to topic (1), preconditioning of some test samples would be done in EBR-II and/or a transient reactor (TREAT) prior to steady state testing. The initial steady state tests would be done in thermal test reactors. Later, confirmatory tests would utilize both FFTF and Safety Test Facility (STF).

5.3.4.1.1 Series I Tests

It is apparent that if sodium bond boiling and/or fuel melting occur as a result of any accident other than a local coolant restriction, a potentially unsafe situation would result. However, there is a large margin of safety between normal operating conditions (peak fuel temperature = 2300°F , peak bond temperature = 1300°F) and the fuel melting point (4300°F) and the bond boiling point (1700°F). Because of this large margin, rather severe accidental power or flow transients could occur without producing either bond boiling or fuel melting; however, such transients may produce other effects in the fuel, bond or clad (excessive swelling, fission gas release, etc.) which would leave the core in a potentially unsafe condition.

The AEC General Design Criteria for Nuclear Power Plant Construction Permits⁽⁶⁾ require that, "core protection systems together with associated equipment, shall be designed to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits." Series I tests are aimed at definition from the safety viewpoint of acceptable fuel damage limits for sodium-bonded carbide and to thus provide essential benchmarks for the design of core protective systems and for safety evaluation.

The initial test program is based on the assumption that these limits are near-to but short-of the boiling point for the bond and the melting point for the fuel. Initial tests would therefore be designed to reach peak fuel temperatures of $\sim 4000^{\circ}\text{F}$ and/or peak bond temperatures of $\sim 1600^{\circ}\text{F}$. (Typical overpower transients starting from full power reach these conditions in 1 to 10 sec depending on the rate of reactivity insertion (see ref 2).) The main variables in these tests are the exposure (burnup) of the fuel at the time of the transient and the type of transient (power or flow). Since the power transients are more severe, these would be studied first. Details of subsequent tests would depend on the results of initial tests. The proposed testing sequence is as follows:

- 1) Steady-state irradiation at normal full power conditions to predetermined exposure. (0; 25,000; 50,000 Mwd/MT);
- 2) Exposure to transient;
- 3) Nondestructive examination;
- 4) Destructive examination of a fraction of the specimens;
- 5) Steady-state irradiation of remainder of specimens over a range of exposures (5,000 - 15,000 Mwd/MT);
- 6) Destructive examination.

A total of 15-20 specimens would be tested in the initial phase of these tests. Long term irradiations subsequent to the transient are not planned because it is the status of the fuel immediately after a transient which is of prime importance to safety. The nature of the damage (if any) produced by the transient will have to be assessed by both out-of-pile examination and in-pile tests in order to make a final

judgement as to whether acceptable fuel damage limits have been exceeded. It will be important to keep in mind that acceptable fuel damage from the safety viewpoint may not be the same as acceptable fuel damage from the economic viewpoint. For this reason, relatively high probability operational transients would be expected to be limited to a narrower range than the more severe low probability transients to which safety criteria must be applied.

5.3.4.1.2 Facilities

Initial testing would be done in sodium capsules in thermal test reactors (GETR, MTR or ETR) with the transients being done in TREAT (unless some acceptable method for accomplishing the transients in one of the thermal test reactors could be worked out). Examinations would be made in existing hot cell facilities. Later confirmatory tests on larger clusters of fuel rods would be done in FFTF (pre-irradiations) and the Safety Test Facility (STF). The STF presumably would be well suited for this work because the transient and subsequent short term steady exposure could be done without moving the fuel.

5.3.4.1.3 Series II Tests

The Series II Tests will investigate the problem of propagation of single pin failures under normal operating conditions. Failures of two types will be studied: clad failures due to excessive fuel swelling and failure (loss) of the sodium bond. Since propagation due to excessive swelling will also be investigated in the development of oxide fuels, the effect of loss of the sodium bond will receive prime attention in these studies.

Two distinct modes of bond loss have been identified: (1) a low-level clad rupture followed by vent plugging, gradual buildup of internal pressure due to fission gases and gradual ejection of the bond through the rupture, and (2) vent plugging and buildup of internal pressure followed by a low-level clad rupture and rapid ejection of the bond. The consequences of the latter sequence of events may be more severe; therefore, it will be studied first.

Such experiments may be performed in one of several ways:

- 1) Direct simulation by expelling the bond at power through application of gas pressure from an external source or by rupturing a prepressurized pin;
- 2) Indirect simulation by rapid insertion of an unbonded pin into an operating test reactor or by a rapid power increase of the reactor with the unbonded pin already inserted.

Simulation of the slow loss-of-bond situation is more straightforward. This would be done by inserting an appropriately defected unvented pin in the test reactor and allowing nature to take its course. The important variables in these tests are the fuel exposure (burnup) and the fuel pin linear power. The critical heat transfer effect of burnup is determined by the thickness of the fuel-to-clad gap when loss of the bond occurs. This can be simulated by varying the gap on unirradiated fuel. Initial tests would be done this way; however, the behavior of the fuel subsequent to bond loss could be significantly influenced by retained fission gases and other characteristics of higher burnup fuel. Therefore, tests with preirradiated fuel will also be done.

The only important effect of bond loss may be excessive fuel swelling; therefore, the bond loss studies could provide the bulk of the information needed with respect to propagation due to excessive swelling. Excessive swelling might also occur as a result of undetected errors in fuel fabrication. Depending on results obtained from bond loss experiments, the excessive swelling experiments would set up to complete the picture of failure propagation potential and appropriate measures for its control. A typical experiment would employ a single fuel pin either fully or partially loaded with pellets known to swell excessively (e.g., hypostoichiometric UC). This pin would be surrounded with normal pins.

All propagation studies will be aimed at answering two main questions:

- 1) What are the qualitative and quantitative effects of the defective pin on the 6 adjacent normal pins?
- 2) Do any of these effects tend to propagate beyond the 6 adjacent normal pins?

The second question is the more important one from the safety point of view.

5.3.4.1.4 Facilities

A total of 10-15 experiments in thermal test reactors is projected. Initially, several single pin bond loss capsule experiments would be done, followed by 5-10 experiments using seven-pin clusters in a capsule-type assembly with an internal flowing sodium cooling system. Several 19-pin experiments in a package or integral loop facility would complete the initial phase. Confirmatory tests in a fast flux environment would be planned for STF with preirradiation in FFTF. Five to ten 19-pin experiments are anticipated. Existing hot cell facilities would be used for post-mortem examinations.

5.3.4.2 In-Pile Tests - Transient Reactors (Subtask 10-2.4.5)

5.3.4.2.1 Series III Tests

Series III Tests are designed to provide information on the behavior of fuel pins and assemblies under conditions which are expected to produce severe fuel damage, i.e., during transients in which fuel melting and/or bond boiling occurs. These experiments will provide base-line data for the definition of design basis accidents involving the core. They will also provide information vital for evaluation and design of protections against propagation of local coolant blockage situations. The phenomena of prime interest in these studies are:

- 1) The mechanisms and dynamics of phase changes and movement of fuel, coolant and clad and the resulting effects on reactivity;
- 2) Pressures created as a result of sodium vaporization and vapor bubble collapse and the effect of these pressures on structures;
- 3) The mechanisms and dynamics of interaction between blocked and unblocked adjacent subassemblies;
- 4) The response of outlet thermocouples, fission product and delayed neutron detectors, reactivity meters and other instrument systems explicitly involved in protection against propagation of local incidents.

Initial experiments would be single-pin experiments in a packaged loop in TREAT. The principal variables are fuel exposure and type and severity of the transient. Within the limitations of the TREAT system, simulations would be made of overpower transients at constant flow and coolant flow transients at constant power. Power transients that could be caused by credible reactivity insertions in the reference reactor generally lead to bond boiling prior to extensive fuel melting (see ref 2). Therefore, the important difference between flow and power transients is the temperature and power density of the fuel at the time bond boiling starts. By adjustment of initial temperatures and the shape of the TREAT power pulse, conditions approximating the two types of transients can be created. Experiments would be highly instrumented. After evaluation of the results of single-pin experiments, several 3 to 7-pin experiments would be performed.

TREAT experiments are not expected to be particularly valuable in developing information on interaction between subassemblies (melt-through) because of the TREAT reactor's lack of ability to sustain power after melting the fuel. These experiments would indicate, however, the most likely conditions immediately following flow blockage (fuel concentrated or dispersed) which would improve calculational models for predicting melt-through times. The measurement of pressures would also help predictions of interactions. Depending on results of these tests, specific capsule-type melt-through experiments may be proposed for a thermal test reactor. More likely, however, resolution of questions in this area would be made with larger-scale experiments in STF.

Scale is expected to be an important factor in determining the magnitude of pressures and other effects when fuel melting occurs. Tests involving the equivalent of a complete subassembly in STF are therefore considered particularly important in this test series. The most serious credible incident that could be contemplated for the reference core is the sudden blockage of coolant to a subassembly while at power. Therefore, STF experiments would concentrate on this accident. A series of 3 to 5 blockage experiments is projected starting with a 7-pin cluster and working up to larger numbers of pins to determine the effect of scale.

These experiments would be coordinated with Series I and II experiments and it is anticipated that at least two of the assemblies fabricated for earlier tests could be reused in these destructive tests.

5.3.4.2.2 Facilities

TREAT and STF have been indicated as the prime facilities for Series III tests. Postmortem examinations would be done in existing hot cell facilities.

5.4 RESEARCH RELATED TO HYPOTHETICAL ACCIDENTS (Task Area 10-2)

The program of research related to hypothetical accidents outlined in the LMFBR Program Plan covers the essential issues well. However, there are several particular areas which are especially important and which should receive first attention. These are:

- 1) Development of analytical procedures and performance of experiments which will yield more realistic estimates of reactivity changes that could result from fuel melting and/or sodium boiling. (Subtasks 10-2.2.1, 10-2.3.3, 10-2.4.4 and 10-2.4.5);
- 2) Development of analytical procedures and performance of experiments which will yield better estimates of the effective work-energy from vaporization of fuel and/or sodium. (Subtasks 10-2.2.2, 10-2.2.4, 10-2.4.5, 10-2.3.4 and 10-2.9.1).

What is most urgently needed are initial experiments in each of these two areas upon which analytical models can be based. The first type of experiment is that type described above under section 5.3, Fuel Failure and Melting-Series III experiments. These will provide an initial picture of fuel and coolant behavior movement which can be used to predict reactivity during severe transients for the sodium-bonded carbide system. Since there is some evidence that this behavior is significantly different than the behavior of metals and gas-bonded oxide,⁽³⁾ it is important to establish as soon as possible the extent and scope of essentially specialized research required for this fuel system (as opposed to generally applicable research, such as that to be done in the area of sodium fires).

In the area of work-energy yield estimation, it is important that TREAT experiments of the type suggested by Miller and Dickerman⁽⁷⁾ be performed at as early a date as is practical. These experiments were suggested in connection with equation of state measurements; however, the method of determining the equation of state involves measurement of the physical work produced by the vaporization of a fuel sample, therefore the experiments would yield directly applicable data. Furthermore, these experiments should include fuel vaporization (and/or melting) in the presence of sodium as well as in a vacuum so that the secondary effect of sodium vaporization can also be determined directly.

Work-energy produced by sodium vaporization from molten or fragmented fuel can also be studied out-of-pile with the type of experiments described under Subtask 10-2.3.4. These experiments are considered especially important for the sodium-bonded carbide system, again because of the results of the one TREAT experiment⁽³⁾ with this fuel.

Completion of these experiments and the development of analytical models based on their results should provide sufficient information for conservative upper-limit type of estimates usable in evaluating containment of the demonstration plant. Their results will also form the basis for definition of additional experiments and theoretical development for larger plants. Because it is not clear at this time whether these experiments will point in the direction of further investigation of Bethe-Tait type of accidents or in another direction, it is not practical to attempt to outline research in this area beyond the initial phases indicated.

5.5 PROTECTIVE SYSTEM (Task 10-3.1)

The purpose of this task is to outline protective system development programs in those areas specific to our reference design which are either not included in the AEC LMFBR Program Plan or are not covered in enough detail in that plan to satisfy our design requirements. A development program for the entire protective system and its components will not be prepared since much of our system development is already described in the Program Plan. The specific areas of interest discussed in this subtask are:

A Method for Protective System Design Development and Evaluation, Outlet Thermocouple Development, Development of Local Plugging Detection, and the Development of a Reactivity Meter.

5.5.1 Protective System Design Development and Evaluation

5.5.1.1 Program Definition

In this task, the protective system will be evaluated to determine how it can best accomplish the goals required of it. These goals, as determined for our reference design, are specified in the AEC General Design criteria as modified for LMFBRs in Volume III of the Task II and III report⁽²⁾ and can be summarized by the following: The protective system combined with the inherent safety characteristics of the core must guarantee that no credible accidents, including those in which portions of the protective system fail to respond, should produce either gross meltdown of the fuel or gross voiding of the core. Included in this evaluation will be an analysis of each credible accident in terms of what is the best way of providing this protection; that is, what and how many reactor parameters must be monitored and connected to the trip circuits. The probability approach will be effectively used to determine how well each accident must be protected in order to reduce to an acceptably low value the probability of it causing gross core damage. This acceptably low value must also be determined. From this probability analysis will come the answers to questions such as: Is a double protective system required? Does a double protective system with separate principle gain significant protection? How many separate trip signals must be provided for each postulated accident? When does too much safety instrumentation become disadvantageous? Is a cocked rod needed during refueling?

5.5.1.2 Description of Current Technology

Protective systems have been designed and successfully employed in several LMFBRs in the United States* and abroad.** It is impossible to

* FERMI, EBR-II, SEFOR

** Dounreay, Rapsodie

know at this time whether similar systems will be adequate for large power reactors; however, it is felt that much of the technology gained will be applicable. Double protective systems appear to be the trend now in both water reactors and sodium reactors. The water reactors employ control rods as one system while the other system usually operates on a different principle such as soluble poison. So far, the separate principle idea has not been carried over to LMFBRs because of the technical difficulties involved; therefore, two completely separate control rod shutdown systems are normally used. The primary system, called the safety rods, is always fully withdrawn during operation and is used for no other purpose but safety. The secondary system consists of the normal shim-regulating rods which are scrambled from any position. Both of these shutdown systems are fast-acting and can detect the malfunction and shut down the reactor before any damage is incurred following most credible accidents. Sufficient excess reactivity is provided in each system to assure an ample shutdown margin during all expected normal and abnormal conditions. With both systems inserted in the core simultaneously, the reactor is subcritical by several percent in reactivity. The number of trips available for any given credible accident varies from plant to plant and accident to accident; however, usually a minimum of two has been provided. The basis for determining the amount of protection provided by the protective system is rather nebulous and probably is a result of the designer's experience and engineering judgement. Cocked control rods have been provided in at least one reactor⁽⁸⁾ to protect against the refueling accident. Other LMFBRs^(9,10) as well as many water reactor systems depend upon a large shutdown margin during refueling to guarantee that this type of accident is not credible. A variety of other shutdown devices and systems have been suggested for backup safety during operation or refueling. These include neutron flux activated mechanisms such as controlled fuel-bundle expansion, meltdown rods, and core differential expansion wedges, as well as liquid poison injection systems (i.e., lithium), similar to boron injection into water reactors. Technical difficulties have rendered most of these ideas impractical up to now.

The detection of local plugging and preventing it from either causing failure or causing the propagation of failure has not been a major consideration in the past. However, since the occurrence of such an accident in an operating LMFBR⁽¹¹⁾ with its resulting problems and delay time in getting back into operation, increased concern has been given to the possibility of avoiding such an incident in future plants. It has been suggested that this might best be accomplished by proper design of the fuel bundle inlets plus additional detection equipment in the protective system.

The inherent reactivity feedback characteristics of the reactor have been considered in the design of the protective systems. The primary contribution from these effects is to provide additional and sufficient time to activate and accomplish the necessary protective action during any credible accident. Metal-fueled fast breeders have depended largely on the negative fuel expansion coefficient to achieve this effect while fast ceramic-fueled breeders will depend on the negative fuel Doppler coefficient. Other effects, both positive and negative, such as material density changes are small compared to these, but are also included in the analysis. The core radial expansion reactivity effect can be made strongly negative by proper core design, e.g., spacers and core clamp. This effect reacts slowly compared to the Doppler since it follows the structure or coolant temperature rather than the fuel temperature. Therefore, it has been suggested⁽¹²⁾ that a large core expansion effect might be quite advantageous in the loss of flow accident.

5.5.1.3 New Information Needed

Since this task is system design oriented rather than a basic research and development program, the information required is primarily input data which will allow evaluations of the protective system or portions of the protective system for determining how well these systems and subsystems perform their function. This system evaluation will be based on the probability approach as briefly outlined in Subtask 10-4.2.

The information needed to successfully complete this evaluation includes the following:

- 1) Failure history and probability data for all components of the protective system;
- 2) In areas in which data is not now available, techniques should be developed for estimating the probability of failure;
- 3) Data on the confidence level of any given detector, assuming it is operating normally. That is, how sure are we that the instrument will detect the change in the parameters that it is designed for, even if it is functioning properly;
- 4) Determination of what is an acceptably low value for the overall probability of failure of the protective system for any accident;
- 5) A final set of specific design criterion for the protective system of LMFBRs, similar to those already in existence for water reactors.

5.5.1.4 Program Description

Two basic analyses must precede the design and evaluation of the protective system. This includes first of all, determining the goals and the requirements of the protective system. These will be established by the development of specific design criteria for LMFBRs which are already part of the Safety Program Plan designated Task 10-4.1. The other area involves the detailed studies of all accidents which are expected to be prevented by the protective system. Again, the Safety Program Plan has completely outlined these studies in the following Subtasks: (10-1.1.1) Summary of Potential Malfunctions and Faults, (10-1.1.2) Definition of Initiating Conditions for Reactor Accidents, (10-1.1.3) Analysis of Initiating Accidents. This finally leads to the specific purpose of this task which is to design and evaluate the protective system, using the information generated in the studies described above.

An analysis of the core protective system (Subtask 10-1.1.4) is included in the Safety Program Plan; however, it is not sufficient to perform the design and evaluation we feel necessary for our protective system. Therefore this area will be developed in some detail for our

reference design. From the results of the accident analysis combined with the specific safety design criteria, the design of the protective system can be determined. At this point, the inherent reactivity feedback coefficients will have already been tentatively set and factored into the accident analysis, and will be changed later only if the protective system evaluation requires it. The protective system design and evaluation will include the following:

- 1) Identify and determine the reactor parameters which will be monitored for detecting abnormal operating situations. This will include estimates of the instrument's operating characteristics, i.e., response times, reliability etc.;
- 2) From the results of the accident studies, determine the protection available for each separate accident, that is, how many independently measured parameters will be effective in preventing the accident from causing damage to the reactor. All protective action for each accident including alarms, power setback and scram should be specified;
- 3) Perform a probabilistic evaluation of the entire protective system. The results of the fault tree analysis which is performed as part of the accident studies, combined with the protective action specified for each accident, will be used as a guide for this evaluation. The evaluation will determine or consist of the following:
 - a) Determine and apply the failure probability data to all components of the Protective System. This will also include allowance for the detectors' confidence level.
 - b) Determine and apply the probability data to the basic malfunction or accident. That is, the probability that any specified accident will be initiated (e.g., control rod ejection, etc.) must be known.
 - c) For each accident (basic malfunction), use probability analysis to determine the final probability that the accident will proceed through this chain of malfunctions and cause core damage or core destruction. The accident fault tree will be used to trace each accident from its initiation, through the protection system and to the final consequences.
 - d) Compare this final probability for each accident with the acceptable value. If it does not meet the acceptable criteria, additional protection must be provided in the protective system. Conversely, if the accident is found

to be overprotected, some trips might be removed if not needed elsewhere. However, the entire list of accidents must be evaluated before the latter can be determined, since these additional trips may be needed to meet the criteria in some accidents while in others they may appear to be redundant.

- e) Redesign the protective system or other parts of the safety system if it is found necessary or advantageous. Furthermore, at this point, it may be necessary to change the internal reactivity feedback coefficients (e.g., Doppler, structural, etc.) to obtain the required safety. If so, it will also be necessary to perform additional safety analysis calculations. The probability analysis will again be performed to determine if the redesigned system meets the requirements. This iterative procedure will be used until the design requirements of the safety system are met.
- 4) From the results of this evaluation several design decisions and criteria will be determined. These include:
- a) the minimum number of scram trips for any accident;
 - b) whether scram trips on several different parameters are preferable to several trips on one parameter;
 - c) on which accidents it is desirable to incorporate a power setback signal;
 - d) whether two (or three) fast acting shutdown systems of the same principle (e.g., scram rods) is safer than two systems of different principle, one fast and the other slow;
 - e) whether the refueling accident requires a cocked rod;
 - f) whether for local plugging, the protective system will be capable of:
 - i) detecting the phenomenon and preventing significant damage to the plugged subassembly, and/or
 - ii) preventing the propagation of damage to adjacent subassemblies;
 - g) whether entrained gas detection is necessary.

5.5.2 Local Plugging Detection

5.5.2.1 Program Definition

The provision of equipment and instrumentation that determines when the flow through an individual subassembly is either blocked or becoming blocked will be necessary on at least the first generation of LMFBRs. Past operating history of experimental fast breeders has proven this to be desirable for preventing long and expensive operating delays due to local fuel failures, while current safety analyses of future large LMFBRs indicate that it will probably be needed to satisfy design criteria. Several techniques involving the measurement of different local parameters appear feasible. These include the measurement of local subassembly flow, channel temperatures, channel pressure, strain in channel walls, and the detection of incipient local sodium boiling by sonic devices.

In our reference design, three lines of protection were provided for the detection of local plugging and the prevention of its consequences from propagating. These are:

- 1) In-Core thermocouples;
- 2) Reactivity Meter;
- 3) Fission Gas and Delayed Neutron Detection.

Protection for two types of local flow blockage, i.e., complete or partial, is provided in the reference design. The partial blockage situation is monitored by a computer controlled in-core thermocouple system which will detect a flow reduction early enough to prevent it from becoming a large plug with subsequent damage to the fuel subassembly. The computer and its data analysis are not considered an integral part of the protective system but are used only to obtain operating information and to increase the probability of detecting local plugging before it can cause damage. With proper core and subassembly design, the most realistic and probable type of plugging will be slow and will allow plenty of time for detection. This detection system is tailored for that type of blockage. It is recognized that a small percentage of

outlet thermocouples may be inoperable during a year's refueling cycle (they can be replaced during refueling). Therefore, a subassembly may be unprotected by this system at any given time. There appears to be no economically feasible way of avoiding this problem. Thus, the basic philosophy is to provide highly reliable thermocouples which will insure one good measurement in a very high percentage of the subassemblies. This system is designed primarily to increase operating efficiency and to protect the plant owner's investment by reducing the risk of extended outages. It is not intended as part of the safety system required for the protection of the public; however, by reducing the probability of local fuel failures, it increases the overall safety of the system.

Complete or partial blockage which immediately causes channel voiding cannot be detected early enough to be prevented by any detection system. Therefore, the function of the detection system is to react early enough to limit the damage to the blocked subassembly(s) and prevent failure propagation.

From the results of our accident analysis, it is evident that the fewer the subassemblies simultaneously blocked, the more difficult it is to make prompt detection. Thus, the blockage of a single subassembly is more difficult to detect and will result in a longer delay than the blockage of several. This occurs because the positive reactivity effect from sodium voiding of one subassembly is so small that it is undetectable until fuel meltdown takes place a few seconds later. The reactivity meter (Section 5.5.4, Reactivity Meter) might separate the change from the normal background noise and expected normal changes; however, a scram on this signal alone is probably not feasible. Therefore, the reactivity change must be used in coincidence with another signal such as delayed neutron activity in the primary sodium or fission product activity in the reactor vessel cover gas which would be detected several seconds after the first gross clad failure. A more definite reactivity signal will occur after the fuel melts and begins moving out of the seed region. This reactivity change is large enough to be confidently detected with the reactivity meter and can be used as a protective action scram signal.

Flow blockage, and subsequent voiding of several subassemblies will avoid these problems because the positive reactivity addition from voiding is large enough to be confidently detected by the reactivity meter. Furthermore, it results in power changes large enough to cause high power and outlet temperature level trips. In such cases, the reactor will shut down before much fuel has melted in the blocked subassemblies, preventing any failure propagation.

5.5.2.2 Description of Current Technology

There is very little past experience in directly determining the occurrence of local plugging in a single subassembly. Although such events have occurred in LMFBRs in the past, their detection locally was not foreseen, but they were eventually detected by the gross core monitoring equipment after significant local fuel damage was incurred. Both the FERMI⁽¹¹⁾ and the SRE⁽¹³⁾ fuel failure incidents were caused by local subassembly plugging and neither were positively indicated until the gross fission product monitoring systems exceeded their limits and shut the plants down. In both cases, earlier indications were available from a few scattered subassembly outlet temperatures and reactivity calculations; however, due to the uncertainties in these measurements, a positive identification of this abnormal condition could not be made. It was suggested later that if individual subassembly instrumentation, i.e., temperature, flow, etc., had been available, both of these incidents could have been avoided.

Thermocouples have been used at the subassembly outlets in several reactors as discussed in the next section (Section 5.5.3, Outlet Thermocouples), but they have not been used to indicate the flow in each subassembly. In most instances, they were used primarily to provide operating information. Direct measurement of flow by some type of flowmeter has never been done on individual subassemblies; however, much of the development of small sodium loop flowmeters will be applicable. This includes such types as permanent-magnet or eddy-current flowmeters. Background information for these and other types is given in detail in the Program Plan, Task 4-7.5, Sodium Flow Sensors for Use on Fuel Assemblies.

5.5.2.3 New Information Needed

Since little work has been done in this area, the new information needed consists basically of the development of a system of components that will confidently detect the flow plugging of each fuel subassembly. This system must have a fast response time in the order of 1 second or less. The components of the system including the sensor must be replaceable, preferably during operation immediately after failure, or at least during each refueling interval. The system must include a continuous automatic evaluation of the data generated and then supply this information to both the operator and the protective system.

5.5.2.4 Program Description

The development of a variety of possible sensors for measuring parameters affected by flow blockage, is an important part of the AEC Program Plan for Instrumentation and Control. None of these have yet been proven for this purpose in an LMFBR; however, direct and indirect measurement of local flow by flowmeters and thermocouples, respectively, appear to be the most promising. Flowmeter development and installation is included in Task 4-7.5. Programs for testing and developing subassembly flowmeters of the permanent-magnet, eddy-current and turbine type are outlined in Subtasks 4-7.5.6, 4-7.5.7, 4-7.5.8 and 4-7.5.9. Flow can also be measured indirectly from temperature data supplied by thermocouples placed in the subassembly coolant channel. The development of these sensors is described in Section 5.5.3, Outlet Thermocouples. Another indirect method of determining flow is to measure the pressure at some point in the subassembly channel. These sensors will be developed as part of Task 4-7.7. Detection of the early physical effects of flow blockage adds another possibility to the list of possible sensors. The detection of incipient sodium boiling falls into this category and attempts to develop this technique are outlined in Task 4-3.4.

The development of an installation design for these sensor systems that permits easy repairs and replacement will be incorporated into the design of the first demonstration plant. Separate development of such a system would be impractical since it must be closely tied to the

specific details of the core and primary system design. A computerized data evaluation system is generally outlined in Task 4-4.2, Applications for Computers, but will require more specific development during the design of the demonstration plant. However, general programs for analyzing flow and temperature data and creating instructions for the operator on the protective system can be developed now and should be included in the computer applications task.

5.5.3 Outlet Thermocouples

5.5.3.1 Program Definition

In light of recent operating experience in several LMFBRs, it appears advantageous and probably necessary to know continuously the characteristics and condition of each fuel subassembly during operation. This implies the provision of some type of instrumentation in all subassemblies that continuously supplies operating data such as temperature, pressure, flow, etc. to a system monitor. The easiest and most reliable instrumentation considering the current state of technology is the outlet thermocouple. One or more thermocouples can be placed in each subassembly to measure the coolant temperature and can indirectly be used for determining changes in flow or power. The problem is to develop long-lived, reliable thermocouples and a method for inserting them in each subassembly during operation, yet have them removable and replaceable at least during refueling. This task involves the development and testing of such thermocouples under the reactor conditions prevalent in LMFBRs. It further includes the study and design development of a method for handling and placing the thermocouples in the reactor. The third, and very important objective of this task is to devise a system for retrieving, analyzing and using the information generated from these thermocouples.

Outlet thermocouples are expected to perform at least two functions. The first is to provide information on the operating characteristics of each subassembly for the purpose of detecting any impending failure condition. This information will further provide an indication of local failure due to flow blockage and voiding. The second prime function of the thermocouples is to provide scram signals to the protective system.

The philosophy here is to permit a rapid response to gross core changes (i.e., power transients, etc.) rather than to scram on changes in any single subassembly. Therefore, only a select few thermocouples will be used in the protective system at any given time. However, as thermocouple reliability is increased it may eventually be possible to scram on signals from any individual subassembly.

5.5.3.2 Description of Current Technology

Thermocouples placed in the outlet of fuel subassemblies have been used in several currently operating sodium-cooled reactors and have produced some rather encouraging results. All of these plants have been small relative to a large fast breeder and hence have required a comparatively small number of thermocouples. Furthermore, most of these plants^(8,9,14) used outlet thermocouples on only a few of the subassemblies rather than on all. However, the French reactor, RAPSODIE, has thermocouples placed in the outlet of every fuel bundle and has had remarkably few thermocouple failures. That is, out of 165 outlet thermocouples, they have experienced only five failures⁽¹⁵⁾ over the lifetime of the reactor until Nov., 1968. This indicates a very high reliability for thermocouples of this type. These thermocouples are all individually connected to the protective system, two per subassembly, and scram on a one-out-of-two basis. In addition the five failed thermocouples were easily removed and replaced during a refueling shutdown. The temperature information from these thermocouples is received and analyzed by a computer. This computer continuously compares the measured value to an expected computed value and automatically scrams the reactor if a given ΔT is exceeded on any single measurement. With the exception of a few problems with the computer⁽¹⁵⁾ at the beginning of operation, this method has reportedly worked well.

In EBR-II and Hallam, a few selected outlet thermocouples were used in the protective action trip circuits to provide early scram protection during abnormal operation. Outlet thermocouples in FERMI were used basically to supply information to the operator for purposes of determining

the conditions locally within some regions of the core. The thermocouples performed these functions satisfactorily and reliably in all the plants. Following the FERMI incident, it was postulated⁽¹¹⁾ that if some method of automatically analyzing the thermocouple data immediately had been available the local meltdown may have been avoided. This was said to be a possibility even with the few outlet thermocouples that were provided.

It should also be emphasized that most current 1000 Mwe LMFBR designs indicate the requirement for outlet temperature measurements in all fuel subassemblies. The purpose is primarily to protect the core from local flow blockage and the resulting core damage. The operating temperature for these thermocouples is somewhat higher (1100-1200°F) than those used in the reactors described above (800-900°F); however, this is not expected to create many additional problems since there is much long term out-of-pile testing and data at these higher temperatures.

5.5.3.3 New Information Needed

The development of thermocouples that operate reliably in high temperature sodium and under conditions of high neutron flux, has progressed successfully over the past few years so that these sensors are now essentially a proven state-of-the-art technique. Therefore, detailed basic development of the thermocouple itself is no longer of prime importance but, instead, tests and development are needed for applying these devices in LMFBRs and obtaining and processing the outlet subassembly temperatures. This development requires information in the following areas:

- 1) Testing of various types of already developed thermocouples in an LMFBR environment, i.e., high integrated fast flux at temperatures of 1000-1200°F, to determine which meet a lifetime requirement of at least 1 year and maintain a high reliability. The thermocouple chosen should have an accuracy of at least ± 1 percent under these operating conditions;
- 2) Methods must be developed for either reducing the temperature drift during irradiation or determining it accurately enough to make a proper adjustment in the output temperatures;

- 3) Develop a method of placing the thermocouple into the subassembly channel. This will involve, as a minimum, a trade-off study and testing of either placement in thermocouple wells filled with an intermediate fluid or direct immersion of the thermocouple into the flowing coolant. Response times of the various methods tested will be an extremely important consideration. These times are required to be on the order of ≤ 1.0 second;
- 4) A method must be developed and tested for efficiently getting the thermocouples in and out of each subassembly either during operation or at least during the refueling shutdown interval. Furthermore, some form of positive identification is needed to determine if the thermocouple is in its proper location;
- 5) A system must be developed and tested that will continuously read and gather this temperature data and then evaluate it in terms of the function required of the outlet thermocouples. This will require some type of computer or possibly two computers, one used as a backup;
- 6) The reliability of the thermocouple measurements must be investigated to determine if it is high enough to permit a scram from signals in each separate subassembly.

5.5.3.4 Program Description

Much of this development is already included in programs which are discussed in the LMFBR Program Plan. Although in some instances, the necessary specific information called out above is not listed identically in the program plan, it will be generated by one or more of the programs outlined. The purpose of this program then is to list the research and development necessary to accomplish the specific purpose of measuring and evaluating the subassembly outlet temperatures even though portions are already covered in the program plan. This will serve to guide the R&D towards the successful accomplishment of this very important goal.

A program for the testing and development of the thermocouple in the LMFBR environment is completely outlined in Task 4-7.3, Temperature Sensor for General Use. A determination of the reliability and life of various types of thermocouples is the purpose of Subtask 4-7.3.4. Irradiation effects will be determined as discussed in Subtask 4-7.3.5. Information needed to determine if either a well design or a direct immersion system can be used will be available upon completion of Subtask 4-7.3.4 and

Subtask 4-7.3.7. The decision on using each individual subassembly thermocouple signal for reactor scram or only to provide detailed operating information for early warning of abnormal operation can only be made after all phases of the thermocouple development program are completed, i.e., Task 4-7.3.

The development of a method for inserting these thermocouples through the reactor primary containment boundary and into each subassembly, will best be accomplished during the detailed design of the reactor primary system. Also at this time, a system of removing and replacing failed thermocouples will be developed. A computer system for retrieving and evaluating this outlet temperature data will be developed as an integral part of the reactor core and safety system design. Most of these development problems will be included and resolved in the design of the demonstration plant. Furthermore, although FFTF may not be prototypical of a large LMFBR, attempts will be made to include much of this program in its design development program.

5.5.4 Reactivity Meter

5.5.4.1 Program Definition

It is proposed, as a result of our protected accident studies, that a reactivity meter be developed which can be used as an integral part of the protective system. Its function will be to continuously measure the reactivity of the core during operation and to generate a scram signal if the difference between this measured and a calculated value exceeds some preset $\Delta\rho$. It should also be designed to scram if the change in reactivity is excessive. It is hoped that such an instrument will permit the immediate detection of small local voids within the core. That is, the reactivity effect associated with the blockage and voiding of a few (or possibly even one) subassemblies or gas bubbles transiting the core will be detected by this instrument. In most cases, this detection will not occur in time to prevent damage to the locally voided subassemblies, but will scram the reactor in time to prevent failure propagation. Reactivity

effects caused by movement of small amounts of fuel resulting from local melting will be detected. This is expected to produce a significant reactivity change some time before possible melt-through to adjacent subassemblies. Detection by the reactivity meter will shut down the reactor in time to prevent melting into adjacent areas.

5.5.4.2 Description of Current Technology

Reactivity meters have been used on several reactors as an operational aid; however, they have never been incorporated into the protective system. A device of this type was described by the French⁽¹⁶⁾, where its uses were said to include (1) control rod calibration, (2) the analysis of dilution effects of poisons, (3) evaluating temperature effects and xenon poisoning, all three of which are essentially measurement analyses. It has more recently been employed as a control device for assisting regulation of the reactor. On Rapsodie⁽¹⁶⁾, the reactivity meter is used primarily for the surveillance of the startup and for stabilizing the reactor at a given power where even the smallest movement of the shim rod is immediately visible. Changes in reactivity of the order of $\sim 0.01\% \Delta k/k$ were discernible with this instrument system.

The reactivity meter, as currently used, operates off any parameter in the reactor which represents neutron power. Neutron flux as measured in the fission chambers was used as the input signal in Rapsodie. By electrical analogy, the instrument solves the reactor kinetic equations and delivers an output signal proportional to reactivity (ρ).

5.5.4.3 New Information Needed

The principle and operation of the reactivity meter appears to be adequately developed; however, its usefulness as an integrated part of the protective system has not been tried and proven. Test data must be obtained that will determine if it is possible to detect small reactivity changes associated with voids in the reactor. This will involve analyses of instrument background noise and the development of techniques to reduce it wherever possible. The instrument must not be so sensitive that

it creates frequent spurious scrams, yet it must be capable of detecting the physical realities within the core. A technique must be developed for continuously monitoring and evaluating the output signal and determining when the reactor should be scrammed. This may involve comparing to a continuously calculated value or simply a rate-of-change procedure.

5.5.4.4 Program Description

Subtask 9-5.1.5 of the Physics Program Plan contains a sketchy low-priority program for the development of a reactivity meter to measure subcritical reactivity. Some of this development may be applicable to the solution of the problem presented here. However, it is not safety oriented and is definitely insufficient for the purpose of this task.

A more conclusive program for the development of a reactivity meter for the protective system must include several areas of testing and analysis. First of all, tests must be performed on reactivity meters that are already developed and available. These tests can be run in either EBR-II, SEFOR, or FERMI. Since the input to these meters is usually neutron power, this signal can, in most cases, be taken off instrumentation that is already available. However, new types of neutron sensors as developed in Task 4-7.1 (Inst. and Control) may be used in conjunction with this task. Reactivity changes over the time interval representative of local voiding will be difficult to simulate in EBR-II and FERMI; however, in SEFOR its pulse rod can be used effectively to simulate sodium voiding transients. The objectives of these tests are to determine the minimum change in reactivity that can confidently be measured by these meters and to further determine if it is sufficient to provide protection from failure propagation. The requirements of the reactivity meters which are necessary to meet this latter objective must be predetermined by a detailed investigation of accidents resulting in local voiding and local melting (Section 5.3, Fuel Failure and Melting, Task 10-2.4). If these currently available reactivity meters are not capable of meeting the protective system requirements, a new type employing techniques such as noise analysis, low frequency oscillation, etc., as suggested in Subtask 9-5.1.5, might prove feasible. Furthermore, it will probably be

necessary in most of these instruments to develop techniques to reduce the level of background noise interference. The basic problem to be solved is to develop a meter sensitive enough to measure small changes in reactivity but yet not cause spurious reactor scrams.

The last major area that must be developed is a system that reads and evaluates the reactivity signal and then decides what to do with it. The best system is probably a comparison of the measured reactivity and a calculated value. This requires the development of a computer system that can accurately calculate the reactivity continuously or at least whenever a change is to be made. Subtask 9-5.1.5 provides a program for some gross computer reactivity monitoring development; however, more emphasis should be placed on it. Other techniques for evaluating the reactivity meter output signal should also be investigated. These include possible scrams on high rate-of-change of reactivity or unexpected changes of reactivity exceeding a preset $\Delta\rho$.

REFERENCES (SAFETY)

for Section 5.0

1. Liquid Metal Fast Breeder Reactor Program Plan, Vol. 10, Safety, WASH-1110.
2. CEND-337, Vol. III, "1000 Mwe LMFBR Follow-on Study: Report of Tasks II and III, Safety and Control Analyses," Sept. 1968.
3. S. J. Stachura and M. Silberberg, "The Application of Experimental Data from TREAT Meltdown Studies to Reactor Accident Analysis," Proceedings of the Conference on Safety, Fuels and Core Design in Large Fast Power Reactors, October 1965, ANL-7120.
4. C. E. Dickerman, et al., "Fast Reactor Safety Experiments on EBR-II Samples in Flowing Sodium Environment," Trans. ANS, Vol. 8 No. 2, Winter Meeting, Washington, November 15-18, 1965.
5. "Quarterly Technical Progress Report on AEC Unclassified Programs, Oct.-Dec., 1964", NAA-SR-10850, Feb., 1965.
6. "AEC General Design Criteria for Nuclear Power Plant Construction Permits," Appendix A, 10CFR50, July 11, 1967.
7. D. Miller, "A Critical Review of Properties of Materials at High Temperatures and Pressures Significant for Fast Reactor Safety," (Discussion), Proceedings of the Conference on Safety, Fuels and Core Design in Large Fast Power Reactors, October, 1965, ANL-7120.
8. L. J. Koch et al., "Hazard Summary Report, Experimental Breeder Reactor II (EBR-II)," ANL-5719, Argonne National Laboratory, May, 1957.
9. "Enrico Fermi Atomic Power Plant - Hazards Summary Report," NP-10458, Power Reactor Development Co., June, 1961.
10. "SEFOR - Facility Description and Safety Analysis Report," Docket No. 50-231, Supplements 1-21.
11. W. J. McCarthy, "A Review of the FERMI Reactor Fuel Damage Incident, etc.," V-2, Int. Conf. on Fast Reactor Safety, Aix-en-Provence, France, Sept. 19-22, 1967.
12. D. B. Sherer, W. G. Meinhardt, "An Analysis of Fast Reactor Transient Response and Safety in Selected Accidents", GEAP-4787, June, 1966.
13. R. L. Ashley et al., "Sodium Reactor Experiment Fuel Element Damage - Interim and Final Report," NAA-SR-4488 and Supplement, Nov., 1959 and June, 1961.
14. "Final Summary Safeguards Report for the Hallam Nuclear Power Facility," NAA-SR-5700, Atomics International, April 15, 1961.

REFERENCES (SAFETY) (Cont.)

15. G. Denielou, "RAPSODIE: First Year of Operation," Paper presented at Int. Conf. on Sodium Tech. and Large Fast Reactor Design, ANL-7520, Nov. 7-9, 1968.
16. G. Chauvet, E. Toumier, P. Villain, "Reactimeter-Its Utilization," V/5, Int. Conf. on Fast Reactor Safety, Aix-en-Provence, Sept. 19-22, 1967.

Figure 5-1
Safety R and D Schedule and Milestone Chart

Task	Calendar Year	70	71	72	73	74	75	76	77	78	79	80
5.0 Safety												
5.2 Core Configuration Safety Studies			1 ▽	2 ▽		3 ▽	4 ▽				5 ▽	
5.2.4.1 Demonstration Plant Phase							6 ▽					7 ▽
5.2.4.2 Commercial Plant Phase												
5.3 Fuel Failure and Melting			8 ▽	9 ▽		10 ▽						
a) Tests in Thermal Reactors								11 ▽	12 ▽	13 ▽		
b) Tests in Fast Reactors (STF)												
5.5 Protective System Development			14 ▽	15 ▽		16 ▽		18 ▽				
5.5.1 Protective System Design Development and Evaluation				17 ▽				20 ▽				21 ▽
5.5.2 Local Plugging Detection				19 ▽								
						22 ▽		23 ▽				24 ▽
5.5.3 Outlet Thermocouples												
				25 ▽			26 ▽					27 ▽
5.5.4 Reactivity Meter												

Figure 5-1: Safety Development Milestone Legend

1. Design criteria for demonstration plant.
2. Documentation of safety characteristics of demonstration plant.
3. Demonstration plant PSAR.
4. Specification of safety tests and observations for demonstration plant.
5. First summary report evaluating startup and operational safety of demonstration plant.
6. Design criteria for commercial plant.
7. Commercial plant PSAR.
8. Completion of single pin series II and III tests.
9. Completion of single pin series I tests.
10. Completion of multiple pin tests in thermal reactor.
11. Complete failure propagation tests in STF.
12. Acceptable fuel damage limits established.
13. Complete large cluster series-III tests.
14. Protective System Requirements Finalized.
15. Credible Accident Analysis Completed.
16. Complete Preliminary Protective System Design.
17. Preliminary Protective System Failure Probability Data Prepared.
18. Complete Final Protective System Design.
19. Choose Plugging Detection Equipment for Demonstration Plant; Preliminary Tests Completed.
20. Complete Design of Plugging Detection Equipment and the Computer Data Evaluation System.
21. Results Available on Operation of Plugging Equipment in Demonstration Plant.
22. Thermocouple Testing and Development Completed.
23. Complete Design of Reactor Thermocouple Installation.
24. Results Available on Operation of Outlet Thermocouples in Demonstration Plant.

Figure 5-1: Legend (Cont.)

- 25. Testing of Reactivity Meter Complete.
- 26. Development of Computer System for Calculating Reactivity is Complete.
- 27. Results Available on Operation of Reactivity Meter in Demonstration Plant.

Table 5-1. Safety R&D
Manpower and Cost Estimate

Task	Man-Years		Costs - 10 ⁶ \$			
			Operating		Other	
	Scientists & Engineers	Other	Labor & Overhead	Materials & Services	Equipment	Facilities
5.0 <u>Safety</u>						
5.2 Core Configuration Safety Studies						
5.2.4.1 Demonstration Plant Phase	30-40	10-15	1.2-1.6	0.3-0.4	--	--
5.2.4.2 Commercial Plant Phase	20-30	7-10	0.8-1.2	0.2-0.3	--	--
5.3 Fuel Failure and Melting						
a) Tests in thermal reactors*	40-60	50-70	2.6-3.9	0.7-1.0	0.3-0.5	--
b) Tests in fast reactors (STF)*	30-50	50-70	2.3-3.6	0.4-0.8	0.4-0.7	--
5.5 Protective System Development						
5.5.1 Protective System Design Development and Evaluation	18-23	6-10	0.7-1.0	0.3-0.7	--	--
5.5.2 Local Plugging Detection	35-45	45-55	2.4-3.0	0.7-1.0	0.4-0.6	--
5.5.3 Outlet Thermo- couples	20-30	30-40	1.5-2.0	0.5-0.7	0.5-0.7	--
5.5.4 Reactivity Meter	10-15	12-18	0.6-0.8	0.3-0.5	0.2-0.3	--
* Irradiation unit costs excluded						

6.0 CORE DESIGN

Items requiring rather extensive research and development in the core design area for the reference design include the fuel element, the fuel subassembly and the control subassembly. The research and development effort needed to produce a vented, sodium-bonded mixed carbide fuel element is outlined in Section 4.0, Fuels and Materials, and is not discussed in this section. The latter two items, fuel subassembly and control subassembly are discussed in this section.

6.1 PROGRAM DEFINITION

6.1.1 Fuel Subassembly

Fuel elements are contained in larger assemblies for convenience. These subassemblies provide the desired thermal-hydraulic environment and mechanical configuration needed to meet the functional and performance requirements desired from the reactor core. For the C-E reference design, research and development is required in the areas of heat transfer, hydraulics, structural-mechanics, manufacturing-fabrication and materials. The functions of these areas in regard to the fuel subassembly are:

1) Heat Transfer

To determine by appropriate analysis and out-of-reactor tests the spatial distribution of temperatures and thermal performance within the fuel subassembly;

2) Hydraulics

To perform appropriate analysis and out-of-reactor tests to assure the desired rate and distribution of coolant flow for removal of heat from the fuel elements and for the hydraulic holddown of the core;

3) Structural-Mechanics

To determine the hardware required to maintain spatial orientation of the fuel elements and assurance of structural integrity under reactor operating conditions;

4) Manufacturing-Fabrication

To determine the feasibility of fuel subassembly configuration and provide appropriate process and controls to assure reliable, structurally sound fuel subassemblies;

5) Materials

To provide data for the selection and to determine requirements and to assure acceptability by testing of materials to be used in the fuel subassembly (excluding fuel and cladding).

6.1.2 Control Subassembly

Shim-regulating and safety control rod subassemblies of the poison type are employed. Methods of packaging, aligning, guiding and cooling of the control material must be considered. Development requirements for the control subassemblies are somewhat similar to the fuel subassemblies but require emphasis in the areas of structural-mechanics and thermal-hydraulics. Functions of these areas in regard to the control subassemblies are:

1) Structural-Mechanical

Since operation requires movement of a poison element bundle within the core, factors such as vibration, wear, galling and structural loads must be evaluated both by analysis and out-of-reactor tests;

2) Thermal-Hydraulics

Variable heat generation and rapid transients associated with control rod insertion requires evaluation by analysis, and development of a mechanical configuration to accommodate this effect.

6.2 DESCRIPTION OF CURRENT TECHNOLOGY

6.2.1 Fuel Subassembly

6.2.1.1 Thermal-Hydraulics

The coolant flow passages in the C-E LMFBR fuel subassemblies

consist of circular ducts containing bundles of finned rods. It is reasonable to assume that the resultant coolant shear stress distribution may be quite different for these types of fuel element configurations than for those found in other types of ducts and passages. Previously, tests conducted at AI have shown that the use of Moody smooth-tube friction factors in conjunction with the equivalent diameter concept was a satisfactory method for prediction of pressure drops across a variety of fuel element configurations.⁽¹⁾

With the present state-of-the-art, accurate determination of pressure losses across fuel rod spacers and entrance and exit hardware is not completely predictable. Attempts to determine these losses using semi-empirical equations have been made.^(2,3) However, results from pressure drop tests across fuel rod spacers similar to the various LMFBR designs indicate that no single correlation can be used to accurately predict the pressure drop across end spacers of various designs. It may be concluded that hydraulic testing of entrance and exit hardware will, no doubt, always be required because of the complex geometries associated with these components. When it is considered that fuel rod spacers and entrance and exit hardware usually represent greater than 50 percent of the total pressure drop across a fuel element, the need for this testing becomes more apparent.

Technology of steady-state heat transfer is adequate from the fuel-element cladding outward, although improvement in mixing correlations would be valuable in reducing hot-channel factor allowance for undercooled and/or overheated flow ligaments or portions of a flow ligament.

Knowledge of the degree of fluid interchange within a fuel element bundle is essential to all reactor designs to establish the extent of excess coolant flow and appropriate hot-channel factors. Knowledge about the mechanism of coolant mixing will enable optimization of fuel element designs and provide information pertaining to circumferential heat transfer variation within fuel bundles, especially finned pin clusters. Accurate prediction of coolant mixing phenomena in the LMFBR fuel design is not possible with existing technology. Experimental testing must be utilized to adequately define the parameters which affect coolant mixing in fuel

assemblies. It is possible that a semi-empirical equation can be formulated from analytical and experimental studies which will adequately predict coolant mixing. Information is required concerning the effect of pin spacing, pin geometry, and coolant flow rates on the mixing process.

The temperature distribution within the fuel element is dependent upon the nature of the coolant flow through the bundle. Velocity profile studies are required to ensure the adequacy of coolant flow within all regions of the fuel assembly wherein hot spots might develop as a result of nonuniform cooling of individual fuel elements.

LMFBR experience to date with the Fermi reactor has shown that, for certain subassembly-nozzle and support-plate seat designs and for certain conditions of minor misalignment between the two bearing surfaces, coolant can leak past the subassembly nozzle. Owing to the low pressure just downstream from the leak, cavitation may be produced.

Common materials of LMFBR construction (Type-300 stainless steel) do not resist pitting damage from cavitation, although Stellite and other hard alloys offer considerable improvement. When cavitation causes this type of pitting, the situation usually gets progressively worse until, at some point, the bearing seat becomes so badly gouged and the leakage flow rate so great that sufficient back pressure is created at the vena contracta of the leak to avoid further cavitation. By the time this equilibrium is reached, the subassembly may lose an excessive fraction of its intended flow, and stainless-steel particles resulting from the cavitation could be distributed throughout the primary sodium system.

6.2.1.2 Structural-Mechanical

Fuel element support systems can be considered either continuous or intermittent, but many variations are possible. With current technology, a clear-cut choice cannot be made and a comprehensive survey of the analytically selected possibilities, with experimental evaluation of several promising candidates, is required.

There is no apparent consensus on the best way to space and support cylindrical fuel elements on a triangular pitch. Continuous straight hollow

tubes, continuous spiral wires, and many types of intermittent support grids are in contention. The qualitative merits and disadvantages of each type have been recorded, but as yet no comprehensive effort to identify the optimized choice for short and long term usage has been reported.

Moreover, there is no agreement on the required axial intervals of intermittent support. The range is 4 to 12 in. with the British PFR design at the conservative extreme. Since pin deflection for a given lateral load varies as the cube of the unsupported length, a factor of three spread in spacer intervals reflects a major spread in engineering judgement of what lateral loads will be present or how much deflection can be tolerated.

Generally, current practice for fixing fuel elements in the core is to anchor them to a bundle structure using mechanical keepers in the lower end caps, tie the bundle structure to an integral part of the main-subassembly hardware, and employ a holddown device to prevent hydraulic forces from lifting the subassembly. Typical holddown devices use a mechanical arrangement in the form of a latch at the nozzle end or a restraining force at the handling end. Also, hydraulic holddown is another possibility. Some hydraulic designs can be defeated by nozzle plugging. Downflow coolant is advantageous in eliminating the holddown problem, though inherent disadvantages generally are considered to outweigh this advantage. Some of the disadvantages are the requirement of higher performance of top-of-reactor seals and the action of thermal convection opposite to forced convection.

In turbulent flow parallel to the longitudinal axis of a fuel element, there is superimposed on the main motion in the direction of the axis, a subsidiary motion at right angles. Vibrations become significant when these small forces are able to supplement each other to produce large forces. Significant vibrations can occur when certain resonance conditions cause large periodic exciting forces or when exciting forces are in resonance with the natural frequency of the system.

A basic reference on the phenomenon of tube vibration with flow parallel to the tubes is available; Burgreen⁽⁴⁾ and coworkers studied arrays

with 1, 13, and 37 tubes in a 6-inch diameter; however, densities per unit length, rigidities, and end connection were varied. Strain gages were used to determine the amplitude of vibration, requiring an assumption concerning the deflected shape of the vibrating rod. Normally, the amplitude of vibration was not constant, with first one tube and then another vibrating. The fact that all the tubes did not have exactly the same natural frequency and that there was interaction on occasion at the supporting ends, caused some scatter in the data. Although self-induced tube vibrations in a high-velocity stream were observed, they are not satisfactorily understood. The results do indicate a type of flow-induced tube vibration which could cause fatigue of the tube elements.

Data are available^(5,6) on self-induced vibrations of finned tubes in transverse flow. However, the results are very conflicting. The fins have a definite but poorly understood effect on tube vibration. Data for preventive design are not available. The usual preventive measure is to increase the tube vibration frequency and to decrease its amplitude by adding constraints (spacers) along the tube length; however, the mere presence of the spacers introduces the possibility of fretting-fatigue problems between spacers and fuel elements. Clearly, a program of testing of specific fuel subassembly design under the full range of anticipated operating conditions is required to verify design adequacy.

Axial and radial loads will be imposed upon the fuel assemblies during operation in the LMFBR. While standard structural calculations are used for design of assemblies, the complexity of the geometries of end hardware require testing of design concepts to verify the calculations. The concept of utilizing large safety factors in providing structural support is not applicable in reactor core design, where compromise of efficient heat transfer, efficient coolant flow, and minimum metal-coolant ratios can greatly affect the efficiency of power production. Further, this loading may significantly affect vibration characteristics and must be considered under that phase of testing.

To assure the desired rate of coolant flow to the fuel-element bundle, the subassembly nozzle must seat firmly and seal adequately against the

core support grid, which constitutes the top of the coolant feed plenum. Deformation of the support plate or subassembly nozzle, or cavitation damage (pitting) of the seat, can seriously disrupt normal coolant service. In Fermi, it became necessary to use stellite in the support plate because the stainless steel seats became pitted--presumably due to cavitation from leakage flow.

The structural integrity of the fuel subassembly depends on the soundness of welds and the strength of the wrapper tube to prevent bulging from the coolant pressure at the upstream end. Hexagonal wrapper tubes are superior to square tubes in resistance to pressure deformation, because of less unsupported width of flat stock for a constant internal area. Cylindrical tubes of the reference design need to withstand a minimum stress.

The design provision for accommodating thermal expansion is acceptable when movement of the fuel subassemblies does not cause an increase in reactivity with temperature. There are several ways to accomplish this. In the Fermi reactor, fuel subassemblies contact their neighbors only at the plane of spacer pads slightly above the core midplane, and the tops of the subassemblies are gathered inward by the holddown device, creating a tight core in the cold condition. Heating then forces the subassembly to assume an S-shape, with a net outward movement of fuel. Other methods employing split-bundled arrangements, etc., are considered. In these designs, the normal curvature resulting from thermal bowing causes a net movement away from the core centerline. However, thermal operating effects, structural changes - creep distortions are not clearly established and must be determined by careful analysis and experimental tests. Mockup tests for determining the effects of core clamping have been performed recently for SEFOR.⁽⁷⁾ Additional data are needed for application toward the C-E LMFBR core clamping design.

Isothermal endurance testing is required to verify the long term structural integrity of subassembly components--chiefly the fragile fuel-element support structure--in the temperature and flow environment for which they were designed. A 1000-hour test in sodium at 125 to 150 percent of nominal flow at maximum service temperature is considered an accepted proof test. Vibrational-fatigue problems in the design would show physical

evidence of inadequate design in less than 1000 hours. Thermal-gradient tests--to commission a subassembly design for reactor service--should include an endurance test under prototypal thermal-gradient conditions with many thermal cycles to demonstrate the ability of the subassembly hardware and support structure to accommodate expansion and contraction of components without deleterious effects, such as fretting, galling or permanent deformation.

6.2.1.3 Manufacturing and Fabrication

The manufacturing and fabrication technology which is required to provide LMFBR fuel subassemblies is partially available. The materials and fabrication technical base for the fuel element are presented in section 4.0. As with all fuel subassembly design, stringent specifications must be established to assure a reasonable control in the design variables that can affect core performance. Items involved are control of design tolerances, weld integrity, materials control, inspection and nondestructive tests, and processes and techniques which will provide low cost reliable subassemblies where a large number of units are involved.

The use of helical wire wrap for fuel element spacing has been used in other systems. The use of circular fuel subassemblies with BeO moderator channels is unique in C-E LMFBR design and will require development.

In general, for the materials considered for the LMFBR fuel subassembly, limitations on size-shape-machinability-formability-weldability-costs, fabrication methods and techniques, and the effect of fabrication on properties are required.

6.2.1.4 Materials

An effective interchange of design requirements and available material properties is required to establish a reliable fuel subassembly that will withstand the environmental influences during (a) fabrication, (b) non-destructive testing, (c) loading, (d) service life, and (e) unloading, cooling and shipping periods. Careful analysis and experimental testing of representative fuel subassemblies are required to assure reliable performance.

The fuel element and subassembly is generally designed to allow provision for some measure of resistance to plastic deformation. Adequate ductility is necessary for fabrication and to accommodate other deformations experienced by the fuel subassembly. Fuel element components must withstand stress and strain cycling. Vibration and fluctuations in coolant pressure, variation in temperature and transients raise the possibility of fatigue failure.

Factors such as radiation induced property changes, corrosion resistance and others must be carefully evaluated by both design and materials consideration before acceptance in the LMFBR system.

6.2.2 Control Subassembly

The control subassembly is the in-core portion of the reactivity control system which consists of the control rod bundle, the housing or guide channel and hardware required for support and coolant control. The drive mechanism is discussed in Section 7.0, Component Design.

Because of the heat generated in high-flux reactors, all fast reactor control rods are immersed in the reactor coolant. Variable control of the coolant flow is accomplished in the EBR-II by opening more flow passages (holes in the inlet nozzle) as the fueled control rod is raised into the core. This is especially important in fueled control rods in which the heat generation changes significantly for the in and out positions. Variable control of coolant flow also is important for poison rods.

Some control rods incorporate "followers" rather than simply displacing sodium in the channels. Examples are void spaces above and/or below the active region and fuel followed by poison. Control rods have been designed as clusters of poison or fuel elements; annular, canned poison; and internally cooled cans.

In most designs, the control subassembly is detachable from the drive mechanism to facilitate fuel handling and replacement, the control subassembly can be handled by the fuel-handling machinery, and the guide tube is rigidly connected to the core-support grid with provision for its removal.

The guide tube also may provide lateral support to the fuel subassemblies or support and spacing for core instruments. The control rods may occupy a core-subassembly position or a special location may be provided. For core-subassembly locations, the guide tube usually is similar dimensionally to a standard subassembly with the control rod sliding or rolling inside.

Thermal-hydraulic design of control subassemblies involves considerations similar to those of fuel subassemblies. Other areas include:

- 1) Choice of poison material;
- 2) Compatibility of poison material with container;
- 3) Pressure buildup by helium liberated from boron [$B^{10}(n,\alpha)$ reaction];
- 4) Integrity of materials under irradiation;
- 5) Variable heat generation with insertion;
- 6) Wear or galling of control rod in guide tube;
- 7) Fail-safe features;
- 8) Snubbing of "scram rod".

Boron carbide, either natural or enriched in B^{10} , is the most popular poison because:

- 1) Experience using it is most extensive;
- 2) Neutron-absorption characteristics are predictable and good;
- 3) Compatibility with sodium is reasonably good.

However, it has some disadvantages, i.e.:

- 1) Radiation damage;
- 2) Evolution of helium;
- 3) Compatibility problems with stainless steel at high temperature--
 - a) carburization at 1300°F
 - b) B-Ni eutectic at 1022°F.

Tantalum has been proposed as a poison material. Although the absorption cross section is low, the density is high. Therefore, on a volume basis,

in currently considered "soft" spectrum fast reactor cores, it is nearly as effective as B_4C . Advantages of Ta are:

- 1) A (n,γ) reaction; thus no helium is evolved;
- 2) A high-temperature refractory metal;
- 3) Compatibility with clean sodium (usable as a bonding agent).

The main disadvantage of Ta is its high chemical affinity at high temperature for oxygen; therefore, it may require cladding.

Wear and galling of the guide tube is a serious problem, because it might lead to failure to operate at a crucial time (a stuck-rod accident). Rollers have been considered, but more research and tests on suitable materials are required. The present practice of loose fits and articulation needs further evaluation.

Fail-safe features include provision for rods to drop under the influence of gravity into their least reactive position in the event of a power failure and means to preclude rod flotation.

In some designs the scrambled rods are snubbed in the drive mechanism. In others, the rods are released and snubbed in the lower part of the guide tube. Both types need review--the first because the drive mechanism may "drag" during the fall; the second, because of potential shock or damage in the core.

Generally, methods are available for designing control rods. Definitive data are needed to establish the compatibility of B_4C in stainless steel, or an alternative cladding is required.

6.3 NEW INFORMATION NEEDED

6.3.1 Fuel Subassembly

6.3.1.1 Thermal-Hydraulics

Information regarding the thermal behavior and hydraulic performance as well as interrelated mechanical characteristics of LMFBR fuel subassemblies

and components is needed in support of design and optimization efforts.

Specific needs are:

- a) Determination by analytical methods and experimentation of the pressure drop versus flow rate characteristics of LMFBR fuel assemblies in order to optimize the design and to provide information for sizing the reactor cooling system;
- b) Determination by analytical methods and experimentation of the vibrational characteristics of LMFBR fuel subassemblies under simulated reactor flow conditions, to ensure that the dynamic behavior of the fuel channel is sufficiently stable to meet requirements imposed by reactor operation;
- c) Determination by analytical methods and experimentation of the details of heat transfer and transport in the subassemblies including the effect of fluid interchange between the channels of proposed fuel assemblies for the LMFBR core. Included in this effort are evaluation of the effect of interchannel fluid interchange and other heat transfer effects on parameters affecting fuel element design and reactor performance, such as the hot-channel factor for coolant temperature rise and element peaking factors, and the development of mathematical models for predicting the degree of fluid interchange in closely spaced finned-rod arrays and the local variations of clad temperature.

A spiral wire spacer has been selected for the LMFBR fuel element spacer design. Additional information is required both by analytical and experimental verification to establish an optimum wire pitch (axial support intervals) from a thermal-hydraulic and mechanical performance standpoint. The spacer design will be evaluated for:

- a) The required axial intervals of support to prevent bowing of elements due to thermal and lateral pressure gradients;
- b) Pressure loss and possible maldistribution of coolant flow associated with the support scheme;
- c) Vulnerability to flow-induced vibration and fretting damage to fuel element clad, subassembly shroud.

A hydraulic holddown scheme has been selected for the core fuel and blanket subassemblies. Analytical and experimental verification of this design is required in the following areas:

- a) The ability of the holddown to perform safely and reliably under normal and abnormal reactor operation;

- b) The possibility of subassembly inlet nozzle plugging and effect on holddown;
- c) Determination of core subassembly grid plate seal configuration, seal leakage, cavitation or erosion.

6.3.1.2 Structural-Mechanical

New information related to the fuel subassembly and associated components is needed in support of the C-E LMFBR design in the following areas:

- a) Establishment of criteria for the fuel element spacer and definition of the properties of material(s) required to meet life requirements. The technology needed for confident design of fuel-element support structures is not available. The advantages and disadvantages of continuous and intermittent grids, the required frequency of intermittent supports to prevent autocatalytic thermal bowing, a means for compensation to radiation induced swelling of fuel-element cladding, the influence of grids on the surface temperature of the fuel-element cladding performance in regard to vibration and fretting, and the hydraulic resistance of support system candidates need much better understanding. A loop facility specifically designed for the endurance testing of fuel subassemblies in a properly imposed thermal gradient is needed. Also needed is in-reactor capability to test, under prototypal thermal gradient conditions, a sufficiently large structurally representative portion of a fuel subassembly;
- b) Establishment of functional requirements and a method of fuel subassembly holddown and core clamping system. Necessary analytical and experimental investigation should be performed to provide assurance that the holddown and clamping scheme will be acceptable for LMFBR usage. For the C-E reference design, a hydraulic holddown principle is employed and the design will require evaluation by both hydraulic and mechanical tests. Core clamping employs a temperature activated principle. Structural loads imposed on the fuel subassemblies by the clamp, and resultant elastic and possible permanent deformations must be evaluated;
- c) Structural integrity of the C-E reference design fuel subassemblies, because of the unique usages of moderator channels, must be evaluated analytically and experimentally. Load simulation tests, thermal transient tests and vibration tests will be required.

Analytical and experimental verification is required in the area of the core-fuel handling interface in the following categories:

- a) Provision for handling stuck subassemblies or design of schemes for prevention of stuck subassemblies;
- b) Effect of bowed or distorted subassembly removal. Allowable bow and core refueling clearances;
- c) The selection of suitable materials for points of contact in the core subassemblies to prevent high temperature diffusion bonding or cavitation-erosion. Structural performance of the fuel subassemblies must be evaluated for load conditions as a result of shipping, handling and refueling insertion and withdrawal, as well as those loads incurred under in-reactor operation;
- d) Since integral BeO moderator channels are employed with the fuel subassembly, a means for venting or management of fission gases must be determined and evaluated by analytical and experimental methods;
- e) Because of the use of long slender fuel elements and the overall length of the fuel subassemblies employed in the LMFBR design, additional information is needed in the area of high velocity flow induced vibrations, both from the standpoint of fuel element spacer selection and overall subassembly structural integrity.

6.3.1.3 Materials

Sodium, high-temperature, and high-energy neutrons produce conditions in LMFBR fuel subassembly hardware not encountered in LWR systems. Only a few reactors that have been built have incorporated some of the service conditions of LMFBR plants.

The applicable areas where special materials considerations exist in the design of the fuel subassemblies are partially listed below:

- a) shroud tube;
- b) spacers (fuel elements);
- c) seal seats;
- d) orifices and nozzles;
- e) fasteners;

- f) spacer pads;
- g) handling head-outlet port;
- h) holddown and support grid (fuel element).

The missions and environments expected for these component areas can be predicted in only a general way for reactors yet to be designed. Even the proposed initial service conditions for FFTF will be more severe than any achieved to date. When fuel burnup, specific power, core temperature and plant availability have been increased to the goals anticipated for commercial plants, the reactor core components will have planned exposures to conditions well beyond present experience. It is expected that quantitative characteristics of material properties will be vital, and a potential for improved materials will develop. Since the plants will be large, small reductions in size and weight of the components will produce important "multiplier" cost savings. Thus economic considerations alone will force close attention to design "factors of safety" for the materials used in the core subassemblies.

The main structural materials of current interest for the LMFBR fuel subassembly are the austenitic stainless steels (mainly Types 304 and 316). The information on material properties required by the designer is outlined in table 6-1. Only a small part of this information is available; generally the basic properties data with some scattered information available in the other categories listed in the table.

Data on the effects of radiation on these materials, have been uneven in completeness. Material irradiation exposures expected in FFTF will exceed presently available data by a factor of 10 to 1000. Similarly, while some data are available for the higher temperatures of interest, little data are available for temperatures between about 600°F and 1000°F. Also, data on dynamic properties and combined effects of temperature and irradiation are sketchy.

The Fuel and Materials Section (4.0) provides an indication of the radiation effects to be expected and some mechanical properties needed for design of subassembly structural elements. However, structural

elements involve different shaping processes, such as casting and forging, and small but possibly important differences in composition. Heavy section welds produce local areas differing from the parent metal in composition, microstructure, heat treatment, and residual stress. Unlike other structural elements, cladding can and will be tested in the precise geometry to be used. Changes in geometry generally affect mechanical properties in ways which are not yet well understood. Irradiation induced swelling, such as that found to occur in the clad, may also prove to be a problem in other structural elements of the core, particularly those in which tight dimensional tolerances are necessary. These problem areas may pose questions of reliability and safety.

Facing materials are used for purposes such as bearings, bearing points, and the load-bearing surfaces of seals. The current program at LMEC should provide information useful for FFTF and early Demonstration Plant design, particularly in conjunction with the collection, review, evaluation, and publishing of handbook data by BNWL. The present emphasis is on the Stellite and Colmonoy alloys, and on the nitriding treatment for stainless steel. Use of cobalt-bearing alloys should be minimized, however.

While present LWR designs appear to offer few materials problems in this area, the anticipated increase by a factor of 10 or greater in core neutron fluence for the LMFBR will result in similar increases in irradiation levels for adjacent components and will require more extensive irradiation data than is available now. Similarly, the increase in core outlet temperatures from FFTF levels will considerably increase the operating temperature of many adjacent components. The indicated maximum temperature will be increased along with both the average temperature and the maximum temperatures for many components. These larger commercial plants will require larger and heavier sections in structural members. This will tend to increase thermal stresses due to both increased thermal gradients and transients.

An effective interchange of design requirements and available material properties is required to establish a reliable fuel subassembly that will withstand the LMFBR environmental influences during (a) fabrication, (b) nondestructive testing, (c) loading, (d) service life, and (e) unloading,

cooling and shipping periods. Careful analysis and experimental testing of representative fuel subassemblies are required to assure reliable performance.

The fuel element and subassembly are generally designed to allow the provision for some measure of resistance to plastic deformation. Adequate ductility is necessary for fabrication and to accommodate other deformations experienced by the fuel subassembly. Fuel subassembly components must withstand stress and strain cycling. Vibration and fluctuations in coolant pressure, variation in temperature and transients raise the likelihood of fatigue failures.

Other factors such as radiation induced property changes, corrosion resistance and others (table 6-1) must be carefully evaluated for both design and materials consideration before acceptance in the LMFBR system.

Table 6-1. Materials Information Required for
Fuel Subassembly Design

Basic Properties

Tensile-compressive-shear strength, hardness

Ultimate-yield, Young's Modulus, Poisson's ratio

Coefficient of thermal expansion, heat capacity, thermal conductivity

Notch sensitivity, ductility

Fatigue life for high cycle stress

Stability of Properties

Effect on above properties of irradiation, temperature, temperature cycling, time. Also effect of these on dimensional stability.

High-temperature (500 to 1400°F) creep rates and stress to rupture.

Fabricated components (welds and heat-affected zones), skin effects, biaxial stress.

Thickness, nominal composition, grain size, heat treatment, impurities, and fabrication variations.

(Continued)

Table 6-1 (Cont.)

Compatibility With Na

Corrosion rate as function of time and of sodium temperature, impurities, and velocity.

Stress corrosion, effect of material transfer to and from components by sodium.

Erosion rate as function of sodium temperature and velocity, material ratios in coupled systems.

Compatibility with Adjacent Materials (including same material)

Self-welding or diffusion bonding as a function of temperature-time-contact pressure. Galling. Chemical compatibility. Coefficient of friction and wear rates (for selected pairs of materials). Fretting corrosion.

6.3.1.4 Manufacturing and Fabrication

Due to the severe operating environment and quality required for the reactor core subassemblies, information on manufacturing and fabrication development is needed in the following areas.

- a) Development for the fabrication of the circular subassembly shroud with welded moderator channels and evaluation of alternate moderator attachment methods. An evaluation of weldability, welding distortions and structural integrity is required.
- b) Methods of fabricating and installing BeO moderator in fuel subassembly moderator channels must be determined.
- c) For the complete fuel subassembly, assembly techniques and methods for controlling straightness and dimensions are required. Standards, quality control measures and inspection methods must be determined to assure a consistently reliable fuel subassembly.

6.3.2 Control Subassembly

R&D information requirements related to the control subassemblies fall into the areas of thermal-hydraulics, structural-mechanical and materials. For the LMFBR, information is needed in the following areas:

- a) Determination by analytical and experimental methods of the hydraulic characteristics of both the shim-regulating and

safety control rod subassemblies. Also an evaluation of control element spacers and end support hardware is required;

- b) Determination by analysis and experimental tests of appropriate means for control rod cooling regulation that will not raise excessive thermal stresses on the control rod shroud and allows a safe operating temperature for the control rod materials under all conditions;
- c) Analytical and experimental investigations of the dynamic and vibrational characteristics are required to assure that the control rod subassemblies will meet the operational requirements imposed by reactor;
- d) The selection and demonstration of safe, reliable operation of the reference control rod holddown scheme. The present reference scheme employs a hydraulic concept with a mechanical backup holddown;
- e) Wear, galling and operating clearance of the control rod bundle within the guide-shroud must be evaluated to establish a design arrangement and materials that will avoid the possibility of a stuck rod accident;
- f) An evaluation by analytical and experimental means is required to guarantee that the control rod bundles drop freely under the influence of gravity when scrammed and to prevent rod flotation under operating conditions;
- g) Other considerations such as materials selection and fabrication, which are needed for the fuel subassemblies, are also required for the control rod bundles, and other hardware associated with the control subassemblies. Data are needed on the compatibility of B_4C with stainless steel and the use of tantalum in the sodium-nuclear environment.

6.4 PROGRAM DESCRIPTION

6.4.1 Fuel Subassembly

The objectives of this activity are to obtain, by analytical methods and out-of-pile experimentation, information regarding the mechanical characteristics, thermal behavior and hydraulic performance of the LMFBR fuel subassemblies and their components and to evaluate the information in terms of LMFBR design parameters. The results of this effort are required in support of design and optimization efforts on LMFBR fuel

subassemblies to ensure an economic and reliable product. The four basic activity areas for this program are as follows: (1) thermal-hydraulics, (2) structural-mechanics, (3) manufacturing and fabrication, and (4) materials selection and performance.

A description of these activities for the fuel subassembly development (excluding the fuel and cladding) is presented in the following.

6.4.1.1 Thermal-Hydraulics

An analytical and experimental program to improve pressure-loss correlations and define fluid mixing factors, local heat transfer effects and a flow-induced vibration threshold for the most attractive fuel element support systems will be performed. The scope of the program will be to develop drag coefficients, vibration threshold limits and mixing coefficients for several support systems in the parametric envelope defined in the reference concept. Also verification and performance characteristics of the hydraulic holddown will be determined by experimentation.

Tests will be performed in a hot water loop, with transparent test sections instrumented with velocity and static pressure taps, vibration sensors, dye-injection apparatus and load cells.

The program will also investigate the problem of support-plate cavitation where the following areas will be investigated: (a) The design and selection of subassembly nozzle and support plate seating surfaces, (b) Evaluation of the possibility of cavitation from typical fabrication tolerances and credible misalignments of the mating surfaces, and (c) Water testing of several promising designs in a loop in which cavitation can be detected by sonic means.

The problem of thermal bowing, which is considered the major reason for fuel element thermal-hydraulic instability, will require the use of an electrically heated fuel element mockup in flowing sodium. The use of this mockup will also be helpful in heat transfer and mixing studies. Testing will also demonstrate long term structural endurance, resistance to fretting, galling and stability of a fuel subassembly under simulated LMFBR conditions.

6.4.1.2 Structural-Mechanical

A program of analytical and experimental evaluation will be utilized employing water testing and sodium testing in the following areas:

(a) Mechanical performance and endurance of the fuel element spacer design,
(b) Performance of the wrapper tube (shroud) design and other fuel subassembly hardware under pressure loads, bending from simulation of core clamping, and simulated refueling loads. Vibration tests and thermal transient tests will also be performed to evaluate structural integrity.

The structural strength of the overall fuel subassembly will be determined for the operational loads which might occur under reactor conditions. Tests will be performed to establish mechanical stability of fuel subassembly hardware and dimensions under simulated thermal soaking and cycle exposure and mechanical loadings.

A structural vibration survey of the mocked-up fuel subassemblies will be conducted to determine natural frequencies and general response of the various subassembly components, particularly the fuel element spacers. Tests will be conducted in water loops with mockup fuel subassemblies to determine other dynamic response characteristics.

Environmental tests of fuel subassembly arrays will be performed in a sodium test loop at temperatures and coolant velocities equivalent to those expected in the C-E LMFBFR.

The problem of lateral support and vertical holddown of the core subassemblies will be investigated. Hydraulic holddown and an alternate mechanical means will be evaluated. Consideration will be given to the system malfunction thermal expansion, loss of flow, pressure reversal, etc.

6.4.1.3 Manufacturing-Fabrication

Because of the stringent operating environment, the reliability required, and the quantity involved in the construction and assembly of LMFBFR fuel subassemblies, specifications, process methods and controls must be established.

Methods of fabricating the fuel subassembly shroud, fuel element spacer, etc., must be developed both for fabrication of test articles and

future LMFBR purposes. Manufacturing processes that provide uniform (material and dimensional) shroud tubes and other components of the fuel subassembly must be established. Suitable welding techniques for attachment of the moderator channels to the shroud tube must be established.

A manufacturing fabrication program for the fuel subassembly hardware (excluding the fuel element) will be performed. The program will be interrelated to the structural-mechanical materials program and will concentrate on the following areas:

- a) Developing methods and process for welding moderator channels to the circular shroud tube;
- b) Developing methods for installing BeO moderator material into moderator channels;
- c) For weld joints, developing welds of adequate strength and uniformity, and methods of inspection and acceptable standards;
- d) Developing suitable methods for fabricating fuel element spacers and methods of attachment to fuel elements or shroud tube;
- e) Determining practical tolerance dimension control, straightness, shrinkage due to welding and other factors which may cause variance in the fuel subassembly configurations;
- f) Developing methods and processes for assembly of the fuel element bundle, alignment of spacers and inspection techniques utilizing test fuel subassemblies;
- g) Monitoring and assisting the Fuel Recycle Program (Section 8 AEC Program Plan) to provide specifications on such items as required tolerances, and guidance on alternative fabrication processes that may be more economical or better able to meet rigid tolerance requirements.

6.4.1.4 Materials

In the design of the fuel subassembly, consideration is required in the areas of material selection from the standpoint of sodium compatibility, mechanical properties, high temperature service and other factors. A program will be performed both analytically and experimentally to assist those needs of the Fuel and Materials effort, discussed in Section 7 of

the AEC Program Plan. Related efforts will encompass the following areas:

- a) Performing a structural-mechanical analysis combined with the results of initial water mockup tests, the functional requirements and material properties required for the components that make up the fuel subassembly will be determined;
- b) Determining basic material requirements for those portions of the fuel subassembly where vibrational or thermal fatigue and thermal shock must be considered in the operational lifetime;
- c) Determining requirements and establishing suitable materials such as spacer pads, and other diffusion or wear resistant coatings associated with the fuel subassembly;
- d) Determining whether tentatively selected materials and welds, etc., will provide satisfactory performance throughout in-core life;
- e) Developing alternate design arrangements or material selection in the event that specific material properties will not be present at end-of-life, e.g., the shroud tube at a neutron fluence of 10^{23} nvt.

6.4.2 Control Subassembly

Because of the long in-core life, and high reliability required of the reactivity control system, R&D will be required before the final design stages of the control subassembly. The development of a suitable control subassembly requires interfaces with the control rod drive mechanism (Components Design, Section 7.0), and safety (Section 5.0).

A program is required to investigate in detail the in-core portion of the control subassembly, both shim-regulating and safety, in the areas of packaging, aligning, guiding and cooling of the control elements. Development areas for the control subassemblies are somewhat similar to the fuel subassemblies, but more emphasis will be placed on structural-mechanical and thermal-hydraulic aspects. In particular, evaluations by water and sodium loop tests will be performed in the following areas:

- a) Determining the effects of vibration, bowing, thermal transients, and flow variation on control subassembly structural integrity. The effect of core clamping, heatup and cooldown, scram forces, etc., will be evaluated by mockup tests;

- b) A means of preventing galling or excessive friction of the control bundle within the guide tube, either by means of hard coated surfaces or roller devices will be developed;
- c) A means to accomodate variable heat generation, a large temperature difference across the guide shroud and rapid transients associated with control rod insertion will be analyzed and developed;
- d) The tantalum and boron carbide materials tentatively selected for the control subassembly elements will be investigated for sodium compatibility, jacketing selection, fission gas management and the ability to meet operational life requirements. This effort will be closely interrelated with the Fuels and Materials program;
- e) A means for preventing control rod float due to upward hydraulic forces will be evaluated and determined;
- f) Since under scram conditions large acceleration forces are imposed upon the control subassembly, the structural integrity of the control bundle, support rod and control bundle end rings will be evaluated by analysis and mockup tests in water;
- g) Cooling requirements of the control rods are variable with inserted position. To achieve a minimum bypass flow when rods are withdrawn and minimize thermal gradients in the shroud-guide, a variable coolant inlet control is desirable. Analysis and tests will be conducted to develop an appropriate coolant control device.

6.4.3 Facilities and Equipment

To perform the necessary testing - development required to determine thermal performance and hydraulic characteristics, the use of ex-reactor test loops will be required. Initial hydraulic testing will be performed in a water test loop capable of simulating velocities or Reynolds numbers equivalent to those expected in the LMFBR fuel subassembly. The test loop will be equipped with three or more test stations to allow simultaneous testing of fuel subassembly mockups.

A sodium test loop will be required to conduct small scale thermal bowing and heat transfer investigations. The test station will have provision for electrical heating to mock up thermal characteristics in the fuel elements, either in partial section or in full scale. The

sodium loop should be equipped with three or more test stations to perform structural-mechanical and isothermal endurance testing. This test loop and associated equipment will be used to verify the long term structural integrity of subassembly components in a temperature and flow environment for which they were designed. If available, existing industry and Government-owned sodium loops are considered adequate to perform the Core Design experimental work in this program.

Specialized test equipment is required for measurement of coolant mixing and fuel element vibration.

Welding and nondestructive inspection equipment will be required to effect assembly of the fuel subassembly and to assure adequate integrity and quality control.

Combustion Engineering manufacturing and materials facilities will be utilized for manufacture, inspection and basic materials test and evaluation. Subcontract developmental efforts will utilize available suppliers' facilities.

Miscellaneous sensing and recording instruments, pressure gages, mechanical loading and associated test equipment will be required.

6.4.4 Manpower and Costs

For the scope of work envisioned, a manpower estimate for the activity areas is outlined in table 6-2. A cost summary itemized for facilities, equipment and operating cost, including manpower, is presented in table 6-2 also.

6.4.5 Schedule

Figure 6-1 summarizes the time schedule, including identification and scheduling of major milestones.

Table 6-2. Core Design R&D

Manpower and Cost Estimate

Task	Man Years		Costs - 10 ⁶ \$			
			Operating		Other	
	Scientists & Engineers	Other	Labor & Overhead	Materials & Services	Equipment	Facilities
6.0 <u>Core Design</u>						
6.1.1 Fuel Subassembly	60-70	90-100	3.9-4.5	0.9-1.3	0.3-0.4	0.2-0.3
6.1.2 Control Subassembly	30-35	50-60	2.0-2.4	0.5-0.7	0.2-0.3	0.2-0.3

Figure 6-1
Core Design R and D Schedule and Milestone Chart

Task	Calendar Year	70	71	72	73	74	75	76	77	78	79	80
6.0 Core Design												
		1 ▽	2 ▽	3 ▽	4 ▽		5 ▽	6 ▽				7 ▽
6.4.1 Fuel Subassembly												
			8 ▽	9 ▽			10 ▽	11 ▽				12 ▽
6.4.2 Control Subassembly												

Figure 6-1: Core Design Development Milestone Legend

1. Begin thermal-hydraulic and mechanical analysis of fuel subassembly.
2. Begin design of fuel subassemblies to be tested in test facility.
3. Begin fabrication of test units for both water and sodium test loops.
4. Begin development of fuel subassembly manufacturing techniques.
5. Complete testing of fuel subassemblies in water and sodium loops.
6. Complete test evaluation and choose fuel subassembly design for demonstration plant.
7. Obtain results on fuel subassembly performance in demonstration plant.
8. Complete analysis and begin design of control subassemblies for testing.
9. Fabricate test control subassemblies.
10. Complete mechanical and thermal-hydraulic testing in both water and sodium.
11. Complete test evaluation and choose control subassembly design for demonstration plant.
12. Obtain results on control subassembly performance in demonstration plant.

REFERENCES

for Section 6.0

1. Brindley, J. H., "Friction Factors for Isothermal Turbulent Flow Through Several Reactor Fuel Element Geometries," NAA-SR-TDR-7662 September, 1962.
2. deStordeur, A. N., "Drag Coefficients for Fuel Element Spacers," Nucleonics Vol. 19, No. 6, pp 74-79 June, 1961.
3. Brindley, J. H., "Fuel-Spacer Drag Coefficients in Sodium," Nucleonics Vol. 23, No. 8, pp 96-97 August, 1965.
4. Burgreen, D., et al, "Vibration of Rods Induced by Water to Parallel Flow," ASME 57-A-94 December, 1957.
5. Putnam, A. A., "Flow-Induced Noise and Vibration in Heat Exchangers," ASME 64-WA/HT-21 December, 1964.
6. Davies, F. V. and Lis, J., "Heat Transfer and Pressure Drop Characteristics of Concentric Arrangements of Helical Coils," Symposium on the Use of Secondary Surfaces for Heat Transfer with Clean Gases, Inst. of Mech. Engr., London, England, p 42 1961.
7. Cochran, J. T., "SEFOR Core Clamping Model Tests, One-Sixth Segment," GEAR 5465 March, 1968.

7.0 COMPONENTS DESIGN: REACTOR VESSEL AND REACTIVITY CONTROL SYSTEM

7.1 DEFINITION

In the LMFBR plant all of the component elements associated with the reactor and reactor coolant systems are of interest. Items such as the steam generator, liquid metal pumps, intermediate heat exchangers, core-component handling system, reactor vessel system, reactivity control devices, etc., will require some degree of development. Since the C-E LMFBR follow-on study was directed toward the heat generation portion of the reactor, this components development program is directed toward the reactor vessel and reactivity control devices. More specifically the areas requiring research and development for these component systems are as follows.

7.1.1 Reactor Vessel

Reliability and safety impose severe demands upon the design and fabrication of LMFBR reactor vessels and components. Relative to LWR systems, higher temperatures and use of sodium coolant impose unusual and demanding requirements in the design, fabrication and operation of LMFBR reactor vessel systems and associated components. The areas related to thermal-mechanical design, material behavior and field fabrication of the reactor vessel require development beyond the present state-of-the-art.

Emphasis in the reactor vessel development Program Plan is on the thorough development of design, performance, and fabrication technology. Analytical design tools are required to better understand material behavior at vessel discontinuities, at locations of thermal gradients and shocks. Vessel closure considerations and methods of support must be refined. Design features most suitable for LMFBR service will require development and testing. Maintenance procedures and inspection and surveillance methods will require investigation and development. Thermal-hydraulic performance of the reactor vessel system requires additional analytical investigation and development.

Analytical and experimental investigation of internal components such as the reactor core radial clamp, core holddown arrangement, core temperature monitors, hermal baffles and shielding, etc., are required.

Additional development is required for the reactor vessel closure with its seals, bearings and mechanisms. The aspects of shop or field fabrication of large LMFBR vessels, in addition to design, handling and testing require further evaluation and development.

7.1.2 Reactivity Control Devices

High core power densities, and requirements for reliability and safety associated with larger LMFBR plants create a need for advanced technology for control devices. The large LMFBR cores impose severe configuration limitations on the interrelated design of control devices and the fuel handling system. Design improvements over existing reactivity control devices include increased reliability, improved maintainability, and reduced unit costs. A development program is required for near term application and longer term more advanced concepts. In general, the following areas will require investigation.

- 1) Environmental testing of bearings, seals, mechanical elements and electrical devices will be performed. Both components and test element configurations will be tested over a range of temperatures (up to 1200°F).
- 2) Development of new reactivity-control drive systems will be aimed toward early demonstration plants. Full scale units will be designed, fabricated and tested under simulated reactor conditions.
- 3) The study of reliability and maintenance, as well as the establishment of quality assurance measures for the design, manufacture, inspection and testing of mechanical hardware will be accomplished.

7.2 DESCRIPTION OF CURRENT TECHNOLOGY

7.2.1 Reactor Vessel

State-of-the-art for reactor vessels has been developed primarily for the LWR industry utilizing ferritic-steel, heavy-wall, high-pressure vessels (up to 2,500 psi) designed for temperatures in the area of 600°F. LMFBR vessels will require designs for low pressure, using thin walls of austenitic materials, probably larger than water reactor vessels, and will be exposed to sodium coolants at temperatures up to 1200°F.

The existing U. S. vessels for sodium service (Fermi, Hallam, EBR-II, and SRE) range in size up to 26 ft dia by 33 ft high and have been designed for operating temperatures up to 1000°F except for SRE, which was operated for short term at up to 1200°F.

LMFBR reactors may utilize the same or similar basic concepts; however, design details differ widely to suit the considerably diverse operating parameters. Reactor components directly reflect these varying requirements in design features or specific operational considerations, particularly to accomodate the accurate indexing and positioning requirements for fuel handling. Presently, reactor vessels are categorized as: the "pot-type" containing the primary system submerged in sodium, as incorporated in EBR-II, and the "loop-type" with the primary system coupled externally to the vessel, as in Fermi, Hallam, and SRE.

Both the EBR-II and Fermi designs incorporate several similar features, for example: under-the-plug refueling with under-sodium transfer and storage within the sodium containment vessel; use of stainless-steel-clad graphite neutron shields; and outer primary tanks. In addition, the problem of dimensional stability and alignment of heavy, multiple assemblies supported on relatively thin-walled high temperature structures or each from a separate reference datum, is of special significance to refueling and is inherent in both designs. However, the physical arrangements differ greatly, particularly in the method of support, reactor-vessel and primary-tank internals, closure head and coolant circulation.

The EBR-II reactor vessel is carried on a structural-steel base, consisting of horizontal beams that distribute the reactor load to the primary tank, which in turn is hung from a steel column structure. The independently supported primary tank cover assembly contains a double-plug indexing system for refueling as well as ports for the primary-system components located within the primary tank. Radiation shielding surrounds the reactor vessel, and sodium within the primary tank is at the nominal core inlet temperature of 700°F.

Both the reactor vessel and primary shield tanks of Fermi are bottom-supported, with the top cover carried by the shell of both vessel and tank. The single indexing refueling plug used is thicker than the EBR-II plug, since the top of the core is not shielded.

No reactor vessels have failed to date. Lack of adequate statistical evidence, by reason of the limited number of vessels with limited operating life, prohibits direct extrapolation of present experience to future LMFBRs. A more comprehensive state-of-the-art is compiled in the LMFBR Components Program Plan, Section 3.

7.2.2 Reactivity Control Devices

The type (shim or safety) and location (top mounted or bottom mounted) of the reactivity control element has a significant effect on the drive system design in regard to speed, requirements for scram time and acceleration, spatial configuration and mechanism details.

Most of the major liquid metal fast reactor facilities built or under construction incorporate electro-mechanical drives, where electromagnets provide quick rod release. SEFOR uses hydraulically operated drives activated by quick opening valves located outside the reactor vessel. Mechanical or pneumatic devices are used, depending on rod weight to assist gravity insertion and increase scram acceleration.

A number of sodium reactor plants also use electro-mechanical control rod drives for regulation and shim purposes, in addition to performing as safety rods for scram.

Dounreay is unique in that the drive mechanisms are not sealed from the reactor environment. All other plants use seals to isolate the drive mechanism power unit, drive train and electromagnet from the sodium and inert cover gas blanket. Thus, only the gripper and extension shaft are exposed to the sodium.

Shock absorption of the control element bundle is provided by hydraulic dash pots located both outside and within the reactor. Internal dash pots use reactor coolant as the fluid medium.

In general, the performance of control and safety rod drives has met operational and safety requirements. However, service reliability and mechanical and electrical maintainability require improvement to meet the operational life requirements expected for large LMFBR plants.

Problems with mechanical elements (similar to those encountered for fuel-handling components) need to be resolved. Methods of sealing need to be improved.

especially with increased stroke lengths. Erratic and large friction loads on reciprocating shafts have been obtained with standard packing glands. Welded, nested type reciprocating bellows have been used with good success. However, reliability must be improved, and spring constants reduced for those cases where the scram is through the bellows seal.

High-temperature sodium, inert cover gas, and radiation environmental conditions preclude the use of conventional bearings and lubricants. Metal-to-metal galling, self-welding, and wear are problems that require care in selection of materials. In general, the criteria applied to bearings for fuel-handling components also are applicable to bearings for drive systems, i.e., hard contact surfaces and base metal highly finished surfaces, and minimum unit loads. Experience indicates that the use of thrust bearings under sodium should be minimized, and sleeve bearings should have larger clearances than those normally used with conventional lubricants.

Methods for alignment, raising and lowering of the gripper as well as the gripper itself are areas of major concern. Drive shafts, generally long and slender, require close tolerances and straightness to assure alignment between the drive unit and the core control element.

Grippers for the engagement of the drive shaft to the control bundle or rod have incorporated cam operators actuated by contact with the control-element assembly as a function of position. The designs must be fail-safe, particularly where the gripper acts as the scram latch in Fermi. Material selection, geometrical arrangement (assuring adequate clearance for minimizing jamming by contaminants), stress concentrations, etc., must be considered in the gripper design. Use of springs for scram assist must be examined closely to assure against any tendency to cause bonding, relaxation at the high operating temperatures, and changes in spring constants due to radiation effects. Internal dashpots which use sodium must be designed to assure the alignment and clearances necessary to avoid jamming and contaminant plugging.

Electromagnet latches used for release of safety rods must be positively actuated, rapid in response, fast acting, and fail-safe to assure negative reactivity insertion, and must be consistent with space restrictions and high operating temperatures. Electromagnets are usually sealed from the environment.

Fermi utilizes a separate scram system at the gripper below the seal, and EBR-II scrams through the seal. The effects of differential thermal expansion of the drive-element with the reactor system (core and vessel) must be considered in designing the drive. Development of improved mechanisms leading to greater design simplicity is expected to provide a more reliable, more readily maintainable system.

Drive-system orientation and spatial configuration have a significant effect on both facility and reactor design as well as on the procedures required for fuel handling. Location of the drive units directly above the core, as for Fermi and EBR-II, required mounting on the refueling rotating plug. As a result, positive interlocks are required to assure control-element placement in positions of maximum negative reactivity prior to removal of the drive units before refueling. The offset Dounreay system eliminates this problem but requires a more complex system located within the deleterious reactor environment.

A more detailed state-of-the-art is presented in the AEC LMFBR Components Program Plan (Section 3).

7.3 NEW INFORMATION NEEDED

7.3.1 Reactor Vessel

The stringent operating environment imposed by the LMFBR creates special problems related to: materials, performance characteristics (thermal, hydraulic and mechanical), dimensional stability and safety.

Data on physical properties must be determined as a function of time and environment for materials that are considered for vessel construction. The effect of fabrication processes and weld zones on material properties must be determined.

Analytical tools for the determination of the effects of rapid thermal transients and thermal gradients associated with the sodium coolant in the sodium containment vessel must be developed. Particular areas of interest are the coolant nozzle, closures and flange connections and other areas of structural discontinuities. In addition to stress analysis, techniques to assure structural

integrity, methods of evaluating the geometric changes due to temperature difference and material plastic flow associated with high operating temperature vessels must be determined.

Use of sodium coolant with its relatively high thermal conductivity leads to more severe thermal transients and flow or temperature maldistribution than those normally encountered in water systems, resulting in greater thermal-stress conditions. The significance of thermal transients is further increased by the greater thermal expansion and lower thermal conductivity of the austenitic steels, as compared with the ferritic steels used for LWR's. Therefore, system designs for both reactor and sodium containment vessels must compensate for these effects by using thermal and flow baffles. Analytical procedures and design arrangements suitable for LMFBR vessels must be developed and proven by experimental evaluation.

Attainment of close dimensional alignment between many operating mechanisms is difficult to maintain because of geometric changes due to temperature differences and material relaxation. Assurance of stable dimensional relationships is required for reliable control-rod movement; predictable geometric coefficients of reactivity under all operating conditions; predictable flow distribution for core coolant, structural and shield cooling; and positive retention of core configuration are needed. Methods of supporting internal interacting subassemblies from a common, fixed, nonshifting datum must be developed. This datum must be maintained at a limiting temperature well below the plastic (creep) range using a material-geometry combination such that there is no displacement due to differential thermal expansion.

The reactor vessel system must provide safety features to prevent ejection of fuel and control rods from the core; to assure submergence of the core in sodium if a rupture occurs in the primary containment; and to assure blast containment. Particular attention must be paid to the structural design of the shield cover and refueling plug. Specifically, the design must accommodate the kinetic energy acquired by various parts involved in a nuclear-power excursion.

Also needed are methods of supporting reactor and sodium-containment vessels and vessel closures. The advantages and limitations of top support vs bottom support and use of the sodium-containment vessel as a support for the reactor vessel must be determined.

Along with the design development, there is need to determine: requirements and methods for providing continuous surveillance and maintenance of the vessel over the 30-year projected operating lifetime; methods for reliable analytical predication of vessel performance as a function of both static and dynamic operating conditions; and the procedures needed to field manufacture the vessel if necessary.

In regard to ASME Boiler and Pressure Vessel Code, Section III, an extension of the code rules for design and analytical procedures to include temperatures in the creep range under steady state pressures and low cycle fatigue is needed.

Additional development and test evaluation for providing radial core clamping and demonstration of the hydraulic core holddown is required. Also, suitable means for introducing and supporting thermocouple instrumentation for the core subassemblies is needed.

The use of flow schemes, bypass cooling and thermal liners to minimize operating temperatures of highly stressed internal members and vessel walls or nozzles must be evaluated by further analysis and experimental verification in the form of hydraulic model testing.

To compensate for radial pressure losses in the inlet plenum and to provide coolant orificing to specific core subassemblies, further evaluation by analysis and experimental tests is required. Schemes for preventing inadvertent dislocation of inner seed, outer seed and blanket subassemblies by use of combination orificing and locating sleeves in the core grid support plenum must be determined. Clearance seals required for the core support grid structure and at subassembly inlet nozzles require both hydraulic and mechanical evaluation and tests.

Thermal bowing and movement or flow induced vibration of the core subassemblies within the reactor vessel can be minimized by appropriate use of circumferential locking devices. Power-driven or thermal-sensitive actuation schemes have been considered. However, additional analytical and experimental evaluation by mockup tests is required.

Because of the necessity to allow clearances for thermal expansion of various vessel internals such as the core barrel, thermal shields, etc., flow-induced vibrations must be carefully evaluated to avoid structural damage. Both analytical and experimental evaluation in the form of hydraulic mockups (reduced scale) are needed.

7.3.2 Reactivity Control

The required functions and system requirements for the reactivity control devices must be closely defined, and reliability must be attained through design, fabrication and testing of the mechanical elements and electrical devices.

Design simplification of state-of-the-art concepts is needed and can be obtained through development of mechanical elements and mechanisms. Such reactivity control devices and systems would be suitable for FFTF and near-term Demonstration Plants.

Future long-term commercial LMFBRs will require new and advanced systems having greater reliability, easier maintainability, faster response times and smaller space.

Basic needs related to the reactivity control devices are:

- 1) To arrive at a mutual understanding of the functional requirements, including suitable criteria for design and fabrication;
- 2) To perform a thorough evaluation of the control devices, location, method of sealing, etc.;
- 3) Development of a more reliable and maintainable mechanism and electro-mechanical elements;
- 4) To determine the behavior of materials considered for use under high temperature sodium and radiation environment;
- 5) To perform studies of concepts for emergency reactor shutdown that are independent of the reactor control system;
- 6) To obtain performance data from development and testing of a prototype reactivity control system.

7.4 PROGRAM DESCRIPTION

7.4.1 Reactor Vessel

The objective of this effort is to establish a sound technological base to permit the detailed design, construction and successful operation of the reference LMFBR reactor vessel and associated components. Technical scope of this activity will be applicable to the C-E LMFBR reference reactor vessel design and related structural and internal components. Basic components of the vessel are the vessel body, supports, attachments, and closure. Vessel internals consist of inner core support, retainers, holddown members, thermal baffles, flow device and radiation shielding.

To develop a sound technological base it will be necessary to provide a firm engineering evaluation of potential problem areas associated with the concept. Initial preliminary design efforts of vessel designer-fabricators will be performed in conjunction with the reactor designer responsible for the overall Plant Design. From this, screening of design possibilities, trade-offs and technical areas warranting continuing development will be better defined. Basic areas of development are presented in this plan.

The task areas envisioned for the development of the reactor vessel at this point in time are summarized in the following:

7.4.1.1 Design Technology Development

This activity will employ all major technical disciplines to permit prediction of the reactor vessel and components under steady state, transient and abnormal operating conditions.

The technology will be compiled in the form of a work book or design manual to allow engineering application for the design, fabrication and testing to obtain safe, reliable and economic LMFBR reactor vessel and associated components. This activity will also provide guidance and will develop new analytical methods that assist in reactor vessel state-of-the-art advancement.

This activity will provide analytical techniques for system simulation that predict component behavior in the areas of thermal, hydraulic, mechanical and

metallurgical considerations. Among the specific design arrangements and analytical methods to be evaluated under this activity are:

- a) Analysis of the vessel structure, closure, cover, flange nozzles etc., to predict maximum stress, deflections, distortion etc., due to static, thermal pressure and dynamic loading;
- b) Methods of determining temperature distributions in structural members due to steady state and temperature transients and that due to radiation attenuation;
- c) Investigations of methods or models to predict vessel behavior under seismic loadings;
- d) Means for predicting cumulative damage resulting from fatigue, creep or ratcheting will be investigated to assure vessel structure life;
- e) Methods of analysis to determine the integrity of structures operating in the elastic-plastic range, such as at structural discontinuities (shell-to-head-joints, closure flanges, nozzle penetrations, etc.).

7.4.1.2 Design Development: Vessel

The activities of this task are to:

- a) Evaluate the reference reactor vessel design (or alternates) in regard to safety, reliability, maintainability, plant performance and fabricability; and
- b) Identify major problem areas associated with the reference design and prepare appropriate plans for analytical and/or experimental evaluation. Typical problem areas will include: (1) a vessel support system that will provide for radial expansion yet minimize piping reaction loads, (2) a vessel support system that will safely withstand horizontal and vertical seismic disturbances, (3) temperature transients and gradients on critical vessel members, (4) prediction of useful life under conditions of plastic creep, thermal fatigue and thermal ratcheting, (5) methods of in-place vessel surveillance and on-line leak monitoring, and (6) selection of flow distribution, hydraulic reaction, flow induced vibrations, wall temperature gradients, gas entrainment, etc.;
- c) Develop techniques and/or hardware to be used for verification of LMFBR reactor vessel requirements. Thermal, hydraulic and mechanical testing will be performed on full size segments and scaled down models. Model and test simulation will be representative of the full size vessel components and will reflect thermal transients, thermal gradients, pressure and mechanical loadings.

7.4.1.3 Design Development: Vessel Closure and Internals

Activities of this task include:

- a) Evaluation of the reference reactor vessel arrangement in regard to closure and basic internals from the aspect of reliability, safety, maintainability, plant performance and fabricability;
- b) Identification and selection of major problem areas associated with the reference design arrangement and preparation of an appropriate plan for analytical and/or experimental evaluation. Typical problem areas to be considered are: (1) sealing of reactor cover gas, (2) dimensional stability of cover, core support grid and reactivity control drive, (3) hydraulic flow profile and closure seal leakages, (4) internal heat generation and methods of minimizing large thermal gradients, (5) maintenance of dimensional stability of core arrangement under thermal or hydraulic variations, (6) effect of shock loadings (seismic or transients), (7) installation, (8) support of temperature or other core instrumentation;
- c) Developing techniques and/or hardware to evaluate the problems and arrive at solutions to meet the LMFBR requirements. Thermal, hydraulic and mechanical testing will be performed on full size segments or representative scaled down models. Water loops and small and large scale sodium test facilities will be utilized.

7.4.1.4 Fabrication Development: Vessel and Internals

Fabrication standards and specifications are required to assure LMFBR vessel and components reliability. Application of current general fabrication processes, nondestructive testing techniques, qualification procedures and process specifications will be evaluated.

Standards and specifications will be verified by fabrication and tests of adequate models. The activities of this task will include the following:

- a) Current fabrication processes will be surveyed to determine LMFBR applicability;
- b) Detailed process specifications and procedures will be prepared;
- c) Process development will be demonstrated by fabrication of selected elements, including a description of the development program, data obtained and evaluation.

Field fabrication of LMFBR vessels will be evaluated based on current use for some LWR vessels. Particular emphasis will be placed on quality control attainable. Specific problem areas to be evaluated are: (a) handling and loads

incurred during fabrication, erection and installation, (b) in-place boring, drilling and machining operations, (c) field welding and heat treatment, (d) field inspection methods (radiographic, ultrasonic, etc.), (e) inert gas shrouding, weld preparation, fit-up, special fixturing and restoration of component surfaces.

Quality control is of utmost importance since many steps of the field fabrication process are irreversible.

Specifications will be developed and used in the fabrication of mockups of the critical areas of a field erected vessel.

7.4.2 Reactivity Control Devices

The objective of the activity is to provide a technological base to permit the design and fabrication of reactivity control drives for commercial LMFBR plants.

Technical scope of the effort will be applicable to the C-E LMFBR reference design and other LMFBR systems with similar operating requirements. Included in the technology development is the drive train, drive shaft, latches, gripper mechanism, scram assist devices, shock retarders, seals, bearings and method for position indication and feedback.

The task areas envisioned for the development of the reactivity control drive system core are summarized in the following:

7.4.2.1 Design Technology Development

Major technical disciplines will be employed to predict performance of the reactivity control system under steady state, transient and abnormal operating conditions. Technology will be compiled in the form of a work book or design manual to allow readily available engineering application for the design, fabrication and testing to obtain safe, reliable and economic LMFBR control rod drive systems.

Analytical procedures, design methods and design features to be evaluated are as follows:

- a) Static and dynamic structural analysis due to thermal, hydraulic and mechanical loadings;

- b) Analytical models that predict operation of drive systems in sodium-sodium vapor environment, dispersion and containment of fission products, wear, fatigue of seals and bearings;
- c) Means for predicting damage resulting from creep, fatigue and materials, etc.;
- d) Methods for predicting the dynamic response of drive system for use in plant control system analysis.

7.4.2.2 Design Development

This activity covers the design of reactivity control drives. This effort will include the description of the design criteria, preliminary analysis, and trade-off evaluations and drawings to permit evaluation of the concept potential.

The specific areas in this task are:

- a) Performance of a preliminary design in sufficient depth to evaluate safety, reliability, maintainability, in-plant performance and fabricability;
- b) Identification of major problem areas associated with the LMFBR reference design and making preparations for analytical and/or experimental evaluations. Some major areas of concern are:
 - (1) Effects of relative movement of core area control members due to differential thermal expansion, (2) thermal and hydraulic dynamic influences from core coolant, (3) space limitations in core, reactor vessel closure, etc., (4) maintainability, cleaning, decontamination, (5) reliability of major components, e.g., bellows seals, bearings, etc.;
- c) Selected critical mechanical and electrical elements and mechanisms of the reactivity drive will be fabricated and proof and life tested in support of preliminary design selection, fabrication technology development and detail design selection. This testing will provide analytical technique verification, verify performance predications, identify geometric and functional limitations and will provide a demonstration of feasibility for LMFBR service. Testing will be performed on the basis of application and design-performance requirements. Typical data to be obtained for items such as seals, bearings, mechanisms and electrical devices are: (1) temperature effects, (2) materials compatability with system environments, (3) reliability and life, (4) capability of inspection, maintenance or replacement, (5) lubrication requirements, (6) requirements of load, speed or torque, (7) friction, wear, (8) thermal effects or heat dissipation aspects, (9) overload limitations, (10) effect of misalignment and abnormal reactor loads, and (11) electrical performance characteristics.

Where possible test facilities used for Core Components Handling Equipment development will be utilized.

7.4.2.3 Fabrication Development

This activity will ensure the development of fabrication methods and specifications for LMFBR reactivity control drives.

Maximum use will be made of the state-of-the-art, which will include fabrication processes employed in the manufacture of both nuclear and nonnuclear components. Standards and specifications will be verified by fabrication and testing of adequate models or mockups. Typical problem areas which may require development are: (1) Development of welding techniques for joining dissimilar metals, (2) Development of surface treating techniques required to reduce friction and the probability of galling, (3) Special machining operations, e.g., hobbing, electrical discharge machining, deep boring, etc., (4) Maintenance of straightness and alignment, (5) Contour milling of special shapes, (6) Non-destructive inspection methods, and (7) quality assurance programs to achieve reliability requirements.

7.4.2.4 Prototype Development

The objective of this activity is to provide a detail design and to fabricate and perform proof tests on prototype reactivity control drives suitable for LMFBR service. Performance analysis, drawings, material selection, specifications, procedures, failure and reliability analysis will be completed in detail. Representative components will be fabricated from the designs. If necessary, iterative designing, testing and designing may be required on components before overall system assembly testing is performed. Tests will be conducted under simulated plant environment and operating conditions.

The designer-fabricator will prepare a test program including the items, conditions, range and variations in operating parameters, and data required. Detail test procedures and facilities best suited will be established by LMEC.

Test duration and conditions will be adequate to assure that a production unit, if fabricated and operated in a manner identical with the prototype model, will perform as predicted to meet plant requirements. All test data will be evaluated by the designer-fabricator of the reactivity control system. This testing is expected to determine performance, reliability, maintainability and operating life of the reactivity control drive assembly and integral parts and subsystems.

7.4.3 Facilities and Equipment

To perform the necessary testing and development related to the reactor vessel the use of small scale loops and laboratory facilities and equipment will be required. However, technology verification and experimental evaluation will be performed only by test on critical portions of the reactor vessel, e.g., supports, nozzle penetrations, closure seals, etc. There will be no attempt at producing a finished design and testing of a complete vessel mockup in this program. Testing of reduced scale models or full scale models of segment or critical areas will be evaluated by water testing for the development of hydraulic data. Testing of full scale elements or subassemblies of the reactivity control devices will be employed.

Existing industry and Government-owned facilities are considered adequate to perform the necessary experimental test work in this program. However, special test equipment is required for measurement of coolant mixing, vibration and structural loading effects.

Combustion Engineering manufacturing facilities can be utilized for manufacture, inspection and test evaluation. Subcontractor developmental efforts will utilize available shop or laboratory facilities.

Miscellaneous sensing and recording instruments, pressure gages, mechanical loading device, and associated test equipment will be required.

7.4.4 Manpower and Costs

For the scope of work entailed, a manpower estimate for the two task areas (reactor vessel and reactivity control drives) is outlined in table 7-1. A cost summary itemized for facilities, equipment and operating cost, including manpower, is presented in table 7-1 also.

7.4.5 Schedule

Figure 7-1 summarizes the time schedule, including identification of major milestones.

Table 7-1. Components Design R&D
Manpower and Cost Estimate

Task	Man Years		Costs - 10 ⁶ \$			
			Operating		Other	
	Scientists & Engineers	Other	Labor & Overhead	Materials & Services	Equipment	Facilities
7.0 <u>Components Design</u>						
7.4.1 Reactor Vessel	65-75	95-105	3.5-4.0	0.8-1.0	0.2-0.3	0.2-0.3
7.4.2 Reactivity Control Devices	55-65	90-100	3.3-3.8	0.8-1.1	0.2-0.3	0.1-0.2

Figure 7-1
Component Design R and D Schedule and Milestone Chart

Figure 7-1: Components Design Development Milestone Legend

1. Complete the analysis and design phase of the reactor vessel and the vessel closure and internal design development program.
2. Complete the fabrication development for the reactor vessel.
3. The testing and evaluation phase of the design development program for the reactor vessel and vessel closure and internals is completed.
4. Finalize the reactor vessel design for the demonstration plant.
5. Development of the reactor vessel design technology is completed.
6. Complete the design technology development for the reactivity control devices.
7. Complete the design development phase of the reactivity control devices program.
8. Fabrication development is completed.
9. The program for the design and testing of a prototype control rod drive is completed and a final design chosen for demonstration plant.
10. Obtain results on reactivity control drive in demonstration plant.

8.0 COMPONENTS DESIGN: FUEL HANDLING SYSTEM

8.1 PROGRAM DEFINITION

The following areas of the reference refueling scheme require research and development work for the scheme to progress to an operating system.

A. The subassembly handling machine (SAHM) will require performance and endurance testing. This machine should be a prototype of the component presented in CEND-337, Vol. I. The machine should be complete as shown in figure 4-5⁽¹⁾ with the exception of the full shield plug which is not required for testing. However, the complete automatic and manual control, full instrumentation, and drive mounting plate bearing and atmospheric seal must be included.

B. The subassembly transfer machine (SATM) requires performance and endurance testing. It should be pointed out that this machine, because of its liquid metal cooling system, may be used in any refueling scheme whether hot cell or under-the-plug since it allows immediate removal of spent subassemblies from the reactor to external decay storage. The test prototype should be as presented in CEND-337, Vol. I, and complete as shown in figures 4-3 and 4-4.⁽¹⁾ It should include the following:

- 1) Complete automatic and manual control and operating systems;
- 2) The carriage on which the machine is mounted;
- 3) An exit port with which to mate and seal during testing.

C. The Spent Subassembly Cask Cart⁽¹⁾ requires performance and endurance testing. A prototype of the machine shown in figure 4-7⁽¹⁾ complete with automatic and manual operating and control systems is required together with interface operating equipment such as:

- 1) Seal valve at the exit port;
- 2) Hoist to insert and remove finned pots.

D. The Equipment Removal Container, figure 4-6⁽¹⁾, is another item which will be required by all LMFBR plants. A container of this type must be

used to remove and install all large components such as pumps, and IHXs. In the C-E reference refueling scheme the container is also used during the interchange of the control rod drive island and the SAHM. This container will require performance and endurance testing. The large seal valve and the hoisting mechanism must be included.

The following R&D items are mechanisms, subassemblies, or technology areas which are critical to the refueling scheme components.

E. The angular pin drive coupling shown in figure 4-5⁽¹⁾ must be tested. The mechanism is used to transmit rotary motion from the SAHM drive motors, which are located in an air atmosphere, to the handling tools within the reactor vessel. An essential part of the mechanism is the diaphragm seal which prevents gas leakage into or out of the reactor vessel.

F. The most unique and essential part of the SATM, figures 4-3 and 4-4⁽¹⁾, is the subassembly decay heat removal system. A model heat transfer system which includes the sodium-filled subassembly transfer pot, one complete closed NaK loop, and an externally finned NaK intermediate coolant tank must be tested. The system must allow vertical movement of the closed NaK loop.

G. An automatic and remote welding system must be developed to be used in the cask cart, figure 4-7⁽¹⁾, for welding cover lids onto the finned pots after the spent subassemblies have been deposited. The lid welding system must include the lid positioning, centering, clamping, rotating, and welding.

H. The heat transfer capabilities of the finned pot system used for canning the spent subassemblies in lead-bismuth prior to offsite shipping must be tested⁽¹⁾. Tests to determine if the potting compound should remain liquid during shipping or whether the compound can solidify and still adequately cool the three subassemblies through heat conduction to the outside surface of the finned pot are required. This work would also include establishing the exact composition of the potting material.

The following items are of a more general nature.

J. Operation of complex mechanisms such as ball screws and grippers after they have been removed from sodium, stored in an inert atmosphere without cleaning, and then returned to sodium service must be investigated. The mechanisms must be designed and developed so that they drain dry upon removal from sodium and the thin film of sodium which remains does not cause a malfunction when they are reinserted into sodium.

K. A mechanism for effecting a positive atmospheric seal at component interfaces must be developed. When one component of the refueling operation mates with another in order to transfer subassemblies between them they must be sealed to each other by a static seal which prevents air from entering and activated gases from leaving the reactor system. In the reference refueling scheme⁽¹⁾ these seals are at the mating surfaces of all of the seal valves which are located at the exit ports, the lower end of the SATM barrel, and the upper end of the cask cart.

L. Sliding metal bearing surfaces of guides, keyways, and telescoping tubes require development since they are integral parts of the working mechanisms of all of the refueling components. These bearing surfaces must be developed to operate under sodium and in sodium-saturated gas atmospheres without the aid of normal lubricants.

M. A large seal valve which prevents atmospheric contamination during equipment transfer with the Equipment Removal Container (ERC) must be developed. As previously pointed out in paragraph D the valve is a critical part of the ERC which must be used by all LMFBR plants for component removal. In the C-E reference refueling concept, valves are required at the reactor, the storage tank, and the bottom of the ERC.

8.2 DESCRIPTION OF CURRENT TECHNOLOGY

A. There is no existing machine which performs all of the functions that the subassembly handling machine (SAHM) performs. The SAHM functions, which are locating, gripping, inserting and extracting, and rotating a subassembly to a deposit point, require four machines in the EBR-II reactor and two machines in the Fermi reactor. The existing component which most

closely performs the SAHM functions is the Fermi OHM (offset handling mechanism). However, the OHM requires a single rotating plug which operates in conjunction with it to locate over the core positions. The Fermi OHM has been used to transfer many subassemblies and has generally functioned satisfactorily, although most of these transfers were performed at a lower sodium temperature than their refueling design temperature (approx. 600°F).

The SAHM features which the OHM does not have and therefore require testing are: (1) lateral gripper movement (this allows elimination of rotating plugs), and (2) component removal from the reactor vessel after refueling.

B. The base of technology for the subassembly transfer machine is approximately equal to that of the SAHM since the subassembly transfer machine is mechanically quite similar to the EBR-II fuel unloading machine which has been in operation for some time. The basic differences between the SATM and the EBR-II machine is that the SATM transfers the subassembly within a sodium-filled pot. Also, it has a closed NaK loop decay heat cooling system. Closed NaK cooling loops are not new; however, their application in a subassembly transfer machine is unique and the system must be proof tested as an integral part of the SATM.

C. The various features and the operating principles of the spent subassembly cask cart have been developed through the refueling schemes for the Dounreay, Rapsodie, and Fermi reactors. All three use a shielded cask cart to transfer subassemblies into and out of the reactor building; however, only Dounreay and Rapsodie can the spent subassembly in lead or lead alloys and this is not accomplished within the cask cart. In the C-E cask cart the operations of canning the spent subassemblies, seal welding the pot lid, and leak testing the sealed pot are performed within the cask cart.

The general technology has been established by these three refueling schemes and found to be satisfactory. However, a problem of maintaining exit port, and rotor alignment because of structural strains caused by shield movement during heating and cooling cycles of the Fermi cart resulted in the redesign and fabrication of a new transfer cart.

Since the C-E design does combine more features than the existing machines the machine will require proof testing.

D. Equipment removal containers (ERC) similar to the C-E design have been developed and used in the EBR-II and Fermi plants.⁽²⁾ The original Fermi design had a steel cylinder with a large seal valve at the bottom and a large rubber inverted cup attached to the top of the steel cylinder. The rubber cup fitted inside the steel cylinder when the ERC was empty. The ERC was placed over the component to be removed and the overhead bridge crane hook was connected to the component through the rubber cup. As the component was raised the rubber cup unfolded out of the steel cylinder and an argon atmosphere was maintained around the component. However, argon pressure within the system could not be controlled closely enough to prevent the rubber cup from alternately ballooning and collapsing.

The Fermi ERC was redesigned by increasing the length of the steel cylinder and using a piston with a double rubber blade wiper, and argon purge between the blades, as the moving atmospheric seal in the top of the ERC. The redesigned ERC also replaced the rolling gate valve at the bottom with a flat plate slide valve. Deflections in the rolling gate valve caused it to malfunction. Operations with the redesigned ERC have been satisfactory. All three of the Fermi primary sodium pumps have been removed from the system and replaced. The ERC has also been used in the OHM repair.

The base of technology is quite well established; however, the C-E design, being a more sophisticated unit, will require proof testing.

E. Angular pin drive couplings and seals have been used extensively in the Dounreay refueling components. The units are referred to as, "hermetic couplings,"⁽³⁾ or, "Articulated drives,"⁽⁴⁾ by the UKAEA designers in the PFR program. No reference can be found to their use in this country. The base of technology seems to be quite well established with the UKAEA but must be introduced into this country.

F. The basic information and technology required to design, fabricate, and operate the closed NaK loop decay heat cooling system for the SATM has

been developed for many years. Liquid metal test loops, both natural and forced circulation, were operated before the advent of liquid metal cooled power reactors. The facet which requires demonstration is the unique combination of the complete system which involves a closed, double hairpin NaK loop with heat transfer surfaces at both ends which move into and out of free surface, natural circulation, liquid metal pots. With this system, decay heat may be removed from a subassembly within the SATM and rejected to the containment building atmosphere without violating the inert atmosphere within the SATM.

G. The technology of remote automatic welding is very well developed as demonstrated by the fact that automatic TIG welding machines are used to weld steam generator tubes to the tube sheet stubs from the inside. In LMFBR steam generators these are critical welds and must be leak tight since a flaw resulting in a leak would cause a sodium-water reaction and plant shutdown.

This established welding technology must now be incorporated into a complete system which will position a lid onto a finned pot, center, clamp, and rotate the assembly while seal welding the lid. Proof testing of this remote automatic system is required.

H. The basic heat transfer correlations for natural convection of liquid metal within a finned pot containing a heat source are established and have been used to design systems for Fermi and others. Also, the technology of using a lead alloy as a potting compound for spent subassemblies within these finned pots for the purpose of both a heat transfer and shielding medium has been established at Dounreay and Rapsodie.

However, the specific lead alloy composition to be used in the C-E design and the performance characteristics of that compound within its system must be proof tested.

J. There does not seem to be much information covering operation of mechanisms in sodium after having been stored out of sodium in an inert atmosphere with a residual film of sodium remaining on the surfaces. Some reference is made in the literature to the effect that components should

be designed so that there are no crevices or corners to accumulate oxides. Also, components should be designed with open structures which may drain freely.^(2,5)

It appears that a complete R&D program is required to establish the factors which effect or control mechanism operation in sodium upon reinsertion after storage in an inert atmosphere.

K. The state of the art for static seals, which are required at the mating surfaces of all of the refueling components, is well established and has been proven through the operation of the liquid metal cooled reactors such as Fermi, EBR-II, Dounreay, Rapsodie and Hallam.

This seal is generally standard rubber or metal gaskets, packing glands, and rubber or metal O-rings. The primary difficulty in the selection of static seals is environment.

Proper groove design and care in fabrication are necessary to achieve good seals with metal O-rings. Surface finish of the groove, in particular, is critical. Lapped surfaces are generally used. Where temperature and atmosphere permit, synthetic rubber or plastics, such as Teflon and nylon, can be used with good results and have the advantage of greater compressibility to assist in filling in irregularities in the sealing surfaces.^(3,6)

Seal materials must be proof tested for the specific environments existing at the component interface locations in the C-E design.

L. Design, materials, and fabrication of load bearing surfaces for machines operating in sodium or sodium-saturated gas atmospheres was a problem area recognized early in the development of sodium technology. As a result considerable effort has been put forth in this area in the past^(2,3) and work is continuing under AEC sponsorship.⁽⁷⁾

Reference 2 treats the subject quite extensively and it is felt that only proof testing of the bearing designs and materials selected is required for the relatively mild C-E refueling conditions.

M. As previously mentioned in paragraph 8.2D the feasibility of the large gate valves has been established at Fermi and EBR-II. However, design

development and testing is required to eliminate the deflection problem associated with the rolling gate valve. It is desirable not to use a flat plate sliding gate valve since they require such a large area to accommodate the slide in the open position.

8.3 NEW INFORMATION NEEDED

The design philosophy for the C-E reference concept is to provide a system requiring a minimum amount of basic R&D. As a result the system adopted is a novel approach to refueling but the components and mechanisms used to accomplish this are improvements and modifications of operating machines. Because of this the majority of the testing required falls in the category of prototype component proof testing rather than basic R&D. The components, SAHM, SATM, SACC, and ERC, items A through D, Sections 8.1 and 8.2, fall into the proof testing category.

E. Angular pin drive couplings and seals will require more basic development work. At this time no detailed design work has been undertaken for any of the components of the reference concept and only rough estimates can be made for the test ranges of the parameters and sizes of component subassemblies which must be studied. After the machine design analysis has been performed the unit sizes can be established and the following test information can be obtained:

- 1) Load variation with change in differential pressure across seal diaphragm;
- 2) Friction load variation with temperature change;
- 3) Change in operating friction load vs numbers of cycles (storage @ 70°F to operation @ 350°F);
- 4) Change in friction load vs continuous operating time;
- 5) Seal diaphragm gas leak rate change vs number of SA transfer cycles at constant operating rpm.

This information is required for all sizes of units which may range from 4 to 12 inches in diameter. The above information will be used for the automatic load limiting and sensing design.

F. The SATM decay heat removal system will be one complete full sized prototype heat transfer loop including SA transfer pot, closed NaK loop, and NaK intermediate coolant tank. The following information will be acquired during the test program:

- 1) Closed loop natural circulation capability in both the lowered and raised position with power levels at 25 kwt and 120 kwt which represent normal and emergency operations;
- 2) Performance tests of system with normal forced circulation (E.M. pump) at 25 kwt and 120 kwt;
- 3) Effects of alternate liquid metal wetting and drying and thermal cycling on tubing material and weld joints. Data on change in material properties, microstructure, corrosion and mass transfer rates vs number of operating cycles;
- 4) Heat transfer and structural data on finned machine barrel which forms outside heat transfer wall of NaK intermediate coolant tank;
 - a. effects of thermal cycling on structure
 - b. heat transfer characteristics of the system at 25 kwt and 120 kwt;
- 5) Necessity for NaK vapor suppression at the top of the intermediate coolant tanks. Operability of mechanisms in an argon atmosphere saturated with NaK and Na vapor at 300°F and 1 atm. through change in required load vs exposure time.

G. The automatic lid welding system will require mostly proof testing after an initially minor amount of development effort to establish the best electrode materials, current, atmosphere, weld rate, mating part weld preparation and tolerances.

H. Two areas require development concerning the finned pot spent subassembly canning system: (1) the specific lead alloy must be established and this alloy must be tested to establish its shielding and heat transfer characteristics:

- a. Alloy physical and chemical properties must be established,
- b. The effect of impurities upon these properties identified,
- c. The effect of the alloy on the pot material with respect to mass transfer and corrosion must be established;

(2) heat transfer characteristics of the sealed finned pot system must be demonstrated. Proof testing must show that the container will remain leak tight while transferring 50 kw (generated internally) to the surrounding atmosphere through natural convection and radiation which simulates a loss of normal cooling accident.

J. An extensive R&D program is required to establish the operability (and the parameters affecting operability) of mechanisms which are used in sodium, removed and stored in an inert atmosphere, and then returned to sodium operation. The following information is required:

- 1) Effect of material surface conditioning and machining finish upon sodium film retention;
- 2) Change in operating friction of mechanism with successive number of insertion-removal cycles;
- 3) Sodium film remelt time;
- 4) Effect of residual sodium film contamination on remelt time and operating friction;
- 5) Effect of oxygen concentration in storage atmosphere on contamination of the residual sodium film; also, the corrosion effects over successive 1-year storage cycles.

K. As pointed out in the previous state-of-the-art paragraph only proof testing of the specific interface seals for the following information is required.

1. Seal face wear vs number of make-break cycles.
2. Change in argon leak rate vs number of make-break cycles.

Relatively mild conditions will be encountered by the seals; they will separate a sodium vapor saturated argon atmosphere @ 5 psig and 300°F from ambient air.

L. As mentioned in section 8.2L a comprehensive program for bearing surfaces is now being undertaken through AEC sponsorship and the results of these studies will be quite adequate for the relatively mild conditions experienced by the C-E refueling components.

M. Again, the large gate valve on the bottom of the ERC is proof testing development work and requires that design changes be made to the item until it functions properly. This does not involve basic R&D.

8.4 PROGRAM DESCRIPTION

Program descriptions for those items which require the more basic R&D, rather than prototype proof testing, are presented. The proof testing depends on specific designs of components which have not been completed at this time. The R&D items, E, F, G, H, and J, Sections 8.1, 8.2 and 8.3, are required before detail design can start.

8.4.1 Item E, Sections 8.1, 8.2, and 8.3 - Angular Pin Drive Coupling and Seal

8.4.1.1 Objective

To provide a mechanism which will transmit motion across a mechanical boundary separating two atmospheres at different pressures while maintaining a positive seal.

8.4.1.2 Scope

The work should be divided into two phases:

- 1) Obtain commercially available component, probably of English manufacture, and performance test these components;
- 2) If a satisfactory component is not available, then one must be designed, fabricated and tested to obtain information outlined under section 8.3E.

8.4.1.3 Justification

The ability to transmit rotary motion through a positive seal eliminates the necessity of providing a long stacked bellows seal in order to obtain linear movement. Seal bellows are relatively expensive and have proven to be a high maintenance item.

The pin drive coupling could be used in any FBR machine in which the power supply must be located in an air atmosphere and the desired mechanical operation must be performed in a location from which air must be positively excluded.

8.4.1.4 Test Facility

Work could be accomplished at any of the facilities listed in section 7.8 of reference 2.

8.4.1.5 Manpower

If only testing of existing commercial components is involved (item 1 of scope) then the following manpower is required to obtain the information outlined in section 8.3E.

Engineering & Scientific	2 man-years
Technician	4 man-years

However, if the components must be designed and fabricated prior to testing then the following must be added to the testing manpower:

Engineering	1 man-year
Design & Drafting	1 man-year

This does not include fabrication since it is assumed that the component will be manufactured by a commercial supplier according to the developed design.

8.4.1.6 Schedule and Cost

See figure 8-1 for schedule and refer to table 8-1 for cost estimate.

Schedule - Scheme 1

Schedule - Scheme 2 (Excluding Testing)

Table 8-1. Pin Drive and Seal Cost Estimate

Scheme 1

Facility Modification	\$ 30,000
Equipment	10,000
Operating	
Materials	15,000
Manpower	
Engineering (2 man-yrs.)	60,000
Technicians (4 man-yrs.)	100,000
Total Operating	175,000
Total Scheme 1.	<u>\$215,000</u>

Scheme 2

(excluding testing)

Manpower	
Engineering (1 man-yr.)	\$30,000
Drafting (1 man-yr.)	25,000
Materials	2,000
Test Units Fabrication (3)	6,000
Total Scheme 2	<u>\$63,000</u>

8.4.2 Item F - Subassembly Transfer Machine (SATM) Heat Transfer System

8.4.2.1 Objective

To establish the heat transfer characteristics of the SATM decay heat cooling system and establish the compatibility of the system atmospheres and materials under nonradioactive operating conditions.

8.4.2.2 Scope

The following work should be accomplished:

- 1) Design and fabricate a prototype test unit which consists of one full sized heat transfer loop including subassembly transfer pot, closed NaK loop, and NaK intermediate coolant tank;
- 2) Conduct thermal performance tests simulating normal and emergency operating conditions;
- 3) Investigate the compatibility of materials of construction with operating atmospheres under normal and emergency conditions;
- 4) Conduct mechanisms operability tests in atmospheres saturated with NaK and Na vapor under normal and emergency conditions.

8.4.2.3 Justification

In the 1000 Mwe LMFBR the high heat generation rate at the time of discharge from the reactor, after 100,000 Mwd/MT average burnup, requires that the subassembly be cooled by liquid metal at least during the first few weeks of the decay period. Since this period includes the subassembly transfer operation out of the reactor, the transfer machine (SATM) should have a liquid metal decay heat cooling system. This heat removal system must have the highest degree of reliability. Such a system has been incorporated into the SATM. The machine has a triple redundant cooling system and completely eliminates the possibility of subassembly overheating due to plant power failure. This machine's complete invulnerability to loss of power is an extremely significant safety feature. The machine is designed so that a subassembly can never overheat, because of a power failure, no matter what phase of the transfer cycle the machine is in at

the time of the failure. The three heat transfer loops will be sized so that natural circulation within them could maintain the fuel pin cladding temperature at a maximum of 1400°F indefinitely.

The machine keeps the subassembly in sodium at all times and maintains a sealed inert argon atmosphere while transferring the subassembly heat to the machine surroundings. A subassembly overheating accident cannot occur in this system because of a loss of power to the machine.

This SATM may be adopted to any FBR with minor modifications.

8.4.2.4 Test Facility

The work could be conducted at any of the existing facilities listed in section 7.8 of reference 2. The test component is a complete unit and requires only services such as electricity, argon gas, sodium and NaK loading and draining facilities and a control and instrumentation panel to record test data.

8.4.2.5 Manpower

The following manpower requirements correspond to the scope items.

Man-years

Item 1 - Design but not fabrication of test unit.

Engineering & Scientific 2

Design & Drafting 2

Items 2, 3 & 4 (concurrently) - Test operations and data reduction.

Engineering & Scientific 4

Technicians 6

Item 3 - Post Test materials analysis.

Engineering & Scientific 1

Technicians 1

8.4.2.6 Schedule and Cost

See figure 8-2 for schedule and refer to table 8-2 for cost estimate.

Schedule

Item 1 (Design and Fabrication)

Items 2, 3, and 4 (Testing)

Item 3 (Post Test Analysis)

Table 8-2. Cost Estimate SATM Decay Heat Cooling R&D

Item 1 (Design & Fabrication)

Manpower	
Engineering	\$ 60,000
Drafting	33,000
Materials	2,000
Test Unit Fabrication	<u>75,000</u>
Total Item 1	\$170,000

Items 2, 3 & 4 (Testing)

Facility Modification	\$ 75,000
Equipment	30,000
Operating	
Materials	15,000
Manpower	
Engineering	90,000
Technicians	100,000
Total Operating	<u>205,000</u>
Total Items 2, 3 & 4	\$310,000

Item 3 (Post Test Analysis)

Manpower	
Engineering	\$ 7,500
Technician	18,000
Materials	<u>2,000</u>
Total Item 3	\$ 27,500

8.4.3 Item G - Automatic Lid Welding System

8.4.3.1 Objective

To establish welding parameters for an automatic remote lid welding machine which may be incorporated into the lid welding system of the subassembly cask cart.

8.4.3.2 Scope

The following work will be accomplished:

- 1) A prototype welding system will be designed and fabricated;
- 2) The system will be used to establish the welding parameters through testing in a simulated operating atmosphere. These parameters will then be factored into the design of the cask cart.

8.4.3.3 Justification

In this cask cart spent subassemblies are canned directly in a lead-bismuth heat transfer medium which is compatible with the thin film of sodium remaining on the subassembly surfaces. This eliminates the necessity of an elaborate subassembly cleaning facility and its inherent malfunctions. The lead-bismuth potting compound also helps to shield spent subassemblies during shipping and greatly reduces the probability of hydrogenous material contacting the subassemblies in the event of an accident.

This spent subassembly cask cart may be incorporated into any FBR refueling scheme with minor modifications thus simplifying the scheme.

8.4.3.4 Test Facilities

Work could be accomplished at any of the facilities listed in section 7.8 of reference 2.

8.4.3.5 Manpower

The following manpower requirements refer to the scope items.

	<u>Man-year</u>
Item 1 (design) -	
Engineering	1/2
Design & Drafting	1/2
Item 2 (testing) -	
Engineering & Scientific	1/2
Technician	3/4

8.4.3.6 Schedule and Cost

See figure 8-3 for schedule and refer to table 8-3 for cost estimate.

Schedule

Item 1 (Design and Fabrication)

Item 2 (Testing)

Table 8-3. Automatic Welding System Cost Estimate

Item 1 (design & fabrication)

Manpower	
Engineering	\$15,000
Design & Drafting	7,500
Materials	500
Test Unit Fabrication	30,000
Total Item 1	<u>\$53,000</u>

Item 2 (testing)

Facility Modifications	\$20,000
Equipment	10,000
Operating	
Materials	5,000
Manpower	
Engineering	15,000
Technicians	11,000
Total Operating	31,000
Total Item 2	<u>\$61,000</u>

8.4.4 Item H, Sections 8.1, 8.2 and 8.3 - The Cask Cart Finned Pot Subassembly Canning System

8.4.4.1 Objective

To develop sufficient backup information to design the spent subassembly cask cart. ,

8.4.4.2 Scope

Work undertaken will accomplish the following:

- 1) Establish the specific composition of the lead alloy to be used as the canning and heat transfer medium. Also:
 - a) Alloy physical and chemical properties must be established;
 - b) The effect of impurities upon these properties must be identified;
 - c) The effect of the alloy on the pot material with respect to mass transfer and corrosion must be established;
- 2) Design, fabricate, and establish heat transfer characteristics of finned pot system;
- 3) Demonstrate that the container will remain leak tight through emergency conditions by natural convection and radiation.

8.4.4.3 Justification

Subassemblies are to be canned directly in a lead-bismuth heat transfer medium which is compatible with the thin film of sodium remaining on the subassembly surfaces. This eliminates the necessity of an elaborate subassembly cleaning facility and its inherent malfunctions. The lead-bismuth potting compound also helps to shield spent subassemblies during shipping and greatly reduces the probability of hydrogenous material contacting the subassemblies in the event of an accident.

The fact that the C-E reference fuel is sodium-bonded and employs a gas vent at the top of the pin makes it advisable from a safety standpoint not to clean the element with water or steam prior to shipment for reprocessing.

The cask cart may be used in any FBR refueling scheme through minor modifications.

8.4.4.4 Facilities

Any of the facilities listed in section 7.8 of reference 2 could conduct this program.

8.4.4.5 Manpower

The following manpower requirements are listed according to the scope items.

	<u>Man-years</u>
Item 1 (establish alloy) -	
Scientific & Engineering	3
Technician	3 3/4
Item 2 & 3 (Design, fabrication & Test)	
Engineering	1 1/2
Design & Technician	2

8.4.4.6 Schedule and Cost

See figure 8-4 for schedule and refer to table 8-4 for cost estimate.

Schedule

Item 1 (Establish Alloy)

Alloy Selection

Design

Facility Modification

Testing -
Properties
Compatibility

Item 2 (Heat Transfer Characteristics)

Design

Fabrication
(1 Test Unit)

Facility Modification

Testing

Time, month

Table 8-4. Subassembly Canning System Cost Estimate

Item 1 (establish alloy)

Facility Modification		\$ 20,000
Equipment		10,000
Operating		
Manpower		
Engineering &		
Scientific	90,000	
Technicians	56,000	
Materials	15,000	
Total Operation		161,000
Total Item 1		<u>\$191,000</u>

Item 2 (heat transfer characteristics)

Test Unit		\$ 15,000
Facility Modifications		50,000
Equipment		10,000
Operating		
Materials	10,000	
Manpower		
Engineering	45,000	
Technicians	30,000	
Total Operating		85,000
Total Item 2		<u>\$160,000</u>

8.4.5 Item J Sections 8.1, 8.2 and 8.3 - Intermittent Operation and Storage of Mechanisms Between Sodium and Inert Gas

8.4.5.1 Objective

To establish the parameters effecting operability of mechanisms used in sodium, removed and stored in argon, and then returned to sodium operation without an intermediate cleaning operation.

8.4.5.2 Scope

Work would cover the following items:

- 1) Tests of material surface conditioning and machining finish to establish their effect upon sodium film retention;
- 2) Investigations of the change in operating friction of the mechanism with the successive number of insertion-removal cycles;
- 3) Tests to establish the effect of residual sodium film contamination on remelt time and operating friction;
- 4) A study of the effect of oxygen concentration in storage atmosphere on contamination of the residual sodium film; also, the corrosion effects over successive 1-year storage cycles.

8.4.5.3 Justification

Development of a storage technique which does not necessitate prior mechanism cleaning would effect a plant capital cost saving, reduce maintenance time, and reduce the safety hazard.

Cleaning and disposal techniques generally involve reacting sodium or NaK with air, water, or some chemical compound. As such, these reactions are potentially hazardous and require extensive safety precautions and procedures. Cleaning and waste disposal involve a potentially greater hazard to operating personnel than any other handling procedure. Methods must be compatible with equipment which is to be reused to avoid mechanism overheating or corrosion damage. Since most sodium or NaK cleaning and disposal operations involve hydrogen evolution and the production of obnoxious fumes, the work must be accomplished outdoors, preferably in

isolated areas, or in areas with forced ventilation. Fire fighting equipment, safety precautions, and protective clothing requirements for cleaning and waste disposal operations are necessary.⁽²⁾

Elimination of cleaning facilities greatly simplifies the operation.

The information developed in this program would be useful to designers of mechanisms and components for all liquid metal operations.

8.4.5.4 Facilities

Any of the facilities listed in section 7.8 of reference 2 are qualified to conduct these tests. All of the tests are bench scale and would not require extensive equipment.

8.4.5.5 Manpower

The following manpower requirements are in accordance with the scope items.

	<u>Man-years</u>
Item 1 - (sodium film retention)	
Engineers	3/4
Drafting & Technician	3/4
Items 2 & 3 - (remelt & friction tests)	
Engineers	1
Designers & Technicians	1
Item 4 - (oxygen contamination)	
Engineers	1 1/4
Designers & Technicians	2

8.4.5.6 Schedule and Cost

See figure 8-5 for schedule and refer to table 8-5 for cost estimate.

Schedule

Item 1 (Film Retention)

Items 2 and 3 (Remelt and Friction Test)

Item 4 (Oxygen Contamination)

Table 8-5. Mechanism Storage Tests Cost Estimate

Item 1 (film retention)

Manpower	
Engineering	\$ 22,500
Technician	11,000
Facility Modifications	2,000
Materials	5,000
Total Item 1	<u>\$ 40,500</u>

Items 2 & 3 (remelt & friction tests)

Test Items	5,000
Facility Modifications	10,000
Equipment	10,000
Operating	
Materials	3,000
Manpower	
Engineering	30,000
Technician	15,000
Total Operating	48,000
Total Items 2 & 3	<u>\$ 73,000</u>

Item 4 (oxygen contamination)

Test Items	1,000
Facility Modifications	20,000
Equipment	10,000
Operating	
Materials	5,000
Manpower	
Engineers	37,500
Technicians	30,000
Total Operating	72,500
Total Item 4	<u>\$103,500</u>

8.5 PROBABILITY OF ACCOMPLISHMENT

As previously stated, the C-E reference refueling concept is a novel approach to refueling but the components and mechanisms used to accomplish this are improvements and modifications of operating machines. As such, there is not a great deal of basic R&D required and that which is required is relatively simple and has been identified in the LMFBR program plan. The R&D which has not already been undertaken through AEC sponsorship and is specific to the C-E concept is straightforward.

An estimate of the probability of successfully accomplishing these R&D programs within the prescribed time and money is 95%+.

8.6 EFFECT OF NONATTAINMENT OF R&D

8.6.1 Item E, Sections 8.1, 8.2 and 8.3 - Angular Pin Drive Coupling & Seal

It is quite unlikely that the outlined program cannot be satisfied since units of this type are being used in the Dounreay Fast Reactor (DFR). However, should they prove unsatisfactory in our particular application, then the only consequence would be a minor redesign of the SAHM in order to substitute another seal-coupling design, possibly similar to that used in the EBR-II fuel unloading machine.

The incremental increase in plant power costs would come from the increased capital cost of the two SAHMs. The increase cannot be estimated at this time.

8.6.2 Item F, Sections 8.1, 8.2 and 8.3 - Subassembly Transfer Machine (SATM) Heat Transfer System

Nonattainment of the R&D required for the wet transfer SATM would require a change in the C-E refueling concept. Decay heat storage would have to be accomplished within the reactor vessel.

The capital cost of in-vessel storage and ex-vessel storage seem to be about the same. With in-vessel storage, the reactor vessel becomes larger in diameter, and the internals become more massive and complicated.

Also, either the reactor vessel cover becomes more complicated because of multiple transfer machine locations over the subassembly storage positions, or the storage locations must be movable within the reactor vessel to line up with one exit port. The latter design requires massive moving machinery in the hot primary sodium. This would present a maintenance problem and would increase the probability of downtime due to refueling equipment malfunction. With ex-reactor vessel storage, the tank requires a sodium heat transport loop with a pump, a heat dump of 5 Mwt maximum capacity, and a cleanup system.

With in-vessel storage, a second shutdown would be necessary or the decaying fuel would have to stay in the reactor the full 12 months between refueling cycles. This approach is undesirable, since it would increase the plutonium inventory charges by \$520,000 for the additional 8-month period.

8.6.3 Item G, Sections 8.1, 8.2 and 8.3 - Automatic Lid Welding System

If the remote automatic lid welding system is not developed, then the lid would have to be welded remote manually.

Another alternative would be to design the container with either a mechanical seal and bolting arrangement or a screw-on lid. The latter two methods are not as attractive from a safety viewpoint since they are more apt to leak.

In any event, nonattainment would not alter the C-E refueling concept.

8.6.4 Item H, Sections 8.1, 8.2 and 8.3 - Cask Cart Finned Pot Subassembly Canning

It is highly unlikely that the scope of work cannot be successfully completed as outlined. Again, the idea of subassembly canning is not new. Both DFR and Rapsodie can the spent subassemblies in an alloy before shipping. However, nonattainment of the R&D would require subassembly cleaning before shipping. This cleaning might necessitate a change from a vented to a sealed fuel pin design.

8.6.5 Item J, Sections 8.1, 8.2 and 8.3 - Intermittent Operation
and Storage of Mechanisms between Sodium and Inert Gas

If it is concluded that the mechanisms cannot be stored after operation in sodium without first being cleaned, then the plant design will be modified to include a mechanism cleaning facility. This would have no effect on the refueling operation or time since the cleaning operation would be done while the plant is operating. However, the plant capital costs would increase by about \$750,000.

Figure 8-6
Fuel Handling System R and D Schedule and Milestone Chart

Figure 8-6: Fuel Handling System Development Milestone Legend

1. Complete research and development and factor results into subassembly handling machine prototype development.
2. Complete research and development and factor results into subassembly transfer machine prototype design.
3. Complete research and development and factor results into spent subassembly cask cart prototype design.
4. Complete research and development effort.
5. Complete analysis and detailed design: Began prototype construction and proof testing program.
6. Complete prototype development and finalize the design for the demo plant.

Table 8-6. Fuel Handling System R&D

Manpower and Cost Estimate

Task	Man-Years		Costs - 10 ⁶ \$			
			Operating		Other	
	Scientists & Engineers	Other	Labor & Overhead	Materials & Services	Equipment	Facilities
8.0 <u>Components Design: Fuel Handling System</u>						
E - Angular Pin Drive coupling and seal	2-3	4-5	0.15-0.25	0.025	0.01	0.03
F - Subassembly Transfer Machine heat transfer system	7	9	0.3	0.1	0.03	0.08
G - Automatic Lid Welding System	1	1-2	0.05-0.06	0.035	0.01	0.02
H - Subassembly Canning System	4-5	5-6	0.2-0.3	0.025	0.02	0.07
J - Mechanism Storage	3	3-4	0.15-0.18	0.015	0.02	0.03
Totals	17-19	22-26	0.85-1.1	0.2	0.1	0.25
Prototype Component Development						
A - Subassembly Handling Machine	2	3 1/2	0.112	0.01	0.4	0.02
B - Subassembly transfer Machine	3	6	0.180	0.01	0.5	0.02
C - Spent Subassembly Cask Cart	3	6	0.180	0.01	0.4	0.02
D - Equipment Removal Container	2	3 1/2	0.112	0.005	0.26	0.08

REFERENCES

for Section 8.0

1. 1000 Mwe Liquid Metal Fast Breeder Reactor Design Study, Task II & III Report, CEND-337, Vol. I, June, 1968.
2. Liquid Metals Handbook Sodium & NaK Supplement (Review Copy) O. J. Foust, Editor, June 30, 1967.
3. Fast Reactor Technology, Plant Design, J. G. Yevick, Editor, MIT Press, 1966.
4. Status of UKAEA's Prototype Fast Reactor, A. G. Frame, ANS-101, 4-67.
5. Design and Operating Experience with the Control and Safety Rod Drive Mechanisms for the Enrico Fermi Fast Breeder Reactor, J. W. Hess, APDA-301, June, 1966.
6. EBR-II Fuel-Unloading Machine: Design and Performance Characteristics, E. Hutter et al, ANL-7201, June, 1966.
7. Static and Sliding Contact Behavior of Materials in Sodium, W. J. Freede, NAA SR-12446.

9.0 PHYSICS

9.1 INTRODUCTION

The objective of any basic physics research and development work should be consistent with that stated in Section 9 of the AEC LMFBR Program Plan⁽¹⁾: namely, to make possible the physics design of a large 1000 Mwe commercial LMFBR plant without recourse to mockup criticals or integral experiments. This will require an accurate prediction of: (1) the spatial power distribution; (2) the effective values of operational reactivity coefficients and their spatial distributions; (3) the reactivity change due to burnup; (4) changes in the power distribution and operational reactivity coefficients during each refueling interval; and (5) control rod worths which are adequate to meet appropriate safety criteria. Prediction of the dynamic behavior of the system is also necessary for proper design of the control and instrumentation system. Achievement of such a prediction capability requires research and development in four basic categories: (1) acquisition of nuclear data; (2) performance of critical experiments; (3) improvement and development of calculational methods; and (4) additional analysis. Generally speaking, each of these areas is covered to some extent in section 9 of the Program Plan. The most significant aspects of each of these areas are discussed in the sections which follow together with a list of the research and development required in each. The R&D work listed in these sections should be emphasized while being carried out within the general context of the program plan document.

9.2 ACQUISITION OF NUCLEAR DATA

9.2.1 Definition of Technical Area

The C-E reference design as described in the Conceptual Systems Design Description⁽²⁾ is a beryllium-oxide moderated, uranium-plutonium carbide fueled cylindrical core with stainless steel cladding and tantalum and enriched boron-carbide control subassemblies. Accurate prediction of the performance of this design therefore requires adequate cross section data for the following nuclides: (1) Be; (2) O; (3) C; (4) U-238; (5) Pu-239; (6) Pu-240; (7) Pu-241; (8) Pu-242; (9) individual fission products; (10) U-235; (11) U-236; (12) Fe; (13) Ni; (14) Cr;

(15) Mo; (16) Ta; and (17) B-10. Although data are presently available for most of these nuclides, values are not yet as accurate as desired for design purposes.

9.2.2 Base of Current Technology

A recent sensitivity analysis by Greebler et al⁽³⁾ shows an uncertainty of approximately 0.25 mill/kwhe in fuel cost for a reactor which is required to satisfy a voided Doppler criterion of $T dk/dT = -0.004$ and, hence, contains a large amount of moderating material (BeO) among its core constituents. The analysis also shows that the current large uncertainty in the value of alpha (i.e., ratio of capture to fission) for Pu-239 between 0.2 kev and 15 kev is the largest single contributor to this overall fuel cost uncertainty. This is followed closely by the uncertainties in $\bar{\nu}$ (i.e., the average number of neutrons emitted per fission) for Pu-239 and uncertainties in the capture cross section for U-238. Values of the average number of neutrons emitted per fission for Pu-239 are about 4 percent uncertain in the range from 0 to 300 kev. There is about 50 percent uncertainty in the capture values for U-238 between 10 and 100 kev. Although this sensitivity analysis was carried out for an oxide-fueled reactor, the same conclusions would be applicable to the C-E moderated carbide breeder design.

The large uncertainty in the value of alpha for Pu-239 is underscored by recent measurements at Harwell⁽⁴⁾ which indicate values in the range from 0.5 to 50 kev which are significantly larger (50 percent to 100 percent) than previous results. To determine any effects of these possible higher alpha values on the performance of the reference design, some calculations were performed in which values were modified in appropriate groups of the multigroup set which was used for analysis. This was done by increasing capture cross sections while holding fission values constant. The modified and original alpha values are listed in table 9-1 together with results of the calculations. The difference in parameter values again emphasizes the importance of ascertaining the proper value for alpha. If the higher alpha values prove to be correct, the reduction in seed conversion ratio alone will necessitate design changes due to differences in the burnup reactivity change and associated control requirements. Reactor performance would also be significantly poorer than currently predicted with respect to breeding performance and doubling time. In addition, the operational reactivity coefficients would be adversely affected.

Table 9-1. Effects of Possible Higher Alpha Values
for Pu-239* on Performance of C-E Reference Design

<u>Parameter</u>	<u>Current α Value</u>	<u>Possible Higher α Value</u>	<u>Percentage Change</u>
Seed Conversion Ratio	1.005	0.894	-11.0
Total Breeding Ratio	1.441	1.292	-10.3
Average Seed Enrichment (% Pu-239 + Pu-241)	10.79	11.14	+3.2
Sodium Void Coefficient (% $\Delta k/k$ -complete voiding of seed and axial blanket)	+1.36	+1.63	+19.8
Voided Doppler Coefficient (-T dk/dT)	0.0042	0.0030	-28.6

* Alpha Values Employed in these Calculations are:

<u>Group</u>	<u>Energy Range (kev)</u>	<u>Original Value</u>	<u>Revised Value</u>
12	25-15	0.41	0.75
13	15-9.1	0.50	1.00
14	9.1-4.3	0.53	1.20
15	4.3-2.6	0.56	1.10
16	2.6-2.0	0.59	1.25
17	2.0-1.2	0.62	1.10
18	1.2-0.96	0.67	0.80
19	0.96-0.58	0.67	1.20

Because of large uncertainties such as those discussed above which are associated with currently available cross section data and the importance associated with removing these uncertainties, measurements of important data should be scheduled on a priority basis. This has been done to some extent in Section 9 of the AEC LMFBR Program Plan and a list of the highest priority measurements applicable to the C-E reference design has been excerpted and is given in table 9-2. The first three entries in this tabulation are: (1) alpha for plutonium 239; (2) the average number of neutrons emitted per fission for plutonium 239; and (3) U238 capture. The importance of these quantities has been pointed out in the preceding discussion. In view of their significance, every effort should be made to obtain these values as rapidly as possible. In particular, the ultimate accuracy measurements for these quantities should be scheduled sooner than indicated in table 9-2, if at all feasible. Since these measurements will influence design activities to a high degree and may also affect the outcome of neutronic analyses contemplated in other sections of the Program Plan, they probably should be placed in a special higher priority category. Such a category should be established to emphasize the importance of these measurements and to insure a rapid acquisition of their values. When these measurements are completed and evaluated, results should be disseminated to LMFBR contractors as quickly as possible so that the higher accuracy may be incorporated into design activities. Establishing a higher priority for these three quantities in no way minimizes the importance of the other measurements in table 9-2 as currently scheduled.

At present, fission product cross sections are also highly uncertain for fast reactor systems. These cross sections are of particular importance in determining the reactivity change due to burnup which, in turn, affects the length of the refueling interval and requirements for the control system. Uncertainties in these cross sections also have a significant effect on calculated values of the fuel cycle cost and doubling time. The sensitivity analysis previously mentioned by Greebler showed these uncertainties to be about 0.06 mill/kwhe for the fuel cycle cost and 1.4 years for the doubling time. In view of these considerations, measurements of these cross sections should be included in research and development activities and given a relatively high priority. Although a measurement of fission product capture (1 kev-2 Mev) is listed in

Table 9-2. Schedule of Highest Priority Cross Section Measurements from LMFBR Program Plan Applicable to C-E Design Effort

Nuclide	Cross Section	Incident Neutron Energy Range	Ultimate Accuracy Needed	Schedule Start/Finish (FY)	Lesser Accuracy Measurement	Schedule Start/Finish (FY)
Pu-239	Alpha	0.1 kev - 1 kev	5%	75/78	20%	69/70
		1 kev - 50 kev	3%	75/78	20%	69/70
		50 kev - 600 kev	5%	75/78	15%	71/75
	$\bar{\nu}$ (E)	1 kev - 3 Mev 3 Mev - 10 Mev	1/2% 1%	74/78 75/78	1-1/2% 1-1/2%	69/72 71/75
U-238	Capture	1 kev - 300 kev	2%	73/77	10%	69/71
		3 Mev - 10 Mev	10%	73/77	20%	69/73
Cr	Capture	1 kev - 150 kev	10%	69/72		
Fe	Elastic Scattering	1 kev - 10 Mev	10%	74/76		
	Inelastic Scattering	1 Mev - 2 Mev	5%	72/75		
	Capture	1 kev - 175 kev 0.175 Mev - 1.5 Mev	10% 20%	74/77	20%	69/72
U-238	Inelastic Scattering	0.1 Mev - 10 Mev	5%	75/80	20%	69/74
	Fission	1 Mev - 10 Mev	5%	71/75		
	Resonance Parameters		10%	75/78	20%	71/75
U-238	Capture	1 kev - 300 kev	2%	73/77	10%	69/71
		3 Mev - 10 Mev	10%	73/77	20%	69/73
	$\bar{\nu}$ (E)	1 Mev - 10 Mev	3%	70/75		
Pu-239	Elastic Scattering	1 Mev - 3 Mev	10%	70/75		
	Inelastic Scattering	100 kev - 10 Mev	20%	70/75		
	Fission	1 kev - 20 kev	3%	73/78	10%	69/72
		20 kev - 300 kev	2%	73/78	5%	69/72
		0.3 Mev - 3 Mev	2%	73/78	3%	69/72
		3 Mev - 10 Mev	5%	73/78	7%	69/72
Pu-240	Capture	0.5 kev - 150 kev	5%	75/78	15%	69/72

Section 9 of the Program Plan, no priority or schedule dates for this are included in the document. This situation should be rectified as soon as possible since these measurements will present difficulty due to the problems involved in isolating individual fission products and in evaluating their yields.

Other cross section measurements scheduled in the Program Plan, although of lower priority than those already mentioned, are also important for the proper prediction of performance factors for the C-E reference design. These are indicated for completeness in table 9-3.

The measurement of standard cross sections to higher accuracy must also be emphasized where these values are necessary to determine other required cross sections from cross section ratio measurements. Typical standard cross sections include (n,α) values for boron-10, elastic scattering values for carbon-12, and the fission cross section for U-235. Other standards which may be applicable include the elastic scattering cross section for hydrogen and the (n,α) reaction for lithium.

The differential nuclear data measurements discussed in the Program Plan also deserve particular emphasis. These include: (1) a determination of the spectrum and yield of delayed neutrons; (2) a determination of the energy yield per fission; and (3) a determination of the yields of individual fission products. The delayed neutron data which is useful in determining the effective delayed neutron fraction is particularly important for safety analyses. Precise evaluation of the energy yield per fission would probably result in a higher performance design due to the elimination of uncertainty factors. Consequently, this quantity should be accurately determined not only for plutonium-239 but also for the other fissionable materials (i.e., U-238, Pu-241, and U-235) as well. A determination of the yields of individual fission products will enable a more precise evaluation of burnup reactivity changes when the fission product cross sections are also available and can be used with more advanced burnup programs.

A particularly important portion of the nuclear data effort is the cooperative evaluation and distribution of evaluated data. This aspect of physics research and development is well underway and considerable emphasis has already been placed upon activities of the Cross Section Evaluation Working Group (CSEWG). The activities of this group should increase in importance until the objectives of the Program

Table 9-3. Other Cross Section Measurements from
LMFBR Program Plan Applicable to C-E Design Effort

<u>Nuclide</u>		<u>Cross Section</u>	<u>Energy Range</u>
beryllium		inelastic scattering	2-10 Mev
oxygen		elastic scattering	4-10 Mev
sodium		inelastic scattering	2-10 Mev
sodium		resonance parameters	3 kev
sodium		capture	1-100 kev
iron		inelastic scattering	2-10 Mev
nickel		elastic scattering	10 kev-3 Mev
nickel		inelastic scattering	5-10 Mev
nickel		capture	1-150 kev
tantalum		capture	1-150 kev
U-235		elastic scattering	1-5 Mev
U-235		inelastic scattering	100 kev-3 Mev
U-235		alpha	1 kev-7 Mev
U-235		$\bar{\nu}$	Up to 3 Mev
U-238		(n,2n)	6-10 Mev
Pu-239		(n,2n)	6-10 Mev
Pu-239		alpha	600 kev-10 Mev
Pu-240		inelastic scattering	threshold-10 Mev
Pu-240		fission	0.5-10 Mev
Pu-240		resonance parameters	0.1-5 kev
Pu-240		$\bar{\nu}$	threshold-10 Mev
Pu-241		fission	1 kev-6 Mev
Pu-241		alpha	1 kev-2 Mev
Pu-241		$\bar{\nu}$	Up to 10 Mev
Pu-242		capture	1 kev-7 Mev
Pu-242		$\bar{\nu}$	0.5 - 10 Mev

Plan have been fulfilled. Continued emphasis should therefore be given to their activities in this regard.

A list of the recommendations concerning nuclear data acquisition which have been discussed in the preceding paragraphs is presented for summary purposes in table 9-4.

Table 9-4. Recommendations Concerning Research and Development in the Area of Nuclear Data Acquisition

1. Establish higher priority categories for Pu-239 α , Pu-239 $\bar{\nu}$, and U-238 capture measurements.
 2. Establish priority and schedule dates for measurements of fission product capture cross sections.
 3. Effectively implement schedule of cross section measurements outlined in Section 9 of LMFBR Program Plan.
 4. Determine spectrum and yield of delayed neutrons.
 5. Determine energy yield per fission for Pu-239, U-238, Pu-241 and U-235.
 6. Determine individual fission product yields.
 7. Place adequate emphasis on evaluation and distribution of evaluated data.
-

9.3 PERFORMANCE OF CRITICAL EXPERIMENTS

9.3.1 Definition of Technical Area

In view of the large uncertainties currently associated with the prediction of neutronic behavior in fast systems, it will be necessary to conduct numerous critical experiments to gain an understanding of the operational reactivity changes which contribute to the overall power coefficient. These include Doppler reactivity changes, coolant void reactivity effects, fuel and structural expansion effects, and possible fuel redistribution during irradiation.² Many of these experiments will have to be performed in plutonium-carbide fueled assemblies to get a proper representation of the spectral effects. They will culminate in a critical facility mockup for the first demonstration plant which will be among several required before construction of the first commercial power plant.

9.3.2 Description of Current Technology

The great majority of experience in the performance of fast critical experiments has been obtained at Argonne National Laboratory. Although this has more recently included a number of smaller plutonium-fueled assemblies, most of the experience to date has been with U-235 fuels. Consequently, it will be necessary to perform many future experiments using plutonium fuel in larger assemblies.

Since the prediction of past critical experiments with available data and methods has not always been satisfactory, a question exists as to whether or not the data or the methods or both are responsible for the discrepancies. In view of this, more experiments are necessary in order to resolve this question. The answers so obtained will have application in the design of actual systems as well.

9.3.3 New Information Needed

A list of the most important basic experiments required is given in table 9-5. Where possible, these experiments should be performed in plutonium-fueled carbide assemblies having volume fractions similar to those of the reference design.

In addition, all experiments should be repeated after including approximately 4.6 percent beryllium oxide in the assembly composition to soften the neutron spectrum. This procedure has been adopted in the reference design to achieve a voided Doppler coefficient (i.e., $T dk/dT$) of -0.004 in order to limit the energy release in the event of the maximum credible accident. If, however, more advanced safety analyses or protective system studies indicate that this additional moderator material is not really beneficial and hence is not necessary, it could be removed with a consequent increase in core performance. Duplicate critical experiments with and without moderator would therefore provide information of interest for either eventuality. Such a procedure would also contribute to an understanding of spectral effects in fast reactor systems.

The most important critical experiments should be performed with significant percentages of the higher plutonium isotopes in the fuel so as to simulate the equilibrium composition of the proposed system (i.e., ~27% Pu-240). The experiments would then include the nuclear effects of these nuclides.

Table 9-5. List of Basic Physics
Experiments in Support of LMFBR Program

1. Spatial dependence of the sodium void coefficient.
 2. Temperature dependence of the Doppler coefficient.
 3. Spatial dependence of the Doppler coefficient.
 4. Neutron spectrum measurements.
 5. Effects of practical control rods.
 6. Fuel migration and expansion versus exposure.
 7. Subassembly bowing effects.
 8. Demonstration plant mockup critical.
-

The sodium worth experiments should emphasize the spatial dependence of the sodium reactivity effect. Doppler experiments should emphasize the spatial and temperature dependence of this effect together with a determination of the effective value of the Doppler coefficient in a configuration with a realistic temperature distribution.

The neutron spectrum measurements should be especially valuable in testing data and methods used for nuclear design. Consequently, the results of these as well as all other experiments should be disseminated to all LMFBR contractors so that they may be used for calculational comparisons.

Critical experiments should also be performed in assemblies containing tantalum, natural boron carbide and enriched boron carbide control zones. Measurements should be made to determine the power perturbations in fuel regions adjacent to the control zones. Relative worths per kilogram in various positions of the assembly should be obtained. Finally, energy and self-shielding effects should be evaluated. Measurements with and without additional moderator will be helpful in isolating the effects of moderator on control rod worths.

Experiments to determine any fuel redistribution and expansion as a function of exposure will probably require considerable planning. Of necessity, these cannot be performed in critical facilities. However, they will be very important in evaluating reactivity effects due to fuel movement during the reactor duty cycle and in determining any possible required action. Such experiments should

also include an analysis of structural expansion effects including those resulting from the bowing of individual fuel rods or fuel subassemblies.

The final experiment shown in table 9-5 is the mockup critical for a demonstration plant of approximately 500 Mwe. This demonstration plant, fueled with sodium-bonded carbide, will be required in the middle 1970's in order to achieve successful commercial operation of the selected reference design around 1980. The critical facility mockup of the demonstration plant is necessary in ~1973 in view of the large uncertainties currently associated with the prediction of nuclear effects and should be sufficiently realistic to permit adequate simulation of the neutron spectrum in the demonstration plant. In particular, it must possess the same geometry and should include realistic axial and radial blankets. The fuel should have the same isotopic composition as that to be used in loading the demonstration plant. This experiment can then be employed to predict all important nuclear effects associated with the performance of the demonstration plant with reasonable uncertainties involved.

9.4 IMPROVEMENT AND DEVELOPMENT OF CALCULATIONAL METHODS

9.4.1 Definition of Technical Area

Proper design of the demonstration plants as well as the first-round commercial plants will require improved two- and three-dimensional computer programs for the calculation of the neutron flux distribution and effective static operational reactivity coefficients. Higher dimensional burnup programs will also be required to permit an accurate determination of reactivity effects during the equilibrium duty cycles of these reactors and also for properly evaluating their startup characteristics during the approach to equilibrium. Such programs should also be efficient enough with respect to execution time that they can be used routinely for design purposes. The availability of such higher dimensional programs will also make possible the isolation of discrepancies between calculation and experiment for various critical assembly measurements. The reasons for such discrepancies cannot yet always be determined since they may be attributable to deficiencies in both data and methods.

9.4.2 Description of Current Technology

Currently available nuclear computer programs can be grouped into two general categories: (1) design packages; and (2) standards programs. The design packages are characterized by reasonably short running times, usually at the expense of accuracy. The standards programs, on the other hand, are more accurate but require excessive execution time. Although present methods can be used in the design of future LMFBR systems, the use of programs in the standards category would be required to achieve the required accuracy. A large effort should therefore be devoted to improving the running time of some of the currently available standard codes so that they may be transferred to the design category with a resultant improvement in the accuracy of current design methods.

9.4.3 New Information Needed

Since the number of areas for improvement and sophistication of current fast reactor design methods is very large, effort in this area can be expected to encompass a long period of time. However, certain items are more important for the immediate goal of designing a commercial power plant for operation by 1980 than are others. The most important of these items are listed in table 9-6. A discussion of each item follows.

Table 9-6. Research and Development Required
in Computational Methods

1. Development of Efficient 2D Diffusion Design Code
 2. Development of 3D Diffusion Code
 3. Development of Efficient 1D Transport Design Code
 4. Improvement of 2D Transport Code
 5. Development of 1D Buckling Synthesis Burnup Code
 6. Improvement of 2D Burnup Code
 7. Improvement and Development of Methods for Cross Section Generation
-

Improvement of one of the two-dimensional diffusion codes such as CANDID⁽⁹⁾ should be given a relatively high priority in this regard. This code could then be used for routine analysis of the reference design which, in actuality, possesses

a hexagonal and not a cylindrical configuration. This type of program is also required for properly analyzing the characteristics of the corners of the radial blanket (i.e., protrusions of the radial blanket above the seed region). The program should also be used for calculating abnormal void reactivity patterns as well as sodium void coefficients. Such a program would also be useful for the analysis of other, odd-shaped configurations in safety analysis work. A large improvement in running time would permit the use of this type of program in reactor optimization surveys and might change some of the conclusions reached with one-dimensional programs.

Code development efforts should also include production of a three-dimensional diffusion program.⁹ This should be available at the same time the demonstration plants are operating and could be used for calculation of measured experimental parameters. The more accurate geometrical treatment would then make possible better isolation of data uncertainties which could be rectified before design of the commercial plants.

Although both³ one- and two-dimensional transport programs are currently available, running times, especially for the two-dimensional program, are high enough to obviate their use in most situations of interest. Both types of programs should, therefore, be improved with respect to running time. A particular effort should be made to reduce the running time of DTF-IV⁽¹⁰⁾, for example, so that it may be used routinely for one-dimensional design calculations. This is important since transport programs are frequently used for cell calculations in the evaluation of heterogeneity effects and as reference calculations to test the validity of diffusion theory for specific applications. Transport theory must also be used routinely to obtain flux distributions near boundaries or where there are strong flux gradients. These uses justify conversion of a one-dimensional transport program from the standard to the design category.

At least one of the unidimensional burnup programs such as SIZZLE⁽¹¹⁾ should be improved in the following manner: (1) the capability should be present for searching on the required excess reactivity in addition to the time-dependent critical poison concentration in a single problem run and both types of searches should have the capability of atom conservation for a variable number of atoms; (2) radioactive decay of the plutonium isotopes should be treated; (3) the treatment of fission products should be such that fission product captures lead

to different fission products; and (4) program dimensions should be extended to permit greater flexibility in the use of different multigroup cross section sets for the core and blanket regions. The programs should also include an option to permit iteration on the initial excess reactivity for a specified burnup until the final eigenvalue converges to a specified input value. After revision, the program should then be converted into a synthesis routine with iteration between radial and axial calculations with the bucklings obtained in one direction being transferred to the calculations in the other direction. This type of program would then include the effects of nonuniform axial depletion in the results.

A two-dimensional burnup program with minimum running time should also be developed. This may be based upon the ASSAULT⁽¹²⁾ program, for example, and should incorporate those features just mentioned for the one-dimensional program. This two-dimensional burnup program will be of particular importance in analyzing the performance of the demonstration plants to refine the data and methods for design of the first-round commercial breeder reactors. The same considerations mentioned above in connection with the two-dimensional diffusion program also apply to the two-dimensional burnup program and strengthen the emphasis for its development.

An improvement of methods for multigroup cross section generation is of paramount importance to design activities since such generation currently occupies a major portion of the time in a design optimization sequence involving many different configurations and fuels. Practical expediency therefore often dictates that relatively few cross section sets be used in such a procedure which compromises its validity to an uncertain extent. Improved cross section generation codes should therefore be developed with an emphasis on both accuracy and speed. Since the MC² program⁽⁶⁾ is probably very likely the best code presently available for this purpose, it should be recoded for optimum execution time. If necessary, optimized machine language versions should be developed for the principal computer systems in use (e.g., IBM 360 or CDC 6600) so that execution time could be further reduced. In addition, other programs should be developed employing bilinear averaging (i.e. $\phi\phi^*$) as well as variational methods. Results obtained with these new programs should then be compared with existing methods and criteria developed for the use of each technique.

An investigation should be conducted to determine the number of multigroup cross section sets required for analyzing a typical reactor system such as the reference design to the required accuracy. For example, is a separate multigroup set required for the axial blankets even though the fuel, structural and coolant volume fractions are the same as for the seed region? What would be the effect of generating a separate set for each region of a zoned core where only the fissile enrichment differs between regions? For an accurate treatment of reaction rates in the radial and axial blankets, should different multigroup sets be used to account for the spatial distribution of the plutonium concentration? In short, what is the gain in accuracy which is obtained by increasing the complexity of the analysis? In this context, cross section changes as a consequence of fuel depletion and higher isotope buildup should be investigated. The necessity of using different cross section sets to account for spatial changes during various degrees of coolant voiding should also be evaluated to avoid the redundant generation of cross sections by contractors.

Other effects which should be analyzed in connection with cross section generation include a determination of the adequacy of using buckling value(s) to represent system leakage during the averaging process. Improved methods should also be developed for the treatment of flux self-shielding effects, heterogeneity effects and resonance absorption effects. Treatment of overlap from contributing resonances of fertile and fissile isotopes should be emphasized.

Averaging methods to generate cross section sets for use in transport calculations should also be considered. The investigation should include an analysis of the effects of utilizing diffusion constants in transport calculations. This aspect of cross section generation is particularly important in view of the necessity of occasionally using transport computations for reference standards and cell calculations for the treatment of heterogeneity effects. In this regard, a study should also be carried out to ascertain the validity and limitations of multigroup diffusion theory for design purposes and to establish the limitations of diffusion theory as a standard. The effects of diffusion theory in the analysis of short cylindrical high leakage cores such as the reference design should be emphasized.

Successful operation of a commercial power plant by 1980 requires additional in-depth physics analysis in three main areas in order to sufficiently develop and refine the present reference design. These areas are: (1) fuel management; (2) control; and (3) safety. All these areas should be emphasized as each has important interfaces with research and development work in the categories of safety and plant design.

Fuel management studies should include a more detailed evaluation of management of the radial and axial blankets. Various fuel shuffling procedures for the blankets should be compared with proper load factor corrections for variations in downtime and considering any comparative changes in the control requirements between schemes. Axial blanket studies should include a comparison of elements which are an integral part of those in the seed and others which are scheduled and manipulated independently of the seed elements. Radial as well as axial zoning should be investigated in more detail with respect to optimization of the fuel cycle cost. In particular, more than two radial seed zones should be considered. A study should be carried out for the reference design in which a subassembly-by-subassembly refueling pattern is evaluated. An analysis in this depth might serve to reveal additional problems for investigation.

Techniques should be developed for analysis of the physics of fast reactor control rods. Requirements for control rod materials should be evaluated in detail. Studies should then be performed aimed at providing guidelines for the composition and location of control mechanisms. This would include consideration of locating control subassemblies in radial blanket regions. The effects of spatial and energy self-shielding for all proposed control materials in fast neutron spectra should be determined. Criteria for rod replacement should be developed. Development of techniques for analysis should include: (1) determination of the peak rod worth with other rods totally or partially inserted; (2) perturbation of the power distribution in adjacent fuel assemblies; and (3) calculation of total shutdown reactivity. The ganging and scheduling of rods for optimum operation should also be considered for a typical LMFBR system.

A large emphasis should be placed on work relating to the analysis and understanding of the physics of safety. Paramount in this work should be the

development of techniques for determining the reactivity of melting and reassembly configurations. In particular, how do cross sections change as a result of changes in energy and spatial self-shielding during various phases of meltdown? What effects do these changes have on the calculated reactivities of the various configurations? More realistic treatment of feedback effects should be incorporated into present safety models. Emphasis should also be placed upon obtaining temperature-dependent material property measurements for use in improved energy release analyses.

REFERENCES

for Section 9.0

1. LMFBR Liquid Metal Fast Breeder Reactor Program Plan, Vol. 9, Physics, WASH-1109 (August, 1968).
2. 1000 Mwe Liquid Metal Fast Breeder Reactor Follow-on Study: Report of Tasks II and III, A Reference Conceptual Design of a 1000 Mwe LMFBR, Vol. I: Conceptual System Design Descriptions, CEND-337 (July, 1968)
3. P. Greebler, B. A. Hutchins, and R. B. Linford, "Sensitivity of Fast Reactor Economics to Uncertainties in Nuclear Data," Nuclear Applications, Vol. 4, (May, 1968)
4. M. G. Shomberg, M. G. Sowerby, and F. W. Evans, "A New Method of Measuring Alpha (E) for Pu-239, "IAEA Symposium Fast Reactor Physics and Related Safety Problems, SM-101/41, Karlsruhe, Germany (October 30-November 3, 1967)
5. D. M. O'Shea, H. H. Hummel, W. B. Lowenstein, and D. Okrent, "Twenty-Six-Group Cross Sections," Trans. Am. Nuclear Soc. 7, 242 (1964)
6. Toppel, B. J., Rago, A. L., and O'Shea, D. M., "MC², A Code to Calculate Multigroup Cross Sections," ANL-7318 (June 1967).
7. "Liquid Metal Fast Breeder Reactor Design Study," CEND-200 (Jan. 1964)
8. 1000 Mwe Liquid Metal Fast Breeder Reactor Follow-on Study: Report of Tasks II and III, A Reference Conceptual Design of a 1000 Mwe LMFBR, Vol. II: Static Design and Performance Analyses, CEND-337 (July, 1968)
9. "ANL-CANDID, A Two-dimensional, Diffusion-theory Code Based on CANDID2D," ANL-7305 (September 1967)
10. K. D. Lathrop, "DTF-IV, A FORTRAN-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering," Los Alamos Scientific Laboratory Report, LA-3373 (1965).
11. D. P. Satkus and H. P. Flatt, "SIZZLE, A Fast or Intermediate Reactor Burnup Code," NAA Program Description, AI-64-MEMO-183 (1964)
12. D. R. Vondy, T. B. Fowler and M. L. Tobias, "Reactor Depletion Code Assault (Two-Dimensional, Multi-Neutron-Group), ORNL-TM-1302, (March, 1966).