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

Approved by B. J. Goulding
Project Engineer

M. W. Croft
Project Manager

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

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PREFACE

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

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

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

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

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7. FUEL AND MATERIALS

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7. FUEL AND MATERIALS

7.1. Introduction

The performance of fuel and other materials is of prime importance to the 1000-MWe LMFBR. The fuel element, composed of fuel and cladding, must be able to achieve a peak burnup of about 140,000 MWd/T. Mixed oxide fuel has demonstrated its ability to endure such a burnup in a thermal reactor, and it may do as well in a fast reactor; however, existing experience has not confirmed that austenitic stainless steel cladding can contain the fuel for such long reactor service.

The primary R&D effort must be directed toward determining the reactor service that can be achieved with stainless steel cladding. In the event that existing commercial alloys cannot maintain their integrity sufficiently, modifications of the stainless steels or another alloy system (e.g., vanadium) must be evaluated.

Fuel pin failure can be expected as the result of one of the following factors:

1. Rupture from excessive fission gas pressure.
2. Rupture from stresses generated by fuel growth.
3. Rupture from cycling of cladding.
4. Growth of cladding.
5. Interaction of fuel and cladding.

In order to predict fuel element lifetime, it is necessary to estimate volumetric changes of the fuel and cladding material as a result of the various forces that occur during reactor operation. This can be done empirically or analytically. The current approach is to attempt an analytical solution, for which mechanical and physical properties and chemical changes must be determined during and after irradiation at temperatures where measurements are very difficult. An intensive effort will be required to support this approach.

A vented pin has been chosen for the B&W design to reduce both the likelihood of pin failure and the amount of steel in the core. Failure by fission gas pressure buildup is eliminated except in cases of vent failure or gas entrapment. Particulate fuel releases its gases more readily than does pellet fuel; thus the fuel entrapment and growth problems are eased.

Information gained during LMFBR development of unvented pellet fuel will be utilized as much as possible. The costs associated with pellet fuel development for the first demonstration plant are assigned herein to the predecessor plant, since the latter also must be qualified for nonvented fuel elements containing pellets.

The EBR-II is required for some early capsule irradiations, and FFTF loops are counted on for the important evaluation of vented elements. This work must produce sufficient assurance to permit the inclusion of test bundles of vented fuel in the predecessor plant. In turn, the test irradiations in the predecessor plant are expected to provide the basis for acceptance of vented, vipac, mixed oxide fuel elements to be used in the 1000-MWe LMFBR.

The primary and secondary systems will need several materials capable of 30-year service. Material properties are not available today to furnish allowable stresses for pressure-bearing parts to the stress analysts; in the absence of data, it is necessary to establish conservative stress levels. To improve economics, a work task is included to determine more realistic allowable stresses.

A large reactor is fabricated from a number of materials in addition to the primary ones already mentioned. Provisions are made herein for R&D for potential needs, even though the LMFBR design has not progressed far enough for many material applications to be identified.

Table 7-1 shows the overall schedule and estimated total costs for fuel and materials R&D.

7.2. Fuel Material

7.2.1. General

A fuel is required for the predecessor plant as well as for the 1000-MWe LMFBR plant. Vipac will not be developed in time

Table 7-1. Overall Schedule and Estimated Total Cost for Fuel and Materials R&D

Task	Calendar year															Total cost, \$ × 10 ³
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
<u>Fuel and Materials</u>						△ 1 2 3						△ 4 5			△ 6	
Fuel material																21,039.5
Cladding																18,800.0
Fuel elements																38,045.0
System materials																14,910.0
Total																92,794.5

Milestones

- △¹ Vipac fuel ready for FFTF irradiations.
- △² Vented element ready for FFTF irradiations.
- △³ Control and shielding materials ready for FFTF irradiations.
- △⁴ Vented vipac element ready for predecessor irradiations.
- △⁵ System materials ready for predecessor irradiations.
- △⁶ All materials ready for 1000-MWe plant use.

to be used as driver fuel in the predecessor plant; therefore, a separate fuel must be developed for each of the two plants.

Since the predecessor plant will follow the first demonstration plant, the R&D devoted to the first demonstration plant is expected to provide sufficient basis for the use of pellet fuel in the predecessor plant.

Mixed oxide pellet fuel has been chosen for the predecessor plant because of extensive prior testing and use in the production of pelleted uranium dioxide. This background is well-known and documented. Consequently, it has been considered wise to extrapolate this experience to mixed oxide fuel for fast breeder use.

The requirements on a fuel for fast reactor service are more severe than those for thermal reactor service. Much longer burnups at high heat ratings are necessary for competitive economics. One consequence is that significant solid fission-product expansion must be tolerated.

The development of vipac particulate fuel for the 1000-MWe LMFBR must be provided for largely by this program; therefore, a detailed description is presented here.

The as-fabricated structures of particulate and pellet fuels have basic differences. Pellet fuels are fabricated by compacting fine powder and then sintering the compact to develop the desired density; the result is a continuous structure of fine grains having intergranular and grain-boundary porosity. The sintering process tends to produce random porosity distribution and a high degree of spherical voids at high density. The microstructures of the individual grains of particulate fuel are similar to those of sintered pellets; however, on a macroscale, the structure is discontinuous, with particle-to-particle interfaces, and most of the porosity is irregular and uniformly distributed between grains throughout the compact. Further, the fabricated structures of the fuel rods differ in that there is a gap between pellets and cladding, but particulate fuel is in contact with the cladding after fabrication.

Much of the apprehension concerning particulate fuels is based on comparison of their as-fabricated structure with that of pellet fuels, particularly regarding the mobility of the particulate structure

and the decrease in effective thermal conductivity to be expected from the lack of bonding between grains. Under the heating conditions experienced in the reactor, steep temperature gradients are produced in low-thermal-conductivity materials, such as the oxides, and massive restructuring occurs. Restructuring of particulate fuels includes sintering of the particles where sufficient temperature is developed, and massive grain growth can occur after sintering. As a result, the as-fabricated structures tend to equilibrate in the reactor to a structure characteristic of low-thermal-conductivity materials maintained in a steep temperature gradient. A collection of irradiated structures of various materials shows a consistency of structural features for materials that experience similar temperature profiles and burnups. The effect of burnup on structure also involves a complex interrelation of temperature-time-temperature-gradient effects on grain growth and void movement.

The characteristic structure of low-burnup fuels exhibits an inner region of columnar grains and an outer region of essentially as-fabricated fuel. Within the columnar grain region there is (typically) an inner porous zone, a central band of high density, and an outer band of small, porous, columnar grains. Equiaxed grain growth often occurs in the outer region adjacent to the columnar grain zone. High-burnup fuels typically show an inner region of long, thin, columnar grains and an outer region of as-fabricated structure which varies in proportion to heat rating and burnup.

The comparison of irradiated structures shows that major structural differences between particulate and pellet fuels exist on a short-term basis. The rate of restructuring has not been established for uranium-plutonium oxide fuels under LMFBR heat rating conditions, but out-reactor studies have shown that major restructuring occurs very rapidly for UO_2 and $\text{UO}_2\text{-ThO}_2$ at much lower heat ratings than those anticipated for fast reactor conditions. Thus, major structural differences may occur only during a startup period. Some degree of structural alteration will continue throughout the lifetime of the fuel, since the system conditions will be non-equilibrium both chemically and thermally.

The major area of interest after this initial restructuring is the peripheral rim of the fuel material, which will maintain its as-fabricated fuel structure for an extended time. In the case of particulate fuel, this rim shows little sintering of particles and reflects the initial, tightly packed granular structure with a rather uniform distribution of inter-connected voids. The outer rim of pelletized fuel shows a high-density, sintered structure in which the majority of the void space is closed.

The essentially unsintered rim of the particulate fuel has been viewed with concern, since such a layer would cause the effective thermal conductivity of the system to decrease. This concern has a firm theoretical basis; however, the extent of the effect and its relationship to performance have not been established for typical fast-fuel compositions and pin configurations under anticipated fast-reactor operating conditions.

The nature and the anticipated behavior of this outer rim, in conjunction with fabrication and processing advantages and other performance characteristics, are the basis for the interest in developing particulate fuel for a fast reactor system. Areas in which particulate fuel is expected to show definite advantages over a corresponding pellet fuel are as follows:

1. Swelling is accommodated.
2. Ridging and ratcheting effects are eliminated.
3. Stress distribution on cladding is improved.
4. Thermal expansion is more uniform.
5. Fission gas removal is enhanced.
6. Fabrication cost is reduced.

Fuel swelling is a major obstacle to reaching the high burnup—about 10% average—desired in fast reactor fuel, and provisions must be made to accommodate swelling in any solid-fuel element. At the present level of technology, the amount of fuel swelling and the preferred mechanism for its accommodation in the fuel element are still uncertain.

The swelling rate data most commonly used were derived from UO_2 irradiations that indicated a fission-product swelling rate of

about 1.7% volume increase for each 1% burnup of metal atoms. Early results of mixed-oxide irradiations have indicated that the net swelling rates are lower for fast reactor pin configurations and operating conditions. Swelling experiments carried out with strong radial restraint have shown much-reduced axial swelling and a strong dependence on temperature. On the basis of the UO_2 swelling data, the total volume change resulting from solid and gaseous fission products may be about 17% at a burnup level of 10%. Most current oxide fuel designs incorporate a 15 to 20% void space in the fuel element to accommodate fuel swelling with a minimum of cladding strain.

7.2.2. Fabrication Development of Vipac Fuel

7.2.2.1. Objective

The objective of this R&D program is to develop the feed material preparation and vibratory compaction processes sufficiently to produce technically and economically acceptable vipac fuel rods for use in large LMFBRs by 1983.

7.2.2.2. Summary of Current Technology

The design and production of a satisfactory fuel element begins with establishing a suitable fuel feed material. In this respect, $(\text{U}, \text{Pu})\text{O}_2$ fuel has benefited greatly from the background of UO_2 technology. The relative insensitivity of the mixed oxides to contamination by moisture and oxygen provides process advantages over such other fuels as carbides and nitrides.

There are three techniques for the industrial preparation of oxide materials:

1. Mechanical blending of individual oxides.
2. Coprecipitation of mixed oxides.
3. Sol-gel preparation of mixed oxides.

The first two techniques are more highly developed, although the last may be sufficiently developed by the time it would be required for commercial plants. Neither mechanical blending nor coprecipitation techniques produce material that is directly suitable for vipac. On the basis

of the present state of the art, only mechanical blending and coprecipitation could be considered. B&W's preference for the coprecipitation technique is based on present homogeneity requirements.

Several processes are available to produce high-density particles from the oxides produced by these techniques: for example, (1) fusion, (2) compaction-sintering, (3) particle agglomeration, and (4) pneumatic impaction. Of these four, only the fusion process is obviously too little-developed for consideration. The development of a process to produce feed material for vipac will center on the use of coprecipitated mixed oxides densified into a suitable vipac material by any one of the latter three techniques above. Primary emphasis will be placed on the compaction-sintering process, with secondary efforts on particle agglomeration and pneumatic impaction. Controlling parameters will be O/M ratio, particle density, control of the Pu/Pu+U ratio, and homogeneity of plutonium distribution.

The vibratory compaction process used for fabricating fuel rods depends on the characteristics of the feed material (particle density, friability, and particle shape), the character of the applied vibration, the classification of particle sizes, and the method of loading the particles. Several of these parameters are interdependent; angular particles require high-energy uniaxial vibration like that obtained from electrodynamic or pneumatic shakers, whereas spherical particles may be compacted best by the use of low-energy biaxial vibration. Previous work has shown that angular particles may be compacted to higher densities (83 to 87% TD) than can now be obtained with spherical particles (80 to 92% TD) in reasonable compaction times. The method of introducing the feed material into the cladding has a strong influence on the mass variation obtainable along the length of the fuel rod. The frequency of vibration and the acceleration imparted to the rod are both influential on packed density and mass variation.

The development of the vibratory compaction process will center on the use of angular particles with a three-diameter-particle system, using electrodynamic vibration. The development of optimum particle sizes, vibration and acceleration sequences, and feeding methods will be carried out with the goal of $85 \pm 1\%$ TD packed density and a mass variation of less than 1% along the total fuel column length.

The degree of intermingling of fuel and blanket materials at their interface will also be defined, and this zone will be minimized.

7.2.2.3. Technical Justification

At the present time, practically no work has been done on producing angular particles from the compaction-sintering process. The slugging and sintering of large quantities of mixed oxide has not been studied. It would certainly be cheaper to slug and sinter than to press carefully controlled pellets which are sintered. The effects of surface cracks, laminations, and other structural faults on the acceptability of the slug and on the production of material with a high particle density need to be investigated to evaluate the feasibility and economics of the process. The packing efficiency achieved with present vipac technology is approximately 88%, and the packed density may be calculated by multiplying the packing efficiency by the particle density. If 85% TD is desired for the packed density, it can be seen that a minimum particle density of 96.5% TD is required.

The pneumatic impaction work that has been done to date has been carried out at BNWL. Their process is capable of producing high-density masses that can be crushed to give material with suitable particle densities. It has not been demonstrated that this process can produce the great volume of material necessary to make it economically attractive, nor has it been demonstrated that the material so produced has the proper O/M ratio or a low enough impurity content (sorbed gas and moisture) for LMFBR fuel. The particle agglomeration process has not been demonstrated in large-scale operations to show its feasibility at "kilogram-per-day" quantities.

7.2.2.4. Scope of Work

It is felt that the single most important step in the vipac process is the introduction of the fuel material into the cladding tube. The feeding of the oxide into the cladding in a uniform, thoroughly mixed, and homogeneous batch will be investigated, and an optimum technique will be developed for a nominal production rate. The anticipated results will be an increased packing efficiency and a reduced mass variation. At the present time, both of these results are necessary to ensure the success of the vipac process.

The three different sizes of particles used in all prior vipac work have been established on an empirical basis. The empirical formula has shown some signs of weakness when small-diameter tubing is used (less than 0.300 inch ID), yielding lower packed densities than expected. Particle sizing relationships for small-diameter tubes need to be established to optimize the packing efficiency. The particle sizing relationship also affects the mass variation achievable in the fuel column, as do the feeding of the oxide and the vibration frequencies and acceleration. The mass variation needs to be minimized to produce more uniform heat generation rates along the length of the rod. Mass variations of less than 10% are required.

7.2.2.5. Relationship to Current or Planned Programs

The National LMFBR Program will not provide the necessary information on particulate fuels. The Gel-Addition Program presently funded at B&W by the USAEC will provide some information on particulate fuels, but it is primarily aimed at spherical particles and their irradiation behavior. It will not provide the in-depth fabrication R&D that is necessary.

7.2.2.6. Major Facilities and/or Equipment Required

The major facilities and equipment needed are available at the Babcock & Wilcox Nuclear Development Center in Lynchburg, Virginia.

7.2.2.7. Budget Estimate

Table 7-2 shows the planned budget and schedule for fabrication development R&D.

7.2.3. Irradiation Performance and Behavior

7.2.3.1. Objective

The phenomenological behavior of mixed oxide vipac fuel will be evaluated in sufficient depth to justify its inclusion in irradiations in EBR-II, the FFTF, and the predecessor plant (as test bundles). The ultimate purpose is to qualify vipac for 1000-MWe use.

Table 7-2. Estimated Costs and Schedule for R&D on the Fabrication Development of Vipac Fuel

Overall schedule and estimated cost, \$ × 10 ³						
Task	FY:	1969	1970	1971	1972	1973
			①		② ③ ④	⑤
Laboratory development						
Pilot run						
Total						

350.0

74.5

424.5

Milestones

- ① Start vipac development.
- ② Start pilot run for statistics.
- ③ Complete vipac development.
- ④ Complete pilot run for statistics.
- ⑤ Finish report.

(a)	Manpower	\$350,000
	Equipment	11,500
	Materials	63,000
	Total	\$424,500

7.2.3.2. Summary of Current Technology

Current information is reported on powder fuel where it has been found available. Data on pellet fuel are included where needed to supplement the data on powder fuel and to provide background for the predecessor plant fuel.

1. Restructuring — Restructuring and void redistribution, center hole formation, and grain reorientation for 83% mixed oxide, powder fuel were described by Bailey in 1967,¹ and Freshley reported on restructuring in UO₂-PuO₂ powder fuel in the PRTR.² They found that time-temperature-dependent sintering effects cause the radius of the sintering boundary to increase with increasing exposure.

2. Fission Gas Production and Release — Lewis³ reports that below 1200 C, fission gas release is negligible for burnups of up to 15,000 MWd/MT (U). In light of this and a review of the theory of fission gas release, it appears likely that fission gas release is mainly a function of the volume of fuel at temperatures above 1800 C, where all of the fission gas formed is released. The percentage of gas

released is also a function of the cyclic power operation, which causes the fuel to crack, releasing the trapped fission gas from the central void (in some cases) and from the grain boundaries and voids opened by the cracks. Cracked surfaces operated in the temperature range of 1300 to 1800 C would be expected to release more gas than cracked surfaces in the lower temperature range because of the gas concentrated in the grain boundaries at the higher temperature.

Several parameters have been considered to affect the percentage of fission gas release from an operating fuel element. These are fuel density, surface area, temperature, stoichiometry, grain growth, burnup, and fission rate.

Effect of Fuel Density and Surface Area on Fission Gas Release: Figure 7-1 shows Notley's data on percentage of fission gas release from UO_2 fuel rods as a function of density and heat rate or $\int \text{kdT}$.⁴ The density varied between 10.42 gm/cm³ (95% TD) and 10.74 gm/cm³ (98% TD). The fission gas release and the extent of equiaxed grain growth increased as the density decreased. The extent of equiaxed grain growth apparently does not affect the percentage of gas released.

Gerhart⁵ carried out a program to study the irradiation behavior of 5% enriched $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_2$ in 40 fuel rods encapsulated in NaK and irradiated in the GETR to burnups ranging between 5200 and 109,000 MWd/MT (Pu + U). The inner surface temperature of the cladding ranged from 705 to 1255 F, and the heat fluxes ranged from 0.5×10^6 to 1.58×10^6 Btu/hr-ft². The fuel rods had 0.190-inch ODs and were clad with stainless steel 0.020 inch thick. The diametral clearance between the fuel and cladding was 0.006 to 0.008 inch for the pellet rods. End voids were included in all but a few rods to prevent high fission gas pressured.

None of the fuel rods failed, nor was excessive swelling apparently observed. The correlation between fission gas release and heat rating or burnup was impossible, because of the number of variables and experimental inaccuracy. The variables affecting the fission gas release were pellet densities, cladding inner surface temperatures, fuel-cladding diametral gaps, and irradiation time. The experimental accuracy of the parameters affecting the fission gas release correlation were as much as 20%, based on thermocouple data.

Figure 7-1. Percentage of Fission Gas Release as a Function of Density and Heat Rate⁴

Gerhart reported that the fission gas release averaged 34.5% for the pellet and 47.1% for the swaged fuel rods.

The Saxton Plutonium Program⁶⁻⁸ is an extensive irradiation program comparing vibratory compacted powder and mixed pellets of $(\text{Pu}_{0.6}\text{U}_{0.4})\text{O}_2$. The burnup of pellets reached 21,550 MWd/MT in June 1967. The first postirradiation data on fuel rods operated at maximum heat rates of 15 kW/ft to a peak burnup of 6100 MWd/MT showed no diametral increase but did show length increases of 17 to 31 mils. The fission gas release of the 94%-dense pellet rods and 87%-dense powder rods were 2 and 10%, respectively.

Brown⁹ has irradiated $(\text{Pu}_{0.20}\text{U}_{0.80})\text{O}_2$ Dynapac Powder fuel in the EBR-II to a peak burnup of 24,000 MWd/MT (Pu + U). A detailed description of the postirradiation examination of fuel rod SOV-6 was presented in ANL reports. Gas release of approximately 65% for an O/M of 2.00 to 2.02 was estimated by volumetric measurements. The calculated pressure in the rod based on 65% gas release is 224 psi.

Bailey¹¹ has irradiated vibratory compacted $(\text{Pu}_{0.015}\text{U}_{0.985})\text{O}_2$ fuel rods to a peak burnup of 26,300 MWd/T of fuel without any failures. These rods were 0.412 inch in diameter and clad in Zircaloy. For a heat rate of 19.2 kW/ft and a burnup of 16,930 MWd/T, the fission gas release was 30%. For a heat rate of 10.4 kW/ft and a burnup of 18,300 MWd/T, the fission gas release was 14%.

Ross¹² has irradiated $(\text{Pu}_{0.05}\text{U}_{0.95})\text{O}_2$ to a peak burnup of 20,000 MWd/MT(M) and $(\text{Pu}_{0.20}\text{U}_{0.80})\text{O}_2$ to a peak burnup of 50,000 MWd/MT(M). For the 5% PuO_2 fuel, fission gas releases of 18 to 36% were reported. Central melting and columnar grain growth were found in the mechanically blended fuel, which had an O/M ratio of 2.00.

Effect of Temperature and Stoichiometry on Fission Gas Release: General Electric¹³ has collected the available fission gas release data on 94 to 97%-dense UO_2 pellets and has correlated this information with the fuel volumetric average temperature (VAT).

Freshley's data¹⁴ for fission gas release as a function of the volumetric average temperature for $(\text{Pu}_{0.02}\text{U}_{0.98})\text{O}_2$ fuel with burnups ranging from 100 to 5000 MWd/MT are given in Figure 7-2. This curve is for vibratory compacted powder of 86 ± 1 % TD.

The results show that 92% of the fission gas is released at a VAT of 2200 C in a fuel rod operated at 27 kW/ft. It is estimated that melting occurs at 20 to 21 kW/ft.

Figure 7-2. Fission Gas Release as a Function of Volumetric Average Fuel Rod Temperature in Vibratory Compacted $(\text{Pu}_{0.02}\text{U}_{0.98})\text{O}_2$ PRTR Fuel Rods¹⁴

Effect of Grain Growth on Fission Gas Release: The extent of equiaxed grain growth is a function of temperature, time at temperature, and fuel impurities. The time to achieve a given grain size increases as the temperature decreases.

The increase in fission gas release with increasing grain growth appears to be a function of temperature rather than grain growth. The grain boundaries retain the gas until cracking along the grain boundaries releases it. An increase in temperature increases the rate of bubble diffusion to the grain boundaries and also moves the isotherm, which, in turn, causes grain boundary movement out to a larger diameter. This causes an increase in fission gas release because of the larger volume of fuel in the temperature region where grain boundary sweeping takes place.

Further Effects of Burnup on Fission Gas Release: Mustelier¹⁷ irradiated ten type 304 stainless-steel-clad rods of $(\text{Pu}_{0.11}\text{U}_{0.89})\text{O}_2$. The only gas release values comparable as a function of burnup gave fission gas releases of 12.5% at a burnup of 4000 MWd/MT (U + Pu) and 24.5% at 8000 MWd/MT (U + Pu). These rods were operated at a peak heat rate of 14 to 15 kW/ft.

Lawton's work on mixed oxides indicates that the percentage of fission gas released is a function of burnup for vibratory compacted and annular pellet fuel;¹⁸ his results are shown in Figure 7-3. Vibratory compacted fuel releases considerably more gas than do annular pellets; the amount released is almost linearly proportional to burnup to about 25,000 MWd/MT. Lawton concludes that about 90% gas release will be obtained from vibratory compacted fuels and 50% gas release from annular pellets at 82,000 MWd/MT.

3. Sodium-Fuel Compatibility—Experimental studies of sodium logging in fast reactor fuels were conducted by General Electric and reported in 1963.¹⁹ It was stated in their report that sodium logging should not be a major criterion for setting performance limits for high-density fuel. They tested fuel performance characteristics to power levels which produced central fuel temperatures above the fuel melting point without significant damage to the fuel element by the sodium.

Figure 7-3. Fission Gas Release Data for Mixed Oxide ($\text{Pu}_{0.15}\text{U}_{0.85}\text{O}_2$) Fuel¹⁸

The 24th quarterly report on the Sodium Cooled Reactors Fast Ceramic Reactor Development Program includes the results of this examination of a fuel-sodium-cladding compatibility test capsule. The majority of the sodium appeared to have reacted with the fuel after isothermal testing at 1400 F for 1007 hours.

4. Burnup Swelling — Tests have shown that fuels increase in volume as a function of burnup.²⁰ This volume increase can result in significant fuel-element distortion and eventual failure of the cladding. Fuel swells relatively slowly until some critical burnup is reached, after which the swelling rate is accelerated.

It is reported that the swelling is caused by the accumulation of fission products in the fuel—primarily fission gases, which occupy more space than the fissioned atoms. The increase in swelling at some particular burnup is the result of internal porosity and the loss of accommodation for part of the fission products by the crystal lattice. The burnup at which accelerated swelling occurs is a function of the temperature, density, and restraint by the cladding. The burnup, for breakaway swelling, decreases with increasing heat rate and fuel density and increases with restraint.

Westinghouse¹⁶ irradiated 60 UO_2 fuel plates to burnups as high as 127,000 MWd/MT(U). The fuel density was 87 to 98% of theoretical. The fuel center temperatures were generally less than 1175 C (2150 F), but in a few plates peak fuel center temperatures up to 2100 C (3800 F) were achieved. It was concluded that the swelling rate was not affected by the fuel temperature at temperatures below 2000 C. Swelling rates of 0.16 vol% per 10^{20} f/cm were indicated until fuel internal voids were filled, followed by 0.7 vol% per 10^{20} f/cm.

Bettis²¹ reported that the swelling rate of dished-end pellets was about 0.82 vol% per 10^{20} f/cm after burnups of 45,000 to 55,000 MWd/t. About 35% of the swelling was accommodated in the end dishes, 49% in filling the porosity, and 16% by diametral expansion of pellets.

Gerhart⁵ irradiated $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_2$ to burnups as high as 109,000 MWd/MT (U + Pu). The swelling rate as a function of burnup could not be correlated because of the small fuel column heights (1.6 to 1.8 inch) and the accuracy ($\pm 5\%$) with which the percentage

change could be measured. None of the 40 fuel rods failed. These rods contained a plenum for high gas release, relatively thick cladding, and a large amount of void space in the fuel of the swaged rods and in the diametral gap (6 to 8 mils) of the pellet rods, which accounts for the lack of failures. The swaged fuel rod with the highest burnup, 109,000 MWd/MT (U + Pu), contained 25% void volume in the fuel. If the assumed fuel swelling rate is 0.7 vol% per 4000 MWd/MT(M) or 10^{20} cm³, the burnup required to just fill the voids would be 143,000 MWd/MT (U + Pu).

Mustelier²² irradiated (Pu_{0.11}U_{0.89})O₂ to burnups of 26,800 MWd/MT (U + Pu) at a peak heat rate of 14 kW/ft and found diametral increases up to 0.5%. A swelling rate of 0.7 vol% per 10^{20} f/cm³ or 4000 MWd/MT (U + Pu) could have been accommodated within the cladding without exceeding these diametral changes.

Nelson²³ and Zebroski²⁴ have irradiated (Pu_{0.25}U_{0.75})O₂ fuel to high burnups. Capsule E2E had a measured burnup of 154,000 MWd/MT (oxide). Irradiations of other similar capsules, E2F, E2G, and E2H, are continuing. In these capsules, the swelling was determined by measuring the increase in fuel column length. According to their data, the swelling rate for capsule E2E was 0.094 vol% per 10^{20} f/cm³. At the heat rate encountered in this pin, 12 kW/ft, no significant columnar grain growth or central void was observed. Equiaxed grain growth started at 0.7 of the radius of the fuel and extended to the center of the fuel. These tests indicated that the swelling rates are affected by the fuel temperature, as shown in the following tabulation:

Swelling Rate Vs Temperature

<u>Capsule</u>	<u>Fuel surface temp, C</u>	<u>VAT, C^(a)</u>	<u>Fuel center temp, C</u>	<u>Swelling rate of (Pu_{0.25}U_{0.75})O₂, vol% per 10^{20} f/cm³</u>
E2G	680	1000	1400	0.055
E2F	860	1200	1800	0.182
E2E	1060	1500	2200	0.094

(a) VAT: volumetric average temperature.

These swelling rates for (Pu_{0.25}U_{0.75})O₂ are lower than those reported for UO₂.

In another phase of the GE program, data were obtained from their capsule E1E. Burnup analyses were made using neodymium, cerium, and heavy elements. The average burnup in these pins was 87,000 MWd/MT (oxide) or 76,500 MWd/MT (U + Pu). Profilometer scans of the pins did not reveal any radial swelling. Neutron radiography and gamma scanning indicated axial expansions equivalent to volume changes of 3.78 and 4.42 vol%, respectively; this is equivalent to about 0.2 vol% per 10^{20} f/cm³.

The GE data²⁵ on swelling rate versus burnup of mixed oxide fuel are shown in Figure 7-4. These data, obtained from thermal reactor irradiations with strong radial restraint, indicate that the swelling rate of mixed oxide is lower than that of UO₂ and that the swelling rate depends on the fuel temperature. They report that the maximum swelling rate is exhibited by fuel in the intermediate temperature range (1800 C). The rate of swelling was found to be 0.3 vol% per 10^{20} f/cm³ for radially restrained fuel pins. Fuel pins with fuel densities and radial restraint closer to prototype conditions displayed a swelling rate similar to that found with burnups of 106,000 MWd/MT (U + Pu). Their swelling volume was based on the axial expansion of the fuel column and the cladding diametral increase. Corrections were not applied for central void formation, internal porosity, or diametral gap.

7.2.3.3. Scope of Work

An R&D program designed to provide the required information will include the following work:

1. Determination of performance capability and operational limits for the fuel material.
2. Translation of overall performance information into optimum fuel system design and operating conditions.
3. Confirmation and assessment of predicted fuel behavior.

The major phases of the irradiation work include evaluation of restructuring and void distribution, fission product generation, fuel-sodium compatibility, and burnup effects such as swelling and gas release.

One of the least-understood and most controversial areas of concern is the effect of the particulate fuel structure on the initiation of central fuel melting, and the degree to which such an

Figure 7-4. Volumetric Swelling in $\text{PuO}_2\text{-UO}_2$, Fuel Irradiated in Thermal Flux²⁵

effect limits the operational heat rating of the fuel system. There is a pressing need for evaluation of the various aspects of this problem.

The thermal behavior of the particulate, mixed-oxide fuel is related to the initial fuel structure, the microstructure developed in the reactor, and the effective thermal conductivities associated with these structures. The initial low thermal conductivity of the as-fabricated particulate fuel indicates a sensitivity of reactor startup conditions. For similar startup conditions, greater temperature gradients can be anticipated in the particulate fuel than in pellet fuel. Irradiation experiments are required to determine the effect of startup conditions on the restructuring rate of particulate fuel.

The evaluation of performance capability of the vipac oxide fuels requires not only this early-life information but also a quantitative indication of the maximum heat rating allowable throughout the core lifetime. Included here are requirements for evaluating the allowable heat rating in relation to central fuel temperature, effective thermal conductivity, physical restructuring, burnup, and fission product formation and distribution.

Determination of the burnup capability of the vi-pac fuel requires irradiation work to establish the effects of various factors that contribute to burnup limitation: (1) swelling behavior, (2) fission product effects, and (3) changes in physical properties.

The information needs regarding swelling behavior are not unique to the particulate fuel but apply also for successful utilization of mixed-oxide pellet fuels. The lack of agreement on fuel swelling emphasizes the need for experimental work in this area. There is a need for definitive experiments to assess the maximum swelling rate of the mixed oxide fuels to at least the 10 at. % burnup level and to determine the effects of operating variables, restraint, and fuel structure on swelling.

The design of particulate fuel elements requires knowledge of the rate of gas release and total gas release as a function of temperature and burnup in order to assess the contribution to cladding deformation.

The formation of solid fission products and their possible migration contributes to the overall problem of fuel homogeneity (including fuel migration) and may contribute to the degradation of fuel properties. The presence of solid fission products at variable concentrations creates a complex phase system in which solid-state reactions may contribute to a decreased density and a lower melting point. There is a need to establish the effect of fission product concentration and reactions on the melting point and density profiles of the fuel pin as a function of burnup.

The particulate fuel structure is susceptible to sodium infiltration before restructuring occurs and after restructuring in the outer, as-fabricated rim. The initially open structure of this fuel presents good possibilities for sodium-logging of the element. Recent General Electric reports on the fuel-sodium reaction at 1400 F must be substantiated and evaluated in the interest of establishing true temperature limits for operation with sodium ingases.

The evaluation of restructuring will require 1-1/2 years' work, during which the effect of heat rate will be evaluated; a total of 17 capsules will be irradiated in a thermal reactor. The effect of heat rate, time, and power cycling on the gas release characteristics

of particulate fuel will be determined in this task, which will require about 18 months to complete the irradiation and evaluation of 11 capsules.

Three years will be required to investigate the compatibility of the particulate fuel with sodium. In the initial part of the task, the effect of heat rate, temperature, and the type of defect on fuel-sodium compatibility will be evaluated in eight capsules; this work can be performed in a thermal reactor. Eventually, two nine-rod clusters containing defective elements will be irradiated in an in-reactor flowing sodium loop (FFTF or GETR).

The swelling behavior of particulate fuel will be evaluated in the EBR-II reactor. Elements operating at three different heat rates to burnups in excess of 100,000 MWd/T will be included in this task. It is anticipated that three 19-rod bundles or portions of three 37-rod bundles can be used for these irradiations. About 8 years will be required to complete the task.

7.2.3.4. Major Facilities and/or Equipment Required

This work will require reactor and hot cell facilities such as thermal reactors (MTR, ETR, or BAWTR); fast reactors (EBR-II or FTR-FFTF); and hot cell facilities. Since the LMFBR effort on pellet fuel will predate the work described in this document, sufficient equipment for fabricating, shipping, handling, and checking of test capsules and elements is expected to be available at several sites.

7.2.3.5. Relationship to National Programs and Technical Justification

The state-of-the-art review has indicated the applicability of most pellet fuel R&D data to particulate fuel development. The data can be utilized because vipac sinters in-reactor to a similar structure except for the rim, and because similar densities are contemplated for both fuels in LMFBR R&D.

The National LMFBR Program has decided to concentrate on pellet fuel. Fortunately, a few tests of particulate fuel have been included, but many more must be performed to establish the potential benefits of particulate fuel. In particular, long-term tests are necessary to establish its increased burnup capability under LMFBR conditions (over that obtainable with pellet fuel). Evaluation of the

benefits that might accrue in fabrication and recycling is in the province of Fuel Recycle.

In addition, tests must cover those aspects not adequately investigated in pellet fuel tests and must provide sufficient statistical information to convince technical and nontechnical management that vipac fuel should be used.

7.2.3.6. Budget Estimate

Table 7-3 shows the planned budget and schedule for this work.

7.2.4. Measurement of Fuel Properties

7.2.4.1. Objective

Accurate prediction of fuel element performance in commercial LMFBRs will be possible only if valid numbers can be assigned to mechanical and physical properties of the fuel from the beginning to the end of reactor service. This task will provide these numbers to the extent and accuracy that is practical. Emphasis is on irradiated properties.

7.2.4.2. Summary of Current Technology

Thermal Conductivity — Thermal conductivity for UO_2 powder was determined by B&W in 1962 as shown in Figure 7-5.²⁶ General Electric produced curves for 85 and 90% dense UO_2 powder in 1966, as shown in Figures 7-6 and 7-7.²⁷ Their curves were made by assuming that the conductivity of pellet and powder fuel is the same at the melting point. Below the melting point, the curve for powder fuel was assumed to fall smoothly and gradually below that for pellet fuel. The deviation between the two curves was adjusted so that the $\int k dt$ for powder from 500 C to melting would be 20% less than that for pellet fuel. Below 500 C, the curve was extended to provide a mean through the out-of-reactor data. The dotted line in Figure 7-6 represents K if K is discontinuous instead of smooth.

Westinghouse curves for 87% dense UO_2 powder are shown in Figures 7-8 and 7-9.²⁸ These curves for startup and operating conditions indicate that densification occurs at approximately 400 C and the thermal conductivity is improved.

Table 7-3. Estimated Costs and Schedule for R&D on the Irradiation Performance and Behavior of Fuel Material

Task	Overall schedule and estimated cost, \$ × 10 ³										Total ^(a)
	FY: 1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	
Irradiation Performance and Behavior of Fuel Material				△1					△2		
Time for restructuring and void distribution											590.0
Fission product generation											780.0
Fuel-sodium compatibility											350.0
Swelling and gas release											1770.0
Total											3490.0

Milestones

- △1 Complete thermal irradiations.
 △2 Complete EBR-II irradiations.

(a) Manpower	\$2,835,000
Equipment	--
Materials	205,000
Irradiation	450,000
	<u>\$3,490,000</u>

Fereday obtained a value of 56 ± 3 W/cm for $\int kdt$ in 3-minute irradiations of 92% TD, swaged UO_2 specimens.²⁹ By extrapolating 20-day irradiation data he derived a value of 68 W/cm for $\int kdt$.

General Electric³⁰ compared the thermal conductivity of UO_2 with $(Pu_{0.2}U_{0.8})O_{1.97}$. They reported that the mixed oxide has a slightly lower thermal conductivity than that of UO_2 below 1800 C, as shown in Figure 7-10.

General Electric revised their $\int kdt$ curve for $(Pu_{0.2}U_{0.8})O_2$ in 1966 as shown in Figure 7-11.³¹ Values of $\int kdt$ from 900 to 2875 C were obtained by integrating the out-of-reactor data for $(Pu_{0.2}O_{0.8})O_{1.98}$.

Freshley¹⁴ reported that fuel melting, as evidenced by the formation of a well-defined region composed of a cellular-appearing structure, commences in vibratory compacted (86% TD) $(Pu_{0.02}U_{0.98})O_2$ fuel rods at 20 to 21 kW/ft under PRTR operating conditions. General Electric²⁴ reported conductivity data for $(Pu_{0.2}U_{0.8})O_{1.98}$ having 95% TD as shown in Figure 7-12.

Gibby of BNWL³² has calculated the thermal conductivity of $(Pu_{0.2}U_{0.8})O_2$ as a function of temperature using a laser heat-pulse apparatus to measure the thermal diffusivity. As shown in Figure 7-10, the thermal conductivity of stoichiometric $(Pu_{0.2}U_{0.8})O_2$ is essentially the same as that of stoichiometric UO_2 up to 1200 C. The UO_2 data of Godfrey³³ and the $(Pu_{0.2}U_{0.8})O_2$ data of Schmidt³⁴ and Hetzler³⁵ are shown for comparison.

Bailey³⁸ has published new thermal conductivity and integrated thermal conductivity curves for stoichiometric, 95% TD $(Pu_{0.2}U_{0.8})O_2$; these curves are shown in Figures 7-13 and 7-14, respectively. Bailey reported that the thermal conductivity of mixed oxide fuel is slightly lower than that of UO_2 . This lower thermal conductivity is more pronounced at temperatures below 1200 C. However, if we compare the UO_2 conductivity (Figure 7-15) with that of the mixed oxide (Figure 7-14), the lower thermal conductivity of the mixed oxide is only evident at temperatures below 800 C.

It has been found that significantly more melting can be expected in 85% dense fuel than in 95% dense fuel operating

at similar power levels. A slightly lower effective conductivity of powder fuel versus pellet fuel at near-identical smeared densities has been observed in thermal and fast flux irradiations. Comparing the 900 C $T_m \int k dT$ of GE for pellets (Figure 7-15) and powder (Figure 7-7), the powder-integrated conductivity is about 5% less than that for pellet fuel. Comparing the GE UO_2 pellet $\int k dT$ curve and the $(Pu_{0.2}U_{0.8})O_2$ curve indicates almost identical 900 C $T_m \int k dT$ values for UO_2 and $(Pu_{0.2}U_{0.8})O_2$.

1. Effect of Density on Thermal Conductivity:

The effect of density on thermal conductivity has been studied by Notley,⁴ who formulated the following equation for UO_2 :

$$k_p = k(1 - \alpha p) \quad (\text{modified Loeb equation}) \quad (7-1)$$

where

- k_p = thermal conductivity of porous UO_2
- k = thermal conductivity of 100% dense UO_2
- p = fractional porosity originally in the UO_2 ;
 $= 100\% - x\%$
 $= \frac{100\%}{100\%}$
- ρ = density
- α = constant = 2.6 ± 0.8

Biancheria related α to the shape of the pores and recommended the use of the following equation for correcting for density:^{36,37}

$$k_p = \frac{K(1 - p)}{1 + p(\alpha - 1)} \quad (7-2)$$

The following values of α are recommended by Biancheria:

<u>Fuel density, %</u>	<u>α</u>
90-100	1.5
85-90	2.0
80-85	2.4
75-80	2.6

Figure 7-5. Heat Rating Vs Temperature

Figure 7-6. Thermal Conductivity of UO_2 Powder (85-90% TD)
by General Electric²⁷

Figure 7-7. $\int kdT$ Vs T of UO_2 Powder (85-90% TD) by
General Electric²⁷

Figure 7-8. Thermal Conductivity of UO_2 Powder (87% TD) by Westinghouse²⁸

Figure 7-9. $\int kdT$ Vs T of UO_2 Powder (87% TD) by Westinghouse²⁸

Figure 7-10. Thermal Conductivity of $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_{2.0}$ as a Function of Temperature (Out-Reactor Data)³⁰

Figure 7-11. $\int kdT$ Vs T of $(Pu_{0.2}U_{0.8})O_2$ (95% TD)^{31,74}

Figure 7-12. In-Reactor Thermal Conductivity of $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_2$ (95%TD)^{24,30}

Figure 7-13. Thermal Conductivity of $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_2$ (95% TD)³⁸

Figure 7-14. $\int kdT$ Vs T of $(Pu_{0.2}U_{0.8})O_2$ (95% TD) by General Electric³⁸

Figure 7-15. $\int kdT$ Vs T of UO_2 Pellet (95% TD) by General Electric²⁷

The correction factors for equations 7-1 and 7-2 are listed below.

Fuel Density, %	α^*	Porosity, p	Equation 7-1 $(1 - 2.5 p)$	Equation 7-2 $1 - p$ $1 + p (\alpha - 1)$
90-100	1.5	0.1 - 0.0	0.85-1.0	0.85-1.0
85-90	2.0	0.15-1.0	0.70-0.85	0.74-0.82
80-85	2.4	0.2 - 0.15	0.52-0.70	0.63-0.70
75-80	2.6	0.25-0.2	0.35-0.52	0.54-0.61

* α is used with equation 7-2 only, since a value of 2.5 is used for correcting equation 7-1.

Asamoto⁴⁰ has measured the thermal conductivity (k) of 82, 86, 91, and 95% TD UO_2 between 800 and 2200 C by the radial heat flow technique. He reported that the difference in k with a given density change at a constant temperature is greater at densities of 90 to 95% TD than in the 80 to 90% TD range. He concluded that the modified Loeb equation was not valid for the temperatures and densities investigated, and that there was no increase in k with increasing temperature. The k of UO_2 is related to both temperature and density by equation 7-3.

$$k = 0.013 + \frac{1}{T(0.4848 - 0.4465 D)} \text{ (W/cm-}^\circ\text{C)} \quad (7-3)$$

for T between 800 and 2200 C and D between 82 and 95% TD.

The conductivity values calculated by equation 7-3 seem to check the unirradiated values. However, other experiments using the same radial heat flow technique have shown that the K does increase with temperature. Since this equation applies only to temperatures up to 2000 C and is based on out-of-reactor data, the modified Loeb equation is preferred.

2. Effect of Stoichiometry on Thermal Conductivity: Experimental values for $\int_0^T k dT$ versus T for irradiated UO_2 are shown in Figure 7-7. The $\text{UO}_{2.00}$ curve is drawn to fit thermocouple data at 1200 C, grain growth data at 1500 C, and melting at 2800 C. The references for the other data are listed on the figure. The thermal conductivity at a specific temperature of a particular composition during

irradiation equals the slope of its line in Figure 7-16. It is apparent from this figure that at high temperatures the average thermal conductivity of hyperstoichiometric UO_2 is still appreciably less than that of stoichiometric UO_2 . However, the O:U ratio affects the melting point as well as the thermal conductivity of UO_2 . BNWL data indicate that the melting point of UO_2 decreases from 2800 C at an O:U of 2.00 to 2400 C at an O:U of 2.2. The thermal conductivity of hyperstoichiometric UO_2 is less than that of stoichiometric UO_2 and less than that predicted by Robertson⁴¹ because of the reduction in melting point. Robertson's data do not show the reduction in melting point.

MacEwan⁴² has indicated that hypostoichiometric urania ($\text{UO}_{1.98}$) still has a higher thermal conductivity during irradiation than hyperstoichiometric urania ($\text{UO}_{2.007}$). This is a further indication that the thermal conductivity increases as the O:U ratio decreases below 2.00.

Bates⁴⁴ has shown that the fission products cesium and ruthenium migrate during operation of the fuel rod. The concentration of cesium across the fuel element radius varies by a factor as high as 1000. The cesium and ruthenium are depleted in the large columnar grain region. Thus the fission products are redistributed across the fuel radius during irradiation in addition to picking up oxygen.

UO_2 has been shown to dissociate at temperatures below its melting point.⁴⁵ The equilibrium composition is a function of both temperature and oxygen partial pressure. When UO_2 or $\text{UO}_{1.75}$ is heated in a hydrogen furnace to 2400 to 2600 C, an equilibrium composition of $\text{UO}_{1.89}$ is obtained. When heated in helium, which has a higher partial pressure of oxygen than dry hydrogen, the equilibrium composition is $\text{UO}_{1.94}$.

Christensen⁴⁶ has demonstrated that the $\int \text{kdT}$ from 500 C to melting decreases as the average O:U ratio increases. The $\int \text{kdT}$ from 500 C to columnar grain growth is constant for O:U ratios less than 2.00 and decreases rapidly for O:U ratios greater than 2.09. The $\int \text{kdT}$ from columnar grain growth to melting reaches a minimum at an O:U ratio of 2.00 and rises for an O:U greater or less than 2.00.

MacEwan⁴² irradiated $\text{UO}_{1.98}$ and $\text{UO}_{2.007}$ pellets to burnups as high as 460 MWd/MT(U). The $\int \text{kdT}$ for the $\text{UO}_{2.007}$

Figure 7-16. $\int k dT$ Vs Temperature (T) for Various Compositions of UO_2 During Irradiation⁴¹

decreased, whereas the $\int k dT$ for the $UO_{1.98}$ remained the same as the un-irradiated value. The latter case indicates that the thermal conductivity of the fuel above 500 C increased to compensate for the loss of conductivity in the low-temperature region for hypostoichiometric fuel.

Rubin⁴⁷ irradiated $(Pu_{0.25}U_{0.75})O_2$ fuel to high burnup (134,000 MWd/MT peak) to compare the behavior of stoichiometric and hypostoichiometric fuels at high burnups and high temperatures. It was concluded that at high burnup the thermal conductivities for O:M ratios of 2.00 and 1.96 are the same.

Baily and Craig⁴⁸ irradiated $(Pu_{0.20}U_{0.80})O_2$ for 5 hours to compare the molten radii as a function of stoichiometry. The fuel with an O:M ratio of 2.00 had an 8.75% greater melting radius than that with an O:M of 1.98. This indicates that at low burnup the thermal conductivity of fuel with an O:M of 1.98 from the surface to the

molten radius is 20% higher than that for fuel with an O:M of 2.00. However, the melting point of $\text{PuO}_2\text{-UO}_2$ with an O:M of 1.98 is 85 C higher than that with an O:M of 2.00. The difference in melting points accounts for much of the difference in molten radii attributed to the difference in thermal conductivity. More recent data by Bailey¹ indicate that the fuel with an O:M of 1.98 has a thermal conductivity from surface to molten radius only 7% higher than that with an O:M of 2.00.

3. Effect of Burnup on Thermal Conductivity:

Dayton and Tipton⁴⁹ report on investigations showing that the thermal conductivity (k) of UO_2 is reduced as much as 50% when irradiated at 600 C (1132 F) and below. These changes in k are observed at low burnups. At temperatures above 600 C, the k changes little during irradiation.⁵⁰ At higher burnups, the in-reactor thermal conductivity is unchanged from the out-of-reactor thermal conductivity.

Westinghouse⁵¹ uses the following equation to correct thermal conductivity for burnup:

$$k = k_0 [1 - \alpha (f/\text{cm}^3)] \quad (7-4)$$

where

k_0 = initial conductivity

k = actual conductivity

α = a constant = 0.016×10^{-20} (initial fuel porosity < 10%)

= a constant = 0 (initial fuel porosity > 10%)

This equation is for fuels with high initial densities (greater than 90% TD) in which the conductivity decreases as a function of burnup caused by the generation of fission gas pores. For fuel densities less than 90% TD, the correction factor is zero, but for densities greater than 90% TD, the correction factor is appreciable.

The UKAEA⁵³ irradiated an instrumented mixed oxide fuel pin and found that the center temperature increased from 1260 to 1500 C in 17 days of irradiation. The center temperature then decreased until it was back to 1260 at the end of the 80-day irradiation. They estimated that a 25% fission gas release in the first 17 days of operation could account for the temperature rise by decreasing the fuel-to-cladding-gap thermal conductivity.

According to Deem and Matolich,⁵⁴ the thermal conductivity of single crystals of UO_2 at 800 and 1200 C was 1.6 and 2.6 times greater than that of polycrystalline UO_2 . May⁵⁵ observed that the thermal conductivity of single crystals of UO_2 was higher than that of stoichiometric polycrystalline UO_2 regardless of the single crystal orientation.

The differences in thermal conductivity between the single crystals and polycrystalline UO_2 were related to differences in the O:U ratio rather than the crystal structure. Oxidizing the single crystals to remove the free uranium negated the increase in conductivity. Conversely, reducing a stoichiometric single crystal of UO_2 restored the high conductivity. Hypostoichiometric polycrystalline UO_2 and hypostoichiometric single crystals having the same stoichiometry have the same conductivity.

May has observed from both in-pile and out-of-pile data that the grain size and crystal orientation of UO_2 have little effect on the thermal conductivity.⁵⁵

Melting and Identification of Molten Boundary in UO_2 Fuel Rods — Bain⁵⁶ conducted 10-minute constant power irradiation tests of UO_2 pellets containing tungsten spheres; he concluded that the boundary between the central zone and the dense band of UO_2 is the limit of melting. According to Lyons,⁵⁷ the limit of melting is indicated by the initial disappearance of porosity moving radially inward, i.e., the outside edge of the dense band.

1. Effect of O:U Ratio: The data on the melting points of various PuO_2 - UO_2 fuels obtained by tungsten filament techniques^{58,59,61} indicate that a maximum occurs in the liquidus line at 5 to 10% PuO_2 . The thermal arrest data⁶⁰ obtained using sealed tungsten capsules did not show this maximum. The effect of initial stoichiometry on the melting temperature was determined by the heated-filament method. These data are presented in Figure 7-17, which also includes urania data for comparison.

The melting points of UO_2 and $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_2 \pm X$ were measured by the thermal arrest technique using tungsten-encapsulated specimens;^{64,24} these data (Figure 7-18) show that the melting

points of both UO_2 and $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_2$ vary as a function of the O:M ratio. Fuel with an O:M ratio of 1.97 to 1.98 has thus far been found to have the highest melting temperature.

The 2390 ± 20 C melting point for stoichiometric PuO_2 observed by Lyon⁶⁰ is in excellent agreement with a value of 2400 C reported by Hough and Marples.⁶² The 2840 ± 20 C melting point for UO_2 observed by Lyon is also in good agreement with the 2805 ± 15 C reported by Hausner.⁶³ Determination of the melting point of stoichiometric UO_2 at 2876 ± 7 C has been reported by Latta and Fryxell.⁶⁴ Lyon attributed the difference in reported melting points to the difficulty in establishing the true temperature calibration scale in the high-temperature range. The dotted line in Figure 7-18 represents the data of Latta and Fryxell.

The most recent data published by Bates indicate that the melting point of UO_2 is 2805 C for an O:U ratio of 2.00.⁶⁵ This experimenter used the heated filament technique and recognized that the O:U ratio probably changed during the tests; he did not report the final O:U ratios.

2. Effect of Burnup on Melting: Several experiments have shown a decrease in the melting point of UO_2 as a function of increasing burnup.^{66,67} These data are summarized in Figure 7-19. Christensen⁶⁶ reported a 130 C increase in the melting point of UO_2 irradiated to 0.3×10^{20} fissions/cm³ (1250 MWd/MT uranium). Further irradiation decreased the melting point until it was 140 C below its initial melting point at 95,000 MWd/MT(U). This peak has been omitted from the data shown in Figure 7-19. Bates⁴⁴ repeated Christensen's experiment and did not obtain the increase in melting point at low burnup. The data of Christensen were obtained with hyperstoichiometric UO_2 ; in the heated-filament technique used, it is quite likely that the O:U ratio decreased, causing the increase in melting point observed at this low burnup.

Thermal expansion data for UO_2 ⁶⁸⁻⁷¹ and PuO_2 ^{72,73} are shown in Figure 7-20. Murry's UO_2 thermal expansion data⁶⁸ agree closely with the Stanford Research Institute (SRI) data.⁶⁹ Conway's⁷⁰ and Christensen's⁷¹ UO_2 thermal expansion data are similar and indicate a lower thermal expansion than earlier data.

Figure 7-17. Results of Tungsten Ribbon Melting Point Determinations
Showing the Effect of the O/M Ratio on Melting
Temperature for $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_2$ ⁶⁰

Figure 7-18. Melting Temperature of $\text{UO}_{2 \pm X}$ and $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_{2 \pm X}$ ^{24,64}

Figure 7-19. Melting Temperature of UO_2 as a Function of Burnup^{66,67}

Figure 7-20. Thermal Expansion of UO_2 , PuO_2 , and 20% PuO_2 -80% UO_2 ^{68-70,72,74}

Brett⁷³ investigated the thermal expansion of PuO_2 having various O:Pu ratios (PuO_2 , $\text{PuO}_{1.98}$, $\text{PuO}_{1.77}$) and UO_2 up to a temperature of 1000 C. All available UO_2 thermal expansion data, including Brett's, agree up to 1000 C. The expansion values for PuO_2 , $\text{PuO}_{1.98}$, and the 1% PuO_2 in UO_2 are the same as the value for UO_2 up to 1000 C. The expansion of $\text{PuO}_{1.77}$ data up to 1000 C is in agreement with the data of Chikalla⁷² and is slightly greater than the expansion of UO_2 . The higher expansion of hypostoichiometric $\text{PuO}_{1.77}$ is caused by the contained $\alpha\text{-Pu}_2\text{O}_3$ changing as temperature is increased.

UO_2 and PuO_2 are soluble in each other and exhibit a linear variation in lattice parameters with composition changes. A weighted average thermal expansion curve for $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_2$, Figure 7-20, was made by McNelly⁷⁴ of GEAP based on GE-NMPO⁷⁰ data for UO_2 and HAPO⁷² data for PuO_2 .

Stoichiometry —

1. Effect of Temperature of Stoichiometry:

A mixed oxide fuel rod, 0.8 inch in diameter,²⁴ was irradiated under steady-state conditions for 55 EFPD at 17.9 kW/ft with an estimated center temperature of 2300 C. The fuel was found to have large columnar grains and a central void. At the peak fuel temperature, the O:U ratio was reduced from 2.00 to 1.98.

Bates⁷⁵ found that during irradiation UO_2 fuels became hypostoichiometric in regions where the temperature exceeded 1500 C. Westinghouse⁷⁶ found that after high burnups, above 50,000 MWd/MT(U), fuels have sufficient uranium metal precipitated in the columnar grain region to produce an average O:U ratio approaching 1.90 for this region. The fuel regions below 1500 C are thermodynamically capable of gettering all the oxygen liberated in the high-temperature fuel region. Thus, an irradiated fuel, depending on the temperature and average O:U ratio of the fuel, varies from hyperstoichiometric in the cooler regions to hypostoichiometric in the high-temperature regions.

It has been found that the most important variables affecting oxygen migration during the irradiation of a fuel rod are the average stoichiometry at beginning of life and the temperature profile established by the initial heat rating and fuel surface temperature.

2. Effect of Initial Average Stoichiometry:

Christensen⁴⁶ irradiated 1-3/8-inch-diameter UO_2 fuel pellets at 30 kW/ft and a fuel surface temperature of 500 C; appreciable center melting occurred. The average O:U ratios of the specimens were 1.90, 1.99, 2.00, 2.02, 2.05, and 2.15. These specimens were irradiated for 18 hours, which is ample time to achieve equilibrium stoichiometry. The postirradiation examination of the specimens has not been completed, but the radial oxygen distribution has been determined for the specimen with an O:U ratio of 2.00. The oxygen relocated slightly; migration occurred from the columnar grains into the adjacent cooler oxide and, to some extent, to the fuel center.

If the initial average O:U ratio is less than 1.99 at a fuel surface temperature of 500 C and a fuel center temperature of less than 1800 C, either the oxygen does not redistribute radially or the redistribution is such that the conductivity decrease in the low-temperature region is equal to the thermal conductivity increase in the high-temperature region.

When the initial average O:U ratio is greater than 1.99 at a fuel surface temperature of 500 C and a fuel center temperature of 1800 C, the oxygen redistributes radially, so that the decrease in thermal conductivity in the low-temperature region is greater than the increase in conductivity in the high-temperature region, and the integrated thermal conductivity is thus decreased. This is shown to be true by MacEwan's⁴² irradiation of $\text{UO}_{1.98}$ and $\text{UO}_{2.007}$ under identical conditions, where the $\int_{200}^{T_c} k dT$ of $\text{UO}_{1.98}$ did not change, but the $\int_{200}^{T_c} k dT$ of the $\text{UO}_{2.007}$ was reduced about 15%.

3. Effect of Burnup on Stoichiometry: It is estimated that 0.3 or 0.7 atom of oxygen is available per uranium or plutonium fission, respectively. The oxygen available from fissioning is taken up by either the fuel, the cladding, or both, depending on the cladding and its temperature.

Westinghouse data⁵¹ in Figure 7-21 show that the moisture content of the fuel dissociates into oxygen and hydrogen during irradiation and is completely dissociated after burnups of several thousand MWd/MT. The burnups required for complete dissociation depend on the moisture content of the fuel, which tends to increase the O:U

Figure 7-21. Reaction of Absorbed Water Vapor in UO_2 Fuel
During Irradiation⁵¹

ratio of the lower-temperature region of the fuel. When the moisture content is around 10 ppm, it has a negligible effect on the O:U ratio, but the effect becomes important with increased fuel moisture content.

Since these two sources of oxygen plus sorbed oxygen in the fuel will be available, it seems likely that the average O:U ratio will increase with burnup. The increase in O:U ratio will be rapid at first, until the available oxygen is absorbed by the fuel and cladding. Then the rate of increase in the O:U ratio will be slowed down to a constant rate, depending on the fissioning rate. In the case where the fuel rod power decreases with burnup, which is true for practically all high-burnup test specimens, the rate of increase in the O:U ratio will continue to decrease, and the decreasing rate of change in the O:U ratio will be proportional to the decrease in fissioning rate.

If the conclusion is correct that the average O:U ratio of a fuel element increases with burnup, then the integrated thermal conductivity from the surface to the center of the fuel will decrease with increasing burnup. If the 0.3 atom of oxygen per uranium fission is absorbed by the UO_2 , then at a burnup of 40,000 MWd/MT(U) (10×10^{20} f/cm³), the average O:U ratio will reach 2.12 from an initial 2.00. Changing the average O:U ratio from 2.00 to 2.12 will decrease the melting point of the fuel, resulting in a decrease in the heat rate to produce center melting. The changing O:U ratio with burnup accounts for some of the decrease in fuel melting temperatures at high burnups, but the individual effects of fission products and changing O:U ratios on the melting point are unknown.

Mechanical Properties — Tottle conducted mechanical tests on UO_2 employing four-point loading in bending at temperatures from room temperature (RT) to 2000 C.⁷⁷ A ductile-brittle transition was found between 1250 and 1500 C. Load deflection curves which were measured from 25 to 1900 C are shown in Figure 7-22. Tottle found that porosity plays a vital role in the mechanism of deformation.

Fortano and coworkers investigated the elasticity and anelasticity of uranium oxides at room temperature.⁷⁸ They found that the elastic modulus of stoichiometric sintered compacts increases with density. The following linear equation was developed for a volume fraction porosity greater than 0.1:

$$E = 22.33.85 (1 - 2.277 P)$$

where P = volume fraction of porosity

They found that the internal friction of $UO_{2.00}$ decreased sharply as grain size increased. Other microstructural features, such as number, size, and position of the pores, contributed to the behavior. Young's modulus for nonstoichiometric UO_2 depended mainly on the history of the sample. Varying the oxidation or the cooling rate of the sample will produce completely different results.

Tottle⁷⁷ also investigated the relaxation of UO_2 . A sample was bent to a deflection of 0.76 mm under a medium rate of straining at 1750 C, cooled to room temperature, inverted in the tools, and straightened by rebending at 1750 C under the same conditions. This procedure was carried out ten times. Apart from increased indentation by the knife edges, no change was observed in the specimen when the test was finished.

Attempts were made to conduct creep tests in a vacuum, but the test machine failed to function reliably enough to permit overnight runs. The only meaningful results are tabulated below.

Temperature, C	Density, g/cc	Load, kg	Total strain, mm	Time, hr	Minimum deflection rate, mm/mm-hr
1000	10.61	41	0.24	6	1.22×10^{-4}
1250	10.62	41	0.81	54.5	2.69×10^{-4}

Bettis has investigated the mechanical properties of UO_2 and other fuels under low-temperature, thermal reactor conditions. Data on low-temperature creep were provided by the KAPL 67-101 tests.

Bettis has also reviewed the out-of-pile mechanical properties of oxide fuels;⁷⁹ they found that steady-state creep rates show a transition from nearly viscous behavior at low stresses to low-power stress dependence at higher stresses. The creep rates and the viscous region were found to be strongly dependent on grain size. It is believed that the steady-state creep above $0.5 T_m$ can be represented by an equation of the following form:

Figure 7-22. Load Deflection Curves Measured at Temperatures From 25 to 1900 C

Load-Deflection Curves for Medium and Slow Strain at 1750 C

Load-Deflection Curves for Medium Strain at 1900 C

Load-Deflection Curves for Slow Strain at 25, 1000, and 1250 C

Load-Deflection Curves for Fast, Medium, and Slow Strain at 1500 C

$$\epsilon_s = A_1 \sigma \exp(-Q_1/RT) + A_2 \sigma^n \exp(-Q_2/RT)$$

where A, n, and Q are empirical parameters. As yet no fuel composition has been satisfactorily characterized with respect to the parameters in this equation except over limited ranges of stress, temperature, and material variables such as grain size. The apparent activation energy for creep of UO₂ indicates that cation diffusion is the rate-controlling process for high-temperature creep.

The strength of oxide fuels under conditions that result in brittle failure is dependent on a number of environmental and material variables. Some success has been achieved in evaluating the effects of grain size and porosity for UO₂. The data are described by the following empirical equation:

$$S = KG^{-a} e^{-bp}$$

where

G = grain diameter

P = volume fraction porosity

S = strength

K, a, b = empirical constants, which have been determined for a few temperatures and modes of fracture

The Bettis review includes a curve for Vickers hardness of UO₂ as a function of temperature (Figure 7-23). Corrections for the effect of spherical, closed porosity on the elastic modulus were found to be as follows:

<u>Young's modulus</u> <u>E_R, kbars</u>	<u>Shear modulus</u> <u>G_R, kbars</u>	<u>Bulk modulus</u> <u>K_R, kbars</u>	<u>Poisson's</u> <u>ratio</u>
2305	875	2127	0.316

The approximate porosity correction applicable for spherical, closed porosity is as follows:

$$\text{UO}_2: E = E_R (1 - 3.02 P + 3.57 P^2)$$

where P is the volume fraction of porosity. Although it was found that the shape and distribution of the porosity have a strong effect on the

elastic moduli, no quantitative expressions have been derived to account for deviations from spherical, closed porosity.

The effect of nonstoichiometry of UO_2 on the elastic modulus of the same material is shown in Figure 7-24. Porosity and grain size affect the strength of UO_2 as shown in Table 7-4a, and the modulus of rupture from 25 to 1250 C is compiled in Table 7-4b.

Table 7-4a. Effect of Grain Size and Porosity on the Strength of UO_2

Material	Sinter temp, C	Test type	Test temp, C	$K \times 10^{-3}$	a	b
UO_2	2000	Bend	RT	409.0	0.119	3.17
UO_2	2000	Bend	1000	23.7	0.837	5.84

Note: $S = KG^{-a}e^{-bP}$

where

S = strength, psi

G = avg grain diameter, microns

P = volume fraction porosity

Table 7-4b. Modulus of Rupture for UO_2 ⁷⁹

Test temp, C	Atmosphere	Avg density, gm/cc	Mean load rate, gm/cc	Modulus of rupture, (a) kg/cm ²
25	Air	10.58	68.0	843 (5)
25	Air	10.44	68.0	598 (5)
25	Air	10.07	117.9	433 (4)
25	Air	10.26	680.4	943 (5)
25	Vac	10.64	86.2	1511 (5)
25	Vac	10.59	285.7	1480 (4)
1000	Vac	10.55	63.5	1239 (4)
1000	Vac	10.60	204.1	1306 (4)
1000	Vac	10.51	1115.9	1044 (5)
1250	Vac	10.60	27.2	1759 (5)
1250	Vac	10.62	195.0	1624 (2)
1250	Vac	10.61	548.8	1317 (5)
1250	Vac	9.85	199.6	1234 (5)
1250	Vac	10.23	426.4	792 (4)

(a) Number of specimens is shown in parentheses.

Figure 7-23. Effect of Temperature on the Vickers Hardness of ThO_2 , UO_2 , and $\text{ThO}_2\text{-UO}_2$ Compositions

Figure 7-24. Effect of Nonstoichiometry on the Elastic Modulus of UO_2 ⁷⁹

The steady-state deflection rate for nonstoichiometric UO_2 can be found in Figure 7-15; and creep data from bending experiments are compared in Figure 7-25. Figures 7-26 through 7-30 provide additional information on creep.

Estimates of the cohesive strength of UO_2 are as follows:

Compound	U_o , kcal/mole	a_o , Å	σ_c , kg/mm ²	$o\text{UO}_2/a_X$
UO_2	-2461	5.4704	21,300	1.0

$$\sigma_c = \frac{y_o U_o}{2V_o a}$$

where

σ_c = cohesive strength

U_o = cohesive energy

y_o = distance between nearest neighbors = $3/4 a_o$

V_o = cell volume

a = range of interatomic forces assumed, y_o

Figure 7-25. Steady-State Deflection Rate of Nonstoichiometric UO_2 at 1300 C

Figure 7-26. Effect of Stress on Steady-State Creep Rate of UO_2 , Compression Testing

Figure 7-27. Effect of Temperature on Steady-State Creep Rate of UO_2

Figure 7-28. Effect of Porosity on Creep Rate of UO_2 at 7000 psi and 1400 C

Figure 7-29. Effect of Oxide Additions on Stress Dependence of Creep Rate of UO_2 at 1400 C

Figure 7-30. Comparison of Diffusion Coefficients Determined by Diffusion Experiments With Those Calculated From Creep Data

7.2.4.3. Scope of Work

It has been shown in the review of the state of the art that considerable data are available on thermal properties from preceding work, but little are available on the mechanical properties of UO_2 and mixed oxides. The thermal properties of UO_2 have been determined out-of-pile and in-pile as much as technology has permitted. The chief limitation has occurred at high temperatures where measurement is difficult. Mixed oxides have received much attention, but vipac has been only lightly represented in test work on $(\text{Pu}, \text{U})\text{O}_2$. Therefore, there is a deficiency in data for vipac mixed oxides, particularly at high temperatures. A total of eight highly instrumented capsules will be required for the determination of $\int k dT$ of particulate fuel in the as-fabricated and restructured conditions.

The LMFBR pellet fuel program will be depended upon to furnish supplementary data on the effects of stoichiometry, burnup, and porosity on thermal conductivity. The LMFBR pellet program can also be expected to provide melting point data.

It may be noted that information on the mechanical properties of mixed oxides is nonexistent, although some out-of-pile data are available on UO_2 . This is the point of departure for the LMFBR pellet program.

The B&W vipac program will follow behind the LMFBR pellet program. It is considered impractical to propose a parallel program for the mechanical properties of particulate fuel. Therefore, B&W proposes only to extend the LMFBR pellet work to particulate fuel. The pellet program will emphasize compressive-creep constants and compressive strength under carefully controlled conditions. In-reactor creep tests will be conducted if feasible and warranted.

This task provides for the determination of compressive strength, elastic behavior, thermal expansion, and creep. The first three will be determined on as-irradiated mixed oxides in the initial work. Elastic behavior and creep will be determined in an FFTF loop.

Three instrumented capsules will be used for tests in the elastic range. Six highly instrumented capsules enclosing AECL-type mechanisms will be used for the creep work.

7.2.4.4. Relationship to Current or Planned Programs

Thermal properties and mechanical properties which will be determined in the LMFBR program will contribute a large amount of data that will be useful in the design of particulate fuel.

7.2.4.5. Technical Justification

Thermal properties must be well-known in order to predict the performance of a fuel, and mechanical properties must be well-known if analytical approaches to fuel element performance are to be used.

7.2.4.6. Major Facilities and/or Equipment Required

The FFTF is desired for all of the in-reactor work. The BAWTR hot cell could be used for evaluations.

7.2.4.7. Budget Estimate

Table 7-5 shows the estimated budget and schedule planned for this work.

7.2.5. Burnup, Swelling, and Proof Testing

7.2.5.1. Objective

The objective of this work is for industry and the AEC to accept the use of powdered, mixed oxide fuel. This goal will be achieved by demonstrating the ability of the fuel to provide a minimum of swelling and mechanical interaction with the cladding to ensure the designed burnup lifetime, 140,000 MWd/T peak or 100,000 MWd/T average.

7.2.5.2. Summary of Current Technology

Many fuel elements containing powdered, mixed oxide fuel with about 2% plutonium have been irradiated by PNWL;^{2,14} however, the conditions of irradiation were typical of thermal reactors.

General Electric has irradiated a total of 20 vipac and pellet mixed oxide fuel pins in the EBR-II at burnups of 1450 to 45,600 MWd/MT (U + Pu); the pins were examined after irradiation.

Table 7-5. Estimated Costs and Schedule for R&D on the Measurement of Fuel Material Properties

Task	Overall Schedule and Estimated Cost, \$ × 10 ³											Total ^(a)
	FY: 1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	
	△1			△2						△3	△4	
<u>Fuel Property Measurements</u>												
Thermal properties												1300.0
Creep												2500.0
Compressive strength												100.0
Elasticity and Anelasticity												175.0
Thermal expansion												85.0
Total												4190.0

Milestones

- △1 Vipac material ready for property evaluation.
- △2 Irradiations begin.
- △3 Irradiations complete.
- △4 Data ready for fuel element model use.

(a) Manpower	\$1,890,000
Equipment	800,000
Materials	500,000
Irradiation	1,000,000
	<u>\$4,190,000</u>

An additional 93 specimens, ranging in burnup from 10,900 to 46,300 MWd/MT (U + Pu) are currently being irradiated in EBR-II.

The first six GE pins listed in Table 7-6 were removed from EBR-II in May 1965 after a short-term irradiation.¹ In November 1966 two fuel rods containing vibratory compacted UO_2 -20% PuO_2 were removed from EBR-II after a burnup of 19,500 MWd/MT (U + Pu). In February 1967 the first pins to achieve greater than 36,200 MWd/MT (U + Pu) were removed. This batch of 12 pins included 11 that had reached burnups of 36,200 to 45,600 MWd/MT (U + Pu) and one at 26,000 MWd/MT (U + Pu). It is expected that the first pins to reach 72,400 MWd/MT (U + Pu) burnup will be removed from EBR-II in mid-1968.

The maximum diametral strain observed on 11 of the 12 F2 series fuel pins ranged from 0.40% (0.0010-inch ∇D); the twelfth was for a fuel pin with an axially restrained fuel column. Excluding the axially restrained specimen and the lower-burnup (3.6 at.%) specimen, the maximum diametral strains all fall in the range of 0.40 to 0.92%. The corresponding fuel pin gross volumetric swelling rates ($\text{vol \%}/10^{20} \text{ f-cm}^3$) are well below the minimum rate observed for the basic swelling capsules; this indicates that fuel swelling is accommodated to a great extent in some manner other than diametral cladding strain. The nearly double strain exhibited by the axially restrained specimen tends to confirm the significance of axial expansion as a mechanism for accommodating fuel swelling.

The UKAEA irradiated 14 fuel pins in the Dounreay Fast Reactor (DFR). The pins were generally 0.200 inch in ID and had 0.015-inch, type 316 stainless steel cladding, but a few pins were 0.275 inch in ID with 0.025-inch cladding. Both vibratory compacted and annular pellet ($\text{Pu}_{0.15}\text{U}_{0.85}$) O_2 fuel pins of about 80% smeared density were irradiated to mean burnups in excess of 33,100 MWd/MT (U + Pu). The heat rates were 9 to 12 kW/ft except for one pin at 22.9 kW/ft. The cladding surface temperatures ranged from 500 to 800 C. Three pins failed—two at 47,500 and 54,500 MWd/MT (U + Pu); the third pin was intact at 47,500, but failed after reinsertion and before irradiation to 61,000 MWd/MT (U + Pu). This third pin contained annular pellets, whereas the other two contained vibratory compacted fuel.

Table 7-6. Results of Postirradiation Examination of UO_2 -20% PuO_2 Fuel Pins Irradiated in EBR-II

Pin No.	O/M	Fuel Form	Design Parameters			Operating Parameters				Postirradiation Results			Fission Gas Release, %	Remarks		
			Cladding		Gaps, in.		Average Smear Density, %	Maximum Linear Power, kw/ft	Maximum Cladding Temperature, F	Peak Burnup, MWD/MT(U+Pu)	Maximum Diameter, in.	Δ Density, %			Δ Length, in.	
			Type	OD, in.	Thickness, in.	Diametral										Axial
F1A	2.00	Pellet	347 SS	0.250	0.015	0.0039	0.49	93	16.1	1086	1450	<0.001 ^(a)	-0.013 ^(b)	11	Not destructively examined To be reirradiated in TREAT	
F1B	2.00	Pellet	347 SS	0.250	0.015	0.0025	0.48	95	16.4	1091	1450	<0.001 ^(a)	-0.008 ^(b)	5		
F1C	2.03	Pellet	Incoloy-800	0.250	0.015	0.0022	0.002	94	16.8	1097	1450	<0.001 ^(a)	+0.002 ^(b)	7		
F1D	1.98	Pellet	Incoloy-800	0.250	0.015	0.0030	0.48	93	16.2	1088	1450	<0.001 ^(a)	-0.007 ^(b)	8		
F1E	1.98	Pellet	316 SS	0.250	0.015	0.0030	0.50	87	14.8	1066	1450	<0.001 ^(a)	-0.017 ^(b)	26		
F1F	1.99	Vipac	Incoloy-800	0.250	0.015	N.A.	0.31	85	16.4	1091	1450	<0.001	-0.013 ^(b)	28		
SOV-5	> 2.00	Vipac	304 SS	0.296	0.021	N.A.		83.4	17.2	1049	19,500	Not significant	<0.03 ^(c)	+0.54 ^(d)		To be reirradiated in TREAT
SOV-6	> 2.00	Vipac	304 SS	0.296	0.021	N.A.		83.3	18.4	1076	20,900	Not significant	<0.03 ^(c)	+0.54 ^(d)		
F2A ^(e)	2.00	Pellet	347 SS	0.250	0.015	0.0025		94.0	16.1	1140	42,000	0.0014	+0.033 ^(b)	49		
F2B	2.00	Pellet	316 SS	0.250	0.015	0.0026		94.6	16.4	1154	43,500	0.0017	+0.026 ^(b)	48		
F2E	1.98	Pellet	316 SS	0.250	0.015	0.0032		94.4	16.7	1161	43,500	0.0021	+0.022 ^(b)	50		
F2F	1.98	Pellet	Incoloy-800	0.250	0.015	0.0030		94.8	16.8	1165	44,000				Failed Local central void	
F2N	2.00	Pellet	347 SS	0.250	0.010	0.0034		92.6	16.6	1160	43,500	0.0022	+0.028 ^(b)	50		
F2P	2.00	Pellet	347 SS	0.250	0.015	0.0030		94.5	9.9	975	26,000	0.0013	+0.029 ^(b)	32		
F2Q	2.00	Pellet	347 SS	0.250	0.015	0.0012		96.7	16.7	1161	43,500	0.0023	+0.032 ^(b)	47		
F2S	2.00	Pellet	347 SS	0.250	0.030	0.0012		96.7	17.5	1184	45,600	0.0043	+0.014 ^(b)	55	Axially restrained fuel column.	
F2U	1.98	Pellet	347 SS	0.250	0.015	0.0034		87.0	16.3	1150	42,600	0.001	+0.017 ^(b)	63		
F2W	1.99	Vipac	347 SS	0.250	0.015	N.A.		83.8	14.1	1092	36,900	0.0011	+0.013 ^(b)	58		
F2Y	1.99	Vipac	316 SS	0.250	0.015	N.A.		83.9	13.8	1084	36,200	0.0010	+0.006 ^(b)	56		
F2Z	1.99	Pellet	316 SS	0.250	0.015	0.0029		87.2	16.1	1140	42,000	0.0016	+0.011 ^(b)	58	Central void	

(a) Exhibited ridging (<0.0002 inch) at some pellet interfaces.

(b) For fuel pin about 35.5 inches long.

(c) For jacketed fuel column.

(d) For fuel column about 11.5 inches long.

(e) Fuel diameter of the F2 series was 0.218 inch and fuel length was 14.2 inches. The fuel for the F2 series contained approximately 94% enriched uranium except pin No. F2P, which contained 40% enriched uranium.

Diametral changes were unpredictable at low burnups, but at burnups above 30,000 MWd/MT (U + Pu), a definite increase in pin diameter was noted at a rate corresponding to about 0.089% $\nabla D/D$ per 10^{20} f-cm³. Pin failures occurred in three out of four cases when the diametral strain on the cladding exceeded 0.8%. Several pins experienced 0.5 to 0.8% strains without failure. No significant differences were observed between the strains caused by annular pellet and vibratory compacted fuel. Some plasticity of the fuel was indicated by the reduction in the diameter of the central void in the annular pellet fuel at high burnup. Some swelling of the cladding also occurred. The maximum $\nabla D/D$ observed was 1.7%.

The UK irradiated a 77-pin subassembly in the DFR. The fuel was (Pu_{0.15}U_{0.85})O₂, and the cladding was type 316 stainless steel. Both vibratory compacted powder and annular pellets were used. The annular pellets had diametral gaps of 0.005 inch and 0.002 inch. The fuel pins were about 0.230 inch in OD by 21 inches long and contained an 11-inch fuel column. The cladding was about 0.015 inch thick. Examination after irradiation to an average burnup of 44,000 and a peak burnup of 50,500 MWd/MT (U + Pu) indicated that all pins were intact and in good condition. The maximum swelling of the pins with annular fuel and annealed cladding was 0.004 inch or 1.4% of the diameter; the maximum was 0.0015 inch or 0.4% for vibratory compacted fuel in cold-worked cladding. Pin length increases of up to 0.090 inch were recorded. Cladding (mid-wall) temperatures ranged from 300 to 500 C. It was concluded that the larger fuel-to-cladding gaps did not significantly reduce the cladding strain at burnups to 44,000 MWd/MT (U + Pu).

7.2.5.3. Scope of Work

To ensure that particulate fuel will perform as required in the 1000-MWe plant, long-term irradiations in a fast flux at high heat rates (representative of 1000-MWe plants) are necessary. These tests must be well instrumented, and an FFTF loop is expected to provide the necessary conditions. During a 3-1/2-year period, ten capsules will be exposed to burnups of 30,000, 50,000, 70,000, 90,000, 120,000, and 140,000 MWd/T. Extremes of stoichiometry (1.95 to 2.00) will be factored into the tests. The center fuel temperature will be

maintained at 400 to 4500 F. The capsules will be designed to accurately determine the amount of fuel swelling in 80 and 85% dense fuel prepared by coprecipitation to a feed fissile plutonium average enrichment of 11.6%. The effect of changes in stoichiometry during burnup will be evaluated.

7.2.5.4. Technical Justification

These irradiations are necessary to confirm the expected behavior of vipac mixed oxide fuel, so that its use in a power reactor can be justified. Previous irradiations have developed the fuel to the point that well-instrumented fast flux irradiations at high heat flux and high center temperature are now mandatory for satisfactory qualification.

7.2.5.5. Major Facilities and/or Equipment Required

An FFTF loop and a hot cell will be used for these tests.

7.2.5.6. Budget Estimate

Table 7-7 shows the estimated costs and schedule planned for this work.

7.2.6. Pellet Fuel for the Predecessor Plant

7.2.6.1. Objective

Sufficient R&D is required to qualify unvented, mixed oxide, pellet fuel for use in the predecessor plant. The demonstration plant, which precedes the predecessor plant, will use pellets, and the LMFBR Program Plan includes extensive R&D to qualify pellets prior to operation of the demonstration plant. This R&D and the operation of the demonstrator are considered adequate to qualify pellets for predecessor plant use.

7.2.6.2. Summary of Current Technology

Pellet fuel R&D was referenced extensively in other summaries in this section in connection with the properties and behavior of powder fuel. Section 7 of the LMFBR Program Plan will provide additional information.

Table 7-7. Estimated Costs and Schedule for In-Reactor Proof Testing of Fuel Material

Task	Overall schedule and estimated cost, \$ × 10 ³										Total ^(a)
	FY: 1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
<u>In-Reactor Proof Testing</u>				△1					△2		
Design capsules	■										160
Fabricate		■									205
Irradiate				■	■	■	■				--
Hot cell							■	■			500
Report									■		70
Total											935

Milestones

- △1 Begin irradiations in FFTF loop.
 △2 Complete irradiations in FFTF loop.

(a) Manpower	\$635,000
Equipment	--
Materials	100,000
Irradiation	<u>unknown</u>
	\$735,000

The R&D required on pellet fuel for the predecessor plant are identical to the requirements for the LMFBR demonstration plant. The R&D work included herein is that expected to be necessary for the demonstration plant. B&W will have further evidence available for the predecessor plant, since it precedes the demonstration plant by eight years.

7.2.6.3. Budget Estimate

Table 7-8 shows the estimated budget and the planned schedule for this work.

7.3. Cladding Material

7.3.1. General

Stainless steels, nickel alloys, and vanadium alloys have shown the greatest promise for LMFBR service. Considerable R&D have been expended on them; however, no alloy has yet demonstrated the ability to survive the high burnups required in a commercial LMFBR.

Excluding corrosion, which is covered in the Sodium Technology Program, failure can be expected to result from one or a combination of the following causes:

1. Fission gas release from the fuel.
2. Fuel swelling.
3. Strain cycling.
4. Swelling of the cladding.
5. Fuel-cladding interaction.
6. Impurities in the fuel.
7. Undetected flaws in fuel element structure.
8. Lack of cladding strength or ductility.

B&W has chosen to eliminate the first cause, except in the case of malfunction, by using a vented fuel element. The use of 85% TD vipac mixed oxide fuel is expected to reduce the effects of fuel swelling. Having virtually eliminated internal gas pressure, a thin, 10-mil cladding appears to be practical; this will reduce thermal stress as well as core metal volume.

Table 7-8. Estimated Costs and Schedule for R&D on Pellet Fuel for the Reference Plant

Task	Overall schedule and estimated cost, \$ × 10 ³													Total
	FY: 1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	
		①		②	③		④							
<u>Pellet Fuel for the Predecessor Plant</u>														
Thermal properties														1,250.0
Fuel swelling and gas release														2,500.0
Fuel-sodium compatibility														1,500.0
Fuel-cladding compatibility														750.0
Mechanical properties														3,500.0
Behavior of Pu, O, and fission products														2,500.0
Total														12,000.0

Milestones

- ① Complete testing to obtain data for FFTF fuel specifications.
- ② Complete testing to obtain design data for FFTF.
- ③ Complete testing to obtain data for first demonstration reactor core fuel specifications.
- ④ Complete testing to obtain additional design data required for first demonstration core.

Strain cycling and fuel and metal swelling remain matters of great concern. These can well be the limiting factors in LMFBR design. Cladding ductility plays an important role in failures by these mechanisms. This is of particular importance, since irradiations to date indicate that almost total loss of elongation may occur after irradiation in fast reactors. There is hope that vanadium is not affected in this way by irradiation.

The loss of high-temperature ductility is believed to be associated with intergranular fracture that results from the accumulation of helium at the grain boundaries. This helium is produced by the $^{10}\text{B}(n, \alpha)^7\text{Li}$ and (n, α) reactions with the constituents of the alloys. Thermal neutrons contribute most of the helium from these reactions, whereas fast neutrons in the 3 to 4 MeV range generate the helium from the (n, α) reactions with the constituents of the alloy.

Most iron alloys may contain as much as 20 to 30 ppm boron, which generally segregates at the grain boundaries; during irradiation this is the source of the helium that accumulates at the grain boundaries. The (n, α) reaction with the other constituents produces helium throughout the structure. This helium can and does migrate to the grain boundaries or other nucleating sites within the grains if the temperature is high enough. The small bubbles of helium coalesce and grow at the grain boundaries and, under an applied stress, cause premature failure of the material.

Few irradiations have been conducted under conditions that are likely to reveal the effects of mixed oxide and/or cladding swelling in an LMFBR. Reports to date indicate that metal swelling is sensitive to temperature in a way that is not amenable to correction by core design. These reports also indicate that all of the materials that have been tested are subject to swelling.

7.3.2. Optimum Chemistry and Heat Treatment

7.3.2.1. Objectives

The ultimate objectives of the cladding development program are (1) to establish specifications for fuel cladding material for B&W's LMFBR, and (2) to provide fuel element design and safety analysis programs with accurate cladding characterization data

so that the resulting fuel element is a highly reliable, safe, and efficient component.

The first requirement is to establish the optimum alloy content, processing procedures, and heat treatment for the cladding in order to obtain adequate strength, ductility, and resistance to detrimental changes during irradiation in an alloy that will resist corrosion in sodium under the design conditions of the 1000-MWe LMFBFR. Mechanical and physical properties will be evaluated so as to ensure that the alloy can carry out the mission prescribed by the Core Design Program.

7.3.2.2. Summary of Current Technology

Tensile Properties — Martin and Weir reported the results of tensile tests on types 316L and 17-4PH steel irradiated at 975 ± 45 F in LAMPRE.⁸⁰ These results, shown in Table 7-9, reveal little change in strength after a fast fluence of 2×10^{21} nvt (> 1 MeV). There was considerable loss in elongation of the type 316L when it was tested at 1475 F. The results of tensile tests on types 304 and 316 from another series of irradiations in LAMPRE (at about 1000 F) to a fast fluence of 1.6×10^{21} nvt⁸¹ are summarized in Table 7-10. At temperatures lower than 932 F, there was essentially no change in the tensile properties of types 304 and 316 stainless steel. At test temperatures of 1300 and 1475 F, there was considerable reduction in total elongation but essentially no change in tensile strengths. Comparing the data in Tables 7-9 and 7-10 reveals that there is little difference in the irradiation behavior of types 304, 316, and 316L stainless steel.

Busboom⁸² conducted a series of irradiations of Incoloy-800, Inconel-625, and Hastelloy-X in the GETR and EBR-II. The results of the tests are summarized in Tables 7-11 through 7-13. These data reveal that after relatively low exposure to fast neutrons, all three materials exhibit very little change in strength, but there is a significant reduction in uniform elongation. In fact, elongations of less than 4% were observed after exposures of 2×10^{21} nvt. Table 7-12 lists the results obtained from specimens exposed in the essentially fast spectrum of EBR-II. The effects are very similar to those obtained in the GETR, where the thermal flux was somewhat greater than the fast flux.

Table 7-9. LAMPRE Irradiations of Types 316L and 17-4PH Stainless Steels⁸⁰

Test temp, F	Fast fluence, nvt	Yield strength, ksi	Elongation, %	
			Uniform	Total
Type 316L				
70	2 × 10 ²¹	34.2	51.7	75.6
	0	33.1	51.0	72.2
932	2 × 10 ²¹	20.5	34.0	44.4
	0	17.8	33.5	44.8
1300	2 × 10 ²¹	18.6	17.7	27.0
	0	15.6	18.3	42.7
1475	2 × 10 ²¹	15.0	4.5	7.3
	0	13.2	14.0	42.3
Type 17-4PH				
70	2 × 10 ²¹	148.0	7.8	12.6
	0	162.0	6.9	10.2
932	2 × 10 ²¹	89.5	3.1	8.0
	0	109.8	2.7	8.2
1300	2 × 10 ²¹	33.5	--	16.2
	0	23.1	4.1	33.4
1475	2 × 10 ²¹	15.2	4.8	37.8
	0	15.9	--	44.0

Table 7-10. Effects of Irradiation in a Fast Reactor on the Strength and Ductility of Types 304 and 316 Stainless Steel⁸¹

Test Temp, F	Yield Strength, ksi	Elongation, %		Fast Fluence, ^(a) nvt
		Uniform	Total	
Type 304				
70	58.5	45.9	63.7	1.6 x 10 ²¹
	65.7	45.5	64.2	0
932	48.7	18.8	24.4	1.6 x 10 ²¹
	42.4	20.6	27.9	0
1300	32.2	11.1	15.1	1.6 x 10 ²¹
	29.3	6.8	28.8	0
1475	22.4	5.7	13.9	1.6 x 10 ²¹
	19.5	0.0	30.7	0
Type 316				
70	66.7	30.5	41.8	1.6 x 10 ²¹
	78.5	35.8	48.7	0
932	39.8	19.6	25.7	1.6 x 10 ²¹
	55.4	22.5	31.5	0
1300	42.5	7.2	11.9	1.6 x 10 ²¹
	42.5	8.2	20.3	0
1475	30.4	3.2	10.2	1.6 x 10 ²¹
	31.1	1.8	35.2	0

(a) Irradiation in LAMPRE at 975 ± 45 F.

Table 7-11. Comparison of 1300 F Tensile Properties of Inconel-625 and Incoloy-800

Material	Fast Fluence, nvt	Yield Strength, ksi	Ultimate Tensile Strength, ksi	Elongation, %	
				Uniform	Total
Incoloy 800	3.5×10^{20} 0	37	51	6	6
		36	54	11	46
Inconel 625	3.5×10^{20} 0	75	90	4	4
		87	106	8	20
Inconel 625	2×10^{21} 0	80	88	1	1
		89	110	7	23
Inconel 800	2×10^{21} 0	34	46	3	3
		41	60	10	49

Table 7-12. Comparison of 1300 F Tensile Properties of Inconel-625 Irradiated in EBR-II and GETR

Place of Irradiation	Fast Fluence, nvt	Yield Strength, ksi	Ultimate Tensile Strength, ksi	Elongation, %	
				Uniform	Total
GETR ^(a)	3.5×10^{20} 0	75	90	4	4
		87	106	8	20
	2.0×10^{21} 0	80	88	1	1
		89	110	7	23
EBR-II ^(b)	3.3×10^{20} 0	73.2	92.5	5.2	5.8
		48.1	96.4	24.3	36.0

(a) Irradiated at 1300 F in inert atmosphere.

(b) Irradiated at 1000 F in inert atmosphere.

Table 7-13. Comparison of 1100 F Tensile Properties of
Inconel-625 and Hastelloy-X

<u>Material</u>	<u>Place of Irradiation</u>	<u>Fluence, nvt</u>	<u>Yield Strength, ksi</u>	<u>Ultimate Tensile Strength, ksi</u>	<u>Elongation, %</u>	
					<u>Uniform</u>	<u>Total</u>
Inconel 625	GETR	3.5×10^{20} 0	74	122	8	8
			91	146	22	23
	GETR	2.0×10^{21} 0	92	114	3	3
			90	146	19	20
	EBR-II	3.3×10^{20} 0	75.8	124.5	13.8	14.1
			51.5	119.0	47.9	49.5
Hastelloy X	GETR	2.0×10^{21} 0	59	93	4	5
			56	111	20	28
	EBR-II	3.3×10^{20} 0	52.3	102.9	14.4	14.7
			26.5	76.0	54.5	57.3

In another program,⁸³ Incoloy-800 and Inconel-625 were irradiated at 1300 F in the GETR to a peak fast fluence of 1.6×10^{21} . The fast-to-thermal ratio in these tests was about 2. Control specimens were aged at 1000 and 1300 F for time periods similar to those for the irradiated specimens. The results of the tensile tests on the Incoloy-800 specimens are summarized in Table 7-14. The irradiation resulted in decreases in both strength and elongation; the total elongation of the irradiated specimens was as low as 1.3%. In the burst tests conducted at 1100, 1300, and 1500 F, the total elongation of both Incoloy-800 and Inconel-625 was less than 1.0% after fast fluence ranging from 1.1 to 1.6×10^{21} . The decrease in strength of both materials during irradiation was greater than the decrease observed for the thermally aged specimens. Aging of the Incoloy-800 had essentially no effect on the properties of the material. Aging of the Inconel-625, however, resulted in an increase in strength with a corresponding decrease in total elongation. In some instances, the total elongation of the aged specimens was less than 1%. Annealed type 304 stainless steel specimens were irradiated at 1000 F in the EBR-II to an exposure of 1.7×10^{22} nvt.⁸⁴ These irradiations resulted in an increase in yield strength but essentially no change in ultimate strength. Uniform and total elongation were reduced at all test temperatures, but the reductions were more pronounced at 1200 to 1500 F. The results are shown in Figures 7-31 through 7-34.

Type 316 stainless steel was irradiated at about 475 to 650 F in the Dounreay Fast Reactor to a fast fluence of about 3.8×10^{22} nvt ($E > 1$ MeV).⁸⁵ At test temperatures of 1100 F and below, irradiation increased the strength and decreased elongation; this effect was attributed to clusters of point defects, which hardened the lattice. The yield strength was not affected at test temperatures of 1200 F and above, but the tensile strength and ductility were reduced. The loss in ductility was proportional to the fast neutron exposure, as shown in Figure 7-35. The residual ductility in all instances was greater than 10%. Similar results were observed for FV 548 and for type 316 stainless steel incorporating about 1.0% niobium.

Nimonic PE16, a nickel-based alloy which hardens by the precipitation of $\text{Ni}_3(\text{TiAl})$, was also irradiated in the

DFR⁸⁵ at about 575 F to a fluence of 5×10^{22} nvt. The elongation at test temperatures of about 1200 and 1300 F was quite low and approached zero after an exposure of about 4×10^{22} nvt.

At ANL, burst tests were conducted on type 304 stainless steel specimens which had been irradiated to as high as 1.1×10^{22} nvt in EBR-II.⁸⁶ The burst tests conducted at 932 F indicated that the hoop strength increased from about 48,000 to about 90,000 psi. The circumferential elongation decreased from about 16 to 0.2%. Subsequent tests conducted at higher temperatures are summarized in Table 7-15.⁸⁷ The circumferential elongation is plotted as a function of test temperature in Figure 7-36.

Kangilaski and Shober⁸⁸ have reported the results of tensile tests on type 347 stainless steel which had been irradiated at about 1140 F to a fast fluence of 2.1×10^{22} . In the tests conducted at 1385 F, the yield strength increased by about 10,000 psi, but the total elongation decreased from 26 to 0.6%. Postirradiation annealing at about 2450 F had no effect on total elongation, but both the yield and tensile strengths were reduced. In tests conducted at 600 F, the yield strength increased from 27,000 to 86,000 psi, while the total elongation decreased from about 35 to 10%.

Chou and Jones have investigated the effect of low-temperature martensite transformation in type 304 stainless steel as a means of improving ductility after irradiation.¹⁰¹ Significantly higher elongations and reduced strengths were found after exposure to 1.3×10^{20} nvt, more than 0.82 MeV, and 200 F when tested at 1200 and 1380 F. Chou is also investigating cobalt alloys for LMFBR use.¹⁰¹ Haynes-25 has been found to embrittle at 1475 after an exposure to 1.3×10^{20} nvt.

Creep and Stress Rupture—At BNWL, in-pile creep tests were conducted at 932 to 1400 F and at stress levels ranging from 10,000 to 30,000 psi.⁸⁹ These tests indicated that the secondary creep rate was identical to that observed out of pile. The creep life, however, was reduced significantly because of the loss in total elongation.

Table 7-14. Tensile Tests of Aged and Irradiated
Incoloy-800

<u>Exposure Temp, F</u>	<u>Test Temp, F</u>	<u>0.2% Yield Strength, ksi</u>	<u>Ultimate Tensile Strength, ksi</u>	<u>Elongation, %^(a)</u>	
				<u>Uniform</u>	<u>Total</u>
<u>Thermal Control</u>					
1000	1100	45.37	73.23	24.5	26.1
1300	"	37.11	56.86	16.5	18.7
1000	1300	33.44	44.48	9.0	15.9
1300	"	26.96	42.22	12.1	19.8
1000	1500	21.55	24.16	5.3	21.4
1300	"	17.33	19.09	9.3	27.5
<u>Irradiated</u>					
1300	1100	20.74	34.45	4.6	5.3
"	"	22.79	43.54	8.0	8.7
"	1300	19.19	20.54	1.9	2.2
"	"	19.68	20.32	1.1	1.3
"	1500	10.2	10.53	1.0	1.3

(a) 1-inch gage length.

Figure 7-31. Yield Strength of Annealed Type 304 Stainless Steel Before and After Irradiation

Figure 7-32. Ultimate Strength of Annealed Type 304 Stainless Steel Before and After Irradiation

Figure 7-33. Total Elongation of Annealed Type 304 Stainless Steel Before and After Irradiation

Figure 7-34. Uniform Elongation of Annealed Type 304 Stainless Steel Before and After Irradiation

Figure 7-35. Ductility of Type 316 Stainless Steel at 1200 and 1300 F
After Irradiation at 475 to 650 F⁸⁵

Figure 7-36. Effect of Test Temperature on $\nabla D/D$ for EBR-II
Type 304L Stainless Steel Fuel Cladding

Table 7-15. Effect of Irradiation on the Burst Properties of Type 304L Stainless Steel⁸⁷

Test Temp, F	Unirradiated		Irradiated ^(a)	
	Rupture Strength, ksi	$\Delta D/D$, %	Rupture Strength, ksi	$\Delta D/D$, %
1112	42.5	13.8	59.1	0.3
			63.4	0.6
			65.7	0.2
			67.1	1.2
			64.3	1.7
1292	33.8	11.1	35.2	8.8
			40.7	1.3
			42.8	3.6
			41.6	5.0
			37.0	1.5
1472	27.6 29.4	13.5 16.7	23.0	3.9
			21.8	1.7
			23.4	2.2
			24.1	3.3
1652	19.3 19.3	15.8 19.5	14.9	4.6
			14.7	2.2
			13.8	2.9
			14.5	3.2
			14.5	3.6

(a) The maximum fluence was 1.2×10^{22} nvt, and the minimum was 5×10^{21} nvt. The irradiation temperature ranged from 700 to 965 F depending on the position along the length of the tube.

Inpile creep tests using INOR-8 were conducted at 1200 F at ORNL.⁹⁰ These tests indicated that the effect of irradiation on creep rate was stress-sensitive. At high stress levels, irradiation increased the creep rate, but at the lower stresses, it had no effect on the creep rate. Postirradiation creep tests indicated that the time to rupture decreased as the total exposure increased.^{90,91} After an exposure of 5×10^{19} nvt, the time to rupture was a factor of about 10 lower than that for unirradiated material.

The creep-rupture life of irradiated Hastelloy-N was found to be reduced by approximately 20 to 30 at 650 C and by 2 to 3 at 815 C.⁸⁹ The loss in rupture life appeared to be controlled by the maximum exposure temperature, regardless of whether it occurred during irradiation or during subsequent testing. Also, the activation energy for creep in the 650 to 815 C range did not appear to be affected by radiation.

Shively investigated the effects of thermal gradients on the stress rupture of cladding.¹⁰² He found that cyclic heating produces large changes in microstructure, strain profile, strain at failure, and stress-rupture life. A spacer wire helix became welded to the tubing in one case, and the tubing failed beneath the wire in another case. Shively considers ratcheting to be a distinct possibility in LMFBRs.

Postirradiation creep tests were conducted at about 1200 F using material that had been irradiated to about 1.35×10^{22} nvt in the DFR.⁸⁵ These tests indicated not only that the time to rupture was significantly reduced, but that the creep rate was increased by a factor of two to three by irradiation.

Effects of Composition and Heat Treatment—

Several programs are in progress to develop an alloy or fabrication process that will yield a structure not affected by irradiation. Martin and Weir⁹² have demonstrated that the effects of irradiation on type 304 stainless steel strongly depend on the level of cold work and grain size—the finer the grain size, the greater is the resistance to irradiation effects. Bell⁸⁵ also reported that a fine grain structure is affected less by irradiation than is a coarse structure.

Levy and Wheeler have demonstrated that the effects of irradiation on Hastelloy-X-280 can be minimized by a proper

solution heat treatment prior to irradiation.⁹³ Roberts and Harries⁹⁴ believe that the loss of high-temperature ductility of the 20 Cr-25 Ni-Cb alloy can be minimized by achieving (before irradiation) a structure containing a fine precipitate within the grains. Such a structure could conceivably be obtained by cold working and aging after a solution treatment. Martin and Weir have demonstrated that this theory may be correct.⁹² They have added 0.2 wt% titanium to type 304 stainless steel, which is believed to cause the precipitation of titanium boride within the grains. This precipitate acts as a nucleus for the generation and collection of helium and prevents the helium from migrating to the grain boundaries.

Irvin, Bement, and Hoagland⁹⁵ have reported that the effects of irradiation on type 348 stainless steel are less severe than its effects on type 304 stainless steel. This resistance to irradiation might be explained by the stabilization of the carbides by the niobium. Similar results were observed by Bell⁸⁵ when comparing type 316 stainless steel with FV-548. (FV-548 is essentially type 316 stainless steel containing a small amount of niobium.)

Cladding Swelling—Cawthorne and Fulton reported that microscopic voids form in stainless steel after irradiation beyond 2×10^{22} nvt in the Dounreay Fast Reactor.⁹⁶ They found a large increase in void formation as they reached 7.8×10^{22} nvt to 7 vol %. Applied stress, cold work, and temperature appeared to be variables in void formation. To date, investigators in the United States have had to use samples with lower neutron dosages, but they have confirmed the trend in void formation.

7.3.2.3. Scope of Work

Data on the effects of irradiation on the properties of the potential LMFBR cladding materials are limited and often contradictory. General Electric (GE) reports that type 304 stainless steel is more resistant to irradiation than are the more complex stainless steel and nickel-based alloys. However, Murphy has demonstrated that even type 304 stainless steel has essentially no burst ductility after an exposure of about 2×10^{22} nvt.⁸⁶ On the other hand, Bell, Irvin, and

Weir found that a structure containing a fine precipitate within the grains is more resistant to irradiation than is a structure void of precipitates.

For all cladding materials being considered (except vanadium alloys), total elongations of less than 1% have been observed in some specimens upon fracture after irradiation. Some fabrication procedures and alloy modifications reportedly reduce the effects of irradiation. However, the improvements that have been observed are based on very limited irradiation exposure. Irradiations of these improved materials to higher fast flux exposures may demonstrate that there are no significant advantages in the modifications.

The effect of irradiation on the properties of the cladding appears to be a function of many different variables, such as alloy composition, grain size, fabrication history, irradiation temperature, test temperature, flux spectrum, and total fast flux exposure. To produce reliable data, the tests must be conducted under conditions that simulate those encountered in the reactor as closely as possible.

Finding an alloy to function satisfactorily as a cladding is of such great importance that a specific work task is included here to supplement the work that will be performed by the LMFBR program on cladding for the demonstration plant.

The timing of the predecessor plant design will enable B&W to analyze upcoming irradiations to 1×10^{23} nvt and more in order to determine the course of development work needed. It has been possible to tailor alloys to meet the requirements of other irradiation effects, i.e., ORNL's work with titanium-modified stainless steel; therefore, there is reason for hope that some yet unknown metallurgical mechanism may be employed to stop the swelling that has been reported in England and the United States and still obtain sufficient ductility for the functions of a vented fuel element. An effort of 11 years is visualized, using a 16-man, steady effort supported by others for short periods of time.

Weldability and fabricability will be assessed as necessary for such specialized operations as melting and heat treatment. Most irradiations will be performed in EBR-II, but the FFTF will be required for some confirmatory, high-flux irradiations. Thirty

capsules will be used in EBR-II and five in the FFTF. This task will determine the selection of a reference alloy.

7.3.2.4. Relationship to Current or Planned Programs

Programs at Argonne, Battelle, Oak Ridge, General Electric, Brookhaven, Westinghouse, and Atomics International may find a commercial or modified alloy that is capable of LMFBR service. In such a case, the funding for this program may be greatly reduced. Programs of principal interest at four of these laboratories are summarized in the following paragraphs from TID-6506:

Argonne National Laboratory—Austenitic stainless steels, refractory metal alloys, and high-strength nickel-base alloys are being investigated in the ANL program. However, only the nickel-base alloys and stainless steels are currently being considered for use with oxide fuels. Alloys of interest included in the program are type 304 SS, Hastelloy-X, and Inconel-625.

The specimens will be irradiated in both the Fermi and EBR-II reactors at 400, 500, 650, and 750 C, to exposures ranging from 1×10^{21} to 1×10^{23} nvt (>1 MeV). Tensile, creep, stress rupture, and tubular rupture tests will be conducted at 25, 550, 650, and 750 C. Other samples of cladding will be obtained from some of the fuel pins also being irradiated in EBR-II and Fermi.

Battelle Northwest Laboratories—Several separate and distinct material irradiation programs are in progress at BNWL in support of the LMFBR. Each of these programs is summarized below.

1. Program A. A series of capsules is being irradiated at 650 and 750 F in the ETR to determine changes in the long- and short-term mechanical properties of 304, 316, and nickel-base alloys as functions of irradiation and test temperatures, flux rate, total exposure, fabrication variables, and coolant environment.

Tensile, creep, and stress rupture tests are being conducted after irradiation and on out-of-reactor control specimens after exposures under similar conditions except for the neutron fluence. Aging studies in a sodium environment will be conducted out

of reactor to determine the metallurgical stability of candidate alloys. Changes in the structure due to the combined effects of irradiation and environment are being determined by means of electron and optical microscopy studies.

One series of capsules was removed in March 1966 after an exposure of 2×10^{20} nvt, and the second series of capsules was removed in 1967 at an exposure of 1×10^{21} nvt.

2. Program B. This program is being conducted to obtain data on the effects of irradiation on the tensile, creep, and stress rupture properties of austenitic stainless steels and high-nickel alloys. Short- and long-term irradiations will be conducted in both EBR-II and Fermi at irradiation temperatures of 700 to 1200 F. The program will attempt to determine the metallurgical stability and ductility of the alloys as a function of irradiation and test temperatures, fluence, variations in chemical composition, prior fabrication history, and irradiation environment. A major effort will be devoted to AISI 304 and 316 stainless steels. Studies on 19-9DL, 20Cr-25Ni, Inconel-625, Incoloy-800, and Hastelloy-X will be somewhat less comprehensive.

3. Program C. This program is for fuel development, with one purpose being to develop cladding materials that will retain mechanical and chemical integrity throughout long-term neutron exposures in extreme fast reactor environments. Intensive studies of the interactions between fuel and cladding materials under reactor operating conditions will have the goals of optimizing fabrication methods and economics and minimizing causes of failure.

Candidate cladding materials of interest to B&W include AISI 304, 316, and 348 stainless steels and nickel-base alloys such as Inconel-600, Inconel-X-750, Hastelloy-X-280, and Incoloy-800. Irradiations will be performed in the ATR pressurized water loop, the ATR high-temperature gas loop, and eventually the FFTF. Exposures greater than 10^{22} nvt (> 1 MeV) will be obtained in the ATR. Goal exposures for FFTF irradiations will be approximately 10^{23} to 10^{24} nvt.

Tensile, creep, and stress rupture tests will be conducted at elevated temperatures. Out-of-reactor control tests will be performed to determine the contributions of thermal cycling, aging, and environment.

General Electric Company, Advanced Products —

The GE program is designed to determine the performance limits of core materials. The program includes the irradiation of two subassemblies in EBR-II and two in Fermi. The subassemblies will consist of individual capsules containing internally pressurized, tubular stress-rupture specimens and sheet tensile coupons. Postirradiation testing will be performed at 1200 F.

Investigations are being conducted on cladding from irradiated fuel pins to determine the effect of fission fragment bombardment or fuel impurities on the distribution and morphology of the precipitates. Out-of-pile thermal control tests on coupons will be run under conditions simulating those obtained in-reactor. Thus, it will be possible to distinguish purely thermal effects from irradiation effects.

Oak Ridge National Laboratory —Several

theories have been advanced to explain the radiation damage mechanism in stainless steels. ORNL is conducting an extensive program to understand these mechanisms better. Their fast reactor irradiations will be conducted in the EBR-II to determine the effects of alloy treatment and processing variables on mechanical and physical properties. The irradiations are designed to:

1. Determine the effect of preirradiation heat treatment and grain size on mechanical properties.
2. Compare the mechanical properties of various stainless steels irradiated at elevated temperatures.
3. Determine the dose rate dependence on mechanical properties of stainless steels.
4. Determine the effect of preirradiation heat treatment and grain structure on mechanical properties.

A very large number of tensile and creep rupture specimens will be irradiated at ambient coolant temperature and 500, 600, 700, and 800 C to various fast fluences and at various instantaneous fast fluxes. Materials to be included in the program are:

1. Type 304 SS.
2. Type M304 SS, believed to be modified with 0.2% Ti.
3. Type 316 SS.

4. Type M316 SS.
5. Inconel-625.
6. Inconel-625M.
7. Hastelloy-X.

The effects of grain size, carbon content, and heat treatment will be evaluated for the 304, M304, 316, and M316 alloys.

7.3.2.5. Major Facilities and/or Equipment Required

EBR-II and an FFTF loop will be needed.

Melting, testing, and fabrications facilities such as those at B&W's Alliance Research Center also are required. Hot cell facilities like those planned for the B&W Nuclear Development Center will be required.

7.3.3. Cladding-Fuel Chemical Interactions

7.3.3.1. Objective

The chemical interactions of the reference alloy with the fuel must be investigated for normal operating and accident conditions.

7.3.3.2. Summary of Current Technology

Rabin, Darmitzel, and Kendall⁹⁷ reported that a localized inner cladding surface reaction found in short-term irradiations should be evaluated for PuO_2 - UO_2 and stainless steel in order to correlate the effect with burnup and to identify the impurities that may cause it. Westinghouse has conducted out-of-pile compatibility tests on vanadium alloys at 800 C for 1000 hours;⁹⁸ the results are not yet available. F. M. Smith of BNWL¹⁰⁰ has reported on tests between UO_2 -25% PuO_2 ; using theoretical densities up to 95% and heating times up to 1000 hours at 650 to 1000 C, he found no evidence of incompatibility whatsoever with helium or sodium bonded materials.

7.3.3.3. Scope of Work

The LMFBR program is expected to provide the test work on chemical compatibility required to qualify a reference cladding alloy for use in the B&W LMFBR; therefore, the funding is in agreement with Program 7 of the LMFBR Program Plan.

7.3.3.4. Relationship to Current or Planned Programs

General Electric continues to investigate compatibility in their Fast Ceramic Reactor Program. Future input from their work may reduce the budget requirements for this task.

7.3.4. Dimensional Tolerances and Thickness Allowances

7.3.4.1. Objective

If a 10-mil cladding is to be used, then permissible tolerances and allowances must be accurately evaluated.

7.3.4.2. Summary of Current Technology

Little effort has been reported on this subject. GE has sought to use welded tubing in its Superheat Program in order to obtain the best surface condition and tolerances; they report that a number of failures occurred in the welds during irradiation.⁹⁹

7.3.4.3. Scope of Work

Seamless and welded tubing of the reference alloy will be evaluated for the tolerances that can be achieved for LMFBR service. Allowances will be established for unavoidable flaws, lack of concentricity, and the like.

Input from the sodium technology area is required to establish corrosion and mass transfer allowances under heat transfer conditions.

We visualize a steady four-man effort over 6-1/2 years; supplementary workers will be needed from time to time for fabrication and test work. The work will start approximately three years after task 7.2.2.2, so that the effort will be expended mainly on the best candidates for the reference alloy.

7.3.4.4. Relationship to Current or Planned Programs

The 10-mil thickness of B&W cladding is expected to make it unique among LMFBR claddings.

7.3.5. Irradiation Effects on the Reference Alloy

7.3.5.1. Objectives

Mechanical properties up to 1300 F for normal operation and approaching the melting point for transients are needed for the reference alloy. We must also determine how the subsequent performance of the alloy (as cladding) is affected by structural changes caused by transients.

7.3.5.2. Summary of Current Technology

Applicable work has been recorded under task 7.2.2.2.

7.3.5.3. Scope of Work

The potential of the reference alloy to resist swelling and retain ductility under long-term irradiation will have been determined in work phase 7.2.2.2. It will then be necessary to obtain sufficient statistical data to establish design stresses.

Numerous mechanical tests must be performed on the reference cladding during and after fast neutron irradiations. Yield strength, ductility, and burst strength can be evaluated at room and LMFBR operating temperatures on specimens after irradiation in the EBR-II and FFTF. Additional tests will be made at higher temperatures (1400 to 2500 F) to provide data to the Safety Program for accident analyses.

Creep is best evaluated in-reactor, because structural changes in the alloy are certain to affect the course of a dynamic phenomenon such as creep. Tests will be conducted in BNWL- or AECL-type creep capsule assemblies. Three capsules for each of three temperatures (1000, 1150, and 1300 F) will be irradiated in the FFTF. Two additional capsules will be irradiated at 1300 F to 5×10^{22} and 3×10^{23} nvt to provide samples for subsequent out-of-pile creep and creep rupture tests at 1500, 1800, and 2100 F. All capsules will be well-instrumented. Two capsules will be used for cycling tests.

7.3.5.4. Relationship to Current or Planned Programs

ANL, BNWL, ORNL, GE, and Westinghouse are expected to participate in cladding irradiations over the next ten years. The results of their work will influence the design of this task.

7.3.5.5. Major Facilities and/or Equipment Required

One or two FFTF loops will be required for this work. Out-of-reactor hot cell facilities with tensile, creep, burst, and stress rupture capabilities will be needed. It is expected that adequate facilities will exist by the time they are needed.

7.3.6. Predecessor Plant Cladding Development

7.3.6.1. Objective

It is hoped that the reference alloy developed in the preceding tasks will be sufficiently proven in time for use in the predecessor plant. Since this is not certain, the LMFBR development of standard alloys for the demonstration plant will be relied upon to determine the proper cladding for the driver fuel in the predecessor plant.

7.3.6.2. Summary of Current Technology

See preceding tasks and Volume 7 of the LMFBR Program Plan.

7.3.6.3. Budget Estimate

The budget and schedule are identical with Volume 7 of the LMFBR Program Plan. Table 7-16 shows the estimated costs and planned schedule for R&D work on cladding materials.

7.4. Fuel Elements

7.4.1. General

To be economically competitive, $\text{UO}_2\text{-PuO}_2$ fuel must be burned at a high heat transfer rate to a burnup of approximately 140,000 MWd/T at the peak burnup location in a 1000-MWe LMFBR. It has been noted that the fuel has been proven capable of producing power to such burnups in "gun barrel" thermal irradiations, but it has not been proven

Table 7-16. Estimated Costs and Schedule for R&D on Cladding

Task	Overall schedule and estimated cost, \$ × 10 ³																Total ^(a)
	CY: 1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983		
Cladding				①		②	③						⑤				
Optimum chemistry, fabrication, and heat treatment							④										
Cladding-fuel chemical interactions																	
Dimensional tolerances and thickness allowances																	
Effects of irradiation on reference alloy																	
Predecessor plant cladding development																	
Total																	

Milestones

- ① Start irradiations in EBR-II.
- ② Complete irradiations in EBR-II
- ③ Selection of reference alloy.
- ④ Start irradiations in FFTF loop.
- ⑤ Complete irradiations in FFTF loop.

(a) Manpower	\$11,750,000
Equipment	2,350,000
Materials	4,700,000
Irradiation	unknown
	<u>\$18,800,000</u>

(b) Irradiation charges in the FFTF are omitted because they are unpredictable at this time.

than any proposed cladding alloy can contain the fuel satisfactorily under the 1000-MWe design conditions.

Whereas the cladding work task will determine to a degree the ability of a selected alloy to meet the design requirements, this task seeks to reduce the stresses imposed on the cladding, so that it will have the best chance of achieving its mission in the reactor. The stresses are reduced by venting the pin to the coolant through a porous plug. The design must provide for a holdup time, so that radioactivity is reduced substantially before the fission gases are released into the sodium coolant. The design of the porous plug becomes a key objective of the R&D program.

Neither vipac fuel nor the vented element is scheduled for use as driver fuel in the predecessor plant or in the LMFBR designs of other contractors. They must be developed in time for use in B&W's 1000-MWe LMFBR; therefore, development of the vented pin with vipac fuel becomes an assignment with special significance to this task.

Other features of the pin design to be considered in this task are fuel separators and retainers.

The complete fuel pin must be subjected to sufficient out-of-reactor tests before proceeding to in-reactor tests. Irradiations under steady-state and transient conditions are necessary to prove the durability and safety of the design.

The driver fuel elements for the predecessor plant will be nonvented and loaded with pellet fuel. The development of these elements will be accomplished by R&D listed in the LMFBR Program Plan. Test elements with vipac fuel and vents will be included in the predecessor plant core.

7.4.2. Fuel Element Design and Vent Development

7.4.2.1. Objectives

A fuel element will be designed in accordance with the concept chosen by the core design task. The most important part of this task will be the development of a porous plug, which will serve to vent fission gases to the coolant. Additional effort will be expended on a backup mechanism, the diving bell.

7.4.2.2. Summary of Current Technology

General Electric considered four mechanisms in their initial work on the Fast Ceramic Reactor Development Program: a capillary tube, a pressure relief device, the porous plug, and the diving bell.¹⁰³ They chose the diving bell as their reference device.

No published results of inpile tests of porous-plug end devices are available. Outpile tests have been conducted on porous plugs of nickel, Cr-Ni 18/8, Cr-Ni 18/10, and iron. Sodium leakage across plugs with pore sizes between 1 and 5 microns occurred at pressures as low as 0.5 atm (at 600 C).

GE has obtained encouraging results with the diving bell in thermal reactor irradiations to 17,000, 47,500, 8700, and 6700 MWd/T, but the AEC recently stopped their vented fuel element program.

7.4.2.3. Scope of Work

Vent Material Evaluation—A material screening program will be conducted to evaluate promising porous plug candidates. The objective is to select several of the most promising materials and to obtain test data where the desired information is lacking. The required information includes mechanical properties, sodium compatibility, and the effects of fast fluence and physical configuration on continuous operation.

Test specimen sizes will conform to standard test specifications where applicable. Most of the tests will be performed on unirradiated specimens. Some specimens will be subjected to fast flux irradiation and to combined irradiation and sodium immersion.

The effects of various amounts of porosity and types of porosity on mechanical properties will be determined. The range of porosity to be investigated has not been decided upon. Data requirements are itemized below.

1. Tensile, compressive, and shear strengths.
2. Modulus of elasticity, modulus of compression, Poisson's ratio.
3. Coefficient of thermal expansion, heat capacity, thermal conductivity.

4. Low-cycle fatigue life.
5. High-temperature creep rates (temperatures from 950 to 1500 F) and stress to rupture.
6. Effects of irradiation on the important mechanical properties.
7. Chemical compatibility with sodium (corrosion rate as a function of time, sodium temperature, sodium impurities, and mass transport effects).

Vent Design, Fabrication, and Test—We must design, develop, and test a device to provide a holdup time before the fission gas is released to the gas plenum, so that extremely radioactive gas is not released to the coolant. The holdup period is now set at approximately five days to provide for radioactive decay of xenon. The holdup device must not only provide a hold time, but it must allow a continuous release of gas so that pressure is not built up within the fuel inside the cladding tube. It is expected that some sort of a semi-permeable plug would provide the necessary holdup time while allowing the gas to pass through to the gas plenum chamber continuously.

The release of fission gas from the interior of the cladding becomes imperative at high fuel burnups because of the possibility of pressure-induced swelling. There are many designs to do this, most of which operate on the basis of a pressure differential (basically, the gas is released in "burps") and do not provide for selective release of aged gas. Because there are so many of these designs proposed, each fulfilling specific needs, the need for a thorough engineering evaluation is apparent. The specific conditions of the reference LMFBR design may negate some of the designs or suggest modifications to others. In any event, a specific design for such a release mechanism must be selected by careful evaluation.

An additional requirement is that only aged fission gas be available for release to the coolant to maintain the lowest level of radioactivity possible in the coolant. It appears that a five-day decay period is optimum in that it provides for the decay of some of the short half-life xenon daughter products, yet is not so long that the design problems become extreme. No known work is being done to develop such a device. It is anticipated that the design and development of a low-permeability membrane should lead to the desired holdup time and release

rate. If this is not successful, one of the other concepts will be pursued; the diving bell has been tentatively selected as the alternate.

The test to determine pressure flow (or diffusion rate) characteristics for the passage of fission gases will be carried out in an unirradiated but high-temperature environment. The flow test functional parameters to be tested include plug material porosity and plug and gas temperatures. Temperatures will range from 950 to 1500 F, and test flow rates will range up to 430×10^6 cc/sec-cm² of plug cross-sectional area.

To enable the vent device to function reliably, consideration must be given to potential mechanisms of porous plug clogging. A filtering device may be necessary to prevent the deposit of both halides and alkali metals in the pores and the passage out of the fuel element of long-half-life isotopes. Tests will be conducted on promising designs to find the best one.

The results of the defect testing programs of other contractors will provide input to the vent development program concerning the allowable extent of sodium ingress through the vent device. The design criteria so set will have a great influence on the effort expended in arriving at a qualified vent device. The other major design criterion—holdup time—will be set by the Core Design Program.

Sufficient testing will be performed to guide the development and substantiation of a model for the inherent fuel element (fuel properties, fuel geometry, and cladding geometry) holdup time characteristics. The parameters of interest include fuel smeared density, linear power rating, active fuel length, axial blanket lengths, fuel outside diameter, and cladding inside diameter. Since the porous plug characteristics are likely to have a strong feedback on the pin holdup time, the effects of internal hydrostatic pressure will be investigated.

The design of the diving bell device (the alternate concept) to provide sufficient holdup time for fission gases is expected to present no insuperable problem.

Fuel Element Design and Fabrication Development—The vibratory compaction method of filling the pin with core and

blanket fuel necessitates consideration of the method by which core and blanket are separated. A separation technique utilizing controlled filling or separators will be developed.

A method for installing the porous plug in the fuel element must be devised, and the necessary length of fuel tube extension must be determined. A spring or other device for retaining the fuel will be needed, unless the vent device performs this function. Finally, a fuel element closure method must be selected.

Evaluation of the effects of fabrication variables on core performance will help to dictate the tolerances to be specified for the 1000-MWe LMFBR. The work that has been performed to date is reported in the cost benefit section of this report.

7.4.2.4. Relationship to Current or Planned Programs

The General Electric program for vented fuel element development was recently cancelled by the AEC. To our knowledge, no other work on this subject is proceeding or planned.

7.4.2.5. Major Facilities and/or Equipment Required

Facilities such as those now existing at B&W's Nuclear Development Center at Lynchburg are adequate for the design and fabrication of vents and fuel elements. Standard mechanical and metallurgical test equipment like that at our Alliance Research Center will support the work.

A sodium loop will be necessary for the out-of-pile tests of the venting mechanism. The AEC is expected to have sodium loops in operation by the time the tests are scheduled to start.

7.4.2.6. Budget Estimate

Table 7-17 shows the estimated costs and the planned schedule for R&D work on fuel element design and fabrication.

7.4.3. Fuel Element Testing—Normal Operation

7.4.3.1. Objectives

It will be necessary to prove that the venting device can operate reliably in the fuel element as designed. It must be

able to withstand the forces and structural changes inherent in normal fast reactor operation and continue to perform its function for the proposed fuel element design life.

The transport of fission products must be evaluated to determine the specific effectiveness of venting.

The incidence of sodium logging will be investigated to define its effects on operation and safety.

Transients to the point of local failure of fuel cladding (by melting or other mechanisms) will be investigated to determine their effects on core design.

Table 7-17. Estimated Costs and Schedule for Fuel Element Design and Fabrication Development

Task	Overall schedule and estimated cost, \$ $\times 10^3$						
	CY: 1969	1970	1971	1972	1973	1974	Total ^(a)
Selection of release mechanism	████████						60
Design of holdup mechanism		████████					90
Selection and test of vent materials		████████████████					310
Fuel element design		████████████████████					350
Fuel element (incl. vent) fabrication development				████████			260
Operational out-reactor vent tests			████████████████				300
Total							1370

Milestone: 1 Fuel element ready for in-reactor tests.

(a) Manpower	\$ 1,010,000
Equipment	250,000
Materials	110,000
Total	1,370,000

7.4.3.2. Summary of Current Technology

Fission product transport for two specimens irradiated in Phase 1 of the GE Fast Ceramic Reactor Program was reported as follows:¹⁰⁴

Cesium Leached From End Plenum of
Specimens IX-1-P and IV-3-P

Specimen	Cesium, micrograms				Fission gas released, %	Burnup, MWd/T
	¹³³ Cs	¹³⁴ Cs	¹³⁵ Cs	¹³⁷ Cs		
IV-3-P	80.2	0.013	19.5	32.3	52.4	10,900
IX-1-P	526	1.95	22.4	26.6	?	47,600

From the burnup and amount of fuel in the two specimens, it was calculated that IV-3-P and IX-1-P yielded 1900 and 6900 micrograms of cesium-133, respectively. Thus 4.2% and 7.6% of the cesium-133 generated in IV-3-P and IX-1-P, respectively, was collected in the reservoir end of the cladding. Furthermore, small but similar quantities of cesium-135 and cesium-137 were found in the reservoir of the two specimens. Since it has been established that fission gas release to the plenum of IV-3-P was 52.4% of the total amount produced, these data indicate that a substantial fraction of fission gas generated in the fuel region escaped to the plenum of IX-1-P and deposited long-lived cesium daughters. Cesium distillation to the end plenum per se appears unlikely, because the cesium-133:cesium-135 and cesium-137 ratios are quite different in the two specimens and also differ from the fission yields. To date, efforts have been unsuccessful in establishing quantitative relationships between the observed isotopic concentrations of cesium in the end plenums of either specimen, but the concentrations are probably related to the diffusion rates and half-lives of the iodine and xenon precursors.

The top portion of IX-1-P, including 0.43 inch of fuel and 0.50 inch (all) of the insulator pellet region, was cross-sectioned longitudinally and sampled. The specimen was mounted horizontally and the top half of the cylindrical specimen was ground off with a metal-bonded, diamond-abrasive wheel. Samples were collected in 0.20-inch increments along the axis. Nine of the samples were collected for initial analysis.

Figure 7-37. Axial Fission Product Distribution in Specimen IX-1-P

Figure 7-37 shows the results of the analysis of the first nine samples for strontium-90, zirconium-95, cesium-134, cesium-137, and cerium-144. There was no appreciable transport of zirconium or cerium across the fuel insulator boundary. However, slight transport of strontium did occur, and cesium-137 concentrations were a factor of five greater than corresponding cesium and zirconium concentrations in the first sample, located completely in the insulator pellet. Furthermore, cesium-134 concentrations (formed from an n, γ reaction with cesium-133) exhibited gross transport into the insulator pellet region and a decided concentration buildup at the interface between the two insulator pellets. Additional samples will be analyzed to establish more precisely the profile of the axial fission product transport in IX-1-P. These data indicate that long irradiations do cause buildup of specific fission products, such as cesium-133 in this case, in selective regions of the fuel and insulator pellets. Additional testing is necessary to establish the distribution and redistribution of fission products under operating conditions and temperature profiles that simulate those expected in the FCR reference design more closely.

The following significant observations were made in the course of the metallographic examination of specimen IX-1-P:

1. Examination of the insulator pellet and the pellet-to-clad gap in the insulator zone showed no evidence of a fission product plug. In the present (cold) condition, a continuous gap from the fuel zone through the insulator zone was observed, so that there was apparently a path for fission gas to reach the end plenum.
2. The central void extended to within 0.020 inch of the insulator pellet and was enlarged to about a 0.045-inch diameter near this pellet. This enlarged void is consistent with the burnup "peaking" observed on the gamma scans.

Analyses taken along the length of specimen IX-1-P for strontium-90, zirconium-95, cesium-124, and cerium-144 are shown in Figure 7-37.

Sodium logging studies are reported in GEAP-4283.¹⁹ Test conditions included purposely thinned cladding, "transient" irradiation to 125% of peak FCR power conditions at over 200 times the expected startup rate, and the incorporation of "captured" sodium, which was purposely exposed to the maximum temperatures expected in

GE's design. The tests were conducted in thermal test reactors. The lack of fuel damage in the test indicates that oxide fuel power levels that result in values for $\int k d\theta$ of 90 to 100 W/cm may be realized without fear of damaging effects from sodium logging. GE concluded that the central temperature of mixed PuO_2 - UO_2 fuels may be brought up to or slightly above the melting point without fear that the accidental ingress of sodium will produce damaging sodium vapor pressures.

Further, it was found that the presence of sodium in the fuel rod increased the thermal conductivity and, consequently, decreased the center temperature of the fuel.

Power cycling produces deformation of the cladding. After 100 cycles from zero to full power, the 6-mil cladding was distorted up to 8 mils. This was considered acceptable from safety considerations, since fuel rod spacing for their design was about 100 mils.

7.4.3.3. Scope of Work

Steady-state fast flux irradiation tests on fuel pins are required to determine the maximum allowable linear power rating, the fission gas release rate, and the attainable burnup. Production-grade fuel pin assemblies will be selected, thoroughly inspected, characterized, and then tested in the sodium environment at elevated temperatures.

Each test specimen will be a single fuel element which is identical in every respect to the reference core element, except that it may be altered to accommodate fission gas collection and measurement. Fission gas collection must be provided for, so that the gas passes through the porous plug to simulate the actual internal fuel region hydrostatic pressure. Each fuel pin must be thoroughly inspected and its character documented before and after testing.

The test environment will include a fast neutron flux and sodium coolant with a temperature profile (axial) as similar as is practicable to the core profile.

The parameters to be tested include linear power, fission gas release rate, and burnup. The linear power must be varied, depending on the test facility axial flux profile, to give a range

of maximum linear power from 12 to 20 kW/ft. Various pins will be run from 10,000 to 100,000 MWd/T.

The required test information includes axial power profiles, cladding surface temperature axial profiles, gas release as a function of power level and burnup, and limiting burnup as determined by cladding failure or fuel melting. Also, cladding strain measurements corresponding to different power levels and burnups will be required.

The tests must be well-instrumented so that the performance of the fuel and cladding can be checked against the expected behavior determined in the two preceding tasks. A sufficient number of tests will be conducted so that the materials information obtained will be valid for use in fuel element model development.

Tests of the vented design must be conducted in a thermal reactor before scarce fast reactor facilities are committed to the purpose. Eight capsules will be irradiated in a reactor like the BAWTR over a period of two years.

After satisfactory completion of thermal irradiations, testing will commence in the FFTF. It is anticipated that two 19-element clusters will be needed to demonstrate the suitability of the vent design. The testing will include thermal cycling periods. Five years will be needed to complete irradiation and hot cell testing.

7.4.3.4. Relationship to Current or Planned Programs

It is expected that there will be competition for the FFTF loops, but the specific schedule conflicts cannot be predicted at this time.

7.4.3.5. Major Equipment and/or Facilities Required

A thermal reactor, such as the BAWTR, and a flowing sodium loop in the FFTF will be needed.

7.4.3.6. Budget Estimate

Table 7-18 is a summary of estimated costs and the planned schedule for R&D work on fuel element testing under normal operating conditions.

Table 7-18. Estimated Costs and Schedule for Fuel Element Testing—Normal Operation

Task	Overall schedule and estimated cost, \$ × 10 ³										Total ^(a)
	CY:	1973	1974	1975	1976	1977	1978	1979	1980	1981	
Fuel Element Testing—Normal Operation					△			△			
Thermal reactor tests											430
Fast reactor steady-state tests											1150 ^(b)
Fast reactor cycling tests											635
Total											2215

Milestones

- △ Start tests in FFTF.
 △ Complete tests in FFTF.

(a)	Manpower	\$1,365,000
	Equipment	460,000
	Materials	190,000
	Irradiation	200,000
		<u>\$2,215,000</u>

(b) Does not include irradiation units for FFTF.

7.4.4. Transient Behavior of Fuel and Fuel Pins

7.4.4.1. Objectives

Basic information is required concerning the behavior of oxide fuels during sudden overpower conditions. The amount of fission gas release accompanying fuel melting is of interest. The dynamics of molten fuel require study, as do the effects of molten fuel on cladding integrity. Prototype fuel elements must be proof-tested to determine their transient power capabilities; testing will be conducted up to the point of fuel element failure. Failure propagation is handled by the safety program. Properties for the equation of state for fuel and cladding will be provided by this subtask.

7.4.4.2. Summary of Current Technology

Transient tests on vipac uranium dioxide fuel elements have been conducted at TREAT in a cooperative effort by ANL and BNWL.¹⁰⁵ They found that the fission energy necessary to produce cladding failure is between 260 and 330 cal/g UO_2 as compared with 290 cal/g UO_2 for pellet-filled elements. Vipac did not melt as much cladding as pellets because of its quick expulsion from the fuel element during the transient. Localized bulging of the cladding was observed with vipac at a fission density 10 to 20% below that required to cause cladding failures in pelleted fuel rods.

Transient irradiations of three non-pre-irradiated, axially unrestrained, mixed oxide fuel specimens were performed in the TREAT facility at the National Reactor Testing Station (NRTS) in Idaho.¹⁰⁶ The fuel pin outside diameter was 0.25 inch. The irradiation was performed to study fuel cladding performance during power excursions simulating non-standard fast reactor operating conditions. Interpretation of the results, which are summarized in Table 7-19, has led to the following conclusions:

1. Axially unrestrained fuel can accommodate the approximately 10% ΔV on melting of oxide fuel during a severe transient without cladding deformation.
2. Thermal stress itself apparently does not cause cladding deformation.
3. Fuel-cladding gap conductivities may be higher during a transient than in steady-state conditions.

Table 7-19. Summary of Results From Test Transients¹⁰⁶

Specimen	Reactor power		Sample energy release, Btu	Calculated temperatures, F			Postirradiation condition
	Peak, MW	Integrated, MW/sec		Fuel		Cladding	
				Peak	Volume avg		
C2A	176	155	56 6	4900 ± 500	4090	1530	No cladding deformation; fuel pin bowed; pellet cracking and grain growth; no central void.
C2B	295	204	69 7	5000 ± 500	4900	1800	No cladding deformation; fuel pin bowed; pellets cracked; central void; extensive fuel relocation and densification.
C2C	258	260	94 10	7700 ± 1000	6100	2500	Cladding melted but remaining cladding not deformed radially; fuel pin bowed during calibration transients; extensive fuel movement and melting.

4. Densification and relocation of the fuel is to be expected once a portion of the fuel has passed through the heat of fusion.
5. Incipient failure of the cladding can be defined by the conditions that would cause melting of the cladding:
 - a. Molten fuel coming in contact with the cladding (high fuel surface temperatures), or
 - b. Vapor blanketing of the cladding by boiling of the coolant.
6. A transient which does not cause the fuel to pass through the heat of fusion will not cause fuel or cladding damage.

From the few transient overpower tests conducted to date, some qualitative results have been obtained; they are as follows:

1. No immediate cladding failure occurs if the power transient does not produce fuel melting.
2. Entrapped fission gas is a primary driving force for molten fuel movement.
3. The fission gas released upon fuel melting contributes to fuel element failure.
4. The primary cause of cladding failure during power transients is the volume increase of fuel that occurs with melting.

Available information in this area has been obtained on low to medium burnups or nonvented fuel pins. No known published information is available on the effects of transients on vent performance or the effects of a vent on the transient performance of fuel and cladding. Also, no known results are available concerning test results or analytical models of the mechanical interaction of fuel and cladding that accompanies reactor scrams.

Robbins reviewed the data available for application to the equation of state of UO_2 .¹⁰⁹ Hern has obtained improvements in enthalpy measurements for UO_2 .¹¹⁰ Rand reports that compositions of UO_2 and PuO_2 -X behave like ideal mixtures.¹¹¹ Vapor pressures for metallic elements have been recorded in the Metals Handbook. Additional data on the thermodynamic properties of metals of interest are believed to exist.

7.4.4.3. Scope of Work

Fuel pins containing powder fuel—80 to 88% TD with previous steady-state irradiations of 10,000, 50,000, and 100,000 MWd/T—will be tested. Overpower will range from that which causes incipient melting to that which causes enough melting to result in immediate cladding failure. Fission gas pressure and fuel movement will be measured.

Prototypal fuel elements will be subjected to transients simulating the credible accident conditions prescribed by the Core Design Program on the transient behavior of fuel pins. The pins that survive these tests as well as the less-stringent tests will be re-irradiated to determine the effects of transient overpower on the useful incore lifetime. Also, outpile thermal shock tests will be conducted to proof-test the integrity of the fuel pin under simulated scram conditions. Tests will be conducted with the spiral ribbon spacer so that interactions between the cladding and the spacer can be observed. Test conditions will be as specified in the Core Design Program.

Data for equations of state will be sought by this subtask for use by the Safety Program in its analyses of the propagation of failure of fuel elements.

7.4.4.4. Relationship to Current or Planned Programs

The data for the equations of state will be needed by all LMFBR contractors; therefore, it is anticipated that a national laboratory will sponsor the work.

7.4.4.5. Major Facilities and/or Equipment Required

TREAT and a loop in the FFTF will be needed. An out-of-reactor loop for thermal shock testing is expected to be available within the mechanical testing facilities established for Core Design tests of the fuel bundle. Hot cell facilities should be available when needed.

Table 7-20. Estimated Costs and Schedule for R&D on the Transient Behavior of Fuel Elements

	<u>Overall schedule and estimated cost, \$'× 10³</u>												
<u>Task</u>	CY:	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	<u>1978</u>	<u>1979</u>	<u>1980</u>	<u>1981</u>	<u>1982</u>	<u>Total</u> ^(a)
<u>Transient Behavior of Fuel Elements</u>													
TREAT tests													360 ^(b)
Measurement of fuel and cladding properties for equation of state													800
Total													1160

(a) Manpower \$ 815,000
 Equipment 220,000
 Materials 125,000
\$1,160,000

(b) Irradiation charges unknown.

7.4.4.6. Budget Estimate

Table 7-20 is a summary of estimated costs and the planned schedule for R&D work on the transient behavior of fuel pins.

7.4.5. Behavior of Defective Fuel Elements

7.4.5.1. Objective

This program is supplementary to the steady-state and transient fuel element tests in that it allows specific provision for testing (steady-state and transient) defective elements. Such elements must be investigated to determine their effect on operational objectives as specified by the Core Design Program.

A defect is defined here as any fuel element characteristic that allows sodium ingress or fuel egress (except that amount of sodium ingress through a properly operating vent mechanism determined to be acceptable). These include defects that occur as a result of fuel element manufacture or fuel assembly fabrication.

7.4.5.2. Summary of Current Technology

General Electric has investigated a mixed oxide fuel element with an 0.005-inch hole located near the midpoint.¹⁰⁷ A plug of solid material formed in the hole (defect), and a 4- to 6-mil increase in diameter was recorded.⁵⁹ Another mixed oxide fuel element—with a diving bell vent—was irradiated by GE to check fission product distribution.¹⁰⁸ Sodium was inadvertently admitted through the vent during fabrication, and a 5- to 6-mil fuel-sodium reaction zone resulted at the periphery of the fuel. The type 316 stainless steel cladding was found to be in excellent condition after 23,500 MWd/T at 18 kW/ft and 1100 F cladding surface temperature.

An unvented mixed oxide fuel element failed during irradiation, and a 34 vol % increase was found at the point of failure,¹⁰⁸ while it was 7.2% elsewhere. This element had been irradiated to 45,000 MWd/T at 21 kW/ft and 1250 F cladding surface temperature. GE found no evidence of cladding damage after cycling the linear power of defective UO₂ elements from 0 to 26 kW/ft.¹⁰⁷

7.4.5.3. Scope of Work

Incore tests (steady-state and transient) of defective fuel elements will be conducted. The effects of long-term sodium logging on fuel property changes are of interest, as are the induced swelling and fuel-cladding compatibility effects. More basic testing is required to discern the effects of sodium vapor generation. The effects of combined vent plugging and high fission gas release (which can occur upon fuel melting) must be investigated.

This program has a strong influence on the establishment of assembly fabrication techniques. It is anticipated that fabrication will result in more defective elements, however infrequent. During the fabrication development, any pin damage that results from the assembly process will be fully characterized, and selected pins will be irradiated to determine the effects of defects on useful incore lifetime.

Defective elements will be subjected to outpile tests to determine the effects of defects on pin integrity under conditions of thermal cycling and shock and long-term immersion in sodium.

7.4.5.4. Relationship to Current or Planned Programs

Defect tests conducted by other contractors will contribute to the overall experience needed to anticipate the effects of defects on 1000-MWe plant operation and safety. Only defects peculiar to the vented fuel design are included in this task.

7.4.5.5. Major Facilities and/or Equipment Required

A thermal reactor such as the BAWTR or GETR will be needed. Hot cell facilities should be available at several sites by the time they are needed.

7.4.5.6. Budget Estimate

Table 7-21 is a summary of the estimated costs and the planned schedule for R&D work on defective fuel elements.

Table 7-21. Estimated Costs and Schedule for R&D on the Behavior of Defective Fuel Elements

Task		Overall schedule and estimated cost, \$ × 10 ³					Total ^(a)
		CY: 1979	1980	1981	1982	1983	
Irradiations in FFTF loop							500 ^(b)
(a)	Manpower	\$ 375,000					
	Equipment	0					
	Materials	125,000					
	Total	\$ 500,000					

(b) Does not include irradiation units for the FFTF.

7.4.6. Fuel Element Model Development

7.4.6.1. Objective

A model is needed on which to base judgements of the behavior of the fuel elements irradiated in this task and to translate this information into design improvements. Long-range burnup effects on the fuel element must be added and coded through FFTF and predecessor irradiations.

7.4.6.2. Summary of Current Technology

Bettis has developed several computer codes based on model development at their laboratory and at KAPL for thermal reactors; of course, these models are not directly applicable to fast breeder reactors. BNWL is in the process of developing a code for LMFBF use.

7.4.6.3. Scope of Work

Information from the BNWL model work will be the basis for a computer code to account for the behavior of the vented vipac fuel element. Corrections to account for the differences between the fuel element to be used in the FFTF and the vented vipac element will be made in this task.

A continuing task from 1969 to 1983 is contemplated, since information from actual tests will be needed to complete the development of the model. In particular, work to improve long-term burnup codes will extend into 1983.

This task will require special hot cell equipment in order to obtain the precise physical, mechanical, and chemical data necessary for an accurate model. The swelling of fuel and cladding must be determined metallographically and by density determinations. The production of solid and gaseous fission products must be determined analytically. The restructuring and distribution of oxygen, plutonium, and fission products can be evaluated by chemistry, metallography, and autoradiography. Mechanical properties and physical measurements must be evaluated by the multiple methods normal to metallurgical laboratories. The equipment is considered special insofar as many measurements must be both remote and accurate to several decimal places. All of the irradiations and most of the testing will be performed and funded by other tasks and subtasks; however, this task will require tests beyond those accounted for elsewhere.

7.4.6.4. Relationship to Current or Planned Programs

Bettis and BNWL model and computer code programs will contribute heavily to this endeavor.

7.4.6.5. Major Facilities and/or Equipment Required

It is expected that at least one hot cell will be available for this work. An electron microscope, a microprobe, and x-ray diffraction equipment, in addition to more standard items, must be available for use on irradiated samples.

7.4.6.6. Budget Estimate

Table 7-22 is a summary of estimated costs and the planned schedule for R&D work on fuel element model development.

Table 7-22. Estimated Costs and Schedule for Fuel Element Model Development

Task	Overall schedule and estimated cost, \$ × 10 ³																Total ^(a)
	CY:	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
						△1		△2					△3				
<u>Fuel Element Model Development</u>																	
Study existing models for application to vented vipac design																	500
Apply data from other tasks in other LMFBR work to vented vipac design																	700
Develop special features to model for vented vipac design																	750
Total																	1950

Milestones

- △1 Model ready for thermal reactor irradiations.
 △2 Model ready for FFTF irradiations.
 △3 Model established for vented elements to be irradiated in predecessor plant.

(a) Manpower	\$1,550,000
Equipment	--
Materials	400,000
	\$1,950,000

7.4.7. Fuel Element Reliability Tests

7.4.7.1. Objective

Sufficient experience in a fast reactor will be obtained to ensure that vipac, mixed oxide, vented fuel elements will perform adequately in the 1000-MWe LMFBR. FFTF tests described in other subtasks will not provide sufficient statistics for such assurance; therefore, additional tests must be performed in the predecessor plant.

7.4.7.2. Summary of Current Technology

Tests conducted in the EBR-II and Dounreay are the only ones that have contributed positively to the reliability assessment required for fast reactor fuel elements.

General Electric removed the first pin listed in Table 7-23 from EBR-II in May 1965 after a short-term irradiation.¹⁰⁹ In November 1966 two fuel rods containing vibratory compacted UO_2 -20% PuO_2 were removed from EBR-II after a burnup of 19,500 MWd/MT (U + Pu). In February 1967 the first pins to achieve greater than 36,200 MWd/MT (U + Pu) were removed.

ANL is investigating cladding materials and fuel element designs in an irradiation program at EBR-II; the details are listed in Table 7-24. Fully enriched UO_2 with 20% plutonia is used, and cladding surface temperatures vary from 550 to 650 C. The peak fuel temperature is in the range of 2700 to 2900 C, and the linear heat rate varies from 16 to 20 kW/ft.

The UK irradiated 14 fuel pins in the Dounreay Fast Reactor (DFR).¹⁸ The fuel pins were generally 0.200 inch in ID with 0.015-inch, type 316 stainless steel cladding, but a few pins were 0.275 inch in ID with 0.025-inch cladding. Both vibratory compacted and annular pellet ($\text{Pu}_{0.15}\text{U}_{0.85}$) O_2 fuel pins of about 80% smeared density were irradiated to mean burnups in excess of 33,100 MWd/MT (U + Pu). The heat rates were 9 to 12 kW/ft, except for one pin at 22.9 kW/ft, and cladding surface temperatures ranged from 500 to 800 C. Three pins failed—two at 47,500 and 54,500 MWd/MT (U + Pu). The third pin was intact at 47,500 but failed after re-insertion and before irradiation to 61,000 MWd/MT (U + Pu). This pin contained annular pellets, whereas the other two contained vibratory compacted fuel.

Table 7-23. Results of Postirradiation Examination of Vipac
 UO_2 -20% PuO_2 Fuel Pins Irradiated in EBR-II

Pin No. (a)	O/M ratio	Design parameters					
		Cladding			Gap, in.		Avg smear density, %
		Type	OD, in.	Thickness, in.	Diametral	Axial	
F1F	1.99	Incoloy- 800	0.250	0.015	NA ^(b)	0.31	85
SOV5	>2.00	304 SS	0.296	0.021	NA	--	83.4
SOV6	>2.00	304 SS	0.296	0.021	NA	--	83.3
F2W	1.99	347 SS	0.250	0.015	NA	--	83.8
F2Y	1.99	316 SS	0.250	0.015	NA	--	82.9

Pin No. (a)	Operating parameters						
	Max linear power, kW/ft	Max cladding temp, F	Peak burnup, MWd/MT(U+Pu)	Max Δ diam., in.	Δ density, %	Δ length, in.	Fission gas release, %
F1F	16.4	1091	1,450	0.001	--	-0.13 ^(c)	28
SOV5	17.2	1049	19,500	NS ^(d)	<0.03 ^(e)	+0.54 ^(f)	--
SOV6	18.4	1076	20,900	NS	<0.03 ^(e)	+0.54 ^(f)	65
F2W	14.1	1092	36,900	0.0011	--	+0.013 ^(c)	58
F2Y	13.8	1084	36,200	0.0010	--	+0.006 ^(c)	56

(a) F1F not destructively examined; SOV5 to be re-irradiated in TREAT; F2W has central void; F2Y has central void. All specimens are vipac fuel.

(b) NA: not applicable.

(c) For fuel pin about 35.5 inches long.

(d) NS: not significant.

(e) For jacketed fuel column.

(f) For fuel column about 11.5 inches long.

Table 7-24. Details of ANL's UO₂ Fuel Pin Irradiations in EBR-II

Rod No.	Fabrication Synthesis	Method Type	Smeared Density, %	Cladding Material	Cladding OD	Cladding Thickness	Loading Date	Removal Date	Fast Flux	Burnup, MWD/T
SOV-5	Nupac	Vipac	82	304	0.297	0.020	4/1/66	11/17/66	2.4×10^{15}	25,000
SOV-6	"	"	"	"	"	"	"	"	"	"
SOV-1	"	"	80	"	"	"	"	5/6/67	"	50,000
SOV-3	"	"	"	"	"	"	"	"	"	"
SOV-7	"	"	88	"	"	"	"	"	"	"
HOV-4	"	"	80	Hast X	"	0.015	"	"	"	"
HOV-15	"	"	"	"	"	"	"	"	"	"
HOV-8	"	"	88	"	"	"	"	"	"	"
TVOV-1	"	"	80	V20Ti	"	0.020	"	"	"	"
SOS-21	Sol gel	"	"	304 SS	0.290	"	7/1/66	9/10/68	2.2×10^{15}	100,000
SOS-22	"	"	"	"	"	"	"	"	"	"
COS-21	"	"	"	Inconel 625	"	0.015	"	"	"	"
COS-22	"	"	"	"	"	"	"	"	"	"
COS-23	"	"	"	"	"	0.020	"	"	"	"
COS-24	"	"	"	"	"	"	"	"	"	"
HOS-21	"	"	"	Hast X	"	"	"	"	"	"
HOS-22	"	"	"	"	"	"	"	"	"	"
SOV-2	Nupac	"	"	304 SS	"	"	8/1/66	10/10/68	"	"
SOV-4	"	"	"	"	"	"	"	"	"	"
SOV-8	"	"	"	"	"	"	"	"	"	"
HOV-2	"	"	"	Hast X	"	0.015	"	"	"	"
HOV-3	"	"	"	"	"	"	"	"	"	"
HOV-9	"	"	88	"	"	"	"	"	"	"

The UK irradiated a 77-pin subassembly in the DFR.¹¹² The fuel was $(\text{Pu}_{0.15}\text{U}_{0.85})\text{O}_2$, and the cladding was type 316 stainless steel. Both vibratory compacted powder and annular pellets were used. The annular pellets had diametral gaps of 0.005 and 0.002 inch. The fuel pins were about 0.230 inch in OD and had about 0.015-inch cladding. Examination after irradiation to an average burnup of 44,000 and a peak burnup of 50,500 MWd/MT (U + Pu) indicated that all pins were intact and in good condition. The maximum swelling of the pins with annular fuel and annealed cladding was 0.004 inch or 1.4% of the diameter, as compared with 0.0015 inch or 0.4% for vibratory compacted fuel in cold-worked cladding. Increases in pin length of up to 0.090 inch were recorded. Cladding mid-wall temperatures were from 300 to 500 C. The fuel pins were about 21 inches long and contained an 11-inch fuel column.

7.4.7.3. Scope of Work

The number of vipac vented fuel elements which must be irradiated in the predecessor plant will be decided in the future. The design of the core and other objectives must be factored into the decision.

7.4.7.4. Relationship to Current or Planned Programs

It is likely that vipac mixed oxide elements will be tested in the future in EBR-II and the FFTF. These irradiations will assist in establishing the reliability of vipac fuel.

7.4.7.5. Major Facilities and/or Equipment Required

The predecessor plant will be used for irradiations. A hot cell will be required for evaluation of the test elements.

7.4.7.6. Budget Estimate

Table 7-25 is a summary of estimated costs and the planned schedule for reliability tests on fuel elements.

Table 7-25. Estimated Costs and Schedule for
Fuel Element Reliability Tests

Task	Overall schedule and estimated cost, \$ × 10 ³					
	CY: 1979	1980	1981	1982	1983	Total ^(a)
Analysis of reliability of FFTF tests						150
Analysis of reliability of vented vipac fuel elements in predecessor plant						200
Total						350

(a) Manpower	\$ 350,000
Equipment	--
Materials	--
	<u>\$ 350,000</u>

7.4.8. Predecessor Plant Fuel Element Development

7.4.8.1. Objectives

It is necessary to provide sufficient materials, mechanical design, and safety data to permit the design, construction, and operation of the predecessor plant. Since the non-vented, pelleted fuel element selected for the predecessor plant differs from the fuel element for the 1000-MWe plant and, consequently, contributes little to the design needs of the 1000-MWe plant, a minimum expenditure is indicated. This will be accomplished by using design data developed by others pursuing the non-vented, pelleted fuel concept in the LMFBR program.

7.4.8.2. Summary of Current Technology

Background is given in Task Area 7-3 of the LMFBR Program Plan (Engineering Tests of Stainless-Steel-Clad Oxide Fuel Elements).

7.4.8.3. Scope of Work

The scope of work is found in Task Area 7-3 of the LMFBR Program Plan.

7.4.8.4. Relationship to Current or Planned Programs

This work will be accomplished by other contractors to the AEC's LMFBR Program.

7.4.8.5. Budget Estimate

Table 7-26 is a summary of estimated costs and the planned schedule for predecessor plant fuel element development.

7.5. System Materials

7.5.1. General

Many different materials will be required for the construction of the primary system, secondary system, and associated equipment of an LMFBR. Some applications of materials have been tentatively identified, whereas others will not be identified until the detailed design stage is entered. Materials must be selected for vessels, control rod drives, shielding, core structure, piping, insulation, wear and bearing applications, instrumentation, and the like. Frequently, there will be insufficient mechanical and physical property data available.

System materials are expected to last for 30 years in many instances. Data are not available to establish ASME Code criteria by which design work can proceed on pressure-containing items requiring such long life in a sodium environment. The properties of materials must be determined for temperatures higher than the maximum operating temperatures, and accidental excursions of temperature must also be reckoned with.

7.5.2. Primary and Secondary Sodium System Structures and Components

7.5.2.1. Objectives

Material properties must be supplied for those materials which have sufficient resistance to sodium and impurities in sodium to be chosen for LMFBR service so that stress analyses and further materials evaluations can proceed.

Table 7-26. Estimated Cost and Schedule for Predecessor Plant
Fuel Element Development

Task	Overall schedule and estimated cost, \$ ×10 ³														Total
	CY:	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	
					①	②		④		⑥					
						③		⑤							
Predecessor Plant Fuel Element Development															30,500

Milestones

- ① Complete engineering tests to resolve fuel movement and fuel-cladding interactions.
- ② Complete engineering tests under expected FFTF operating conditions.
- ③ Complete engineering tests on defected fuel element behavior and accentuated failure.
- ④ Statistical proof testing on simulated FFTF fuel elements compiled in EBR-II.
- ⑤ Complete additional engineering tests for first demonstration reactor fuel elements.
- ⑥ Complete statistical proof testing for first demonstration plant fuel elements.

Table 7-27. Summary of LMFBR Structural Material Test Program
Carried Out at MSAR¹¹³

Program		Facility
<u>Phase I — Austenitic Steels</u>		
Task 1	Oxygen excursions of 316 and 304 SS at 1200 and 1400 F followed by creep-rupture tests in clean sodium.	High-oxygen sodium (50 ppm) loop and clean sodium loop.
Task 2	Carbon excursion of 316 and 304 SS at 1200 and 1400 F followed by creep-rupture tests in clean sodium.	High-carbon sodium loop and clean sodium loop.
Task 3	Long-term (10,000 and 20,000 hr) exposures of 316 and 304 SS in clean sodium followed by creep-rupture tests in clean sodium.	Static pots and clean sodium loop.
Task 4	Oxygen and carbon excursions on modified austenitic steel followed by creep-rupture tests in clean sodium.	High-oxygen, high-carbon, and clean sodium loops.
Task 5	Synergistic studies of carbon-oxygen exposures followed by creep-rupture tests in clean sodium.	High-carbon, high-oxygen, and clean sodium loops.
<u>Phase II — Ferritic Steels</u>		
Task 1	Exposure in decarburizing environment followed by creep-rupture tests in clean sodium.	Static pots and clean sodium loop.
Task 2	Long-term exposures (10,000 and 20,000 hr) followed by creep-rupture tests in clean sodium.	High-oxygen loop and clean sodium loop.
<u>Phase III — Other Candidate Alloys</u>		
Task 1	Creep-rupture tests in clean sodium.	Clean sodium loop.
Task 2	Oxygen excursion followed by mechanical testing in clean sodium.	High-oxygen loop and clean sodium loop.
Task 3	Carbon excursion followed by mechanical testing in clean sodium.	High-carbon loop and clean sodium loop.
<u>Phase IV — Extensometer development</u>		
Phase V	Development of techniques for measurement of impurities in sodium	

7.5.2.2. Summary of Current Technology

Principal responsibility for development of the properties of alloys in sodium has rested with the Mine Safety Appliances Research Corporation (MSAR).¹¹³ In early testing of types 304 and 316 stainless steel, they minimized such parameters as ΔT , velocity, and material thickness in order to determine environmental effects. Recent work has been carried out with these parameters approaching LMFBR reactor system conditions, and excursions in system chemistry are being tested under prototypal conditions. Croloy 2-1/4 has been extensively tested by MSAR, and vanadium and cobalt alloys have now been factored into their program; the extent of their program is given in Table 7-27. A collection of their data for Croloy 2-1/4 and type 316 stainless steel is presented in several topical reports. Data on tensile, yield, elongation, creep, rupture, impact, and fatigue tests in sodium are included therein.¹¹⁶

Atomics International has reported on the bi-axial stress rupture properties for types 304 and 316 stainless steel in sodium (see Figures 7-38 and 7-39).¹¹⁵ General Electric has also measured biaxial stress rupture for several austenitic alloys;¹¹⁵ they included a special heat treatment (carbide agglomeration) in their program. MSAR recently compared types 304 and 316 for use in sodium.¹¹⁴ No clear advantage was cited for either alloy.

Babcock & Wilcox has thoroughly reviewed past and existing material programs of the USAEC and others, and the objectives of these programs have been discussed with the cognizant investigators. A current review was obtained at the Second Working Group Meeting on Mechanical Properties of Materials in Sodium in June 1968.¹¹⁵ The findings were that the quantity, type, and quality of data specifically needed to design safe, reliable, and economical LMFBR systems are not available.

7.5.2.3. Scope of Work

The ASME Code is the most widely used and accepted code for the design of long-lived systems, but its present stress-intensity values do not take into account the effects of sodium on the mechanical properties of the materials of construction, nor does it

Figure 7-38. Biaxial Stress Rupture Properties of Cold-Worked and Annealed Type 304 Stainless Steel in Sodium

Figure 7-39. Biaxial Stress Rupture Properties of Cold-Worked and Annealed Type 316 Stainless Steel in Sodium

ensure a specific long design life, such as 30 years. This program is intended to provide enough properly documented quantitative data to develop stress-intensity values that will account for the effects of sodium service and a 30-year design life. The stress-intensity values will be developed to be compatible with the ASME Boiler and Pressure Vessel Code, Section III—Nuclear Vessels.

The mechanical property tests required to establish the allowable stress-intensity values include tensile, creep, stress-rupture, and fatigue tests. The ASME Code committees, in cooperation with ASTM, require that the data from these tests be obtained under controlled, specified conditions before they will accept the data as being valid for design purposes. All of the mechanical property tests proposed will be conducted so that they will be acceptable to the ASME Code if stress intensity values for particular environments are ever established and specified in the code.

Creep tests will be conducted at stresses approaching the design stress to determine the minimum creep rate, primary creep rate law, and times to the end of primary and secondary creep.

The stress rupture tests will be conducted primarily at stresses below the yield strength. Maximum times to rupture will exceed 10,000 hours for all materials and test temperatures.

Fatigue testing will cover the 10 to 100,000 cycle range, using cantilever-type specimens; variables will include cyclic rates and hold times.

Tensile testing will include true-stress, true-strain tests over a range of strain rates up to 200 reciprocal seconds as well as standard tensile tests at temperatures from 70 (RT) to 1500 F.

The initial mechanical property tests will be conducted on as-received heats of one austenitic stainless steel (304) and one low-alloy steel (Croloy 2-1/4). These data will be used to establish base line properties for the particular heats acquired.

Extremes of mass transfer (uniformly carburized stainless steel and decarburized low-alloy steel) can also be tested before the upper limits of mass transfer have been determined.

Finally, specimens which have composition gradients representing long-term sodium effects will be produced by gaseously carburizing or decarburizing. Mechanical property tests of these specimens will be conducted in air using the same equipment and procedures as described above.

All of the data generated will be evaluated and correlated to determine the effect of long-term sodium mass transfer on mechanical properties. It is not intended to evaluate the effect of long-term aging or similar phenomena, but only the effect of the compositional changes that are caused by the sodium. Past experience with long-term, elevated-temperature exposure indicates that aging and similar phenomena will have little effect on mechanical properties, compared to the effect of compositional changes. It is estimated that this program for the evaluation of two materials with two modifications (carburizing or decarburizing) of each will require the following tests:

<u>Type of test</u>	<u>Number required</u>
Standard tensile	150
True-stress, true-strain	150
Stress-rupture	200
Creep	150
Fatigue	800

In addition to these tests, some miscellaneous tests, such as hardness and impact studies, will be conducted.

7.5.2.4. Technical Justification

The initial analysis for designing any high-temperature component uses ASME Code allowable stresses (or acceptable non-code values for non-code components) and any ductility criteria that may apply to evaluate the mechanical loads, pressure, etc., on the structure. If any special corrosion allowances are required, sufficient thickness is added so that the anticipated corrosion will not cause stresses greater than those allowable at any time in the life expectancy of the component.

The designer then proceeds to evaluate the various segments of the component for problems like thermal stresses, vibration and/or mechanical fatigue, and pressure excursions. If any

of these analyses show that the component is questionable as designed, then the designer must eliminate the condition by any means available, even if it sometimes results in redesigning the component. Many techniques are available for minimizing thermal stress, vibration, fatigue, and pressure excursion stresses, but there is no general solution to all of these problems. For example, the designer cannot simply increase the thickness of the material at the points of distress. Rather, he must determine how to minimize these stresses by changing the design configuration or operating characteristics of the component. This is largely a problem of analytically determining the stress distributions and relating them to the mechanical properties of the materials of construction.

In order to be assured of the adequacy of his design, the designer must be aware of such mechanical properties as yield strength, tensile strength, creep rate, stress-rupture strength, modulus of elasticity, coefficient of thermal expansion, Poisson's ratio, true-stress-true-strain curves, fatigue life, and others. Data on all of these properties exist to some extent, but the designer of an LMFBR system needs to know (1) whether these data can be used to design a component which is expected to operate for about 30 years (250,000 hr), and (2) whether these data account for the effects of long-term exposure of the materials of construction to sodium.

As an example, the data used to establish the ASME Code allowable stresses were obtained from tests having a maximum of 10,000 to 20,000 hours' duration. The specimens were almost always tested in air, so that none of these data can be considered directly applicable to LMFBR systems. Furthermore, in order to set allowable stresses, the code data have been extrapolated about one order of magnitude in time to 100,000 hours. The code has done its job reasonably well, but it does not ensure a specific long life (30 years, 250,000 hours), nor does it account for the effect of sodium mass transfer on mechanical properties.

Therefore, this subtask will serve to upgrade the present ASME Code to ensure that the LMFBR systems will operate safely and reliably throughout their 30-year design life.

No funding is provided for core structurals, since it is expected that austenitic stainless steels will be used. The

unirradiated and irradiated properties of these stainless steels and other potentially useful alloys will be provided by LMFBR cladding tasks.

7.5.2.5. Relationship to Current or Planned Programs

None of the existing USAEC-sponsored mechanical property programs will generate the data requirements cited above; therefore, the proposed program, which is intended to generate these data, is not directly related to any of the existing programs. In developing the Program Plan (section 5 on Sodium Technology), the LMFBR Program Office has outlined a basis for planning mechanical property tests in sodium. Task 5-1.2, Carbon Transfer, calls for a program designed to determine how much carbon transfer is tolerable (based on effects on mechanical properties) and to develop the technology for controlling carbon transfer within the allowable limits.

According to the LMFBR Program Plan, carbon transfer is considered to be the major potential effect of sodium on the mechanical properties of thin members. Decarburization can cause significant loss of strength, and carburization can cause complete embrittlement. These effects can be deeply penetrating at LMFBR temperatures and times because of the high diffusivity of carbon. The Long-Range Plan of the LMFBR Program Office calls for half of the effort to be spent on carbon transfer in the determination of allowable limits of carbon content in three or four selected alloys and on determining the corresponding allowable range of carbon activity. In addition, the plan calls for work to characterize carbon transfer kinetics as a function of activity driving force and sodium system parameters.

Therefore, the proposed program will not duplicate the anticipated activity of any existing USAEC program, and it is in full agreement with the LMFBR Program Plan.

7.5.2.6. Major Facilities and/or Equipment Required

Nearly all of the facilities required for this program are now available at B&W's Alliance Research Center. Furnaces and metallographic equipment are available, including the standard tensile, hardness, impact, creep, stress-rupture, and fatigue

machines required for the mechanical property tests. The one major item not available now is the high-strain-rate, true-stress-true-strain equipment. Equipment is available to generate high strain rates, and B&W is developing equipment to measure true-stress-true-strain. It is anticipated that the equipment will be available for this program before it is completed. Little or no outlay of funds from this program would be required to develop the facility. Some irradiation testing is planned for the FFTF, also.

7.5.2.7. Budget Estimate

Table 7-28 is a summary of estimated costs and the planned schedule for R&D work on primary and secondary sodium system structures and components.

7.5.3. Control, Shielding, and Moderator Materials

7.5.3.1. Objectives

Little effort has been expended on the technology of control rod and shielding materials for fast breeder reactors. Material properties must be determined so that design work on the LMFBFR can proceed in the most efficient and economical manner.

7.5.3.2. Summary of Current Technology

Ideally, a control material for an LMFBFR would (1) have a high epithermal and fast neutron absorption cross section, (2) be compatible with the fast reactor environment, (3) be inexpensive and easily fabricated into useful shapes, (4) be compatible with the coolant, and (5) not produce any objectionable products as a result of a neutron-absorption reaction. There are no known ideal materials for fast reactor control. Two materials are considered to possess many of the properties of ideal materials, but both of these have serious deficiencies. The deficiencies can be tolerated in certain circumstances; however, they are serious enough to warrant investigations of alternate materials that are potentially less objectionable.

Boron and tantalum are the two materials currently considered the least objectionable. The principal problems with boron are its burnout characteristics and gas generation. The attractively high cross section of boron becomes less desirable when we

Table 7-28. Estimated Costs and Schedule for R&D on Primary and Secondary System Structures and Components

Overall schedule and estimated cost, \$ × 10 ³																
Task	CY:	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	Total
								△△						3	4	
Primary and Secondary System Structures and Components																
Establish test parameters based on LMFBR design		█														110
Characterize test material chemistry and properties			█	█	█											160
Produce specimens with simulated mass transfer effects			█													115
Determine mechanical properties at extremes of mass transfer				█	█	█										340
Determine mechanical properties of specimens with simulated mass transfer effects					█	█	█									340
Develop design criteria						█	█									145
Prepare design and alternating stress intensity values							█	█								150
Irradiation testing of reactor materials not tested in cladding program									█	█	█	█	█	█	█	740
Total																2100

Milestones

- 1 Complete development of design stresses for unirradiated materials.
- 2 Begin FFTF irradiations.
- 3 Finish FFTF irradiations.
- 4 Complete design stress formulation for irradiated materials

(a) Manpower	\$1,710,000
Facilities	0
Materials	390,000
Irradiation	(b)
	<u>\$2,100,000</u>

(b) Cost of irradiation units is unknown.

consider the fact that each boron-10 atom (approximately 19% of the atoms in natural boron) is transmuted into an atom of helium and an atom of lithium as a result of absorbing a neutron. Neither of these transmutation products has any significant neutron-absorbing capabilities. Further, the lithium is a potential reactant with the cladding material that maintains the boron or boron compound in the necessary geometric arrangement in the reactor. The helium generated must be accommodated either within the control rod or by some mechanism to release it to the coolant system, so that excessive pressure is not generated inside the control rod. It has been found in one of our design concepts that the life of a boron control rod might be less than two months because of gas generation. The short life and possible gas release problems of boron detract appreciably from its desirability for control purposes. For use as shielding material, it may be possible to overcome or accommodate the shortcomings; however, even in shielding applications boron materials pose some serious design and fabrication problems.

The high cost of tantalum and its neutron worth as a control material detract from its universal application for control and shielding purposes in an LMFBR. The heating rate in tantalum is approximately 3 times that of B_4C , which creates a need for more sodium flow to cool control rods made of this material. The advantages of tantalum, however, may be worth tolerating some of its disadvantages for certain applications. There is no appreciable gas generation in tantalum during absorption of neutrons, and the products of neutron absorption maintain a high neutron absorption cross section. Therefore, the limiting factor in control rod life would be embrittlement. Tantalum is stable in sodium, it is easy to fabricate, and the extra control rod mass associated with the higher density of tantalum causes greater scram acceleration against the hydraulic forces of the flowing sodium.

In the area of shielding materials, boron carbide (B_4C) and borated graphite are the generally accepted candidates, but helium generation can be a limiting factor here as in control rod use. Neutron shield elements might have to be replaced as often as every 8 months because of gas generation if a suitable means of accommodating the gas cannot be developed.

Figure 7-40. Thermal Conductivity of TSX Graphite

Special concretes offer inexpensive shielding where temperature and stress permit their use.

The literature on potential control rod and shielding materials is brief. Bonilla and Friedland reported on the thermal conductivity of boronated graphite and boron carbide.¹¹⁷ Baker of BNWL found that the irradiated thermal conductivity of graphite was lowered significantly by irradiation, as shown in Figure 7-40. Davidson (PNWL) determined radiation effects on boronated graphite exposed in the Enrico Fermi reactor.¹¹⁸ The neutron spectrum was principally thermal. Expansions to 3.3% in the temperature range of 370 to 500 C were obtained at a dose of 2.5×10^{21} neut/cm². Gates and Neidner irradiated boronated graphite in the ETR in a flux more representative of fast reactors.¹¹⁹ Contractions up to 1% were found by Cox and Woodruff in TSX graphite after 29×10^{20} nvt fast flux at more than 500 C. A literature review of the effect of irradiation on boron carbide was made by

Evans;¹²⁰ he found no basis for extrapolating low-temperature (500 to 750 F) data to higher temperatures. Irradiation tests of concrete were reported by J. Rappeneau, who found its behavior satisfactory for Rapsodie.¹²¹

7.5.3.3. Scope of Work

Numerous tests of materials are necessary because of the sparse data available. Concurrent assessment with the Physics and Core Design programs is essential to prevent waste of development funds.

Tantalum will be investigated to determine the effect of sodium on its structural integrity; its mechanical and physical properties will be evaluated in both unirradiated and irradiated conditions. It may be necessary to can the tantalum in stainless steel. Accurate determinations of swelling will be obtained for tantalum and boron carbide at temperatures of 1100 to 1500 F for exposure to 10^{24} nvt. The gas release from boron carbide will be determined, and B_4C will also be evaluated for shielding use. Here we need data on its properties alone and mixed with graphite.

The tabulation below is a summary of the target test conditions and the properties to be investigated for the shielding materials referenced in B&W's LMFBR reference concept. All of these items will be determined for a liquid sodium environment.

<u>Material</u>	<u>Temp, F</u>	<u>Fluence, neut/cm²</u>	<u>Properties^(a)</u>
Boronated graphite (25 wt% B)	1000-1300	10^{23}	1-7
Serpentine aggregate	800-1100	10^{21}	1, 4, 7, 8
Ordinary concrete	200-800	10^{21}	1-4, 7, 8

(a) The numbers in this column indicate the following properties:

1. Shielding effectiveness.	5. Oxidation rate.
2. Mechanical properties.	6. Spontaneous energy release.
3. Allowable stresses.	7. Gas evolution and release.
4. Dimensional integrity.	8. Water retention.

Other materials, such as hafnium and the rare earths(samarium, europium, gadolinium, dysprosium, and erbium), will be investigated for their possible application as absorber materials, with appropriate moderating materials, to neutron shielding. Titanium, zirconium, and lithium hydride will be evaluated as moderators.

7.5.3.4. Relationship to Current or Planned Programs

BNWL is known to be investigating control and shielding materials for use in the FFTF.

7.5.3.5. Major Facilities and/or Equipment Required

The FFTF and a hot cell will be needed for irradiation testing. Some evaluations can be performed in the ETR and a thermal reactor like the BAWTR. Conventional materials and laboratory test equipment will be used for much of the work.

7.5.3.6. Budget Estimate

Table 7-29 is a summary of the estimated costs and a schedule of the R&D work planned on control and shielding materials.

7.5.4. Material Properties as Affected by Reactor Transients

7.5.4.1. Objectives

In addition to the properties required for normal reactor behavior, we must evaluate high-temperature exposure effects on the properties of materials under accidental transient conditions. This information is needed in order to estimate the extent of damage caused by the accident and to determine whether components would be usable after an accident.

Mechanical and physical properties are required for structural component materials up to a temperature of about 2500 F. Properties for the equation of state at temperatures above the melting point are also needed, and the effect of exposure to 1500 F for hours or days and to 1500 to 2500 for seconds to minutes must be assessed.

Table 7-29. Estimated Costs and Schedule for R&D on Control and Shielding Materials

Overall schedule and estimated cost, \$ × 10 ³															
Task	CY: 1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	Total ^(a)
			△1	△2	△3		△4	△5					△6		
Control and Shielding Material															
Control															810
Shielding															850
Total															1660

Milestones

- △1 Complete review of status of control and shielding materials.
- △2 Complete design of experimental program.
- △3 Start material development.
- △4 Complete material development.
- △5 Start irradiations.
- △6 Complete irradiations.

(a) Manpower	\$1,260,000
Equipment	250,000
Materials	150,000
Irradiation	(b)
	<u>\$1,660,000</u>

(b) Cost of irradiation is unknown.

7.5.4.2. Summary of Current Technology

Data on unirradiated mechanical properties exist for the stainless steels in small amounts up to 1500 F for steady-state conditions. Design stresses are provided in Section VIII, but not in Section III, of the ASME Code. Above 1500 F, only sparse data are available. The bulk of available data were accumulated by Simmons and Cross. Life fraction studies of Guarnieri¹²² and others provide a beginning basis for evaluating the serviceability of stainless steel that has been exposed to the transients. The data on the thermal cycling effects that may be encountered during the transients are also insufficient.

The scarcity of physical property data for structural materials in the higher temperature range indicates the need for some R&D.

The information on the equation of state for liquid stainless steel and other structural materials that may be used in the core has not been collected so that its adequacy might be evaluated.

It will be more difficult to ascertain the effects of transients on irradiated materials, and existing data are very sparse.

7.5.4.3. Scope of Work

Data requirements are itemized below.

1. Physical properties for the equation of state.
 - a. Structural materials.
2. Mechanical and physical properties up to the melting point.
 - a. Control rods.
 - b. Meltdown pans or core catchers.
3. Mechanical and physical properties to 2500 F.
(exposure time will be seconds)
 - a. Fuel rod spacers and supports.
 - b. Subassembly wrapper cans.
 - c. Grid plate or core support structure.
 - d. Flow nozzles.
4. Mechanical and physical properties to 1400-1500 F.
(exposure time will be hours or days)
 - a. Control rod drive extensions.
 - b. Holddown structure.
 - c. Instrument detector well materials.
 - d. Reactor vessel and interior surface of cover structure.
 - e. Core vessel and baffle or apron attachments.

- f. Thermal shields.
- g. Intermediate heat exchangers (IHXs).
- h. Emergency core coolers.
- i. Primary pumps.
- j. Primary pool piping.
- k. Reactor vessel support members
- l. Primary pool IHX flow control equipment.
- m. Fuel storage rack.
- n. Internals for fuel handling machine.

7.5.4.4. Budget Estimate

Table 7-30 is a summary of the estimated costs and a schedule for R&D on the effects of reactor transients on material properties.

7.5.5. Other Materials

7.5.5.1. Objectives

Bearing, seal, spring, wear, and other materials will be required for the primary and secondary systems. Many of these needs will not be identified until detailed design work is started. In order to have the materials available in time, it is necessary to proceed with evaluation tests.

7.5.5.2. Summary of Current Technology

The Liquid Metal Engineering Center (LMEC) has been testing a number of materials in sliding friction.¹²³ They also plan to test seals.

7.5.5.3. Scope of Work

The principal work that can be identified at this time is material design studies, which will be carried out in cooperation with Core Design and Components. The results of test work conducted by LMEC will be utilized as much as possible. Some of the materials will be used in neutron and/or gamma radiation; hence, some of the testing will involve irradiated materials. Funds for this work are included in the budget.

The plan of action as set forth in the Program Plan includes the following items:

- 1. Compile and examine existing data.
- 2. Develop additional data.

Table 7-30. Estimated Costs and Schedule for R&D on Material Properties as Affected by Reactor Transients

Task	CY:	Overall schedule and estimated cost, \$ × 10 ³									Total ^(a)
		1973	1974	1975	1976	1977	1978	1979	1980	1981	
Material Properties as Affected by Reactor Transients							△1	△2		△3	
										△4	
Determination of mechanical and physical properties											750

Milestones

- △1 Complete mechanical and physical property determinations to 1400-1500 F (item 4).
- △2 Complete mechanical and physical property determinations to 2500 F (item 3).
- △3 Complete mechanical and physical property determinations to melting point (item 2).
- △4 Complete physical property determinations for equation of state (item 1).

(a) Manpower	\$540,000
Facilities	130,000
Materials	80,000
Irradiation	--
	<u>\$750,000</u>

(b) Irradiation costs unknown.

3. Perform proof tests and verify material performance.
4. Develop new and improved materials.
5. Obtain additional information from the applications of candidate materials.

7.5.5.4. Relationship to Current or Planned Programs

The LMFBR Program Plan is expected to provide 95% of the R&D needs. Because the predecessor plant will be built after the demonstration plant, it is expected that some additional test needs will become apparent through experience with the demonstrator; these will require specific funding from this subtask.

7.5.5.5. Major Facilities and/or Equipment Required

The FFTF and a hot cell will be needed in addition to a conventional materials laboratory.

7.5.5.6. Budget Estimate

Table 7-31 is a summary of estimated costs and a schedule for R&D work on other materials associated with the reference LMFBR design.

7.6. Cost Benefit Studies

7.6.1. Fuel Fabrication Uncertainties

In addition to the influence of calculational uncertainties on physics and economic performance predictions, further uncertainties are encountered in translating design specifications into reactor cores. The effect of these fabrication uncertainties on the physics performance of the core has been the object of a low-level study. Obviously, fabrication uncertainties represent problems for all disciplines, e.g., thermal, mechanical; but the scope of this study was limited to an assessment of fuel cost penalties pursuant to the perturbations in physics performance that result from reasonable tolerances and uncertainties during fabrication. Consideration of the major fabrication uncertainties was further limited to the fissile enrichment and smeared fuel density in the pin. Variations in structural and coolant materials also affect physics performance, but since these are expected to be much weaker than fuel variations, they were not calculated.

Table 7-31. Estimated Costs and Schedule for R&D on the Other Materials Associated With the Reference LMFBR Design.

Task	Overall schedule and estimated cost, \$ × 10 ³													Total
	CY: 1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	
		①	②	③			⑤			⑥		⑦		
<u>Other Materials</u>			④											
Brazing alloys														1,800
Facing materials														1,600
Bearing materials														1,600
Spring materials														2,400
Nonmetallics														300
Organics														1,000
Inorganics														2,000
Total														10,700

Milestones

- ① Handbook of design information published (1st ed.).
- ② Material needs for first demonstration plant identified.
- ③ Complete testing to provide additional information for demonstration plant.
- ④ Establish operating limits for bearings, seals, and facings in sodium (ref. Sodium Technology).
- ⑤ Complete characterization of reaction between sodium and nonmetallic materials (ref. Sodium Technology).
- ⑥ Start tests for the predecessor plant.
- ⑦ Complete tests for the predecessor plant.

The cost benefits ground rules adopted in the evaluation of calculational uncertainties were also applied in the economic evaluations for this study. The additional ground rules used in the analysis of fabrication uncertainties are as follows:

1. The uncertainty range of interest here is only the error inherent in determining or measuring the physical properties of fabricated fuel.
2. The small uncertainties in the plutonium isotope percentages were combined to give maximum and minimum fissile content. This variation in composition was merged with the (Pu/Pu+U) variations to determine the fissile enrichment uncertainty.
3. Enrichment and density uncertainties are independent variables.

It was assumed that process line control would be successful in holding enrichment and density within a 2σ band about a specified value. Only the error in determining this nominal value is of any consequence, since, statistically speaking, every high value has its low value counterpart, and the core average will be the nominal value.

Measurement uncertainties, based on brief experience to date with FFTF pellets for qualification pins, were used in the calculations. The calculations and methods were assumed to be precise, so that they accurately predict the operational performance of the reactor. The fissile enrichment uncertainty for coprecipitated material was $\pm 0.74\%$. For mechanically blended materials, an uncertainty of $\pm 2.2\%$ is forecast. In addition to an enrichment uncertainty, the determination of pellet characteristics is expected to allow a $\pm 0.6\%$ absolute uncertainty in smeared density. However, the pellet dish and diametral gap could not be factored in properly, thereby making this value slightly optimistic. To cover the pessimistic side and to indicate the overall trend, smeared density uncertainties of $\pm 2\%$ and $\pm 3.8\%$ were evaluated. The list of the cases examined is shown in Table 7-32.

The results for the singular effect of a given enrichment or density uncertainty are shown as cases 1 through 9 in Table 7-32. A missed enrichment (cases 2 - 5) affects initial criticality and breeding ratio, both of which affect core lifetime. A low enrichment means that

the design core lifetime (or power output) will not be achieved. Since the plutonium investment is essentially the same for the shorter core lifetime, the fuel costs climb rapidly. For an enrichment 2.2% low, a 0.4 mill/kWh penalty was found. On the other hand, high enrichments needlessly increase the amount of fissile material committed, since the increased core lifetime made possible by higher reactivity cannot be allowed because of the peak burnup limit. A similar situation exists for density cases 6 through 9, since low densities also reduce core lifetime, and high densities increase the plutonium content and breeding ratio in a core that is already capable of reaching peak burnup.

Table 7-32. Economic Penalties for Enrichment and Density Variations

Case No.	Description	Fissile Enrichment, %	Smeared density	Fuel cost penalty, mills/kWh
1	Reference	11.63	85.0	--
2	Coprecipitated (+)	11.72	85.0	0.01
3	Coprecipitated (-)	11.56	85.0	0.16
4	Mech. blend (+)	11.89	85.0	0.04
5	Mech. blend (-)	11.39	85.0	0.40
6	Density +2%	11.63	87.0	0.02
7	Density -2%	11.63	83.0	0.51
8	Density +3.8%	11.63	88.8	0.04
9	Density -3.8%	11.63	81.2	1.34
10	Combined penalty	11.89	85.6	0.03

These results made one fact very clear: an enrichment and/or density that is lower than the nominal design value cannot be tolerated, even for very small uncertainty ranges. The only alternative is to modify the enrichment specification to a new elevation, so that the least-favorable combination of enrichment and density uncertainties will be equal to the calculated design position. Applying first the $\pm 0.6\%$ density uncertainty, it was determined that a fissile enrichment of 11.72% would be required to cover the possibility that the actual smeared core density could be 84.4 instead of 85%. Further, an even higher enrichment, 11.80%, would have to be specified for the coprecipitated process to ensure the minimum enrichment of 11.72% required to

protect against a low density. The price for this insurance, however, would be paid if the high side of the enrichment and density uncertainties were to occur simultaneously. Therefore, an enrichment of 11.89%, combined with a smeared density of 85.6%, was run in case 10 to evaluate the maximum penalty that might occur in attempting to avoid "below nominal" performance. The penalty for the adjustment, 0.03 mill/kWh, seems to be a reasonable price to pay to avoid a possible penalty of about 0.2 mill/kWh.

In conclusion, it should be pointed out that the 0.03 mill/kWh penalty is based on very small uncertainties, especially in the case of density, so that this value probably represents a minimum. Therefore, the development of fabrication processes must be monitored and guided to prevent excessive uncertainties. This could easily occur with only a small increase in uncertainties because of the "doubling" process required in their correction. Finally, it must be remembered that the fabrication uncertainties must still be superimposed on the calculational and method uncertainties to truly evaluate the total penalty involved. Given careful analysis, this may well dictate that fabrication uncertainties be reduced even below those assumed in this study.

7.6.2. Cladding and Core Structural

It is possible that stainless steel cannot satisfy the lifetime requirements of the 1000-MWe LMFBF. In that event, it may be possible to substitute a vanadium alloy as the cladding and core structural members.

Vanadium alloys are much more expensive and possess somewhat higher fast neutron cross sections than those of stainless steel alloys. In addition, the impurity limit in sodium is expected to be low enough to make hot trapping mandatory. Although stainless steels are expected to perform satisfactorily in sodium containing up to 10 ppm oxygen, the limit probably must be reduced to 5 ppm for vanadium alloys.

An increase in core fabrication and operating costs is expected as a result. Stainless steels and vanadium alloys were compared by Westinghouse for a mixed-carbide core. They found that vanadium is competitive in an analysis that requires large assumptions. A similar analysis should be made for a mixed-oxide fast reactor.

The effect of fuel element cladding thickness on power generation cost was investigated during the follow-on program for stainless steel; results are shown in Figure 7-41. The importance of success in the R&D program for 10-mil cladding is revealed by the successively higher power generation costs for 15- and 20-mil cladding. The change from 10- to 15-mil cladding increases power costs by approximately 0.5 mill/kWh.

7.6.3. Fuel Elements

The principal objective of fuel element research and development will be the achievement of an average burnup of 100,000 MWd/T at a heat rate high enough to establish a competitive LMFBR economy.

The effect of burnup, together with pin pitch, on minimum fuel cost is assessed in Figure 7-42. Roughly 0.5 mill/kWh would be lost if a burnup of only 50,000 MWd/T is achieved with the reference pin size. Figure 7-43 shows that the effect of burnup and pin pitch on doubling time is 3-1/2 years for the decrease from 100,000 to 50,000 MWd/T.

A linear heat rate of 14 kW/ft is found to increase power generation costs by 0.05 mill/kWh over that obtained with 17 kW/ft. On the other hand, a linear heat rate of 20 kW/ft would decrease costs by approximately 0.03 mill/kWh if a 0.360-inch-diameter pin were used. See Figures 7-44, 7-45, and 7-46 for the variation in power cost with linear heat rates of 14, 17, and 20 kW/ft, respectively.

Figure 7-41. Power Generation Cost for Various Fuel Element Cladding Thicknesses

Figure 7-42. Effect of Pin Pitch and Burnup on Minimum Fuel Cost

Figure 7-43. Effect of Fuel Element Diameter and Burnup on Doubling Time

Figure 7-44. Power Generation Cost Variation With Fuel Element OD and Pitch at a Maximum Linear Heat Rate of 14 kW/ft

Figure 7-45. Power Generation Cost Variation With Fuel Element OD and Pitch at a Maximum Linear Heat Rate of 17 kW/ft

Figure 7-46. Power Generation Cost Variation With Fuel Element OD and Pitch at a Maximum Linear Heat Rate of 20 kW/ft

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8. FUEL RECYCLE

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8. FUEL RECYCLE

8.1. Introduction

8.1.1. General

Fuel recycle operations cover all the phases of fuel handling, fabrication, disassembly, fuel processing, assembly, and shipping that are required to sustain reactor operation. These procedures start at the reactor site and include handling and shipping operations to ensure the proper transmittal of spent fuel assemblies to the reprocessing site without danger to operating personnel or the general public. They continue through the reprocessing and refabrication stages and include all material and component fabrication. The operational procedures conclude only when new fuel assemblies have been safely delivered back at the reactor site and properly stored in readiness for refueling the reactor. All procedures, facilities, components, equipment, and techniques required for accomplishing these tasks safely, reliably, and economically must be defined, designed, and developed to a high level of excellence if the 1000-MWe LMFBR plant is to become a commercial reality.

The responsibility of the fuel recycle programs discussed in this section is to accomplish all of the necessary development work. These programs are very dependent on other development programs—especially those of core design, fuel and materials, physics, and safety (sections 6, 7, 9, and 10, respectively). A most important requirement is design engineering. The basis for all the programs discussed is B&W's 1000-MWe conceptual design established in the Follow-On Study. Many of the programs result from extensive past experience and are supported by laboratory and testing work; however, other programs are not so well-supported and are based on sound reasoning, good engineering judgment, and related experience. In general, the fuel recycle requirements received far less emphasis in the Follow-On Study than did

reactor site facilities. It is assumed, however, that all areas of the subject had been engineered to the Preliminary Design stage.

The following programs will emphasize the development of the technology and industrial capability necessary for the production of LMFBR fuel assemblies and for the reprocessing of the fuel material. Although the B&W program essentially parallels that given in the National Program Plan, there are several major differences. First, this plan does not attempt to develop all alternates. Second, it has been assumed that many problems of fuel recycle will be solved in programs not funded by the government. Where possible, these are noted, and no estimate is made for their accomplishment. The third major area of difference results from the B&W reference design specifics; the following items are especially important:

1. The fuel material is particulate, mixed $\text{PuO}_2\text{-UO}_2$, vibratory compacted.
2. Each fuel pin is vented.
3. The fuel assembly and spacer design is somewhat different (ribbon spacers) than those more commonly proposed.
4. No secondary storage or disassembly of spent fuel at the reactor site is proposed.
5. Shipment of spent fuel assemblies (one per cask) will be accomplished in a sodium-filled shipping cask.
6. A minimum storage time (70 to 90 days) at the reactor site is proposed.
7. Handling, storage, and disassembly of spent fuel assemblies must be incorporated as head-end procedures at the reprocessing site.
8. A greater number of remote handling procedures must be developed at all site locations because of the recycled plutonium fuel.

All of B&W's proposed programs for the development of fuel recycle operations make maximum use of information that we believe will come from other programs. The accomplishments of the

FFTF, Enrico Fermi, and the English reactors will be very valuable, and coordination to the maximum extent possible is considered not only necessary but mandatory. For both near and long terms, the LMFBFR demonstration plants will also be important, and early and continuous cooperation between the designers (through the Program Office direction) will ensure that all LMFBFR fuel recycle operations will be developed to the highest possible level.

8.1.2. Summary Budget and Schedule

The following tables summarize the estimated budget and schedule required for the successful design, construction, and operation of fuel recycle operations required for B&W's 1000-MWe reference design. Table 8-1 gives a total dollar breakdown by calendar and fiscal year, and the tabulation below gives a summary by manpower, facility, and equipment requirements. The estimated costs do not include construction of major facilities for testing, but they do include allowances for adaptation and rental of existing facilities. No effort was made to compare the total demand on these facilities with available space or schedule.

The budget provides estimates for the engineering and design efforts required to define and produce the models, test specimens, and prototypes needed for detail development. It includes the manpower for engineering, test planning, test operation, and test analysis. The cost of manufacturing the required equipment, test apparatus, and instrumentation is also included.

The R&D plan and schedule is based on major use of design, fabrication, and testing experience and technology that will be produced in other LMFBFR programs. Some of the programs are dependent on results to the extent that they will not start in the next few years. Most of the programs are specifically planned for input to the 1000-MWe plant rather than the demonstration plants; these can be noted in Table 8-1.

Table 8-1. Summary of Estimated Costs and Schedule
for Fuel Recycle R&D

Task	FY CY	Estimated cost, \$ × 10 ³												Total
		<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	<u>1978</u>	<u>1979</u>	<u>1980</u>		
		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	
<u>Fuel Recycle</u>														
Fuel material preparation		190	470	570	470	520	420	420	180	--	--	--	--	3,240
Fuel assembly fabrication		50	200	1250	2000	1750	1900	2400	2400	1350	400	--	--	13,700
Fuel recycle operations at the reactor site		160	185	180	140	--	--	--	--	--	--	--	--	665
Fuel shipping		--	--	--	--	150	180	370	400	400	350	150	--	2,000
Fuel reprocessing		--	180	970	720	720	570	620	670	470	410	370	--	5,700
Waste management		110	110	320	320	240	240	370	190	180	170	--	--	2,250
Miscellaneous related activities		--	--	240	210	210	210	190	120	120	100	100	100	1,600
Total		510	1145	3530	3860	3590	3520	4370	3960	2520	1430	620	100	29,155

Summary Budget Estimate

Manpower	\$15,195,000
Facilities	4,860,000
Equipment	<u>9,100,000</u>
Total	\$29,155,000

8.2. Overall R&D Requirements

8.2.1. Fuel Material Preparation (Commercial Development)

8.2.1.1. General

Objective — The objective of this project is to acquire the technology needed to produce LMFBR fuel systems utilizing recycled fuel from the LMFBR demonstration units and from the commercial units that will follow. Inasmuch as B&W has selected particulate (mixed oxide) fuel for its initial reference design,¹ the near-term objective of the project is to develop the technology required for all of the recycle feed material and fuel fabrication capabilities required of the fuel. The long-range objective is to develop this technology to the degree necessary for a safe, competitive, and reliable utility-owned LMFBR. An economic evaluation of the use of particulate fuel in the 1000-MWe plant is also an objective of this project.

Summary of Current Technology — For the reference LMFBR design, B&W's initial choice is slightly hypostoichiometric coprecipitated uranium-plutonium oxide particulate fuel, vibratorily compacted into small-diameter, stainless steel pins. Physically blended oxides and sol-gel materials are also receiving consideration; however, coprecipitated materials have top priority at this time. The reasons for selecting this material are well-articulated in B&W's proposed program for particulate fuel development.¹

Coprecipitation is the most widely used of all methods for preparing plutonia-urania feed material; it has been developed extensively in this country and abroad.² The technique has been used in this country on a pilot-plant scale at various commercial and laboratory organizations, including B&W. It has been used in conjunction

with physical blending to produce improved homogeneity in low-plutonium fuels.^{2a} In addition to better homogeneity, coprecipitation has these advantages:

1. Adaptability to a wide range of PuO_2/UO_2 ratios.
2. Low process losses and scrap recycle.
3. Good process quality control.
4. Simplicity of process and adaptability to continuous operation.

Various methods of coprecipitation have been employed. The customary laboratory techniques involve batch operations, the addition of the precipitant, NH_4OH or ammonia gas, to the acidified feed solution, and precipitation at an elevated temperature to a final pH of 7 to 9. In production-scale operations, continuous coprecipitation techniques are generally employed. The feed solution and precipitant are metered to an agitated vessel held at an elevated temperature (50 to 80 C). The pH of the effluent slurry is monitored continuously and is generally maintained at a pH of around 9. This technique has been used in pilot-plant operations at B&W. After the precipitate is filtered (either a centrifuge or a rotary-vacuum filter is used in plant-scale operations), it is calcined at about 750 C for several hours in a reducing atmosphere to convert the precipitated ammonium diuranate to UO_2 .

Oxide powder prepared by coprecipitation is usually used for pellet production and is not suitable for vibratory compaction, since large-diameter particles are not formed. Therefore, in the production of a particulate feed material from either coprecipitated or mechanically blended oxides, an intermediate step is required to produce particles of the required size and density. Some of the most promising routes studied are (1) fusion, (2) compaction-sintering, (3) particle-agglomeration, and (4) pneumatic impaction processes.

The compaction-sintering and the pneumatic impaction processes produce angular particle feed material suitable for fabrication in the tubing sizes of interest for fast reactors. Current B&W plans are to follow the compaction and sintering technique. Although this method involves compacting the powders by pressing in a

mold, the overall fabrication process is simpler than pelletizing, for the following reasons:

1. The addition of binders or lubricants to the powders is not required for compaction of feed material.

2. Dimensional control is not required in the compaction process, since the goal is densification only (longer life for compaction dies, less concern about material characteristics and pressing parameters).

3. Inspection of compacts before or after sintering is not required, since chipping, lamination, and other pressing defects are not important.

The fabrication of particulate fuel by vibratory compaction is not new to the nuclear industry. R&D studies carried out at B&W and elsewhere over the past ten years have provided much background information that is applicable to fast-reactor fuel problems.³ Much of the initial effort was directed toward the high-density loading of tubes with urania fuels. The maximum densities obtainable ran to about 90% TD* in 0.5-inch-diameter tubes and 84% in 0.2-inch-diameter tubes;⁴ however, the density range of interest for fast reactor fuels is readily obtained by vibratorily compacting particulate fuel material.

B&W has been engaged in the development of vibratory compaction technology since 1959.⁵⁻⁷ In the interim period, over 5000 rods have been fabricated here by vibratory compaction in various development programs. Previous work was concerned with thermal designs (0.400 inch diameter); however, more recent efforts have been concentrated on fast-reactor pin configurations.⁷ As part of the fabrication studies, nondestructive techniques have been developed for determining fuel mass uniformity along the length of the rod. Automated processing has been developed for some stages of the vibratory fabrication technique. Additional B&W studies have established the optimum particle size ranges for maximum packing efficiency, and

- - - - -
*TD: theoretical density.

progress has been made in evaluating the economics of a crushing process to produce the optimum particle size spreads.

Technical Justification — The development of extremely reliable remotely operated and automated systems for feed material preparation and fuel fabrication is essential to the success of the B&W LMFBR program. The initial or near-term objectives will be met by demonstrating the technological feasibility of the developed feed material preparation and fuel fabrication methods. Since near-term fuel loads and process uncertainties do not warrant initial investment in a heavily shielded, remotely operated fuel fabrication plant, a directly operated plant has been postulated for the recycle of demonstrator fuel. The scope of work outlined for the fulfillment of near-term capability requirements must be performed in order to adapt the feed material preparation and fuel fabrication methods (developed under the fuel materials program) to a scale of operation capable of fuel recycle requirements.

Following satisfactory demonstration of the technological feasibility, the processes must be optimized to meet the long-term fuel recycle requirement of reliability in remote operation and favorable economics. Experience gained during near-term operations will be most helpful. The scope of work outlined for the fulfillment of long-term objectives is essential to the successful evolution of the required technology with the minimum investment.

There has been no economic evaluation of pellet fuel versus particulate fuel for use in LMFBRs under the present operating conditions. There is considerable controversy and debate as to the relative merits of each type of fuel, the effects of long-term burnup, cladding-fuel interaction, gas release, and fabrication cost. Much of this is due to fragmentary programs or programs with the established goal of showing a particular fuel form in an advantageous position. In order to resolve the issue, a complete evaluation of both fuels in an unbiased manner, using the established 1000-MWe LMFBR conditions, is necessary.

Relationship to Current or Planned Programs — The National LMFBR Program for the development of equipment

and facilities needed for feed material preparation and fuel fabrication is not oriented in the direction of the reference fuel design, nor is the Program of sufficient depth for its requirements. With respect to the development of specific fabrication techniques, however, the National Program is expected to yield useful information. Useful information on the design and operation of remote facilities for fuel fabrication is also expected to be gained from the TURF facility operations at ORNL.

8.2.1.2. Feed Material Preparation

Objective — The objective of this subtask is to determine the near- and long-term facility requirements for producing LMFBFR feed material from reprocessed reference LMFBFR spent fuel.

Scope of Work — The scope of work outlined below is considered essential to the establishment of the required near-term feed material preparation capability in the recycle of reference LMFBFR fuel.

1. The determination of radiation levels of feed-materials process solutions: Utilizing information developed in task 8.2.3, the intrinsic radiation of fully decontaminated,* recycled plutonium stock solution and recycled uranium stock solution will be determined. Since direct fabrication of fuel during near-term operations is intended, the selection of recycled or virgin uranium will be indicated.
2. The determination of unit shielding requirements, if any, for direct handling of feed material process equipment.
3. The development of remote handling procedures (should only be necessary for equipment maintenance, etc.).
4. The development of automated equipment and process control instrumentation.
5. The determination and development of required analytical support capabilities (establishment of quality control procedures).

- - - - -

* Decontaminated as much as possible in three cycles of operation in the fuel reprocessing plant.

6. The design of plant-scale equipment meeting near-term capacity requirements, utilizing a combination of automated equipment, remote manipulators, and minimal glove box procedures.

The scope of work outlined below is considered essential to the establishment of the required long-term feed material preparation capability in the recycle of reference LMFBR fuel.

1. The determination of radiation levels of feed material process solutions.

2. The determination of shielding requirements (heavy shielding and completely remote operations will be required).

3. The optimization of plant, process, and equipment design; this includes developments in automation, remote manipulation, process control instrumentation, and sampling procedures.

Major Facilities and/or Equipment Required — See section 8.2.1.3 for this information.

Budget and Schedule Estimate — See section 8.2.1.3 for budget and schedule estimates.

8.2.1.3. Fuel Fabrication Development

Objective — The objective of this subtask is to determine the near- and long-term facility requirements for the fabrication of reference fuel elements containing recycled LMFBR fuel.

Scope of Work — The scope of work outlined here is considered essential to the establishment of the technology required for the fabrication of near-term recycled reference LMFBR fuel.

1. The development of unit shielding requirements, if any, for the direct fabrication of fuel elements containing recycled fuel.

2. The development of remote handling procedures.

3. The development of automated equipment and process control instrumentation.

4. The development of nondestructive testing techniques and equipment.

The following scope of work is required to establish the technology required for fabricating long-term recycle reference LMFBR fuel:

1. The determination of shielding requirements (should be the same as for feed material preparation).
2. The development of process control and nondestructive testing equipment for remote operation.
3. The optimization of plant, process, and equipment design.

Major Facilities and/or Equipment Required — Major research and development facilities, such as those available at B&W's Nuclear Development Center (NDC) or those at Oak Ridge National Laboratory (ORNL) will be required. These facilities must be adaptable for the development and optimization of the plant, process, and fabrication equipment required.

Budget Estimate —

Manpower	\$1,940,000
Facilities	350,000
Equipment	<u>770,000</u>
Total	\$3,060,000

Table 8-2 is a summary of estimated costs and a schedule of research and development for fuel material preparation.

8.2.1.4. Economic Evaluation

Objective — The objective of this program is to provide an unbiased economic evaluation of the total fuel cycle cost when particulate fuel is used in the 1000-MWe plant. Necessarily, this involves a comparison with pellet fuel.

Scope of Work — The program will depend on the fabrication program and the irradiation performance and proof testing program for input to determine any cost or performance penalties

Table 8-2. Summary of Estimated Costs and Schedule for
the Fuel Material Preparation R&D Program

Task	FY CY	Estimated cost, \$ × 10 ³								Total
		<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	
		<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	
Fuel Material Preparation (Commercial Development)										
Feed material preparation		95	235	285	235	260	210	210	--	1530
Fuel fabrication develop- ment		95	235	285	235	260	210	210	--	1530
Economic evaluation		--	--	--	--	--	--	--	180	180
Total		190	470	570	470	520	420	420	180	3240

to be assessed to either type of fuel. The input used will consist of fabrication and feed material preparation costs (per unit of finished fuel), material costs, capital and plant costs, and any reactor operating penalties associated with lower irradiation performance of one of the types of fuel. Alternatives available to circumvent the operational penalties will be investigated, and their feasibility will be evaluated by calculations. Detailed work will include the establishment of material preparation and fabrication flow sheets (based on the fabrication development program), plant capacities selected, and the calculated cost of preparation, fabrication, scrap recovery, and reconversion. These costs, in addition to the operating costs in the reactor complex, will be used to determine the fuel cycle costs for pellet and particulate fuel.

Major Facilities and/or Equipment Required — No facilities or equipment will be required.

Budget Estimate — Manpower: \$180,000

Schedule — This project, because it depends on the results of the fabrication development and irradiation performance and proof-testing projects, must begin after the former is completed and after required results are obtained in the latter. (See Table 8-2.)

8.2.2. Fuel Assembly Fabrication (Commercial Development)

8.2.2.1. General

Objective — The objective of this program is to develop the technology necessary for the production of fuel pins and fuel assemblies at the levels required for the LMFBR demonstration plants and the large commercial plants that will follow. The program is dependent on the design and development of the fuel assembly details that will be accomplished in programs on core design,⁸ basic fuel material development,⁹ and on the development of fuel material preparation processes (section 8.2.1). This task includes particular consideration of the fabrication of cladding material and the necessary procedures and controls to permit quantity, quality, and safe production and assembly of fuel pins. Of major concern will be the parallel efforts required to advance the quality control, inspection, and quality assurance techniques.

Figure 8-1. Fabrication Operations Flow Sheet for the Oxide Pin Fuel Assembly

Figure 8-2. Detailed Fabrication Operations Flow Sheet for the Oxide Pin Fuel Assembly

Summary of Current Technology — The B&W

reference 1000-MWe plant specifies coprecipitated uranium-plutonium oxide particulate fuel, vibratorily compacted into small-diameter, stainless-steel pins. Each pin also contains a measured length of depleted uranium material (both above and below the vertical length of fuel material), which is also vibratorily compacted. The pin is fitted with a positive closure at its lower end in the form of a welded end cap and with a porous vent plug at the upper end. A minimum gas void is provided directly below the vent plug.

The fuel pins are subsequently assembled into a bundle (331 pins) with appropriate pin spacers, a solid wrapper can, and adapters at both ends. A complete description of the reference assembly is given in the reactor core description.¹⁰ Numerous fabrication and assembly procedures are to be considered; they are shown on the flow sheets given in Figures 8-1 and 8-2. Figure 8-2 shows which operations will require remote techniques.

The current technology available on cladding, vent design, pin assembly, and fuel assembly is discussed in sections 6 and 7. The emphasis was mainly on support of the core design and materials development programs, and most of the information given is for laboratory and pilot-line quantities. Commercial technology and techniques for cladding production and assembly work has generally not kept pace. Generally speaking, commercial fabrication has been limited to that developed for LWR fuels; in most cases the materials, equipment, techniques, quantity, and quality of production are not advanced enough to satisfy LMFBR requirements. The major role of this program is to develop the current technology to the level of perfection that will be needed for LMFBR demonstration and commercial plants.

Although major developmental efforts are necessary to advance technology to the commercially competitive levels required for the LMFBR fuel assembly, the technology and operating experience that have been gained from fuel assembly operations already developed will be very valuable. Wherever possible, B&W will make maximum use of existing equipment, facility design, and techniques. The major complication to be considered is the presence of the plutonium recycled fuel and, therefore, the existence of excessive radiation

(greater than acceptable levels). The experience with facilities such as the EBR-II Fuel Cycle Facility is especially applicable. Also important will be the experience being gained from the development of plutonium fuel pilot-line equipment and facilities. Development results from the planned FFTF fuel assembly programs will also be of major importance.

Technical Justification — Much of the ground work required to advance fuel assembly fabrication technology to the production and quality levels required for the long-range LMFBR program will be established in near-term programs of detailed design and development. This basic work can influence the requirements of fuel assembly fabrication programs considerably. The details of remote and semiremote procedures and equipment can be quite costly, but—more important—this cost could be pushed beyond practical limits if serious consideration is not given to the problems during early design and development stages.

At the present time existing programs and programs defined in the Program Plan¹¹ apparently do not place sufficient emphasis on the problems of the remote fabrication of fuel assemblies during the design phases. Also important are the problems that result from other LMFBR fuel recycle requirements; major among these is that of disassembly prior to fuel reprocessing. Reasonable solutions to these problems may involve revised fuel assembly details and, therefore, revised production requirements.

Relationship to Current or Planned Programs — The National Program for LMFBR plants is expected to yield valuable information pertinent to this program. However, there is no known program already in existence that satisfies these requirements.

8.2.2.2. Fuel Pin Cladding Development

Objective — The objective of this task is to develop the laboratory and pilot-line techniques and the equipment for fuel cladding production to the level required for commercial LMFBR plants.

Summary of Current Technology — Stainless steel tubing has been specified in the B&W reference fuel assembly design. The technology required for the production of this material is

relatively well-advanced; however, the material thickness specified is significantly thinner than that being produced for other reactor systems. Basic development programs have been assigned in the fuel and materials section of this report (section 7) to produce material of the quality required. Further developmental effort is required here to ensure that tubing with all the required quality characteristics can be produced.

Scope of Work — The first effort of this task will be to review and evaluate fuel pin cladding criteria as applicable to tubing production. Part of this effort will consist of researching current programs, such as FFTF, and refining the requirements as necessary. Most of the input will be derived from the core design and fuel and materials development programs (sections 6 and 7). At the minimum, the following items will be delineated:

1. Type of tubing (drawn or welded).
2. Material specifications.
3. Sheet thickness and tolerances (including flatness, thinning, width, and edge preparation).
4. Limitations on fabrication techniques, tools, and lubricants.
5. Tubing diameter and tolerance.
6. Ovality tolerance.
7. Straightness tolerance.
8. Welding specifications.
9. Inspection and quality assurance requirements.
10. Cleaning specification.
11. Quantity requirements.
12. Shipping requirements.

After criteria have been defined, cooperation between designers and tubing fabricators will be required. This will be accomplished by bid-to-specification requests. These requests will require a thorough review of the fabricator's current technology and capabilities as compared to the LMFBR tubing criteria requirements. Existing facilities, tools, equipment, procedures, and experience will be reported. Each request will require a preliminary estimate and schedule

for needed R&D and a description of facility and equipment requirements. The major result will be a bid for a pilot-line run of tubing.

On the basis of the results of the fabricator-designer joint studies, new development requirements will be defined in detail. It is anticipated that contracts will be let to at least two fabricators to produce pilot-line quantities of tubing. Special adaptation of their existing facilities and tooling will be necessary, but major emphasis will probably be on inspection and quality assurance. Because of the small quantity and rigid quality requirements, a premium price is postulated for this effort. Each fabricator will be required to report his results in detail. Deviations from specifications will be reported along with the fabricator's reasons for deviation and his recommendations for specification revision and further development requirements. The fabricators will also produce a detailed description of production-line requirements and estimates and schedules for production-line construction. Schedules and estimates for production runs of tubing will also be submitted.

All of the foregoing efforts will be evaluated. Actual and anticipated deviations from specifications will be reviewed against fuel assembly design requirements and core design effects. Coordination between all parties will result in production specifications for the LMFBR tubing. Cost estimates will be prepared for the overall economic evaluation.

Technical Justification — The need for a pilot-line run of fuel pin cladding to LMFBR specifications cannot be underestimated. From past experience we have found that many deviations from specification are often required. Most of these may be avoided by early cladding fabrication efforts which will allow sufficient time for the development of production equipment and techniques. Without early efforts, the economic goal of the LMFBR program may not be attained.

Major Facilities and/or Equipment Required — As discussed in the scope of work, we plan to use the facilities and production equipment of two tubing fabricators. Adaptation of these will be required for the pilot-line run of LMFBR tubing.

<u>Budget Estimate —</u>	
Manpower	\$ 950,000
Facilities	1,000,000
Equipment	1,500,000
Total	\$3,450,000

Table 8-3 is a summary of estimated costs and a schedule of the work planned for fuel assembly fabrication R&D.

8.2.2.3. Fuel Pin Assembly Development

Objective — The objective of this task is to develop the facilities, equipment, and assembly techniques to permit quality and quantity production of fuel pin assemblies. Considerable emphasis will be placed on the development of inspection and quality assurance technology.

Summary of Current Technology — The basis for this project will be the detailed design efforts accomplished to advance the B&W conceptual fuel pin design. Also of basic importance will be the development results of the core design and fuel and materials development programs (sections 6 and 7); these programs must produce a firm design backed by laboratory proof of principle and feasibility.

In many respects, the B&W reference pin assembly design is very similar to most fuel pins now being built. However, there are important differences that will have major influence on fabrication techniques. Among these differences are diameter, cladding thickness, cladding material, fuel material, and end fittings. The B&W reference design specifies a 0.280-inch-OD pin using 0.010-inch-thick cladding of type-304 stainless steel. The fuel material is particulate mixed $\text{PuO}_2\text{-UO}_2$, vibratory compacted to a theoretical density (TD) of 85%. Each pin also contains a 14-inch length of vibratory compacted, depleted, UO_2 blanket material at each end of the active fuel material. The bottom end of each pin is sealed with a solid end cap, and the top end is closed with a porous vent plug.

The available technology for vibratory compaction of the fuel and blanket materials is discussed in section 8.2.1. The

development programs described in that section should advance the present technology to the extent necessary to produce LMFBR fuel pins. Programs for the development of other assembly requirements have not been undertaken, however. The technology of today will be very valuable for this work, but in most cases it will not be sufficient. Especially important will be inspection and quality assurance requirements, but techniques, welding codes, assembly fixtures, and even the facilities need major developmental effort. Many remote handling methods are required because of the recycled plutonium fuel, and in most cases, existing equipment has not been designed for remote procedures.

Scope of Work — The first effort in this task will be to review and evaluate the fuel pin design to establish production criteria. Existing facilities and equipment will be assessed.

On the basis of the results of the definition and capability task above, a detailed work program will be written for the new development efforts. Major input will come from programs in core design, where test specimens will be produced. Detailed estimates and schedules will be prepared for the development of facilities, equipment, and techniques.

Engineering design efforts will be devoted to new and upgraded details of the necessary equipment. Facilities will be investigated in detail, and a pilot-line design will be developed; the cost of its construction will be estimated. Maintenance and safety requirements will be studied, as will the economic aspects of the work. Related methods and problems and the feasibility of pilot-line scale-up to the full production facility will be determined. The completion of this task will produce the specifications necessary for construction of the facility and the fabrication of detailed components.

The results of the preceding tasks will determine whether a new facility will be built or an existing one adapted; the latter should satisfy pilot-line requirements. The equipment, tools, instrumentation, and inspection equipment required for this project will be fabricated; fuel pin parts will also be fabricated (fuel pin cladding will be supplied from the pilot run described in section 8.2.2.2).

Assembly components, such as welding apparatus and fixtures, will be perfected in research facilities. Tools for handling, transfer, and inspection will also be so developed.

After all the individual component developmental efforts have been completed to the greatest extent possible, the facility and all of its equipment, tooling, and procedures will be assembled and checked out. Pilot-line runs of the fuel pin assembly will be completed using simulated fuel material, but all other techniques of assembly, handling, inspection, and quality assurance will be tested in conditions as near as possible to actual conditions. After the equipment has been refined as much as possible, the production results will be evaluated. The specifications of assembly equipment and fuel pin design will be assessed, and performance results will be reported; new or revised specification requirements will be documented. Any requirements for further R&D will be described, and estimates and schedules for its accomplishment will be prepared.

The results of all of the preceding tasks will be evaluated to determine what revisions to the fuel pin design, if any, are required. Coordination with core design and fuel and materials programs will produce revised specifications commensurate with fabrication requirements.

Additional pilot runs will be carried out to the new specifications. Final runs will use new (and as clean as practical) plutonium-uranium material as final proof of operation and for final checkout of all inspection and quality assurance equipment.

The accomplishments of all of these efforts will be evaluated. Coordination between all groups will produce production specifications for all details of the fuel pin assembly procedures. Final cost estimates and schedules will be developed for the overall economic study.

Technical Justification — The reference LMFBR fuel pin incorporates many features which are considerably different from those common in other designs: Among these are the thinner cladding material, the smaller diameter, tighter tolerance requirements, vibratory compacted particulate fuel material, and the porous plug top

end cap; each of these features will affect assembly technology. Even more important justification for an extensive R&D program, however, lies in the requirements for remote operation and the rigorous quality and quality control requirements of the LMFBR.

Major Facilities and/or Equipment Required — Major adaptation of an existing facility is required to permit the construction of a pilot line for developing fuel pin assembly techniques and equipment.

Budget Estimate —

Manpower	\$1,800,000
Facilities	1,500,000
Equipment	<u>1,000,000</u>
Total	\$4,300,000

Table 8-3 is a summary of estimated costs and a schedule of R&D work planned for fuel assembly fabrication.

8.2.2.4. Fuel Assembly Development

Objective — The objective of this task is to develop the technology required for fuel assembly operations. This includes the development of facilities, equipment, tooling, and techniques to permit quantity and quality production of fuel assemblies for a commercial LMFBR plant.

Summary of Current Technology — The current technology on the production of fuel assemblies has advanced very rapidly for PWR plants. As with fuel pin technology, the information is very valuable, but in most cases it is not applicable for the B&W reference fuel assembly. The use of recycled plutonium fuel will require remote techniques not now in common use. The most useful information will be that from the EBR-II Fuel Cycle Facility, and we will use as much of it as possible. Many of the reference fuel assembly details, including spacer design, fuel pins, and end fittings, are essentially completely

new items. Concentrated design and engineering effort must be completed early to support this R&D effort.

We expect to obtain additional background information from other fast reactors to support this program. The Enrico Fermi and the upcoming FFTF will be especially important. A continuing program to research these facilities should supply valuable data as this project progresses.

Scope of Work — This project effort is designed to parallel the fuel pin effort (8.2.2.3) as nearly as possible. Similar task efforts are required—they include all the review, evaluation, design, fabrication, and development steps. It is planned that the facility selected for fuel pin assembly will also be used for this project. Minimal duplication of tooling, equipment, inspection apparatus, and quality assurance work is required, although each project obviously requires some items especially designed for it. The specialties required for this project are estimated here: Of major importance are the fixtures required for assembly, which will be major equipment items and will require remote actuation. Tolerance, quality, and quality assurance are again paramount. Cost estimates will be prepared for the overall economic study.

Technical Justification — The quality and quality assurance requirements of the LMFBR necessitate the extensive development programs described. The reference fuel assembly—and for that matter any fuel assembly proposed for LMFBR use—has many unique features and relatively stringent assembly tolerance requirements. Some equipment requirements will be solved in nonremote research facilities, as proposed, but actual assembly conditions must be simulated as closely as possible before the quality and reliability of the operations can be assured.

Major Facilities and/or Equipment Required — See section 8.2.2.3.

<u>Budget Estimate —</u>	
Manpower	\$1,200,000
Facilities	500,000
Equipment	<u>1,500,000</u>
Total	\$3,200,000

Table 8-3 is a summary of estimated costs and a schedule of the R&D work planned for fuel assembly fabrication.

8.2.2.5. Quality Assurance Development

Objective — The objective of this task is to develop procedures for the quality assurance, inspection, and nondestructive testing techniques needed for quantity and quality production of LMFBR fuel assemblies.

Summary of Current Technology — Of particular value to this effort will be the inspection techniques that have been developed in the EBR-II and the procedures that evolved because of the accident at the Enrico Fermi reactor. Although the procedures and tools have been improvised in many cases, and quantity production has not necessarily been an objective, the technology developed appears to be extremely worthwhile. Also of extreme importance will be the plutonium fuel material pilot-line technology that is being developed in several facilities in this country, including B&W. Much of the current effort is base work for the production of FFTF fuel assemblies, and many of the referenced procedures, techniques, and quality assurance mechanisms are expected to be advanced during the fabrication of these assemblies. The results will probably supply sufficient knowledge and equipment for limited production of LMFBR fuel assemblies, but it is not anticipated that full development will be approached. Automated equipment, designed and proven for fully remote operation, will still be an important requirement.

Scope of Work — The work in this area must parallel and include extensive interplay with other tasks of the fuel assembly program. Close cooperation is also required with other development programs (core design, fuel and materials, safety, and the like).

The preliminary work phase will be one of concentrated review and research. In combination with similar programs to establish fuel assembly criteria, this task will establish inspection, nondestructive testing, and quality assurance requirements for each assembly phase. Proposed and existing equipment and techniques will be thoroughly reviewed, and their applicability to our requirements will be evaluated comprehensively. The availability of equipment, the R&D requirements, factors of quality and reliability, and costs will be studied in detail.

Programs for the development of individual components, component assemblies, techniques, reliability, repeatability, and maintenance procedures will be planned and carried out. This task will require the fabrication of equipment for the pilot-line programs of other fuel assembly tasks. Components will be developed first on an individual basis at research facilities, but final development will be a part of the combined program.

The results of all of these efforts will be evaluated; accuracy and repeatability will be especially important. The results will be compared with fuel assembly criteria, deviations will be documented, and additional R&D requirements will be defined. Cost estimate studies will be prepared for the overall economic evaluation.

Technical Justification — Quality and quality assurance requirements are paramount if a safe and reliable LMFBR plant is to be built. The prescribed programs must be conducted in close cooperation with the development of the fuel assembly in order to reach this goal.

Major Facilities and/or Equipment Required — Research facilities like those at the National Laboratories or the Alliance Research Center are required for the development of individual components. A major facility is required for the pilot-line proof testing; this facility is discussed in sections 8.2.2.3 and 8.2.2.4.

<u>Budget Estimate —</u>	
Manpower	\$1,500,000
Facilities	100,000
Equipment	<u>1,000,000</u>
Total	\$2,600,000

Table 8-3 is a summary of estimated costs and a schedule of the R&D work planned for fuel assembly fabrication.

8.2.2.6. Economic Evaluation

Objective — This task will provide an economic evaluation of all fuel assembly fabrication operations.

Scope of Work — This project will depend on each of the associated fabrication and quality assurance programs to assess costs and performance penalties. The input will comprise all facility, equipment, preparation, operation, and maintenance cost data acquired on the fabrication and assembly of LMFBR fuel assemblies containing recycled plutonium fuel. Estimated costs of any additional required R&D will be included. The details of the work on this project will include assembly and evaluation of these cost data and the development of preparation, fabrication, and inspection flow sheets. Plant capacities and maintainability factors will be established, and such features as scrap recovery and waste disposal will be included.

Major Facilities and/or Equipment Required — None.

Budget Estimate — Manpower: \$150,000

Table 8-3 includes a summary of the estimated cost and schedule of the work planned for the economic evaluation of fuel assembly fabrication R&D.

Table 8-3. Summary of Estimated Costs and Schedule for
the Fuel Assembly Fabrication R&D Program

Task	FY CY	Estimated cost, \$ × 10 ³										Total
		<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	<u>1978</u>	<u>1979</u>	
		<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	<u>1978</u>	
Fuel Assembly Fabrication (Commercial Development)												
Fuel pin cladding		50	200	1050	1500	550	100	--	--	--	--	3,450
Fuel pin assembly		--	--	200	300	600	1000	1000	800	300	100	4,300
Fuel assembly		--	--	--	100	300	400	800	1000	500	100	3,200
Quality assurance		--	--	--	100	300	400	600	600	500	100	2,600
Economic evaluation		--	--	--	--	--	--	--	--	50	100	150
Total		50	200	1250	2000	1750	1900	2400	2400	1350	400	13,700

8.2.3. Fuel Recycle Operations at the Reactor Site

8.2.3.1. General

Objective — The overall objective of this program is to develop the technology necessary to make possible the safe, reliable, and economical handling of new and spent fuel at the reactor site. The programs described will provide the basic data needed to design equipment, procedures, and facilities.

Scope of Work — The requirements established in B&W's 1000-MWe LMFBR reference plant conceptual design description will govern the scope of work for this program.¹² The reference plant specifies that spent fuel will be shipped in sodium-filled casks and that new fuel assemblies will utilize dry, inert-gas-filled shipping containers. Since the design does not specify that operations such as cleaning, disassembly, inspection, or secondary decay storage are carried out at the reactor site, these procedures have been delegated to the reprocessing site, and they will be discussed in that section (8.2.5). An additional reduction of scope in this area is a result of the proposed methods for loading and unloading new and spent fuel assemblies. For clarity and convenience, the R&D requirements for these procedures are discussed in section 3.2.5, Core Components Handling Equipment.

The R&D programs required for on-site fuel recycle operations are thus reduced to cover the shipping and receiving of casks, loading and unloading of containers from the transporters, and certain cask moving and storage procedures within the shipping and receiving building.

The new fuel procedures start with receipt of the transporter in the receiving area. At this point, the reference plant concept specifies that one or more new fuel assemblies will be contained in a bulk fuel transport container; each assembly is individually enclosed in a smaller container in a dry, inert-gas environment. The details of the procedures for handling new fuel are as follows:

1. Receive the shipment in the receiving area.
2. Remove the bulk container from the transport vehicle and secure it at the unloading station. (An alternate is to move the complete unit to storage.)

3. Remove the outer container. (The present container design envisions a longitudinally split tank, very similar to new fuel shipping containers now in use.)

4. Inspect each individual fuel assembly container for damage and leakage.

5. Raise the individual fuel assembly container horizontally and lower it to the guide and support rollers adjacent to the new fuel transfer tube.

6. Move the complete unit into the tube after its outer valve has been opened.

Spent fuel handling procedures commence at the point at which the inner shipping cask container was lowered into the shipping cask. At this time, all procedures have been completed to ensure that the spent assembly (one per container) has been properly sealed in the inner container. The cask is located directly beneath the spent fuel cell in the transfer tunnel, and it has already been disconnected from the transfer valve. In detail, the following procedures will now be considered:

1. The shipping cask is moved in the transfer tunnel.

2. Install the cask closure lid and perform final leak detection and radiation monitoring.

3. Operate the shielding doors that separate the transfer tunnel from the shipping area.

4. Move the shipping cask into the shipping area.

5. Load the shipping cask onto the transport vehicle.

6. Maintain cooling of the shipment during all the preceding operations, and transfer the cooling system power source to the transport vehicle's system.

7. Perform final inspection and check out all systems before transport from the site begins.

Summary of Current Technology — Apparently, very little new technology is required for fuel recycle operations at the

reactor site. The major problem will be to define basic fuel information. Other problems are caused primarily by the presence of plutonium fuel, the use of the sodium-filled shipping cask, and the cooling requirements imposed by the high decay heat load generated by the spent fuel. Some extension of state-of-the-art technology is required, but in general, detailed design, mockup, modeling, and testing should solve all problems. A thorough safety and economic analysis must be performed to meet the overall objectives of the program.

There is no precedence for the specific combination of operations proposed in the reference design. All equipment, procedures, and facilities are state-of-the-art, but on the basis of basic data results, they could require adaptation for remote or semiremote operation and therefore require some development.

8.2.3.2. Development of Basic Fuel Information

Objective — The objective of this task is to develop the basic new and spent fuel data needed to establish criteria for the design and development of the facilities, equipment, and procedures for fuel recycle operations at the reactor site.

Summary of Current Technology — Information on fission-product concentrations, energy spectra, and fission product decay heat rates is presently inadequate and must be refined to improve the basic data for use in the design of handling equipment. The need for this information is discussed in the LMFBR Program Plan and is exemplified in the trade-off study on shipping casks which was performed earlier in the Follow-On Study.¹³ Although the major deficiency is in information on spent fuel, new fuel data are also important. In this area, recycled fuel information is needed to permit the design of handling procedures and the new fuel shipping container.

Scope of Work — The major effort required in this task is the evaluation and possible extrapolation of new and spent fuel data for use in establishing criteria for the design and development of fuel handling equipment. Interfaces with other areas of development are apparent, as shown below.

1. Basic fuel data (e.g., fission product activity levels, energy spectra, and decay heat generation rates in spent fuel at the burnup, specific power, cooling time, and neutron energy) of interest to B&W's LMFBR design will be established in physics programs.¹⁴

2. Core design programs will establish basic assembly design data, i.e., vented or non-vented fuel pins.⁸

3. Plant design programs will determine overall design technology, safety, economics, and such related factors as plant fabrication and maintenance.¹⁵

As indicated, both vented and non-vented fuel assembly designs must be considered, as well as several possible variations of fuel material, density, and burnup. Worst-case data must be defined for the particular operations in question; the goal will be the definition of the criteria for the design and development of the equipment, procedures, and facilities.

In the area of new fuel handling at the reactor site, it is proposed that shipping containers be moved into a segregated corner of the shipping and receiving building. The requirements for remote or semiremote operations and equipment will be established on the basis of information obtained for recycled LMFBR fuel. The criteria to be developed are those relating to shielding of the storage area, isolation or shielding of the building itself, handling-time limitations, the need for cooling, criticality factors (need for spacing of stored containers), and safety.

The procedures for handling spent fuel at the reactor site have already been discussed. The required criteria are those for isolation or shielding of the building, the shielding door (between building and transfer tunnel), limitations on installing the shipping cask lid, the need for a guaranteed power source for the shipping cask cooling system, criticality and fire hazards, and overall safety.

Technical Justification — The lack of basic fuel information for use in the design and development of equipment, procedures, and facilities for fuel handling is a well-recognized problem. This has been discussed in the Program Plan, with emphasis on near-term development. The B&W reference design handling systems are specific

for that individual reactor and are not in any way standardized; this situation will necessitate programs that are somewhat slanted to secure specific information for the reference design. This is not to say that no basic fuel information will be secured in programs such as Fermi, EBR-II, FFTF, or others, but rather that the scope of work defined here will still be required to evaluate and extrapolate this information for our specific design. In other words, detailed information, as it applies directly to the operations in question, must be known in order to attain the overall objective of safe, reliable, and economical fuel handling.

Relationship to Current or Planned Programs — As discussed under "Technical Justification," the scope of work in this program is required for the specific B&W fuel recycle operations at the reactor site. There is no known program—either current or planned—that will accomplish these results, basically because the requirements are for the specific and individual design. Therefore, B&W (and probably each of the designers) requires this task in order to set criteria for equipment design and development.

Major Facilities and/or Equipment Required — None.

Budget Estimate — Manpower \$85,000.

Table 8-4 includes a summary of estimated costs and a schedule of R&D work on fuel recycle operations at the reactor site.

8.2.3.3. Development of Handling Equipment

Objective — This task will be carried out to develop the equipment, procedures, and facilities required for fuel recycle operations at the reactor site. Successful completion of this program will depend on establishing criteria in task 8.2.3.2.

Summary of Current Technology — The overall requirements have already been discussed in some detail. It was pointed out that detailed design, mockup, modeling, and testing would probably solve all of the problems that exist. The reference design requires a specific combination of operations, and, hence, specific equipment and facility designs. Other than this combination, however, and any new requirements that may be determined as the design and operating criteria are formulated, all of the B&W reference design is state-of-the-art technology. The criteria that have been postulated must be considered, and programs must be planned to ensure that the equipment will be

developed to the safety, reliability, and economical levels needed for the LMFBR commercially competitive plant.

The Program Plan reviews core components handling equipment in considerable detail; the systems that are discussed (EBR-II, Fermi, Dounreay, Hallam, and others) have established useful technology. Equally important background data are available from the design and successful operation of water reactor plants and numerous hot-cell facilities. Especially applicable will be equipment and procedure technology that is adaptable to the remote movement of a heavy shipping cask or to the remote installation of the heavy cask lid. Also important will be the data on guaranteed power supplies for cooling systems.

Scope of Work — A review of design requirements relative to performance, environment, reliability/availability, operation, safety, maintenance, design standards, and interfaces with other systems makes up the first phase of this work. In parallel, a literature survey will be conducted to determine what equipment and procedures should be considered in the design phases. A preliminary design and engineering analysis will then be carried out to adapt the equipment. In addition, all mockup, modeling, and testing requirements will be determined. The program is divided into three tasks, as follows:

Task 1 — New Fuel Handling Requirements: In this task the criteria for handling will be reviewed thoroughly, and the functional requirements will be set for all equipment and procedures. Through the literature survey and preliminary design stages, the equipment components will be selected and combined into functional units. Minimal testing will probably be required for new fuel handling, with the main emphasis on design. The important areas will be the removal of the bulk container from the new fuel shipment and unloading of the individual containers; it is postulated that this will be a remote procedure. All equipment envisioned is state-of-the-art, but its remote use, operated from the large crane in the shipping and receiving building, will require some modeling and testing to perfect techniques and increase the confidence level. Remote inspection of the individual containers for leakage may also present some problems; techniques for remote

inspection will be developed by model testing. Other procedures, such as loading the container on the guide and support rollers and moving the container into the transfer tube, should not require modeling or testing. For these systems, detailed design and proof testing at the time of installation should solve any problems.

Task 2 — Spent Fuel Handling: In this task the criteria review, literature survey, and preliminary design will be conducted to permit the selection of components and their combination into functional units. The operations of moving the shipping cask, operating the shielding door, and loading the cask onto the transport vehicle will probably not require modeling or testing. The detailed design and proof testing done during the building stages should solve any problems. The remote installation of the cask lid, using equipment in the transfer tunnel, will require some testing, however. Of particular importance will be the lid seal and the positive fastening requirements. The interface with cask design (section 8.2.4) is obvious, but the procedures for accomplishing these important tasks will be developed here. Inspection techniques and related instrumentation and other equipment will be paramount. The use of a full-size mockup of the cask lid and the upper cask section is proposed. Tests can be conducted in a clean facility where the prototype equipment and the mockup can be installed and operated. The procedure will be fully developed, with major emphasis on reliability, availability, safety, and maintenance.

Task 3 — Economic Evaluation: The on-site fuel recycle operations discussed in this section are relatively minor contributors to capital cost and appear to have only small effects on fuel recycle costs; however, these effects must be considered. This evaluation will provide detailed data on the economics of equipment and facilities developed, and new procedures will undergo time studies—the results will be fed into overall fuel recycle cost studies.

Technical Justification — The need for these programs should be apparent. Once again, the goal is to develop specific systems for the individual reactor design. The knowledge obtained will, of course, be of general benefit to all reactor designers, especially those who may propose to ship spent fuel in sodium-filled casks.

Table 8-4. Summary of Estimated Costs and Schedule
for R&D on Fuel Recycle Operations at the
Reactor Site

Task	FY CY	Estimated cost, \$ $\times 10^3$				Total
		<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	
		<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	
Fuel Recycle Operations at the Reactor Site						
Development of basic fuel information		85	--	--	--	85
Development of handling equipment						
1. New fuel		30	90	70	--	190
2. Spent fuel		45	95	90	120	350
3. Economic evaluation		--	--	20	20	40
Total		160	185	180	140	665

Relationship to Current or Planned Programs — Many current and planned programs may be developing components or even partial systems that will be useful to this project. Such development information will be researched in the preliminary stages of the program.

Major Facilities and/or Equipment Required — This program requires a miscellaneous R&D facility to develop the cask lid closure, sealing, and inspection equipment and procedures.

Budget Estimate —

Manpower	\$310,000
Facilities	50,000
Equipment	<u>220,000</u>
Total	\$580,000

Table 8-4 is a summary of estimated costs and a schedule of the R&D work planned for fuel recycle operations at the reactor site.

8.2.4. Fuel Shipping

8.2.4.1. General

Objective — This program will provide the information needed to design the shipping casks and to establish the procedures required for safe, reliable, and economical shipment of new and spent LMFBR fuel assemblies. Safety and economy are the two major nontechnical factors that affect shipping cask design and influence the procedures followed during shipping. Safety considerations require adequate protection from radiation, avoidance of accidental criticality, and reliable containment of radioactive materials during both normal and accident transport conditions. Economy requires optimization of the conditions established for safe shipment of fuel in order to minimize the effects on fuel cycle costs.

The Interstate Commerce Commission and the Atomic Energy Commission have set up regulations for the shipment of solid radioactive fuel; their regulations define normal and abnormal

shipping conditions. The shipping cask design and the cooling system will be developed using these criteria as limits; however, if this R&D program reveals that existing regulations are either too stringent or not stringent enough, changes that reflect the findings will be recommended.

The reference plant concept calls for transporting spent fuel in sodium-filled casks and new fuel in dry, inert-gas-filled containers. Since the shipping requirements for new fuel are not as stringent as those for spent fuel, this effort will be directed mainly toward the development of equipment for shipping spent fuel.

This program is particularly dependent on the basic fuel information that will be developed by the program described in section 8.2.3, "Fuel Recycle Operations at the Reactor Site."

8.2.4.2. Technology and Shipping Cask Development

Objective — The objective of this task is to establish the technology necessary for safe, reliable, and economical shipment of LMFBR fuels. This work includes determination of all of the criteria pertinent to cask and shipping container design, carrier standards, liability coverage, shipping regulations, and international safeguard requirements, as well as their effects on fuel cycle costs.

Summary of Current Technology — Irradiated fuel has been transported from various reactors to off-site processing or examination facilities in the United States and in Europe for a number of years. With the advent and growth of a commercial nuclear power industry, the frequency with which spent fuel is transported to reprocessing sites has increased. This fuel has generally undergone long decay periods (from 6 months to 2 years) before shipment. The long cooling periods resulted largely from the lack of economic pressure under government ownership of special nuclear materials, with its attendant low leasing charges.

There has been essentially no experience to date in the shipment of fast-reactor fuels; however, preliminary studies have shown that the conditions under which LMFBR fuels will be shipped are an order of magnitude more severe than those experienced with LWR fuels.^{11,13} A prime factor accounting for this is that under private ownership of fuel there will be considerable incentive for the rapid recycle of

LMFBR fuel. Other factors that cause more severe shipping conditions are the higher specific power and higher burnups associated with LMFBR fuels.

The dissipation of decay heat from spent LMFBR fuel during shipping is a very important problem associated with spent fuel shipping. To date, all of the large shipping casks fabricated for off-site shipments have been designed to use water as the primary coolant.¹¹ Preliminary studies have shown that an intact LMFBR fuel subassembly of high burnup would require cooling for about 1 year before it could be shipped in such casks.¹¹ A recent study conducted at B&W indicates that an LMFBR fuel subassembly could be shipped in a properly designed sodium-cooled cask after a 70-day cooling period.¹³ This cooling period will have to be rechecked when we acquire more accurate data on fission product concentrations, energy spectra, and fission-product decay rates.

The shielding used in almost all domestic shipping casks is lead encased in steel (carbon or stainless). Because of potential problems with lead shielding (melting, etc.), the British use all-iron or all-steel casks, which are larger and heavier than comparable lead-shielded casks but less expensive. Investigations conducted in this country have shown that the use of depleted uranium as shielding material could reduce cask weight and enhance the safety factor.

To date, regulatory procedures have been geared more or less to the transportation of thermal reactor fuels. Present day shipping casks have been evaluated by the AEC and found acceptable to hold up to 16 megacuries of activity and capable of dissipating up to 100 kW of decay heat.¹¹ Such casks could not handle more than one assembly of closely packed LMFBR fuel elements, nor could they carry LMFBR fuel that was cooled for a period of less than 1 year.¹³

Scope of Work — The following efforts are considered essential to the establishment of required LMFBR fuel shipping technology:

1. Develop the basic heat transfer data necessary for the accurate calculation of fuel and cladding temperatures of

Table 8-5. Summary of Estimated Costs and Schedule
for R&D on Fuel Shipping

Task	FY CY	Estimated cost, \$ × 10 ³								Total
		<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	<u>1978</u>	<u>1979</u>	<u>1980</u>	
		<u>1973</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	<u>1978</u>	<u>1979</u>		
Fuel Shipping										
Technology and shipping cask development		<u>150</u>	<u>180</u>	<u>370</u>	<u>400</u>	<u>400</u>	<u>350</u>	<u>150</u>		<u>2000</u>
Total		150	180	370	400	400	350	150		2000

LMFBR fuel elements arranged in the optimum configuration in shipping casks of advanced design.

2. Carry out engineering design, fabrication, and proof testing of prototype LMFBR cask and container components. Test the validity of the conclusions reported in reference 13.

3. Conduct studies to determine the mutual impact of new LMFBR cask designs and transport regulations. Recommend changes in ICC and/or AEC regulations if revisions are warranted.

4. Conduct a study of shipping problems, including nuclear criticality.

5. Study the potentiality of in-transit accidents and their consequences, and make recommendations on safety standards and insurance coverage.

6. Continue cost studies: feed newly developed cost data into overall fuel cycle economic trade-off studies.

Technical Justification — In order to utilize the reference LMFBR fuel handling procedures, a cask is required for shipping highly irradiated spent fuel from the reactor to a reprocessing facility. The data available now are not sufficient to determine the specific requirements of such a cask nor to design one.

Budget Estimate —

Manpower	\$1,000,000
Facilities	500,000
Equipment	<u>500,000</u>
Total	\$2,000,000

Table 8-5 is a summary of estimated costs and a schedule of the R&D work planned for fuel shipping.

8.2.5. Fuel Reprocessing

8.2.5.1. General

This task will establish the technology necessary to reprocess, safely, reliably, and economically, fuel that has

been discharged from the reference 1000-MWe LMFBR, with initial emphasis on providing near-term capability for recovering fuels from demonstration reactors.

Existing commercial fuel processing plants are not designed to reprocess the reference LMFBR fuel. Included in this consideration are facilities proposed to be completed by the time fuel from the demonstration and predecessor plants is due for reprocessing. The initial loads of spent LMFBR fuel are expected to be ready for reprocessing in the late 1970's or early 1980's.*

The only commercial reprocessing plant existing today is the Nuclear Fuel Services plant at West Valley, New York. This plant has the capacity to process approximately 1 tonne of low-enriched uranium fuel daily. A recent ORNL investigation indicated that, with modifications, a fast reactor fuel capacity of 1.4 tons of 10% plutonium per day would be technically feasible in this plant; this would be equivalent to a throughput of 0.65 tonnes of reference LMFBR fuel per day.

Continuing liaison will be maintained with the NFS plant, as well as the reprocessing industry in general, to ensure that any future process improvements are appropriately recorded. At the appropriate time the NFS plant will be re-evaluated; the General Electric Midwest Fuel Recovery Plant (now under construction) will be similarly evaluated. Liaison will also be maintained with any additional commercial facilities that may be constructed in the next decade. (Both Allied Chemical and the National Lead Corporation have announced their intentions to enter the field.)

8.2.5.2. Preliminary Assessment of LMFBR Fuel Reprocessing Requirements

Objectives — The objectives of this assessment will be to determine as nearly as possible: (1) the nature and condition of the spent fuel received at the fuel reprocessing site, (2) the chemical and physical requirements of the reprocessed fuel, (3) the processing operations necessary to convert the incoming spent fuel to the finished

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* See the LMFBR Development Schedule.

product, and (4) the effects of these considerations on the reprocessing plant requirements.

Summary of Current Technology — The determination of the exact requirements for a facility for reprocessing the reference LMFBR fuel is beyond current technology. However, it is probable that existing facilities, such as the Fuel Recycle Facility at EBR-II, may be suitable for experimental programs to develop information on the problems of reprocessing LMFBR fuels on a commercial basis.

Scope of Work — The following scope of work is considered essential to the assessment of LMFBR fuel reprocessing requirements:

1. Determination of Problems Related to Spent Fuel Design Characteristics: The reference fuel is particulate $\text{PuO}_2\text{-UO}_2$ fuel contained in individual, vented fuel elements. The amenability of this design to the contemplated "shear-leach" reprocessing scheme should be verified. This design has a possible problem in the disposal of sodium. The vented design could allow sodium coolant to work into the fuel. Extra head-end procedures will have to be developed to dispose of this sodium. Unreacted sodium would pose a menace to subsequent fuel dissolution operations.

In addition to the reference fuel design, pellet fuel contained in nonvented pins (from the B&W predecessor plant) may also have to be reprocessed in the same plant. The additional problems associated with the pellet fuel design must be thoroughly studied. One such problem will be the disposal of larger quantities of fission gases from non-vented fuel during shearing operations. The head-end off-gas system design may be severely affected by this eventuality.

The effects of spent fuel design characteristics on the fuel dissolution operation should also be assessed. Observations by others have indicated that the fuel preparation methods significantly affect fuel dissolution rates; e.g., coprecipitated $\text{PuO}_2\text{-UO}_2$ dissolves more easily than mechanically blended fuel. Reportedly, the differences diminish with increased burnup. However, these differences could still effect major variations in the design of dissolution equipment.

2. Determination of Problems Related to Fuel Burnup and Length of Cooling Time Prior to Reprocessing Spent Fuel: These two factors affect the rates of both fission-product activity and heat generation in the spent fuel. These characteristics of the spent fuel, in turn, multiply its handling problems. High fission-product activity imposes stringent requirements on the off-gas handling system and on the decontamination capabilities of the solvent-extraction system. High heat generation rates will necessitate giving considerable attention to the development of rapid and reliable fuel handling equipment and sophisticated cooling mechanisms in the reprocessing systems. These problems accentuate the extreme importance of the determination of fission-product levels in spent LMFBR fuels.

3. Determination of Problems Associated With Reprocessing Failed Fuel: There will probably be two kinds of failed LMFBR fuel to be reprocessed: that which can be handled routinely in the reprocessing equipment and that which will require special head-end handling procedures. Attempts should be made to determine the percentage of failed fuel requiring special handling as well as the type of special handling provisions needed. Special cleanup and decontamination equipment may have to be developed and incorporated into the reprocessing plant design.

4. Determination of Problems Related to LMFBR Spent Fuel Shipping Methods: As previously pointed out, present fuel shipping plans are predicated on the use of sodium coolant in the shipping casks. This will present various problems attendant with handling, if fuel elements in the reprocessing plant are contaminated on the outside, and possibly on the inside, with sodium. Inert gas systems will have to be designed for all areas where sodium is expected to be encountered. Areas included will be the fuel storage cell (for storage under sodium) and the fuel element disassembly cell.

5. Determination of the Purity Requirements of the Reprocessed Fuel: At least two different kinds of considerations affect the purity requirements of the reprocessing plant product. Two in particular are the nuclear specifications of recycled fuel and the radioactivity of recycled feed material. The required fission-product

decontamination factors (DF) for recycled LMFBR fuel are allegedly quite low (10 to 20).¹¹ Heavy shielding and remote operating procedures would be required to refabricate recycled fuel decontaminated to this degree. Such plans for the long term include a remotely operated, heavily shielded fuel refabrication facility. Near-term objectives conceivably could be met by direct fuel fabricating procedures, provided that sufficiently high fission-product DFs were obtained in fuel reprocessing. This is because the higher isotope levels of plutonium in first-generation LMFBR spent fuel may be low enough that heavy neutron shielding is not required. The fuel product purity requirements must be determined for all of the concepts.

6. Determination of the Pu-U Separation Requirements of the Reprocessing System: Some of the factors that affect the plutonium separation requirements of the fuel reprocessing system are as follows: (1) composition and quantity of fuel required for core loading, (2) feed material preparation methods, (3) disposition of reprocessed uranium, (4) reprocessing schemes (core and blanket together or separately), and (5) criticality control. Economics will govern decisions with respect to many of these considerations.

7. Determination of the Processing Operations Required for Recovery of Desired Product(s): Only after these determinations (items 1-6) have been made will it be possible to determine the kinds of operations required in the reprocessing plant and their efficiencies. From the fission-product decontamination factors thus obtained, the number of solvent-extraction cycles can be determined (knowing the DF per cycle). The solvent-extraction partition cycle requirements will most likely depend on two considerations: (1) whether to process the core and blanket together or to separate them in the head-end system and process them separately, and (2) whether to partially or completely remove the plutonium from the uranium product in the solvent-extraction partitioning cycle.

Technical Justification — As of the present time, the exact requirements of a facility for reprocessing reference LMFBR fuel are not known. Therefore, a preliminary assessment of

these requirements is necessary before it can be decided in which directions to proceed in the program.

8.2.5.3. Head-End Operations

Objectives — The objectives of this task are to determine and develop the process and equipment design requirements in the head-end preparation of reference LMFBR fuel for reprocessing.

Summary of Current Technology — Existing reprocessing facilities in the United States are not designed for production handling of sodium-laden fuel assemblies wherein sodium is present inside the assembly and in the shipping cask. The most common procedure, in both existing and known proposed fuel recycle schemes is to clean fuel assemblies at the reactor and ship them in water-filled shipping containers. Reprocessing facilities have not, therefore, been assigned the responsibility for the sodium removal task, as proposed in the B&W reference concept; their tasks normally begin with decladding head-end procedures. Apparently, the only major exception is the English Dounreay reactor, where the discharged fuel assemblies are shipped to a reprocessing site in lead-filled cannisters without any on-site cleaning.

The decladding may be accomplished mechanically, chemically, or in numerous combinations of the two. The totally mechanical methods include extrusion, rolling, milling, and hydraulic jacket expansion.¹⁶ The mechanical methods generally used in combination with chemical operations are sawing and shearing. The combination processes are more widely used in this country (NFS and the Midwest Fuel Recovery Plant, MFRP, are set up for "shear-leach" operations).

In the shear-leach operations the extraneous hardware is sawed off, and then the elements are sheared (individually or in clusters) into pieces about 2 inches long. Following this the cladding and fuel are separated by dissolving the fuel away from the cladding hulls. The hulls are then sent to waste disposal, generally after rinsing.

Successful methods have also been developed for separating the cladding and fuel by dissolving the cladding away, leaving the fuel intact. Such a method involving the caustic-nitrate dissolution of aluminum cladding has been successfully demonstrated in one

of the government-owned reprocessing plants.¹⁷ Chemical decladding can be conducted on intact elements; however, chopping the elements into smaller pieces would generally speed up the subsequent dissolution operations.

For head-end dissolution of spent fuel for Purex solvent-extraction processing, concentrated nitric acid (about 10 M) is used. Irradiated thermal reactor fuels containing UO_2 and $\text{UO}_2\text{-PuO}_2$ solid solutions dissolve readily in nitric acid. However, higher nitric acid concentrations and additions of hydrofluoric acid (HF) are required to dissolve pure PuO_2 and mechanical mixtures of PuO_2 and UO_2 .¹⁸ Sufficient experience with fast reactor fuels is not yet available to predict the degree of difficulty that will be encountered with LMFBF fuels. However, it is reported that the difficulty in dissolving $\text{UO}_2\text{-PuO}_2$ mixtures increases with burnup and temperature.¹⁹

In addition to these procedures, numerous other secondary or intermediate steps are involved in aqueous processing head-end preparation, e. g., acid boiloff, feed adjustment, and clarification. The feed adjustment operations follow dissolution, and they are conducted to put the dissolved fuel solution in a condition that is amenable to the subsequent solvent-extraction operations. Operations included are acid adjustment, plutonium valence adjustment, and digestion to put certain fission product elements in their least-extractable form.

A significant additional step in the head-end treatment of LMFBF fuels will be sodium removal. If present shipping plans are followed, the reference fuel assembly (as it is removed from the cask inner container) will be contaminated with sodium. Procedures for sodium removal using moist air or steam have been demonstrated at EBR-II and Fermi.¹¹ Techniques will have to be developed and tested to cope with sodium-logged, failed fuel elements.

Scope of Work — This task includes the operations associated with shipping cask handling and removal of the fuel assembly from the shipping cask, as well as the subsequent assembly cleaning, storage, and disassembly. Disassembly includes the chemical and mechanical processes applied to the individual fuel pins to recover the contained fertile and fissile materials.

The functional and safety requirements for equipment and procedures will be determined through studies of existing technology and technological surveillance. Using the information gained during this period, design studies will be conducted to select equipment items and combine them into a functioning system. All the operations must be accomplished remotely, and a study will have to be conducted to establish the sequence in which these operations should proceed. State-of-the-art equipment is envisioned; however, its remote use will require some modeling and testing to perfect techniques and establish reliability.

Since present plans call for shipping and storing LMFBR reference fuel assemblies in sodium, equipment and procedures will have to be developed for handling the shipping cask safely and for sodium disposal. This may not be necessary in the case of early demonstrator fuel, since the cooling time at the reactor might be long enough to allow the use of other coolants. However, since the current shipping design requirements are predicated on the use of sodium coolant, it is easy to see that the early development of systems to handle this cooling medium at the reprocessing plant is mandatory. The first head-end step requiring technological development is the removal of sodium from the external surfaces of the spent fuel elements. This is true whether the elements are processed immediately upon withdrawal from the shipping cask or after storage in a sodium pool at the reprocessing plant.

In addition to this external sodium, we may also have to consider internal sodium. This could result from the use of the vented fuel elements (reference design) or from the anticipated occasional ruptured element. Complete removal of this sodium will require soaking the elements in water or some other suitable reactant. Proof tests using chemical systems developed to cope with both types of sodium problems will be required.

Since modifications of existing equipment will probably be used in processing demonstrator fuel, it seems that a wise choice would be to base the development of future mechanical head-end operations on today's most advanced technology. Thus, the maximum long-term benefits could be reaped from near-term experience. For the

long term, a reliable, continuous shear-leach operation is highly desirable. With such a system, the severe problems of criticality, heat dissipation, and gaseous fission-product control would be simplified considerably. The development and proof testing of mechanical head-end equipment for both near- and long-term objectives are required. Efforts expended to satisfy near-term objectives would be directed toward the development of auxiliary equipment conveyors, manipulators, monitors, and instrumentation to incorporate with existing mechanical head-end systems in commercial reprocessing plants. Efforts expended to satisfy long-term objectives would be concentrated on the development of an optimized shear-leach process, integrating all of the process steps that involve sodium removal shearing, off-gas control, fuel dissolution, and heat dissipation.

To optimize the dissolvers for the reference LMFBF fuel, accurate information is needed on the dissolution behavior of the fuel. Basic dissolution data must be collected on capsule specimens of the reference design which have been irradiated in the exposure conditions anticipated for the reference LMFBF fuel. Data will include dissolution rates and fission gas evolution rates as a function of various dissolvent parameters, i. e., temperatures and concentrations.

Various continuous dissolution mechanisms and equipment designs will be evaluated. Prototype test sections will be constructed, with which many of the designs and mechanisms can be verified or eliminated under cold-test conditions. Hot pilot-plant studies are also anticipated. The ultimate continuous dissolver design should be readily adaptable to either a modified head-end system for near-term application or to inclusion in the continuous shear-leach process referred to earlier for long-term application.

Few problems are anticipated in connection with the design of feed adjustment equipment; current designs should be adequate. In any event, at this stage of the dissolution cycle, there should be no discernable difference between the characteristics of fuel solutions, regardless of fabrication history. Therefore, advancements in this area of the National LMFBF Program should be easy to translate into reference plant process and equipment design data.

Technical Justification — This program is necessary to ensure that a reprocessing facility will be available to handle spent LMFBR fuel assemblies that have been discharged from a reactor and shipped in sodium without any on-site cleaning or disassembly. The problems associated with handling such fuel assemblies involve safety, performance, reliability, maintenance, and economics. Definition of the requirements is essential before designers can establish components and tests to ensure the safety and reliability of the proposed procedures.

Relationship to Current or Planned Programs — The information needed to design the components for this program will come from other programs; e.g., physics will provide criticality and radiation intensity data, fuel and materials will provide material data, sodium technology will develop cleaning methods, core design will provide fuel assemblies designed to facilitate the operation developed by this program, instrumentation and control will be responsible for supplying the instrumentation required by this program, and the components section will supply information on cooling systems and mechanisms for operation in sodium-contaminated areas.

Major Facilities Required — None.

8.2.5.4. Solvent-Extraction System Process and Equipment Development

Objectives — In this subtask, solvent-extraction system process and equipment design requirements for the recovery of reference LMFBR fuel will be developed, with initial emphasis on the development of near-term capability for recovery of fuels from demonstration reactors.

Summary of Current Technology — A summary of technology pertinent to this work is presented in the National LMFBR Program Plan (task 8.3.1).¹¹

Scope of Work — The following scope of work is considered essential to the development of solvent-extraction systems capable of recovering reference LMFBR fuels.

1. Design of a Conceptual System Based on Existing Commercial Reprocessing Plant Designs: This design should incorporate the minimum extrapolation of existing technology—only that necessary to meet near-term reprocessing requirements. The work will utilize the results of the preliminary assessment of LMFBR fuel reprocessing requirements and the evaluation of existing commercial facilities discussed in sections 8.2.5.2 and 8.2.5.1, respectively. The net effects of this design effort will be to pinpoint the process steps in need of development and the parameters to be studied, and to establish priorities.

2. Development of Optimum Solvent-Extraction Operating Schemes for Reprocessing Near-Term Fuel Loads: Conceivably, near-term loads would be reprocessed in an existing commercial facility modified as required. The various processing schemes available will be limited by the capabilities of existing equipment. Laboratory studies will be conducted to develop the optimum flow sheets that can be used in an existing facility. The following flow sheet variables will be studied: stream concentrations, relative stream flow rates, the number of contacting stages or equilibrium transfer units, contact time, temperature, and the types of reductants and methods of introducing them. The selection of parameters for these studies will be governed by such considerations as criticality limits, radiation exposure limits, the required fission-product decontamination and plutonium-uranium separation factors, and the need for plutonium valence control.

3. Development of Improved Solvent Treatment Methods: Hot-cell studies will be initiated to test the capability of existing methods for removing deleterious degradation products from severely exposed organic extractants. Methods of removing plutonium and uranium from degraded solvent will also be studied. Initial efforts will be concentrated on Purex extractants (tributyl phosphate plus kerosene-type diluents). The results of these studies may apply equally to near- and long-term objectives.

4. Development of Solvent-Extraction Systems to Meet Long-Term Requirements: Initial efforts will be directed toward the development of optimum equipment designs and solvent-extraction

flow sheets. Equipment design efforts will be concentrated on improving the performance of high-speed contactors. Since solvent degradation is directly proportional to the duration of the solvent's exposure to the highly radioactive feed solutions in the contacting equipment (all other variables remaining unchanged), the amount of degradation can be reduced by shortening the contacting time. Performance studies with simulated LMFBR solutions will be carried out to check new designs and to improve the operability of systems incorporating high-speed contactors and any other equipment innovations.

Flow sheet investigations will include studies of the new organic extractants which are more resistant to radiation, such as those mentioned in Task 8.3.1 of the National LMFBR Program Plan.¹¹ Flow sheet demonstration tests will be conducted in mini-scale equipment on the most promising extraction schemes, incorporating some of the more promising extractants.

Technical Justification — To meet near-term reprocessing requirements within the time available, an independent pursuit of all possible LMFBR fuel reprocessing techniques would be most impractical. Since near-term fuel reprocessing plans are predicated on the use of existing commercial facilities, the process techniques that warrant the maximum effort will be indicated in the prescribed initial design study.

In our judgement, the most prudent approach is also being followed in the scope of work outlined to meet long-term objectives. This position is supported by the following considerations:

1. Much of the development work that was carried out to meet near-term objectives will also be applicable to long-term objectives.
2. There are a number of new avenues for exploration that promise to yield technology of far greater capabilities than present aqueous technology.
3. Less funding is required to explore these avenues than to bring other technologies within the capability range of present aqueous technology.

4. Backup capability in government-owned reprocessing plants exists only in aqueous technology.

Budget Estimate —

Manpower	\$3,230,000
Facilities	500,000
Equipment	<u>1,970,000</u>
Total	\$5,700,000

Table 8-6 is a summary of estimated costs and a schedule of the R&D work planned for fuel reprocessing.

8.2.6. Waste Management

8.2.6.1. General

Objectives — The objectives of this task are to identify all the waste handling problems associated with reference LMFBR fuels, and to develop the best conceivable technology for the treatment and disposal of wastes generated in reprocessing reference LMFBR fuels.

Scope of Work — The B&W reference plant design, as established in the 1000-MWe LMFBR Follow-On Study, specifies on-site spent fuel handling procedures that are considerably different from those envisioned in the LMFBR Program Plan. These procedures are described in detail in the CSDD for Reactor Support Systems¹² and in abstract form in section 8.2.3 of this report. The major difference is that radioactive wastes are not generated in large quantities at the reactor site by operations like disassembly or secondary storage. Therefore, no on-site waste disposal falls within the scope of fuel recycle operations. Gaseous and sodium wastes that are generated at the reactor site are generated during spent fuel handling and cask loading in the spent fuel cell (see section 3.2.5.6). Disposal problems have been treated as part of the reactor cover gas and reactor sodium cleanup systems, although many of these problems will be solved during the following programs for reprocessing site waste disposal.

The general scope of this task covers all of the aspects of the treatment and handling of radioactive wastes generated in the reprocessing of spent LMFBR fuel. There are essentially three

Table 8-6. Summary of Estimated Costs and Schedule
for R&D on Fuel Reprocessing

Task	Estimated cost, \$ × 10 ³										
	FY 1970	FY 1971	FY 1972	FY 1973	FY 1974	FY 1975	FY 1976	FY 1977	FY 1978	FY 1979	FY 1980
	CY 1970	CY 1971	CY 1972	CY 1973	CY 1974	CY 1975	CY 1976	CY 1977	CY 1978	CY 1979	Total
Fuel Reprocessing											
Preliminary assessment of LMFBR fuel reprocessing requirements	50	100	150	150	100	50	--	--	--	--	600
Head-end operations	40	290	190	190	160	190	220	160	140	120	1700
Solvent-extraction system process and equipment development	90	580	380	380	310	380	450	310	270	250	3400
Total	180	970	720	720	570	620	670	470	410	370	5700

major types of wastes to be disposed of: gaseous, liquid, and solid. The major source of radioactivity in the wastes will be the fission products released from the fuel during reprocessing operations. Other sources will be transuranic elements, sodium (coolant used during fuel shipping) and to a lesser extent (n, α) reaction products in the cladding.

Many of the problems unique to the reprocessing wastes from B&W's reference LMFBR fuel will be characteristic of the wastes generated in the reprocessing of all LMFBR fuels, e.g., the disposal of copious quantities of fission products. Other problems uniquely related to reprocessing wastes from reference LMFBR fuel may or may not be characteristic of other LMFBR fuels; namely, the problems related to processing vented particulate oxide fuel, the disposal of contaminated shipping cask coolant (sodium), and the characteristics of wastes produced in an aqueous facility designed to reprocess reference LMFBR fuel. All of the problems will be magnified by the combination of high burnup and a short cooling time before reprocessing.

Waste disposal will undoubtedly be the greatest single problem area confronting LMFBR fuel reprocessing considerations. As stated above, the technology required to manage LMFBR wastes properly is largely a function of spent fuel characteristics and the reprocessing scheme used. Thus, the problem of determining the technology needing development has been simplified somewhat by the fact that the reference fuel design and the type of spent fuel reprocessing technology have already been selected. A distinct advantage of this approach is the fact that no compromise in the direction of effort is necessary.

As in the past, the major responsibility for developing waste management technology will be shouldered by the national laboratories. The current research and development programs are well complemented by the National LMFBR Program Plan. The plan of action for this task is to see that adequate coordination of effort is established with the programs conducted at the national laboratories (and other sites) to ensure the generation of technology meeting B&W's LMFBR waste management requirements.

Summary of Current Technology — Since its origin, the great preponderance of technology on the management of radioactive wastes has been accumulated at the AEC separation facilities and the national laboratories. In over 20 years of operation, over 65 million gallons of radioactive wastes have been handled. Almost all of these wastes have accrued from LWR wastes cooled for longer than 150 days. All of this, however, is a "drop-in-the-bucket" compared to the number of curies of fission-product wastes that will require processing in a viable LMFBR industry.

The sum total of industrial experience in re-process plant waste management has been accrued at NFS, which has been in operation for about two years. To a lesser degree, the industry has grappled with some of the problems of waste management at fuel fabrication plants across the country.

The state of the art and current work on the management of the three kinds of waste (gaseous, liquid, and solid) are well summarized in section 8 of the National LMFBR Program Plan.¹¹

8.2.6.2. Development of Waste Facility Requirements

Objective — This subtask will develop safe, reliable, and economical procedures for the treatment and handling of radioactive wastes generated in reprocessing reference LMFBR fuels.

Summary of Current Technology — Two kinds of waste are usually generated in processing spent reactor fuel in a fuel reprocessing plant—gaseous and liquid.

The predominant fission products in the gaseous wastes are xenon, krypton, iodine, and tritium. In today's plants the iodine is removed by reaction with silver nitrate in a charcoal absorber. Following this separation and concentration of the krypton and xenon, the tritium is released to the atmosphere without storage. The xenon and krypton are stored for 150 days to allow for xenon decay, and then the krypton is released to the atmosphere.

Liquid wastes fall into three classes: high-, intermediate-, and low-activity. In current practice the high-activity wastes may or may not be treated before being sent to underground

storage. At NFS they are concentrated somewhat and neutralized for storage in mild steel tanks under perpetual care; the treatment is determined by the mode of storage. Thus, untreated high-activity wastes are stored in stainless steel tanks. Millions of gallons of high-activity wastes are already stored in this manner. Obviously, more satisfactory high-activity waste treatment procedures will have to be developed for LMFBR wastes.

General Electric has elected to convert all of the radioactive wastes from the spent fuel processed in its Midwest Fuel Recovery Plant to solid form for storage in liquid-cooled vaults. Other methods of reducing high-activity wastes to solid products are being developed at the various national laboratories.

Intermediate-activity wastes are being concentrated by evaporation at NFS. An acid recovery operation is also included (necessitating less caustic for neutralization and less total volume for storage). Evaporator bottoms are added to high-activity wastes, and the condensate is added to low-activity wastes.

Low-activity wastes may be concentrated by evaporation, ion exchange, or several other comparable procedures. They may also be neutralized and discharged to rivers in controlled amounts governed by 10 CFR 20 limits.

Scope of Work — The following scope of work is considered essential to the success of this subtask:

1. Study the Waste Facility Requirements for Treatment of Near-Term Reference LMFBR Spent Fuel Reprocessing Wastes: This study should be incorporated with the near-term fuel reprocessing plant design study recommended in Task 8.2.5. In the course of this study, the disposition of fission products, plutonium, and other transuranic elements will be postulated. Utilizing the basic information obtained in Task 8.2.3 and this study, the following can also be postulated:

- (a) The degree of decontamination and holdup time required for gaseous fission products.
- (b) Shielding requirements.
- (c) The decay heat rates of the various effluent streams.

These requirements will be determined for vented fuel (reference design) as well as for non-vented fuel. Off-gas requirements are expected to be more stringent for non-vented fuel head-end processing.

2. Coordination of Activities With National Laboratories to Ensure That Studies Pertinent to This Task's Objectives are Included in Their Work: The parameters for these studies will be determined by the design study discussed above.

3. Determination of the Functional Requirements of Systems and Equipment Handling Near-Term Fuel Wastes: These requirements will be determined after an evaluation of the results of the work performed at the national laboratories. It can safely be predicted that better methods of treating fission-product off-gases will have to be developed; e.g., retention equipment for xenon and krypton and the fixation method for tritium will be improved.

4. Consolidation of Near-Term LMFBF Waste Handling Requirements With Capabilities of an Existing Commercial Reprocessing Facility: By comparing the existing plant capabilities with LMFBF waste handling requirements, the necessary modifications will be determined.

5. Determination of the Functional Requirements of Systems and Equipment for Handling Long-Term Reference LMFBF Fuel Loads: In compliance with the ultimate objective of the program (to establish economical overall fuel cycle operations), procedures in the treatment and handling of waste streams will be developed and modified to complement optimum disposal methods. Data on developments in ultimate fuel disposal methods will be derived from liaison with the national laboratories.

6. Development of Selected Promising Waste Treatment Processes: Laboratory development studies should be carried out by industry on selected techniques, primarily for the treatment of low-activity wastes. Promising new techniques include reverse osmosis and permeable membrane separation.

8.2.6.3. Development of Waste Disposal Methods

Objective — This subtask will ensure the development of safe, reliable, and economical procedures for the disposal of the wastes resulting from the reprocessing of reference LMFBR fuels.

Summary of Current Technology — As mentioned in section 8.2.6.2, the fission products common to all reprocessing plant offgas systems are xenon, krypton, iodine, and tritium. The xenon is disposed of through holdup for decay, whereas the iodine is trapped on adsorbents or reacted and buried. Finally, the krypton and tritium are released to the atmosphere according to the limits of 10 CFR 20. The long-lived fission products will be much more concentrated in LMFBR fuels. Present methods of disposal will suffice for iodine; however, better methods of disposal will be necessary for ^{85}Kr and ^3H . Krypton will be held for longer times before its release to the environment, and methods of fixation of ^3H will be improved, since release to the atmosphere will not be acceptable.

Since underground storage of high-activity liquid wastes is becoming less feasible, many methods are now being developed to convert them into solids. Many studies are now in progress at the national laboratories to determine the optimum methods of conversion and ultimate disposal of the solidified wastes. Containment in geological formations has undergone intensive study.

Scope of Work — The major part of this subtask is the maintenance of a continuing liaison with the national laboratories and others engaged in development work on waste disposal. The development of waste disposal methods, especially with respect to high-activity wastes, will no doubt continue to be a major responsibility of the national laboratories and separation facilities. The main purposes of liaison will be the coordination of objectives between industry and the national laboratories and the acquisition of information for engineering design and economic studies.

To round out the objectives of this subtask, a continuing economic study should be undertaken to determine the economic advantages and disadvantages of present waste management procedures and of the new procedures that are developed as sufficient data are accumulated.

Technical Justification — As pointed out in the general discussion, many of the problems unique to the management of wastes generated by LMFBR spent fuel reprocessing may or may not be characteristic of all LMFBR wastes. However, since the reference fuel design has already been selected, it behooves B&W to investigate these problems and the development required to solve them. It is recognized that much of the development work required to solve these problems will be performed under the direction of the National LMFBR Program (at the national laboratories). A great deal of this work will be in the area of ultimate waste disposal; much of the work is beyond the scope of this task. Thus, technical liaison will be heavily depended upon in this area.

Since the reference fuel design, fuel shipment mode, and reprocessing method have all been selected, it is possible to identify much of the waste treatment technology required and the direction of efforts to acquire it. Concentration of initial efforts in this area should best serve the interests of this program.

Relationship to Current or Planned Programs — As previously discussed, the responsibility for performing the major waste disposal development work will probably fall to the national laboratories. A great deal of this work has already received considerable attention; the objective has been to develop the optimum waste disposal systems and methods. These programs are continuing and must provide near-term technology for the early LMFBR demonstration plants as well as the longer-term ultimate results required for commercial plants.

Major Facilities and/or Equipment Required — Laboratory facilities are required for the development of selected waste treatment processes.

Budget Estimate — The activities under each of the preceding work scopes are essentially study efforts coordinated with the activities of the national laboratories for two-way exchange of requirements and feasibility. The estimated costs of this work are as follows:

Manpower	\$1,600,000
Facilities	160,000
Equipment	<u>490,000</u>
Total	\$2,250,000

Table 8-7 is a summary of estimated costs and a schedule of the R&D work planned for waste management.

8.2.7. Miscellaneous Related Activities

In this task the problem areas concerning accountability, criticality, and international safeguards will be examined, and the activities required in these areas to minimize the fuel cycle costs potentially attributable to each area will be determined.

8.2.7.1. General

Accountability, criticality, and international safeguards are quite appropriately lumped together because they are closely related. Accountability is answerable to each of the other two subjects, as well as to fuel cycle economics. An example of the latter is the fact that an accountability discrepancy of 1% could result in a loss of \$800 per day for a 1000-MWe reactor containing 20% plutonium fuel burned to 50,000 MWd/tonne.* Careful accounting for fissionable materials also aids in preventing the accumulation of a critical mass. Once fissile material is designated as "safeguarded" fuel, the International Atomic Energy Agency is obligated and authorized to follow the fissile material and any bred fissile material or fractions thereof, ad infinitum. Inspectors of the IAEA maintain continuous surveillance to prevent the diversion of fissile material away from lawful, peaceful use.

8.2.7.2. Accountability

Objective — The objective of this subtask is to ensure that accurate and reliable accountability procedures are available for the various operations used to recycle reference LMFBR fuel.

* See Task 8-5.3 of National LMFBR Program, reference 11.

Table 8-7. Summary of Estimated Costs and Schedule
of R&D Work for Waste Management

		Estimated cost, \$ × 10 ³										
Task	FY	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	
	CY	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	Total
Waste management		110	110	320	320	240	240	370	190	180	170	2250
Total		110	110	320	320	240	240	370	190	180	170	2250

Summary of Current Technology — The major sources of accountability discrepancies are sampling, chemical analyses, and the measurement of weight and volume. Poor sampling is usually due to nonhomogeneity of the material being sampled. When such a discrepancy occurs across a single fuel recycle operation, it is considered a process loss. Problems in analyses often result in shipper-receiver differences. Erroneous measurements of weight or volume also result in material balance discrepancies. Improvements in the accuracy of these determinations will depend largely upon improvements in instrumentation. This is especially evident in remotely controlled systems.

Scope of Work — The following scope of work is considered necessary to ensure the best accountability procedures for recycle operations with reference LMFBR fuel:

1. Conduct a study to determine the major sources of accountability error.
2. Propose and develop new, improved analytical procedures.
3. Examine the suitability of alternate methods of obviating shipper-receiver difference, e.g., an outside analytical party.
4. Evaluate the present methods of burnup determination and recommend the most promising methods for further development.

Technical Justification — Most of the items covered in the scope of work are also covered in the National LMFBR Program.¹¹ However, B&W has a continuing vital concern in many of the problems in this area. Past experience has indicated that considerable duplication of effort is beneficial in this regard.

8.2.7.3. Criticality

Objective — This subtask will maintain surveillance over the development of criticality information and will establish the required liaison to ensure that information pertinent to all of the ramifications of B&W's reference fuel cycle operations are being generated.

Summary of Current Technology — Although the literature contains much calculated information on critical compositions for fuel mixtures of interest to the B&W program as well as the national program,^{11,16,20} sufficient experimental data on plutonium-uranium compositions is still not available. Because of these uncertainties, overconservative approaches are required in setting operating limits.

Scope of Work — The following scope of work is recommended for this subtask:

1. Conduct a study to determine which fuel-recycle operations are in the greatest need of more accurate criticality information.
2. Establish liaison with the National LMFBR Program effort to ensure that the information required to meet B&W's objectives is receiving the proper priority of attention.

Technical Justification — The scope of work outlined in section 8-5.2 of the National LMFBR Program Plan is quite comprehensive and appears to provide for the generation of ample criticality information for B&W's design needs.¹¹ The scope of work described above would coordinate B&W's efforts with the National Program and help to ensure the timely acquisition of critical information pertinent to reference LMFBR fuel recycle requirements.

8.2.7.4. International Safeguards

Objective — The objective of this activity is to determine the effect of IAEA safeguard procedures on the design, construction, and operation of the various systems and operations involved in recycling reference LMFBR fuel.

Summary of Current Technology — By international agreement, the IAEA has the authority to follow all of the operations conducted on safeguarded fissile material. The objective of this surveillance is to prevent the diversion of effective nuclear materials for other than peaceful uses. The IAEA has established regulations governing reactor operations, and work is currently underway to extend these regulations to fuel reprocessing plants and to fuel shipment.

Scope of Work — The following scope of work is planned:

1. Conduct a study of IAEA regulations to determine the potential effect of international safeguard activities on reference fuel recycle operations.

2. Establish liaison with the National LMFBR Program operations responsible for continuous surveillance of the development of International Safeguard procedures.

Technical Justification — There is no reason for concern about this subtask's overlapping National Program efforts, since the National Program will not be studying the effects of International Safeguard activities on the economics of reference LMFBR fuel recycle development. This problem will be the concern of B&W alone.

Budget Estimate —

Manpower:

1. Accountability	\$1,250,000
2. Criticality	200,000
3. International Safeguards	150,000
Total	\$1,600,000

Table 8-8 is a summary of estimated costs and a schedule of the R&D work planned for miscellaneous activities related to fuel recycle.

8.2.8. Fuel Recycle Cost Benefits

The success or failure of the R&D in the fuel recycle program plan can have a significant effect on the fuel cycle costs of our LMFBR. The three major R&D areas in the plan involve fabrication, shipping, and reprocessing, which affect over 50% of the total fuel cycle costs. Finally, the effect of uncertainties is felt in two ways—the direct cost and the fissile inventory charges associated with the time required for the operation in question.

The costs associated with the fabrication area are by far the largest of the three. Included are material preparation, pin

Table 8-8. Summary of Estimated Costs and Schedule of R&D Work on Accountability, Criticality, and International Safeguards

Task	Estimated cost, \$ $\times 10^3$										Total
	FY 1971	FY 1972	FY 1973	FY 1974	FY 1975	FY 1976	FY 1977	FY 1978	FY 1979	FY 1980	
CY	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Total
Miscellaneous related activities											
Accountability	150	150	150	150	150	100	100	100	100	100	1250
Criticality	60	30	30	30	30	10	10	--	--	--	200
International safeguards	30	30	30	30	10	10	10	--	--	--	150
Total	240	210	210	210	190	120	120	100	100	100	1600

fabrication, assembly fabrication, and testing and inspection. For simplicity this discussion will treat only the uncertainty in the total fabrication cost, rather than those of the individual components.

The relationship between the fuel cost uncertainty and the uncertainty in fabrication cost is shown in Figure 8-3. For a core fabrication cost uncertainty of $\pm 50\%$ (i.e., \$135 to \$405/kg, (see BAW 1328, Vol 1), the corresponding fuel cycle costs range from 0.56 to 0.95 mill/kWh, which is equivalent to a variation of $\pm 26\%$ around the nominal value of 0.755 mill/kWh.

The effect of a fabrication time variation on fuel cycle costs is also given in Figure 8-3. For the reference core, the time required to fabricate one third of the core and radial blankets was assumed to be 6 months. If the fuel is purchased as needed, the average residence time at the fabrication plant for the entire fissile loading would be 3 months. A $\pm 50\%$ uncertainty in effective fabrication time, therefore, would range from 1-1/2 to 6 months; the corresponding range in fuel cost is $\pm 2\%$.

The uncertainties in shipping costs arise from a number of sources, many of which are described in the shipping cask trade-off study.¹³ In brief, the factors are decay storage time, type of shipping cask, freight and insurance costs, and shipping distance. Obviously, many of these factors cannot be resolved at this time. The effect of variations in shipping cost (\$/kg) and shipping time are shown in Figure 8-4. The similarity of the curves is coincidental. Since shipping costs are a small fraction of the total fuel recycle cost, an uncertainty in this area has little effect on the total.

Storage of spent fuel prior to shipping is closely related to the shipping time and to the reprocessing costs, which will be discussed later. If the storage time is short, the spent fuel must be shipped in sodium. If water shipping is required, then much longer storage times are required. Similarly, the reprocessing costs will be affected by the activity of the fuel when it arrives at the reprocessing plant. The limiting factor tends to be the amount of gaseous fission products which can be released to the atmosphere.

Figure 8-5 shows how fuel cost is affected by varying the spent fuel decay storage time. If a full year was required, because of

either shipping or reprocessing considerations, then the fuel cycle cost could increase by as much as 0.08 mill/kWh.

Reprocessing charges are incurred on a rate-time basis. The reprocessing time is determined by the fissile enrichment, the batch size, and the plant throughput capacity. The rate (in \$/day) is a function of the type of reprocessing used, the activity of the fuel, and the mode of shipping; i. e., if the fuel assemblies are shipped in sodium, then the head-end costs will probably be higher than if gas or water were used. It is obvious, then, that reprocessing costs will depend on the spent fuel storage time, the type of shipping, and the particular LMFBR design under consideration.

Two modes of variation in the reprocessing costs are considered and presented in Figure 8-6. In one case, the unit reprocessing cost is assumed to be a constant \$30,000/day, and the throughput (and hence reprocessing time) is varied. An additional penalty is incurred because of the greater pre-inventory charges associated with the longer reprocessing time. A similar curve showing the variation of reprocessing cost as a function of unit reprocessing costs at constant throughput is also given in Figure 8-6.

The interdependence of the elements of the fuel recycle process clearly indicates the need for an integrated approach in the development of an optimized fuel cycle.

Figure 8-3. The Effect of Fabrication Costs on Fuel Cycle Cost

Figure 8-4. The Effect of Shipping Costs on Fuel Cycle Cost

Figure 8-5. The Effect of Spent Fuel Storage Time on Fuel Cycle Cost

Figure 8-6. The Effect of Reprocessing Costs on Fuel Cycle Cost

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9. PHYSICS

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9. PHYSICS

9.1. Introduction

The objectives of the work in physics technology are to develop and test the calculational methods required to determine the nuclear performance characteristics of the LMFBR accurately. Although we could use available reactor physics methods, the resulting design would necessarily be conservative because of the many uncertainties in the existing methods. These uncertainties would be reflected in a design which would deviate significantly from the optimum.

A typical calculational uncertainty, which has widespread repercussions, is that associated with the prediction of the initial k_{eff} of the reactor. Current calculational methods can predict the initial k_{eff} of a PWR to $\pm 0.01 \delta\rho$. This uncertainty is typically equivalent to a $\pm 3.3\%$ in enrichment and approximately $\pm 10\%$ in lifetime.

With existing LMFBR calculational methods, the uncertainty in the prediction of the initial k_{eff} is in the range of ± 0.02 to 0.03 , which is equivalent to ± 4 to 6% in fissile enrichment. With this uncertainty, the beginning-of-cycle k_{eff} of the B&W reference LMFBR, which is nominally 1.01 , might actually range from 0.98 to 1.04 . Even if the uncertainty is reduced to ± 0.01 , the reactor would be barely critical if the uncertainty were on the low side. In this case, the lifetime would be reduced by 100% . The nominal enrichment could be increased by 1% to ensure that the minimum k_{eff} be at least 1.01 . However, if the uncertainty were actually $+0.01$ the maximum k_{eff} would be 1.03 . It is clear that deliberate overloading to ensure that the k_{eff} will be at least 1.01 is unfeasible because of the possibility that the maximum k_{eff} will exceed the available control rod worth. With a ± 0.03 uncertainty, the maximum possible k_{eff} is 1.07 . Overloading the core will increase the fuel costs and decrease the breeding ratio and, hence, increase the doubling time.

Table 9-1. R&D Cost and Schedule: Summary of Physics
Technology Area (9.2)

		Cost, \$ × 10 ⁶															
		Fiscal year															
Task	No.	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	Total
Theoretical methods development	9.2.1	1.4	1.9	2.4	2.8	3.0	3.0	2.6	2.1	1.5	1.2	1.0	0.9	0.9	0.8	0.8	26.3
Experimental support of theoretical methods development	9.2.2	3.1	4.7	5.7	6.2	6.5	6.2	5.8	5.0	2.0	1.9	1.9	1.7	1.7	1.3	2.5	57.3
Design studies	9.2.3	0.6	1.0	1.4	1.6	1.6	1.5	1.3	0.9	0.5	0.4	0.4	0.6	0.8	1.0	1.0	14.6
Shielding	9.2.4	0.7	0.8	0.8	0.9	0.9	0.8	0.8	0.6	0.3	--	--	--	--	--	--	6.6
Total		5.8	8.4	10.3	11.5	12.0	11.5	10.5	8.6	5.1	3.6	3.3	3.4	3.4	4.3	4.3	104.8

Uncertainties in the calculated reactivity coefficients could require even greater modification to ensure acceptable performance. These modifications invariably result in less economical performance.

The incentive for reducing the calculational uncertainties associated with the design of an LMFBR is clear. The following sections describe the R&D program designed to achieve this goal; wherever possible the cost benefits of the R&D will also be discussed.

Physics R&D is divided into four main categories or tasks, the first three of which have a profound effect upon each other. The first task, theoretical methods development, includes the formulation of new calculational techniques and the development of computer codes. Experimental support of theoretical methods constitutes the second task. In this task, basic cross-section measurements will provide cross-section libraries for computer codes. Clean critical experiments that can be used to bench-mark calculation methods, while at the same time providing additional basic data, will be performed. Just before the plant is licensed, an engineering mockup critical experiment (duplicating the heterogenities of the reference core as closely as possible) will be performed to verify the validity of the design methods for licensing purposes.

In the third task, continuing design studies will be undertaken so that the reference designs can continually evolve by taking advantage of the progress in the first two tasks and in the other technology areas. In addition, these design studies will provide guidance for the theoretical and experimental R&D task and for the other technology areas. The principal effort in this task will involve fuel management studies, which will result in the development of refueling schemes that combine economy with acceptable safety characteristics.

The final task includes all shielding R&D. In general, these methods and experiments are different from those which would be encountered in the other physics tasks. They are therefore presented separately. Table 9-1 is a summary of the physics R&D cost and schedule; the four tasks are listed, along with the report part that describes each one.

9.2. R&D Requirements

9.2.1. Theoretical Methods Development

The theoretical methods development task provides the tools with which the reactor physicist analyzes LMFBR designs; three major subtasks comprise the task: The first of these, static physics, involves the development of methods for computing steady-state parameters, such as power distributions, breeding ratio, reactivity coefficients, and the like. The second subtask, reactor kinetics, covers the transient behavior of the reactor following various perturbations. The third subtask is the development of methods that will permit different fuel management schemes to be analyzed on a more-or-less automated basis. Of course, these subtasks are closely related to each other and to the other tasks in the physics technology area.

In all cases, unless specified otherwise, the terms "methods" and "models" are used interchangeably. Also included is the effort required to program the theoretical models into the computer codes, which are the end products of the R&D. It is further assumed that all computer programming is in Fortran IV to ensure the availability of the codes at the greatest number of installations.

9.2.1.1. Static Physics

Objectives — The objectives of this subtask are to develop calculational methods for determining the following static physics parameters:

1. Fissile loading (criticality).
2. Power distribution.
3. Reactivity coefficients.
4. Burnup (MWd/T).
5. Control requirements.

These parameters are governed by the core geometry and the fuel management scheme employed. Therefore, the methods developed to analyze them should be compatible with the fuel management methods (discussed in Section 9.2.1.3).

Summary of Current Technology — Group cross-section generation from point data also presents a problem area. Perhaps the most promising development in theoretical physics methods is

the ENDF/B program. The idea of a "best set" of point cross-section data for all prospective reactor materials has tremendous potential. However, ENDF/B is not suitable for design work in its present state. The steps involved in producing a microgroup library tape suitable for use in a code like MC² are extremely cumbersome and time-consuming. ENDF/B tapes must be corrected, since the data for different isotopes (which were generated by different evaluators) are not necessarily consistent in format, and, in addition, they contain many errors. Considerable computer time is required to perform the corrections. The time required to produce, from the ENDF/B tapes, a single ENDF/B merge tape containing only the desired isotopes is also excessive. Furthermore, errors in the ENDF/B and ENDF/B merge tapes can only be detected when the MC² or an equivalent is run. Thus, the considerable calendar and computer time required just to produce a corrected MC² library, as well as the already long running time for MC², limits the usefulness of the ENDF/B system. MC² is an expensive code to run — it requires as much as 20 minutes of computer time per composition on a CDC 6600. In addition, MC² is a fundamental mode calculation; hence, it cannot treat non-asymptotic regions properly.

Resonance integral treatment is tied closely to cross-section generation. Single-level theory is well developed. STRIP,² which also treats overlapped interference phenomena, is probably the best computer code of this type. Multi-level theory is not as well developed.

Many few-group and multigroup computer codes now exist. The majority of these codes are one-dimensional, although a number of two-dimensional codes are available. BISYN,³ a 2-D synthesis code, and SCRAMBLE,⁴ EXTERMINATOR-2,⁵ PDQ-7,⁶ and GAMBLE-5,⁷ which are true 2-D codes, are all limited in groups or space points. PDQ-7 for example, allows 60,000 mesh points but is limited to five groups. SCRAMBLE and EXTERMINATOR permit more groups at the expense of mesh points. In this case, the fine structural detail is lost, and significant calculational error can result in regions having a steep flux gradient. PDQ-7 is also capable of three-dimensional operation; it has the added advantage of hexagonal geometry.

Two-dimensional transport codes are available, as are Monte Carlo codes. However, the limited mesh and group capability and generally excessive running times make codes of this type unsuitable for current design work. Their greatest usefulness is in providing bench-mark calculations in certain types of problems.

Reactivity coefficients are generally computed with either perturbation theory or by the δk_{eff} method. Numerous one-dimensional and several two-dimensional codes are currently available for survey work. The following reactivity coefficients are considered the most important:

1. Sodium Density Coefficient: Assuming that basic cross-section data are obtained in Task 2, existing methods are reasonably good as long as the density change and, hence, spectral and leakage effects are small. Both perturbation theory and the δk_{eff} method are valid.
2. Sodium Void Coefficient: The effects of total loss of sodium, especially locally, are not calculated accurately by current methods. The spectral and spatial perturbations make the applicability of perturbation theory questionable. Two-dimensional or possibly three-dimensional criticality calculation methods are required to treat the leakage component of the void effect accurately and to show local voiding properly.
3. Doppler Coefficient: Accurate knowledge of the Doppler coefficient is a vital factor in the operational and safety performance of the reactor. In addition, the basic resonance integral theory required for accurate Doppler treatment will be valuable for steady-state cross-section averaging.

Scope of Work — A central organization must take steps to ensure that the basic data on the ENDF/B tapes are correct and in a consistent format and that any updating of data is also correct. The user should not be faced with seeking out and finding errors every time a new isotope on existing tapes or new data on revised tapes are used.

Methods for preparing merge tapes and micro-group library tapes for the cross-section generation codes must be

streamlined. In its present form, the code which merges tapes requires that each basic ENDF/B tape must be read in its entirety, even though the data for only one of the isotopes on a tape may be desired.

Methods must be developed to detect the presence of errors at each step in the generation of the microgroup library to avoid the unnecessary loss of both calendar and computer time.

Two levels of sophistication in group cross section are anticipated. The most sophisticated methods, such as a spatial MC², would be reserved for the generation of basic libraries, i.e., multigroup libraries for typical LMFBR compositions with and without sodium. The code would use as source data the ENDF/B point data file.

Given the basic multigroup libraries generated with the ultra-sophisticated method, design methods that can efficiently utilize these libraries will be developed. These methods will be rapid enough to make parametric design calculations feasible, while at the same time, they will be sufficiently accurate to generate meaningful results.

Multi-level resonance models will be developed, programmed, and compared to determine whether or not this degree of sophistication is required.

Multi-dimensional, multigroup, diffusion-theory criticality codes will be developed. The usefulness of variational methods in two- and three-dimensional synthesis calculations will be determined. This type of approach is potentially much more rapid than the conventional multi-dimensional techniques. However, the conventional 2-D and 3-D codes are required to provide a bench mark for the less exact design calculations and to serve as the tool with which final design calculations are made.

Provisions will be made for the automatic calculation of reactivity effects, such as sodium void and Doppler. The technique may be based on perturbation theory or k_{eff} calculations. Second-order perturbation theory will be investigated to determine its usefulness.

Transport and Monte Carlo calculational techniques will be investigated. So far, these methods have been limited in

Table 9-2. R&D Cost and Schedule, Physics Technology
Area: Theoretical Methods Development (9.2.1)

		Cost, \$ × 10 ⁶															
		Fiscal year															
Task	No.	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	Total
Static physics	9.2.1.1	1.0	1.3	1.6	2.0	2.2	2.2	1.8	1.5	1.0	0.8	0.6	0.5	0.5	0.5	0.5	18.0
Kinetics	9.2.1.2	0.2	0.3	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.3	0.3	5.5
Fuel management	9.2.1.3	0.2	0.3	0.4	0.4	0.4	0.4	0.4	0.2	0.1	--	--	--	--	--	--	2.8
Total		1.4	1.9	2.4	2.8	3.0	3.0	2.6	2.1	1.5	1.2	1.0	0.9	0.9	0.8	0.8	26.3

both mesh and group capability because of the generally long computer time required. With the new generation of faster and larger computers, these methods may become useful in design work. A potentially very important application of Monte Carlo methods is in the investigation of heterogeneities. Since many of the fast criticals in the near future will be made up of plates, resolution of the heterogeneity effect will be a major step toward obtaining useful information.

Budget and Schedule — The budget and schedule for this subtask are summarized in Table 9-2.

9.2.1.2 Kinetics Methods

Objective — The objective of this subtask is to provide kinetics methods for use in operational and safety analyses. These codes will be coupled with reactivity feedback models to provide accurate representation of reactor transients. Multigroup space-time kinetics codes are needed for transients in which significant perturbations occur in the flux spectra and spatial distribution.

Much emphasis has been placed on the possible consequence of post-meltdown criticality and subsequent disassembly. Methods that will accurately determine the energy release of such accidents are needed.

Summary of Current Technology — The applicability of point kinetics models to reactor transient studies in certain types of perturbations, i.e., small local changes or more widespread changes, is unquestioned. Thus, for a broad spectrum of perturbation, the existing point kinetics codes are completely adequate.

However, large groups of perturbations, particularly local sodium voiding, cause significant changes in both local flux levels and energy spectra. The analysis of the transient instigated by the blockage of flow to a single fuel assembly with its subsequent voiding and possible propagation to adjacent assemblies, is perhaps the most critical or important problem facing kinetics methods development. This type of problem clearly calls for space-dependent kinetics methods. Therefore, two-dimensional and, if possible, three-dimensional methods are needed. The latter could either be a true 3-D code or possibly a synthesis of 1-D and 2-D methods.

Existing methods do not provide the multi-group capability required by the sodium void problem. The spectral effect involved with this perturbation requires multigroup treatment if the spectral and leakage effects are to be properly accounted for.

The majority of the existing fast reactor kinetics codes (FORE II,⁸ AIROS,⁹ etc.) employ the point kinetics model. Several spatial kinetics codes of limited capability are currently available. WIGLE,¹⁰ a PWR one-dimensional, two-group kinetics code, has been modified for use in the LMFBR at a number of installations (B&W,¹¹ G.E., etc.). No satisfactory 2-D multigroup, space-time, diffusion-theory code is now available. QX-1,¹² a quasi-static multigroup code under development at ANL as part of the ARC system, has not yet been released. In addition, the code is programmed for the IBM 360 series computer. Software peculiarities may also make conversion to other machines difficult. Thus WIGLE, which is limited to two energy groups, is the most generally available space-dependent transient code. Since WIGLE was written for use in PWR analyses, other limitations also exist.

Two types of codes, transport and perturbation theory, are available for use in analyzing post-meltdown criticality accidents: AX-TNT,¹³ a coupled hydrodynamic neutron transport code is limited to spherical geometry, which is not too well suited to the low-L/D cylindrical geometry now proposed. MARS¹⁴ is a two-dimensional, perturbation-theory code which makes use of distributed core material self-worths.

Scope of Work— The leading contenders for kinetics methods will be evaluated so that a more concentrated development plan can be formulated. The two methods are the multigroup quasi-static model and the multigroup space-time-diffusion model. The model selected will then be developed first in a one-dimensional form and then expanded to two and possibly three dimensions. As the code development progresses, studies will be conducted to determine the limits of its applicability for different kinds of transients. The effects of such variables as number of groups, number of space points, and length of time steps, will be investigated, so that errors associated with varying levels of detail can be determined.

Two levels of sophistication are required:

The first is one of extreme detail and accuracy; these methods are used for final design calculations and as occasional bench marks for less-detailed methods. The second level of sophistication is required for survey and preliminary design. In this type of calculation, accuracy is sacrificed for speed and economy. In general, many calculations are performed in early design stages. Therefore, the large amounts of computer time required for the sophisticated methods would make their use prohibitively expensive.

Both types of reassembly codes will be refined. A two-dimensional version of AX will be developed to serve as a bench mark for design methods. The behavior of the core during the time between initial fuel failure and the formation of a critical configuration will be investigated. This is necessary if the initiations of reactivity insertions are to be meaningful.

Budget and Schedule — The budget and schedule for this subtask are summarized in Table 9-2.

9.2.1.3. Fuel Management Methods

Objective — The objective of this effort is to develop techniques for calculating the effects of a wide variety of fuel management schemes on the physics and safety characteristics of the reactor and on the fuel cost. The ultimate goal is a two-dimensional or possibly three-dimensional point depletion code that will not only calculate the pertinent physics and safety parameters of a given core design, but will also shuffle the fuel in and out of the reactor in a variety of ways and calculate the fuel cost of each cycle. However, an adequate near-term goal is a one-dimensional, zone-depletion code that calculates the physics and safety parameters of a given core design upon command, shuffles the zones in and out of the reactor, and calculates the fuel cost of each cycle.

Summary of Current Technology — Many available codes do a combination of the calculations mentioned above. None of them, however, do all of the calculations. Examples of some of the available codes and their advantages and/or disadvantages are listed below:

1. SIZZLE-DHRG3 Package:¹⁵ A one-dimensional, multi-group, region-depletion code. Cross sections are collapsed from 68 groups, using one spectrum for the core and one for the blanket (not one for each region). There is no fuel management scheme tied into this code. Reactivity coefficients are calculated with difficulty. Fuel management and fuel costs must be done by hand.

2. PDQ-7:⁶ A two- and three-dimensional, five-group, point-depletion code. The effect of spectrum on cross sections is accommodated with curve fits. No fuel management scheme is provided. The code cannot always be used with accuracy on fast reactors, because there is no provision for group-skipping, which could give an error as large as 0.5 to 1.0% in k_{eff} . PDQ-7 is not suited for calculating reactivity coefficients conveniently.

3. 2DB:¹⁶ A two-dimensional region-depletion code. Cross sections are not calculated but are read in as constants with no provision for changing the cross sections due to spectrum changes. Also, fuel shuffling and fuel cost must be done by hand.

4. RAPFU:¹⁷ A fuel management code that calculates equilibrium fuel cycle economics using estimated mid-life number densities obtained from a two-dimensional code such as BISYN² or SCRAMBLE.⁴ There are five different fuel management schemes, but the necessity of using two codes for one job is a drawback. No provision is made for altering the flux energy spectrum or spatial distribution with burnup.

5. FARED:¹⁸ A B&W fuel management code now being written. It will contain many of the things mentioned previously as necessary for a good fuel management code. It is a one-dimensional, region-depletion code with space for up to 200 different fuel mixes. There will be provisions for a criticality search on the boron concentration and plutonium enrichment. The code will also shuffle the fuel and calculate the fuel costs for as many cycles as wanted. FARED will eventually be improved to perform two-dimensional, point-depletion calculations.

6. ASSAULT:¹⁹ A two-dimensional region-depletion code with search and fuel management options. Fuel costs can be calculated. Cross sections are not altered with burnup (rather slow).

These codes are capable of computing the individual pieces of information required for the accurate design of an LMFBFR. However, invariably, large amounts of data must be handled in going between the codes. A few-group, point-depletion code can do a reasonably accurate job of predicting burnup. However, the group detail is not adequate for calculating reactivity coefficients.

A fuel management code must do the required calculations because, although a good fuel management scheme will minimize fuel costs, it may adversely affect the physics and safety characteristics of the core from one cycle to the next. A shuffling scheme that increased the feed enrichment to get a required cycle time would have a deleterious effect on the sodium void and Doppler coefficients, while it might not necessarily affect the physics characteristics (particularly the peaking factors). On the other hand, a shuffling scheme that is advantageous from a safety point of view may increase the peaking factors, which would lower the thermal power output and give higher fuel costs. Peaking factors are particularly important to cost calculations, since some fuel management schemes offered by industry carry a warranty on the thermal energy produced.

There are many other effects. Power flattening to lower the peaking factors may increase the attainable average burnup for a given maximum burnup. However, flatter power has another safety effect in that it causes the energy release to be increased in the event of core disassembly. Also, transient fuel melting may be less widespread for an unflattened core.

Scope of Work — These problems in fuel management must be considered in deciding on a fuel shuffling scheme. Analytical methods are being developed to provide input data for plant and core design and for fuel cycle evaluations. However, while two-dimensional codes are becoming available, none of them contain the information necessary for an accurate fuel management study.

The following are just a few of the calculations that a complete fuel management code should be able to do:

1. Two-dimensional point depletion; (allow fuel movement and rotation).

2. Calculate cross sections and flux at various time steps to take into account spectrum changes with fuel depletion.
3. Criticality search on boron concentration to provide control rod withdrawal.
4. Automatic power flattening.
5. Criticality search on plutonium enrichment so that initial enrichment is enough for a complete cycle's operation.
6. Sodium void and Doppler coefficient calculation.
7. Fuel rotation and/or shuffling within the core at any time step.
8. Fuel shuffling out of the core for reprocessing.
9. As many cycles with fuel shuffling as needed.
10. Fuel economics of each cycle.

The code should also be able to handle many different types of fuel shuffling schemes. Various schemes remove portions of the core by zones or in a scattered pattern, with or without blanket material; rotate the core and blanket elements, if necessary, to provide for the even buildup of plutonium; and later replace into the core the fuel previously removed, if that is necessary to avoid a reactor control problem due to high core reactivity and the uneven buildup of plutonium.

The fuel management code to be developed in this subtask will largely automate the design process. The code will search out the initial fissile loading required to give a specified k_{eff} for a given control rod pattern, while at the same time minimizing power peaking. Depletion will begin when the initial criteria are satisfied. At specified intervals, fuel will be moved, removed, or added, as dictated by the fuel management scheme. At specified times, the sodium void, Doppler, and expansion coefficients will be computed. Fuel cycle costs will be generated for each batch of fuel based on a specified out-of-reactor history, e.g., fabrication time, decay time, etc. Special emphasis will be placed on reducing data handling on the part of the user to a minimum.

Based on sensitivity studies performed in task 3, the minimum number of groups required for depletion and for the calculation of reactivity coefficients will be determined.

The reactivity coefficient calculations may be δk_{eff} or perturbation theory. These routines may be coupled independent modules of the greater design code or integral parts of the code.

Budget and Schedule — The budget and schedule for this subtask are summarized in Table 9-2.

9.2.2. Experimental Support of Theoretical Methods

The ultimate objective of both theoretical methods development and its experimental support is to reach a state of knowledge wherein reliable design calculations can be made for fast breeder reactors, utilizing only a library of basic microscopic cross sections for the core and blanket constituents. Of course, this state will not be reached in the development stages of fast breeder technology, and integral data from experiments will be required. The experimental program outlined in this section has two objectives: The first is to provide direct measurements of nuclear properties where needed to finalize the reactor design. The second is to provide a source of experimental data that can be used through numerous comparisons to guide the development of theoretical methods. The experiments required to meet these objects are discussed in the following sections. They consist of measurement of basic microscopic cross sections; a study of clean, uniform, critical assemblies over a range of compositions applicable to the LMFBFR; a study of special features of practical power reactor designs; and also mockup critical experiments.

9.2.2.1. Cross-Section Procurement

Objective — The objective of this activity is to determine the basic microscopic cross sections and differential nuclear data for all of the significant reactor core and blanket constituents (i. e., the fissile, fertile, diluent, structural, coolant, and control materials). The ultimate goal is to create a library of point data (through cross-section measurements, evaluations, and development of nuclear theory) of sufficient accuracy and complete enough to permit the reactor designer to predict all neutronic processes without the need for supporting integral measurements. However, an adequate near-term goal is to achieve a degree of quality and completeness sufficient to interpret clean

critical experiments and reactor mockups and to extrapolate these data to realistic power reactor designs.

Summary of Current Technology — The status of technology in the area of cross-section procurement is summarized in section 9.1.1 of the LMFBR Program Plan. Sensitivity studies have been performed on a variety of LMFBR concepts to determine the effect of cross-section uncertainties on the crucial nuclear properties.²⁰ Based on this work, a master list of cross-section requirements has been prepared; it shows the accuracy required and the priority of the items. It is expected that this compilation will be revised and updated periodically as more sophisticated sensitivity studies are performed and as new and possibly contradictory data indicate that a change in priority is warranted. As an example, the current high-priority effort to resolve the discrepancy in the alpha value for ^{239}Pu in the keV range can be cited.

Substantial progress has been made in the area of cross-section evaluation and compilation. LMFBR contractors are cooperating to assemble evaluated data libraries and computer programs in standardized format (e.g., ENDF/B). If these efforts are coordinated as described in subtask 9.2.1.1, then it will soon be possible to make meaningful comparisons between calculations and analyses performed at different laboratories.

Scope of Work — We agree, generally, with the scope of work delineated in section 9.1.1 of the LMFBR Program Plan. However, it is evident from the exhaustive list of requirements that not all needs can be satisfied. This is especially so because we do not believe that the academic community can be persuaded to assume any substantial part of the load. For these reasons, we offer the following suggestions and recommendations:

1. Move tantalum to priority 1 in view of its growing use as a control material.
2. Increase support to outside groups that have demonstrated a capability and interest in practical cross-section work, such as the underground bomb test group at Los Alamos.

3. Accelerate the construction and startup of the Oak Ridge LINAC and the FNG at Argonne National Laboratory.

4. Exert stronger control over cross-section work funded by the DRDT and increase support of LMFBR-related measurements, if necessary, at the expense of more fundamental cross-section studies.

5. Increase emphasis on effective cross-section measurements by activation and reactivity measurements in critical assemblies. To avoid conflict with other priority critical experiment work, a simple, zoned critical facility could be assembled and calibrated especially for the purpose of measuring effective cross sections.

6. Review the list of cross-section needs and priorities frequently and thoroughly, giving due consideration to the inputs from reactor vendors.

7. Expedite ENDF/B data evaluation efforts and determine a means for reducing the cost and time required to use ENDF/B data in design calculations. This effort ties in closely with the development of design codes in task 1 (section 9.2.1.1).

Our final comment has to do with the tentative listing of fission products in the list of cross-section needs tabulated in the Program Plan. The cross section of the gross fission products in the keV range should be determined because of its large uncertainty and very significant effect on the nuclear properties of the reference LMFBR, such as void coefficient and breeding ratio. Differential measurements are preferred, but it is likely that an integral approach will have to be taken in this case. The subject is discussed further under "Engineering Critical Experiments."

Budget and Schedule — The budget and schedule for this subtask are summarized in Table 9-3.

9.2.2.2. Clean Critical Experiments and Analysis

Objective — The objective of this activity is to develop a fundamental understanding of the neutronics of simple, large, dilute-plutonium-fueled fast reactor systems through the study of clean, uniform, critical assemblies over a range of compositions applicable to

Table 9-3. R&D Cost and Schedule, Physics Technology
Area: Experimental Support of Theoretical
Methods Development (9.2.2)

Task	No.	Cost, \$ × 10 ⁶															Total
		Fiscal year															
		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
Cross sections		1.0	2.0	2.5	2.5	2.5	2.0	2.0	1.5	1.0	0.5	0.5	0.5	0.5	0.5	0.5	20.0
Clean criticals		1.5	1.9	2.2	2.4	2.4	2.4	2.2	1.0	0.8	0.8	0.8	0.8	0.6	0.4	0.2	20.4
Engineering criticals		0.6	0.8	1.0	1.3	1.6	1.8	1.3	1.0	0.9	0.7	0.6	0.6	0.6	--	--	12.8
Demonstration mockup		--	--	--	--	--	--	0.3	1.5	0.1	--	--	--	--	--	--	1.9
Commercial mockup ^(a)		--	--	--	--	--	--	--	--	--	--	--	--	--	0.4	1.8	2.2
Total		3.1	4.7	5.7	6.2	6.5	6.2	5.8	5.0	2.8	2.0	1.9	1.9	1.7	1.3	2.5	57.3

(a) If required.

the LMFBR demonstration and reference plants. The integral data obtained from these experiments are used to substantiate and refine basic cross sections and analytical methods, so that design calculations can be performed with confidence.

The work covered under this general title is, by definition, applicable to all LMFBR designs and concepts and supplies a body of basic technology upon which a viable nuclear industry can be constructed. Studies that mock up or simulate certain characteristics peculiar to a specific power reactor design are excluded. Experiments in this category are included in section 9.2.2.3, "Engineering Critical Experiments," and section 9.2.2.4, "Mockup Critical Experiments."

It is convenient to subdivide the activity into the following tasks: (1) parametric survey, (2) temperature effects, (3) ^{240}Pu effects, and (4) heterogeneity effects. Although the LMFBR Program Plan covers a broader scope, we restrict ourselves here to the concept favored by B&W, i.e., plutonium cycle, oxide fuel, and sodium coolant.

Summary of Current Technology — With the exception of some zoned critical experiment studies (e.g., ZPR-III, assemblies 45 and 45A²¹) and fragmentary information from Europe (ZEBRA²²), there are essentially no integral data on the properties of large, dilute-plutonium-fueled assemblies in the range of interest to demonstrator and reference LMFBR plants. Activity to correct this deficiency is underway with the construction of the ZPPR facility at Idaho and the containment modifications to ZPR-VI and IX at Illinois to permit fueling with plutonium. All facilities should be in operation before the end of 1968, so we can expect to move ahead quite rapidly in the very near future.

Relevant work on small plutonium-fueled assemblies in ZPR-III²³ and on large, dilute- ^{235}U -fueled assemblies in ZPR-VI and IX should be mentioned. These studies have served a very useful purpose in elucidating some of the problems in fast reactor physics not directly related to plutonium and in exploring the domain of plutonium concentration at the upper end of the range of our current interest. Furthermore, the operation of these facilities has provided

considerable experience and insight into the practical problems of performing experiments on large plutonium systems. The experience has also been very useful in developing and refining experimental techniques, such as those required for Doppler coefficient, sodium void coefficient, and neutron spectrum measurements. As a result of this knowledge and experience, we can expect very rapid progress when the facilities for large plutonium experiments mentioned above become operational.

Scope of Work — The experimental plan is to construct a series of clean, uniform, critical assemblies covering a range of variables encompassing those of the B&W demonstrator and reference plants. Although not delineated in detail in the LMFBR Program Plan, it is believed that the scope of work contemplated in the National Laboratory Program will be adequate for our needs. However, to ensure proper coverage of all contractor and vendor requirements, we will set down the B&W requirements in general outline.

It is not necessary (or desirable) to be more specific at this time, since, on the one hand, there are many constraints on the compositions of critical assemblies that can be constructed for the national program, and, on the other hand, minor differences in composition usually will not detract significantly from the usefulness of the results to B&W. An ideal arrangement is to create a steering committee (composed of members from the program office, national laboratory, and reactor manufacturers) which will select the specific critical assemblies to be studied under this phase of the Program Plan.

The relevant parameters of the B&W reference design that determine the base case are given below:²⁴

Geometry	Pancake
L/D ratio	0.29
Enrichment, % (Mass $^{239}\text{Pu} + ^{241}\text{Pu}$)/(Mass U + Pu)	11.6
Volume fractions	
Fuel (oxide)	0.44
Steel	0.17
Sodium	0.35
Control	0.04
Core volume, liters	6516

Because of the large core volume, it will be necessary to use the ZPPR facility for most of these clean critical experiments. However, because of constraints on fuel and its form and other practical limitations, it is not possible to construct ideal clean, critical assemblies. Therefore, the recommended strategy is to perform the basic survey work—designated parametric survey—in the most practical way possible. Then, in supplementary experiments, the effects of temperature, ^{240}Pu , and heterogeneity are explored so that these limitations in the parametric survey can be corrected for.

1. Parametric Survey: The following cases are recommended. In all cases, the core is fully blanketed, and measurements with and without sodium are envisioned. The first series of experiments covers the effect of composition and spectrum on the basic nuclear properties. The second series of experiments explores the effect of geometry and shape—particularly on the void coefficient. The combination of the two series should provide enough data to enable theoreticians to separate infinite medium and leakage properties as much as they can be separated. The compositions to be considered are listed below.

- Case I: Reference composition.
- Case II: Higher enrichment ($\approx 15\%$).
- Case III: Lower enrichment ($\approx 10\%$).
- Case IV: Reference enrichment, larger Na fraction.
- Case V: Reference enrichment, smaller Na fraction.
- Case VI: Reference enrichment, smaller steel fraction.
- Case VII: Reference composition plus BeO.

The geometries to be considered are as follows:

- Case VIII: Right cylinder geometry.
- Case IX: Annular geometry (or slab).
- Case X: Modular core.
- Case XI: Segmented core.

The following measurements are to be made in each core or in selected cores as noted:

1. Critical conditions - size and mass.
2. Sodium void coefficient - total and spatial distribution.

3. "Single-pin" Doppler coefficient - center in all cores and asymmetrical positions in selected cores.
4. Neutron spectrum, including low energy component - in center of selected cores.
5. Reaction rates leading to breeding ratio - spatial distribution in all cores.
6. Reaction rates leading to effective cross sections - in center of selected cores.
7. Fine structure reaction rates across plates and bunching experiments leading to corrections for heterogeneity - selected cores.
8. Kinetic parameters (β/l , zero power transfer function, etc.) - selected cores.

2. Temperature Effects: The motivation for general studies of temperature effects is to obtain an experimental basis for extrapolating the results of the room-temperature critical experiments just described to the operating temperature of the reference reactor. Two types of information are needed: (1) core temperature deficit, and (2) coolant (Na density) and fuel (Doppler) reactivity coefficients at operating temperature. Since the effect of temperature on the sodium coolant is a relatively straightforward density effect, sufficient information on this aspect of the problem probably will be obtained under the sodium void coefficient measurements just described. The effects of fuel temperature, interactions, and spectral changes are considerably more complex, however, and warrant more detailed study.

The experiments are to be performed in a critical facility equipped with a heated central zone. Although not stated in the Program Plan, we understand that plans are being made to modify ZPR-IX to provide this capability. If not, any of several other zoned critical facilities could be modified for the purpose.

The following cases are to be studied:

Case	Test zone	Heated buffer zone	Driver zone
I	Single pin (^{238}U)O ₂	Reference composition	Ref comp
II	Single pin (^{238}U)O ₂	Different enrichment	Ref comp
III	Single pin (U-Pu)O ₂	Reference composition	Ref comp
IV	Single pin (U-Pu)O ₂	Different enrichment	Ref comp
V	Single pin (different enrichment)	Reference composition	Ref comp
VI	Cluster (U-Pu)O ₂	Reference composition	Ref comp
VII	Cluster (U-Pu)O ₂	Different enrichment	Ref comp
VIII	Cluster (U-Pu)O ₂ (different enrichment)	Reference composition	Ref comp
IX	Cluster (U-Pu)O ₂ (different enrichment)	Different composition	Ref comp
X	Cluster (U-Pu)O ₂ (different enrichment)	Reference composition	Ref comp + BeO

In each case the change in reactivity on heating the sample in the test zone is to be measured, with the buffer zone both heated and cold. In selected cases, measurements are also to be taken at a number of intermediate temperatures. Case III will also be repeated for nonuniform heating of the sample, and, if the results are different, the program should be expanded to include this as another variable. In selected cases the Doppler coefficient will also be measured by the activation method. Finally, the neutron spectrum incident in the test zone shall be measured in several cases, both hot and cold.

3. ^{240}Pu Effects: Most of the plutonium fuel to be used in the ZPPR facility for the parametric survey described under item 1 will be low in ^{240}Pu content (about 12%), whereas the fuel that will be used

in the reference LMFBR will contain 20 to 30% ^{240}Pu (about 26% for B&W's design at equilibrium). Since the ^{240}Pu content has a very significant effect on all of the basic nuclear properties (e.g., void coefficient, breeding ratio, and the like), it is necessary to provide an experimental basis for extrapolating the critical experiments performed in the parametric survey to the case of realistic fuel composition.

The experiments envisioned here can be performed in a zoned critical assembly to minimize the quantity of high- ^{240}Pu material required; an inventory of about 100 kg should be sufficient. Any of the ANL fast critical facilities (ZPR-III, VI, or IX) could be used for these experiments, as could any one of several other privately operated facilities.

The following cases are to be studied:

Case	^{240}Pu	Test zone composition
I	Low	Reference
II	High	
III	Low	Higher enrichment ($\approx 15\%$)
IV	High	
V	Low	Lower enrichment ($\approx 10\%$)
VI	High	
VII	Low	Larger Na fraction
VIII	High	
IX	Low	Reference + BeO
X	High	

In all cases the reactivity worth of ^{240}Pu is deduced from the pairs of comparative reactivity measurements listed above. In addition, the measurements numbered 2 through 7 under the parametric survey (item 1) will be repeated for the cases of high ^{240}Pu and a few cases of low ^{240}Pu rechecked for consistency.

4. Heterogeneity Effects: The fuel in the ZPPR parametric survey experiments is in the form of thin plates, and there is no entirely satisfactory way to introduce oxygen (corresponding to the oxide fuel in the reference reactor) in a homogeneous manner. Therefore, the

parametric survey experiments will require a correction for heterogeneity that can amount to 1% or more in k.

The study is largely analytical, but experiments should be performed to substantiate the analysis wherever possible. These experiments can also be performed in a zoned facility. Following are some approaches that can be suggested:

1. Bunching experiments: The fuel, diluent, and other plates are deliberately bunched or clustered to increase the degree of heterogeneity and to test correction under adverse conditions.

2. Fine structure measurements: Foils of different materials are distributed throughout the stack of plates and are activated to produce a spatial plot of reaction rates.

3. Direct comparison: In a few cases, it should be feasible to procure a moderate quantity of homogeneously mixed material—e.g., intermixed steel and (U-Pu)O₂ powder—and make a direct reactivity comparison between a sample of the homogeneous mixture and the equivalent in ZPPR plates.

Budget and Schedule — The budget and schedule of this subtask are summarized in Table 9-3.

9.2.2.3. Engineering Critical Experiments and Analysis

Objective — The broad objective of this activity is to study a number of the more important special features of LMFBR cores that are common to most practical power reactor designs. The experiments in this category are distinct from "mockup experiments" in that they employ a more fundamental approach and are of general applicability; yet they differ from "clean critical experiments" in that the critical assemblies tend to be more complex than those used for basic neutron studies and generally explore non-asymptotic properties. Thus, the engineering criticals bridge the gap between clean criticals and mockup criticals. The need for such experiments is even more apparent when it is recognized that a very appreciable volume fraction of a power reactor core operates under non-asymptotic conditions.

Summary of Current Technology — Useful and definitive information in this area is virtually non-existent at this time.

Scope of Work — Although the definition is not rigorous, it is convenient to categorize the following types of experiments as engineering criticals: (1) blanket studies, (2) zone loading studies, (3) pin-type heterogeneity, (4) control rod studies, and (5) lifetime effects (burnup physics).

1. Blanket Studies: The ultimate objective of this work is to develop analytical methods for optimizing breeder blanket designs. Through definitive measurements, calculational procedures and input cross-section data are checked and improved methods are derived.

The general approach is to construct a two-zone critical assembly in which typical core materials are placed in a central zone and surrounded by representative blanket materials in the outer zone. Attention is focused on the non-asymptotic region around the interface between the two zones and in the blanket zone at distances far removed from the core-blanket interface.

The measurements consist mainly of flux traverses through the core and blanket with foils of ^{239}Pu and ^{238}U , from which the breeding ratio can be deduced, and other foil materials such as ^{238}U (fission), ^{232}Th , ^{235}U , gold, and manganese, which provide information on the change in neutron intensity and spectrum through the non-asymptotic region. The measurements are taken with different core and blanket compositions, with and without sodium. In some cases, plutonium will be added to the blanket to simulate end-of-life conditions and the effect on power distribution, breeding ratio, void coefficient, etc., will be determined. Additionally, the change in reactivity following the removal of a small volume of sodium is measured to determine the spatial variation of the local void coefficient in non-asymptotic regions and near boundaries.

2. Zone Loading Studies: Since it is likely that all fast breeder power reactors will be zone-loaded into two or three zones of different enrichment in order to flatten the power distribution, another source of non-asymptotic behavior will be encountered in the core around

the boundaries between the zones. Experiments similar to those described above are required to test the adequacy of calculational methods under these extreme conditions. The same general approach can be followed, but in these studies, the measurements of flux and power distribution across the interfaces will assume greater importance.

3. Pin-Type Heterogeneity: As noted earlier, there is a rather large difference in heterogeneity between the ZPPR clean critical experiments and the reference power reactor. At one extreme we have plutonium and ^{238}U metal alloy foils in plate form, sodium in plate form, and no really satisfactory way to add oxygen. At the other extreme, we have plutonium and ^{238}U in pin form, oxygen in the pins, and sodium between. The difference in heterogeneity can be responsible for as much as 1% or more in reactivity and, for such local effects as sodium voiding, much larger differences can be expected.

The experiments can be performed in a zoned critical facility. The central zone contains a cluster of about 400 reference-reactor-type fuel pins canned in sodium, equivalent to a typical fuel assembly. The outer driver zone contains ZPPR plate-type materials of approximately the same average composition. The following measurements are taken with the pin cluster in the central zone (these are compared directly with the same measurements with the equivalent in ZPPR plates replacing the pin cluster): reactivity, sodium void coefficient, breeding ratio, and the usual capture and fission rates.

4. Control Rod Studies: The control rods required for safety shutdown and shim control in fast breeder power reactors are another source of non-asymptotic behavior. In addition, a super-lattice of control rods and sodium holes (formed when control rods are withdrawn) is imposed on the uniform pin lattice, forming another level of heterogeneity.

Again, the proposed experiments are performed in a zoned critical like that described in item 3. Sufficient fuel pins are available to simulate a reference reactor fuel assembly or a control assembly in the central zone. The following measurements are envisioned: reactivity worth of candidate control rod materials (e.g., B_4C , Ta, rare earths), reactivity worth of sodium holes, local power distributions with control rods inserted to different positions, local sodium

void coefficient with control rods in and out, and local breeding ratio with control rods in and out.

5. Lifetime Effects (burnup physics):²⁵ The objective is to improve our understanding of the effects of fuel burnup (up to 100,000 MWd/T) on the nuclear properties of fast breeder reactors, particularly those properties related to safety (void and Doppler coefficients) and economics (reactivity, core lifetime, breeding ratio, and doubling time).

The first task is to determine the effective cross section of bulk fission products. This is accomplished by irradiating dilute samples of fissile material to create real fission products and making precise reactivity measurements on the irradiated samples. The pile oscillator measurements are performed in a zoned critical assembly in which the incident neutron spectrum can be varied over a wide range.

The second task is to measure the effect of fission products on the basic nuclear properties of fast systems. The first step is to prepare a number of simulated fission product mixtures comprised of readily available non-radioactive elements and isotopes (physicum) and compare their reactivities and effective cross sections against the sample of real fission products. Having determined the most suitable mixture, kg quantities are prepared so that a clean critical assembly of the type described previously (or zoned critical) can be loaded with simulated fission products. This permits direct measurement of the effect of fission products on the excess reactivity, void coefficient, Doppler coefficient, breeding ratio, etc.

The third task is to determine the adequacy of methods used to predict the changes in the higher isotopes of plutonium with prolonged burnup. This is accomplished by irradiating samples and making precise reactivity comparisons against unirradiated standards. Since the fission product effects have been studied separately, corrections can be made for their contributions, and the reactivity due to higher isotope changes can be isolated.

All of the work described in this section can be performed in ZPPR, ZPR-III, ZPR-VI, or ZPR-IX. However, if there is a schedule conflict, these experiments can also be performed in a simple zoned facility, several of which are available at installations other than the national laboratories.

Budget and Schedule — The budget and schedule for this subtask are given in Table 9-3.

9.2.2.4. Mockup Criticals for Predecessor Plant

Objectives — The objectives are to construct a close mockup of the reactor core of the B&W predecessor plant and to perform direct measurements of its nuclear properties in order to confirm the adequacy of the final design and to obtain safety-related information required by the regulatory agencies.

It is assumed that the ZPPR facility and associated fuel will be used for this experiment, for the reasons discussed below. Therefore, the qualification is added to the objective that the mockup simulate the reference core as closely as possible, within the constraints of the facility and the materials available.

The mockup critical experiments are required for two reasons: to confirm the final design of the reactor and to provide firm safety-related data to the regulatory agencies. Both requirements stem from the fact that our knowledge and understanding of fast reactor physics is not expected to be adequate by the time these data are needed to permit highly precise and reliable design calculations from first principles. Of course, this is the ultimate objective of the studies categorized as "Basic Technology" and it is probable that the need for mockup criticals will eventually vanish, as it has in the water reactor field. But in our judgment, the reactor vendor would take an unnecessarily large risk if the mockup critical is deleted from the program, particularly at this stage in the development of fast breeder technology. This decision should, of course, be reviewed periodically as the LMFBR Program Plan develops and as the difference between B&W's demonstration plant and those of other reactor vendors become clearer.

The LMFBR Program Plan does anticipate the need for mockup critical experiments and has tentatively scheduled the first such experiment for 1971. The schedule for succeeding mockups is not clearly established in the Program Plan, and it is evident from the discussions that the Program office hopes that one demonstration plant mockup will be sufficient. We share their hope that, by careful

planning of the clean critical experiment program, it will not be necessary to build a variety of mockup criticals. However, it is our opinion that if a second or third demonstration plant is worth building, it will have to be different from its predecessors—thereby requiring its own specific mockup critical. Since the B&W mockup critical will not be required until the mid-1970's, however, the matter need not be resolved for some time.

Summary of Current Technology — A number of mockup criticals have been constructed in the ZPR-III facility to support the design of relatively small fast reactors such as Fermi, EBR-II, SEFOR, and currently FFTF. These experiments have generally met their objectives. Although there is presently no direct experience with fast reactor mockups of the size and composition range of the demonstration plant, it is likely that there will be adequate experience with large critical experiments (and possibly one or more other demonstration plant mockup) by the time these mockup criticals are initiated.

Scope of Work — Although the B&W predecessor plant has not yet been designed in detail, its general features can be predicted well enough now to enable us to scope the R&D program. The demonstration plant is expected to be a scaled-down version of the 1000-MWe reference design, having essentially the same power density, core geometry, and material composition (except for enrichment). It will probably be zone-loaded into two regions of different enrichment, rather than three as in the reference design. The scope of work required for the mockup critical is outlined below.

1. Initial Criticality: Load the core to mock up the demonstration plant with all materials other than fissile present and with control assemblies in the "out" position. Then add fissile material gradually until criticality is achieved. This will probably require six to ten steps, starting with approximately one fourth of the fissile material. Calculations of the inherent neutron source for the particular fuel arrangement should be made to determine whether an additional startup source is required. The inherent source added with fuel should be accounted for in the approach-to-critical curve. One way to do this would be to plot reciprocal counts per unit source versus fissile mass. It is

assumed that the best startup detectors will have been determined in previous similar ZPPR startups. Preliminary indications of rod worth may be obtained from the startup curves with and without one or more rods in. After criticality is achieved, shim the fuel to obtain uniform loading in each zone, and obtain the critical mass. Increase the fuel loading and obtain $\Delta K/\Delta M$ at critical using a control rod plus a period measurement. Additional preliminary information on rod worth may be obtained with rod drops. Accuracy required: $\pm 1\%$ in the critical mass.

2. Source Level Determination: Determine the position of a control rod for steady-state conditions at some low power level. Increase the power by a factor of 10, and again note the control rod position for steady-state conditions. Continue and determine the power level above which the control rod position is independent of power.

3. Control Rod Worth: This experiment is conducted to determine the differential and integral reactivity worth of each control rod by the rod-bump-period method and to investigate rod interactions. Increase the loading so that the core is critical with one rod in. Pull the rod to give approximately a 100-second period, and measure the period after a sufficient wait time. Calculations of required wait times should be made before starting the experiment. Balance the core with another rod and repeat this procedure. After the rod is calibrated, repeat, using another rod to balance, to check for rod interaction. After a rod at each unique position is calibrated, increase the loading and repeat with different combinations of rods in. Continue with additional loading increments to determine rod worth as a function of fissile loading and of the position of other rods. Accuracy required: $\pm 1\%$ in rod worth.

4. Shutdown Margin: Determine the shutdown margin (stuck rod) by inserting all control rods (except the most reactive one) and increasing the fuel loading until criticality is achieved. Insert the remaining rod and shim to criticality to obtain the total shutdown margin. Remeasure the differential and integral reactivity worth of several control rods and compare with item 3. Return to the loading of item 1 and check the shutdown margin, using the pulsed neutron technique. Accuracy required: 0.5% in k.

5. Power Distribution: With the loading of item 1, perform a full map of the power distribution through the core and blanket with all control rods withdrawn. Insert selected control rods fully and measure the resulting perturbation in the local power distribution. Insert selected control rods partially and measure the axial power peaking factors to determine the worst case. In selecting the technique to be utilized in power distribution measurements, advantage should be taken of techniques developed in other work, e.g., ZPR-III, ZPPR, SEFOR. At this point, however, it is visualized that a packet of uranium and plutonium foils would be used to represent the fuel in the core and only uranium foils would be used in the blanket. The activity from a particular fission product would then be determined by gamma counting after irradiation of the foils at selected locations in the core and blanket. The relative yield of the fission product counted must be known for ^{238}U and all of the plutonium isotopes. Accuracy required: $\pm 5\%$ in power distribution.

6. Void Coefficient: Measure the reactivity change after removing a small volume of sodium against a calibrated control rod. Determine the differential reactivity worth of sodium by this method, throughout the core and blanket. Insert selected control rods and re-measure the differential sodium worth in the vicinity to determine the effect of control rods. When all the differential measurements are completed, determine the change in reactivity caused by removing all of the sodium from the core and the blanket. If the change is small, evaluate it against calibrated control rods; if large, evaluate it against the increase or decrease in fuel loading required to regain criticality. Accuracy required: $\pm 20\%$ in differential worth, $\pm 0.5\%$ in k in total worth.

7. Doppler Coefficient: Create a small void region in the center of the core and measure the single-pin Doppler coefficient, using a sample having approximately the same composition as the fuel. Repeat the measurement in a void region at the core-blanket interface; repeat the measurement in a void region adjacent to a fully-inserted control rod. Accuracy required: $\pm 20\%$ in Doppler coefficient.

8. Isothermal Temperature Coefficient: The objective of this experiment is to determine the variation in multiplication constant

with core temperature. The data are required to reduce the results of other experiments to a given temperature and will also provide another check point with physics calculations. Establish thermal equilibrium at a temperature that is typical for most experiments and determine the critical position of a calibrated control rod. Repeat at a temperature that is considerably higher than any expected in the experimental program. The increased temperature can be obtained by reducing the cooling provided for decay heat or by electric heaters. Care must be taken to establish thermal equilibrium and to reproduce any thermal gradients that may exist in the core.

9. Neutron Spectrum: Measure the neutron spectrum in the three locations specified under item 7. Use the proton recoil method above a few keV and resonance foil methods at lower energies. Considerable work has been done to develop the proton recoil method.^{26,27} When performed, these measurements should utilize the latest techniques, including those for gamma discrimination. Data should be taken at several gas pressures with both hydrogen and methane counters. Accuracy required: $\pm 10\%$ in the relative spectrum.

10. Reaction Rates: Measure the reaction rates of all materials in the mockup critical (e.g., plutonium isotopes, ^{238}U , sodium, iron) for fissions and/or captures in the three locations specified under item 7. Use the activation technique for isotopes that activate and a reactivity oscillator for those that do not. Use ^{239}Pu and ^{10}B as standards for activations and reactivity measurements, respectively. Accuracy required: $\pm 2\%$ in activations, $\pm 5\%$ in reactivity.

11. Breeding Ratio: Determine the differential, core average, and blanket average breeding ratios by measuring the ratio of captures in ^{238}U to fissions in ^{239}Pu throughout the core and blanket, using foil activation techniques. Use calculated values for the effective alpha. Accuracy required: $\pm 2\%$ in breeding ratio.

12. Kinetic Parameters: Perform a power-noise experiment, utilizing two detectors with cross correlation analysis, to obtain ℓ/β .²⁸ Also determine the zero power transfer function by oscillating a poison rod sinusoidally at several different frequencies. The mean

power level and core temperature are to be constant during the experiment. Analyze the results to obtain β and the fraction of fission neutrons produced by ^{238}U .²⁹

13. ^{240}Pu Effects: Confirm the validity of the correction for ^{240}Pu content deduced from earlier studies by replacing the plutonium in the central zone with plutonium having a high- ^{240}Pu content and by determining the change in reactivity.

14. Heterogeneity Effects: Confirm the validity of the correction for heterogeneity deduced from previous studies by bunching measurements in selected fuel assemblies and by fine structure activation measurements across the plates in selected fuel assemblies. Consider in special detail the region in and around a typical control rod assembly.

15. Control Rod Mockup: If the ZPPR control rod assembly mockup is not adequate, i.e., the measurements under item 14 reveal larger heterogeneities than predicted or other unexplained phenomena, it will be necessary to fabricate an accurate model of the real control rod assembly and replace one of the control rod assembly mockups. Then, certain of the measurements already outlined will have to be repeated in the more realistic environment.

It is assumed that the Zero Power Plutonium Reactor (ZPPR) critical facility at the National Reactor Testing Station in Idaho and the associated supply of plutonium fuel will be used for these mockup critical experiments. There is an unresolved question as to whether an adequate mockup can be constructed in this facility because of the high degree of heterogeneity introduced by the plate-type fuel forms. There is also a question as to whether a high-temperature critical facility is needed. However, these questions cannot be finally answered until the success in interpreting the clean critical experiments and the supplementary studies of temperature effects and heterogeneity effects, as previously outlined, is determined.

Budget and Schedule — The budget and schedule for this subtask are summarized in Table 9-3.

9.2.2.5. Mockup Criticals for Reference Plant

It is assumed that the nuclear properties of the 1000-MWe reference plant will be similar enough to those of the predecessor plant to obviate the need for another mockup critical. Within the scope of this report, this is a good assumption, since we are defining the R&D required for the present reference plant. However, this may be somewhat unrealistic, since there is a 7-year lapse between the demonstration plant and the reference plant. If major design innovations are made during the intervening period, another mockup critical may be required.

9.2.3. Design Studies

A continuing effort to upgrade the reference design and incorporate the results of the experimental work and more sophisticated calculational methods is required. We must know how uncertainties affect reactor performance, so that the direction and goals of the overall R&D program can continually be re-evaluated. Cost benefit studies will determine the necessity of the R&D.

Five reactor vendors are now engaged in the 1000-MWe Follow-On Studies. Each study will produce a reference design and a set of R&D requirements. The B&W reference design is of course based on existing calculational methods and basic data, i. e., current technology. There is no way to predict the methods and cross sections that will constitute 1980 technology.

Thus, the design may be conservative or optimistic, depending upon the uncertainties in existing methods, the direction and magnitude of which may not even be known. When the data from large plutonium-fueled criticals become available, a great deal of the uncertainty should be removed. The reference design should be reoptimized from time to time as new data become available. New problem areas requiring additional R&D may then be revealed.

Parameters generated during the physics design effort form an essential input to the areas of core design, instrumentation and control, safety, and fuel cycle analysis. Power distributions, combined with linear heat rate limits, determine the core size. Reactivity

coefficients are provided for use in safety analyses and control studies, and the mass balances over life provide input for the fuel cycle analyses.

The major uncertainty in the design area and one that has a profound effect on all aspects of the design is that of fuel management. For example, calculations of power distributions and reactivity coefficients are empty gestures if they are not computed for a workable fuel management scheme. The optimum fuel management scheme may indeed be a compromise between the requirements of safety- and design-oriented parameters on the one hand and economics on the other. If peak burnup becomes a limiting factor, then great emphasis will be placed on achieving the lowest practicable peak-to-average burnup. At the same time, this may tend to increase the peak-to-average power and produce unfavorable reactivity coefficients.

9.2.3.1. Sensitivity and Parametric Studies

Objectives — The objectives of this subtask are as follows:

1. Determine the sensitivity of the LMFBR to the results of R&D performed in the other physics subtasks and in the other eight technology areas. Since improved data in one area may reveal unexpected deficiencies in other areas, it is expected that R&D goals may be redefined and new R&D programs established as the design effort progresses.

2. Perform parametric surveys to ensure that the results of the R&D programs have not shifted the design optima, and to provide assistance to other technology areas. Examples of the latter include aiding in the selection of fuels and materials and providing data or fuel composition as functions of burnup for Fuel Recycle.

3. Determine how sensitive the design results are to the degree of sophistication of the design effort. This has not been possible so far because of the lack of methods accurate enough to serve as a basis for comparison.

4. Continually upgrade the reference design in the light of the results of objectives 1 and 2.

Summary of Current Technology — A considerable number of studies of this type, in varying degrees of detail, have been performed in recent years; thus, preferred ranges of reactor parameters have been established. However, these optima may change as R&D results are generated. An interesting example is the trend of LMFBFR core design away from extreme spoiling to improve the voiding characteristics. Improved cross-section data have indicated that the sodium void coefficient for a given core may be greater than originally thought. In addition, more realistic economic considerations resulted in an increased penalty associated with spoiling. Recently published designs are largely optimized in terms of economics, and the major safety emphasis is placed on partial sodium voiding, rather than on total core voiding.

Scope of Work — The B&W reference design will be the basis for the sensitivity studies. The applicable results of the other 1000-MWe design studies will be utilized, and, modifications will be made where desirable. New cross-section data and new design computer codes will be used following their verification with bench-mark calculations of critical experiments.

Studies to determine the degree of sophistication required for accurate LMFBFR design will be conducted. The number of neutron energy groups, number of spatial points, and methods of group cross-section generation will be tested. This information can then be utilized to streamline design methods.

The core will be modified in the light of R&D results in the other technology areas; sensitivity studies will, in turn, provide guidance to the other areas. Special emphasis will be placed on core design, safety, fuels and materials, and fuel recycle.

Budget and Schedule — The budget and schedule for this subtask are given in Table 9-4.

9.2.3.2. Fuel Management Studies

Objective — The objective of this subtask is to evaluate a variety of fuel and blanket management schemes and to determine the effects of these schemes on the overall reactor system. Power distributions throughout the cycle directly affect the core size.

Table 9-4. R&D Cost and Schedule, Physics Technology
Area: Design Studies (9.2.3)

		Cost, \$ × 10 ⁶															
		Fiscal year															
Task	No.	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	Total
Sensitivity studies	9.2.3.1	0.3	0.5	0.6	0.8	0.8	0.7	0.6	0.5	0.3	0.2	0.2	0.2	0.2	0.2	0.2	6.3
Fuel management studies	9.2.3.2	0.3	0.5	0.6	0.6	0.6	0.6	0.5	0.2	--	--	--	0.2	0.4	0.6	0.6	5.7
Analysis of operating reactors	9.2.3.3	--	--	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	2.6
Total		0.6	1.0	1.4	1.6	1.6	1.5	1.3	0.9	0.5	0.4	0.4	0.6	0.8	1.0	1.0	14.6

The distribution of plutonium and fission products in the core and the radial blankets can have a profound effect on the reactivity feedbacks.

The desirability and, indeed, the feasibility of moving (shuffling) fuel assemblies about in the core, as opposed to in-place refueling, must be determined. Although it is not now known what the maximum allowable fuel burnup is and whether the limit is set by fuel or cladding swelling, some upper limit will no doubt be determined. Just as reduced power peaking factors can dramatically reduce core size, decreased peak burnups can increase the average discharge burnup with an accompanying decrease in fuel costs.

Summary of Current Technology — The non-economic effects of fuel management have received relatively little attention in past design studies. This is no doubt due in part to the lack of methods sophisticated enough to provide the detail required for a meaningful evaluation. In general, one-dimensional and even fundamental-mode depletion methods have been used to determine the behavior of the reactor with fuel burnup. Reactivity coefficients are computed at various times in life, and core compositions are obtained with the methods above. Thus, the heterogeneities associated with partial refueling have been largely ignored. For example, the voided worth of a once-burned fuel assembly surrounded by twice- and thrice-burned assemblies may well be considerably different from that of a thrice-burned assembly surrounded by once- and twice-burned assemblies.

During B&W's 1000-MWe Follow-On Study, a two-dimensional fuel management study was carried out. In this study control rods and individual fuel assemblies were shown discretely. The results are given in Volume 2, section 4 of the combined Task II and III final report.²⁴

It is clear that the local peaking associated with various control rod patterns and different levels of burnup in the individual fuel assemblies is not adequately shown in one-dimensional calculations. This is especially true of power gradients across assemblies.

An ideal fuel management scheme will reduce the power peak and at the same time reduce the peak-to-average burnup.

The first effect allows a reduction in core volume, and the latter permits a higher average discharge burnup for a given limiting peak burnup. Thus, both effects tend to reduce power costs. However, the fuel management scheme is also subject to the constraints that cycle time be achieved and that available control rod worth not be exceeded.

In reality, reduced power peaking and reduced peak-to-average burnup may be mutually exclusive. High-enriched outer zone assemblies are not sufficiently depleted to permit their movement into an inner zone without excessive peaking. Similarly, an assembly in a steep flux gradient may have a 25% difference in burnup on opposite sides. If the assembly is rotated 180 degrees to equalize burnup, the low-burnup side is now in the highest flux, and the power peak will increase.

Scope of Work — Using the B&W reference core as a basis, different fuel management schemes will be investigated. In-place refueling, in which the fuel assemblies remain in a fixed position throughout their residence in the core, appears to be feasible. A further variable on this basic scheme is the fraction of fuel removed at the end of each cycle, i.e., one half, one third, one fourth, etc. How long are the radial blanket assemblies kept in the reactor? In shuffling schemes, the time between shuffling is very important. The cheapest and safest scheme may be to remove fuel every 6 months instead of every 12 months. However, it may be desirable to remove fuel every 6 months but to rotate it 180 degrees every 3 months. The removal of the blanket elements does not necessarily occur at the same time as the removal of core elements, and it may be that if the core elements are removed in a scatter pattern, the best way to remove the blanket is by zones. The various shuffling possibilities must be studied in some detail before deciding on the best one.

An alternate refueling scheme would involve the rotation and movement of assemblies within a zone in addition to partial refueling. Once more, the fraction of fuel replaced and the blanket management would be variables.

Although out-in fuel movement does not appear to be feasible at this time, it will be investigated, using new methods,

to determine whether the advantages of this refueling scheme outweigh the penalties associated with increased peaking.

These fuel management schemes will be investigated in the light of economics effects and in terms of the effect of the given fuel management scheme on the safety characteristics.

Factored into the economic survey will be the different refueling times expected for the different schemes, the effects of the remainder of the cycle, i.e., fabrication reprocessing, etc., and the relative attainable peaking factors and maximum-to-average burnup.

Studies have shown that it may be possible to remove fuel directly from the core with no in-vessel storage time being necessary.²⁴ This would limit the on-site storage time to the shipping time and/or reprocessing limitations and lower the fuel cost. More studies, not related to fuel management per se are necessary to see if the spent fuel storage time can be eliminated.

An important part of this program will be the evaluation of the effects of the various refueling schemes on the safety characteristics of the core. The close proximity of fuel assemblies with varying degrees of burnup and varying burnup patterns could affect the sodium void and Doppler coefficient. An additional factor that can have a significant effect on the void coefficient is the buildup of plutonium in the radial blanket adjacent to the core.

These studies will be performed with two-dimensional point-depletion methods, so that the depletion within fuel assemblies can be determined. Otherwise, it is impossible to calculate the rotation and movement of assemblies within a zone. Depletion may take place in relatively few neutron groups. However, the calculation of the reactivity coefficients should be in as many groups as is found necessary in subtask 9.2.3.1. The successful completion of this subtask will, of course, depend heavily upon the other tasks of the physics technology area. Input from RAPSODIE, SEFOR, and PFR will be factored in.

The Core Design Program Plan (6) discusses the need for fuel management studies purely in terms of economics. Since the safety and operating characteristics are also dependent upon

the fuel management scheme it would seem to be more efficient and meaningful to combine the two design studies.

Budget and Schedule — The budget and schedule for this subtask are summarized in Table 9-4.

9.2.3.3. Analysis of Operating Reactors

Objective — This subtask is carried out to verify the validity of the theoretical methods and supporting experimental results generated in other tasks of the physics technology area by calculating the parameters of existing and planned LMFBRs. Power coefficients, such as Doppler and core expansion and burnup characteristics, which cannot be directly simulated in cold critical experiments, are the chief objects of study in this subtask.

Summary of Current Technology — The only mixed oxide LMFBR currently operating is RAPSODIE, a 24-MWt plant in France. It is, of course, extremely small. SEFOR, which will be operating in the near future, is approximately the same in size and power. SEFOR will provide invaluable data on the role played by the Doppler coefficient in fast reactor transients.

No significant operating experience from a large (600-MWt) LMFBR will be available until after the completion of the British PFR in 1971 and the French PHENIX, which is scheduled for criticality in 1973. U.S. and German demonstration plants are scheduled to follow.

Of the plants just mentioned, only RAPSODIE, SEFOR, PFR, and possibly PHENIX will provide significant input to the B&W predecessor plant. The remaining demonstration plants and the B&W predecessor plant will aid the B&W reference design.

Scope of Work — Data from the operating LMFBR will be analyzed in great detail to verify that the design methods are indeed capable of simulating an operating reactor accurately. Power coefficients resulting from core expansion, fuel assembly bow, sodium expansion, and fuel temperature will be analyzed for both the unirradiated core and at various stages of burnup thereafter.

Depletion or burnup studies will attempt to predict the end-of-cycle plutonium and uranium masses and the fission

product content. The predicted values can then be compared with the actual discharged values.

Budget and Schedule — The budget and schedule for this subtask are summarized in Table 9-4.

9.2.4. Shielding

Shielding methods, although not essential to the design in a "go-no-go" sense, can nevertheless have a significant effect on the economics of the plant. In the B&W reference design, as much as 6% of the direct capital costs is either directly or indirectly associated with shielding. Hence, there is a strong incentive to reduce the uncertainties and, as a result, the conservatism in shield design.

The shielding task is divided into three main areas. These are theoretical methods development, cross-section procurement, and experimental verification.

9.2.4.1. Shielding Methods Selection and Development

Objectives — The objectives of this subtask are two-fold:

1. To investigate the various theoretical methods of predicting neutron and gamma attenuation, to choose the methods that seem most promising for fast reactor shield design calculations (both survey and detailed), and to develop and refine these methods into efficient calculational tools. Quite accurate neutron attenuation calculations are needed to determine heat generation rates (and, therefore, cooling requirements) and also the proper location of low-power neutron flux instrumentation. The primary criterion for accuracy, however, is economy. In many cases, the plant size is quite sensitive to the neutron and biological shielding requirements. As a result, significant cost savings could be realized by improving calculational accuracy, thus reducing conservatism in shield design.

2. To develop efficient and accurate theoretical methods for calculating the activation in various key parts of a fast reactor system caused by coolant activation, coolant impurity activation, and entrained fission products. The latter source of activity is

especially important in fast reactor plants that utilize vented-to-coolant fuel. This effort will also provide for accurate and efficient theoretical prediction of radiation (neutron and gamma) penetration through simple and complex shield irregularities.

Summary of Current Technology — It is generally felt that Monte Carlo techniques provide the most accurate theoretical measure of neutron attenuation. These methods are primarily in a developmental stage at present and are not being used as design tools. The next best method is generally considered to be the S_n discrete ordinates technique, preferably including higher order anisotropic scattering. Of all the codes of this type, ANISN (Oak Ridge³⁰) and DTF-IV (Los Alamos³¹) are considered to be the state of the art. ANISN is considered to be somewhat more flexible than DTF-IV. Theoretically, diffusion theory is generally regarded as inapplicable because of the assumption of isotropic scattering at all points in the medium. This assumption should lead to overly optimistic flux attenuation, especially at deep penetrations. Removal-diffusion techniques, which utilize removal theory for higher-energy neutrons, and diffusion theory for lower energies have shown promise. The Swedish NRN³² code is generally regarded as the most flexible and accurate of the removal-diffusion codes. The British have reported optimism toward the use of this method for fast neutron attenuation calculations.

Point-kernel integration techniques and the simplified source geometry equations of Rockwell continue to be used extensively for LMFBR shield design calculations. Common to both of these methods are the uncertainties associated with buildup factors. As with neutron attenuation calculations, Monte Carlo techniques are generally regarded to be the most accurate methods for gamma attenuation calculations. The discrete ordinate codes are prime candidates for achieving accuracy and efficiency in gamma transport calculations, which generally is not possible with either Rockwell's equations or kernel integration. DTF-IV and ANISN seem to be the likely choices for present shield design purposes.

The problem of system activation analysis for fast reactor systems has not been emphasized as it should be. Most

calculations are performed by hand, using many simplifying assumptions; a logical theoretical approach must be applied.

Carroll, Sisman, and others at ORNL have done experimental and theoretical work on the release of fission products from UO_2 fuels.³³⁻³⁶ Atomics International has published literature regarding an approach to predicting fission product deposition in fast reactor systems. Their approach is based on an experimentally determined "system depletion factor."³⁷

The most accurate duct streaming calculations employ Monte Carlo methods. The ORNL method uses the discrete ordinates code ANISN to obtain differential albedos for use with a Monte Carlo technique.^{38,39} This code traces histories through arbitrarily shaped ducts with an arbitrary number of right-angle bends. Good agreement with experiments has been reported with this method.

Scope of Work — The selection and development of appropriate and accurate neutron attenuation methods are considered more important than such studies for gamma attenuation, since the gamma source terms are dependent on neutron interactions. The theoretical methods that will be investigated for neutron attenuation problems include diffusion theory, removal-diffusion techniques, neutron transport theory (discrete ordinates) and Monte Carlo methods. The ranges of validity of the less-advanced methods will be determined (i.e., diffusion and removal-diffusion), since there seems to be widespread disagreement as to the applicability of these methods to fast reactor shield calculations. More efficient one- and two-dimensional S_N codes will be emphasized. The efficiency of the numerical methods of the discrete-ordinates codes will be improved; the Monte Carlo method will be further refined; and a systematic basis for important functions will be evolved.

For gamma attenuation calculations in fast reactors, the following methods will be evaluated for optimum accuracy and efficiency: simplified source geometry and point-kernel with empirical buildup factors, gamma transport (discrete ordinates), and Monte Carlo techniques.

Models will be developed and tested against the experimental results that will be generated in subtask 9.2.4.3.

Table 9-5. R&D Cost and Schedule, Physics Technology
Area: Shielding (9.2.4)

		Cost, \$ × 10 ⁶															
		Fiscal year															
Task	No.	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	Total
Shielding methods	9.2.4.1	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	--	--	--	--	--	--	--	1.6
Shielding cross sections	9.2.4.2	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.2	0.1	--	--	--	--	--	--	2.4
Shielding experiments	9.2.4.3	0.2	0.3	0.3	0.4	0.4	0.3	0.3	0.2	0.2	--	--	--	--	--	--	2.6
Total		0.7	0.8	0.8	0.9	0.9	0.8	0.8	0.6	0.3	--	--	--	--	--	--	6.6

Efficient and accurate models for predicting coolant activation (^{24}Na , ^{22}Na), coolant impurity activation (K, Li, P, S, Sn, Ti, etc.), and component activation as functions of space-energy-dependent system flux values and system characteristics are required. Of course, good results depend on an accurate knowledge of initial coolant impurity concentrations. Reliable and efficient methods are required to predict fission product release, transport, and deposition for a fast reactor system. These should include the effects of fuel composition, temperature, and system characteristics. Especially important for vented-fuel plant concepts are the effects of the venting devices themselves, which ideally should impede the release of fission products as long as possible to allow short-lived products to decay out.

The complicated problems of accurately calculating radiation penetration through shield irregularities, such as ducts and voids, need to be resolved. The primary problem seems to be economy, since the most accurate methods require long computer running times. Large sodium-filled ducts and annular voids should be stressed.

Sophisticated models that can be used to benchmark less accurate and more economical design methods will be developed. The design codes can then be optimized on the basis of accuracy versus computer time utilization.

Budget and Schedule — The budget and schedule for this subtask are summarized in Table 9-5.

9.2.4.2. Cross Sections and Nuclear Data

Objective — This effort is directed toward improving the accuracy of neutron and gamma ray interaction cross sections and other nuclear data of interest to the shielding community. The efforts to assemble the evaluated data in a consistent set, such as the ENDF/B, and to improve methods for the extraction of data for input to shielding codes will continue. Cross-section-sensitivity studies will be conducted to determine the most critical areas of cross-section accuracy desired for shield design calculations. The various cross-section weighting schemes will be thoroughly investigated, and the methods that are most applicable to attenuation problems will be developed.

Summary of Current Technology — Regardless of the calculational model used for an attenuation calculation, the end result is only as reliable as the cross sections and nuclear data employed. The ENDF/B data is generally regarded as the best available for neutron cross sections. It has been observed, however, that the sodium resonances are still not well defined, and that the cross sections of iron, nickel, and others contain inconsistencies over large portions of the spectra. The Physics portion of the LMFBR Program Plan contains much information regarding the state of the art on the accuracy of neutron cross-section measurement.

Recent measurements of fast fission mass chain yields of ^{239}Pu have been performed by Anderson of LASL.⁴⁰ Work in the area of capture gamma ray spectra is underway at Gulf General Atomic, and the results of this effort should be available by October 1968. Texas Nuclear has done extensive work involving cross-section and angular distribution measurements of gamma rays produced by neutron interaction in various elements. The latest published data on photo-interaction cross sections covers elements 1 through 100 in the energy range from 0.001 to 100 MeV.⁴¹ This is a compilation of point data for use in a group-averaging code.

Work is in progress at Los Alamos to develop a code for the construction of photon production cross sections from the ENDF/B tape. Ferguson of USNRDL is developing a gamma-ray interaction cross-section retrieval code.⁴²

In the area of cross-section sensitivity studies, one-dimensional S_n perturbation calculations are now being done by Prezbindowski of Purdue University.⁴³

Scope of Work — Particular cross sections recommended for attention are presented below. Some of the cross sections presented are acknowledged in the Physics LMFBR Program Plan and others are omitted. The entire list is presented here for completeness.

<u>Element</u>	<u>Cross section</u>	<u>Energy range, MeV</u>
Oxygen	$\sigma_{n,n}(E;\theta)$	4 to 16
Oxygen	$\sigma_{n,G}(E;E_{\gamma})$	<10
Silicon	$\sigma_{n,n'}(E;E')$	2 to 14
Calcium	$\sigma_{n,M}(E;E',\theta)$	3 to 16
Iron	$\sigma_{n,M}(E;E',\theta)$	3 to 16
Iron	$\sigma_{n,G}(E;E_{\gamma},\theta_{\gamma})$	Thermal to 10
Iron	$\sigma_{n,\alpha}(E)$	Thermal to 10
Nickel	$\sigma_{n,\gamma}(E;E_{\gamma})$	Thermal to 175
Nickel	$\sigma_{n,\gamma}(E)$	Thermal to 10
Sodium	Resonance data	~3 keV
Sodium	$\sigma_{n,G}(E;E_{\gamma},\theta_{\gamma})$	Thermal to 10

Other nuclear data measurements recommended for concentration of effort are mass chain yields from the fast fission of ^{239}Pu , photon production data, and photon interaction data. Efforts will continue to put photon production data on ENDF/B. Codes should be developed to extract the data for input to gamma-transport codes. Gamma-ray interaction cross-section retrieval codes will also be developed. A very important area of work is in cross-section sensitivity studies using perturbation methods. Such studies will be conducted before detailed programs are recommended for cross-section measurements, other than obvious cases.

The effects of cross-section weighting schemes for multigroup sets used in shielding calculations will be studied. The choice of energy-group structures will be evaluated.

Budget and Schedule — The budget and schedule for this subtask are summarized in Table 9-5.

9.2.4.3. Experimental Verification

This subtask consists of three different types of shielding experiments, grouped according to the type of facility used.

These are experiments performed at (1) critical facilities designed primarily for physics experiments, (2) at a low-power shield mockup facility, and (3) at operating fast reactors. Since the types of experiments are different, an Objective, Summary of Current Technology, and Scope of Work will be presented for each.

Objectives — The objectives of the three types of experiments are as follows:

1. To obtain as much experimental data as possible relevant to shield design calculations and/or shielding calculational methods verification from existing or planned critical experiments. The existing critical assemblies and those planned for construction have been geared primarily for experimental measurements of interest to the core physicist. Although the large attenuation ranges encountered in a 1000-MWe plant would not be possible in such low-power critical facilities, many other shielding parameters can be measured.

2. To provide for a fast reactor shield facility designed expressly for shielding measurements. The facility should be flexible enough to provide for a variety of attenuation and streaming experiments with a variety of source spectra. The facility would be a type of bench-mark experiment, serving as a standard to test the various shield calculation techniques of many contributors. Also, radiation damage studies could be performed. At this stage, it is felt that a thermal reactor with a suitable thermal neutron filter might be useful.

3. To utilize, to the maximum possible extent, operating high-power fast reactors. All high-intensity neutron and gamma flux measurements that are not possible with low-power fast critical experiments would have to be performed with high-power facilities. The required measurements could be quite difficult and costly, depending on the particular reactor configuration.

Summary of Current Technology — A very limited number of measurements have been performed on existing fast critical facilities. Also, it is felt that not enough emphasis has been placed on such measurements with the existing facilities or those proposed for future operation. Although other programs will be required

for complete experimental backup to shielding design, the full utilization of critical facilities for obtaining information of interest to the shielding community could represent significant cost reductions in the overall shielding R&D program.

No fast reactor shield test facility exists today or is definitely proposed for construction. Techniques for low-intensity measurements of gamma heating have been developed for thermal systems, however, and these could be applied to fast experiments as well. The proton-recoil proportional counter techniques used for neutron spectral measurements also function at low-intensity levels.

Preliminary blanket and shielding studies are being planned with the ATSR, although the program is not well defined. These measurements, however, are planned to extend only to the outer edge of the blanket. Also, one of the alternatives suggested for ZPR-III is a study of blanket and shielding characteristics.

A few measurements of interest to the shielding community have been carried out on the Fermi, Dounreay and EBR-II reactors. These include radial and axial neutron and gamma flux distributions, core and blanket neutron spectral measurements, heat generation measurements, studies of radiation damage to shielding materials (graphite is currently being irradiated in the blanket region of EBR-II), gamma spectral measurements, and others.⁴⁴ Confirming measurements like those mentioned above will contribute to a knowledge of the confidence levels attributable to shield design calculational tools.

Scope of Work — The kinds of experiments proposed for the different facilities are discussed below.

Critical facilities can provide the following measurements:

1. Detailed blanket neutron leakage spectrum and angular distribution.
2. Detailed blanket gamma leakage spectrum and angular distribution.
3. Effects of core and blanket burnup and fuel recycle on items 1 and 2.
4. Source spectral measurements (included in physics measurements).
5. Nuclear heat generation measurements.

6. Streaming measurements, neutron and gamma.

These measurements will be limited by the physical arrangement of the critical assembly itself, which (it is assumed) is not designed specifically for such purposes.

The fast reactor shield test facility could be used to perform the following kinds of experiments:

1. Attenuation experiments: Neutron attenuation experiments will be performed in various geometric arrangements through a number of candidate fast reactor shield materials. These should include borated graphite, boron carbide, sodium, serpentine aggregate, and ordinary concrete, to mention a few. Measurements like these in simplified geometries would then serve as standards for testing the accuracy of fast reactor attenuation calculation techniques.

2. Streaming Experiments: The facility will provide for neutron and gamma streaming experiments under a variety of geometric arrangements. Attenuation measurements through large, sodium-filled, multibend ducts should be undertaken, as well as large annular duct attenuation measurements. Provisions should be made to mock up reactor top shield plugs with their accompanying penetrations.

3. General: The reactor power for the facility would be adequate to cover deep penetration measurement requirements. The accurate simulation of axial and radial neutron leakage spectrum would be extremely important; this could be accomplished by using steel "filters." Detailed neutron spectrometer and integral flux measurements would be carried out in support.

Integral neutron flux measurements will be required over all neutron energies to provide for flux maps throughout each shielding mockup. Measurements utilizing sandwich foil resonance detector techniques, hydrogen-filled proportional counters, and proton-recoil proportional counter techniques would probably be required. Also gamma dose rate measurements would be made where pertinent.

All of the measurements pertinent to shielding that can be made with the shield mockup facility could, of course, be accomplished with a fast critical facility. However, the measurements could be

obtained more readily in the shield mockup facility, since shielding measurements would have low priority on an existing or proposed fast critical facility.

The experimental information obtainable from operating high-power systems will, of course, be a function of the availability and accessibility of areas in the reactor for the installation of test specimens, mock shields, or sensors.

Radiation damage measurements, bulk attenuation studies, radiation streaming measurements, spectral and angular neutron and gamma blanket leakage measurements, and others would be desirable wherever possible.

Budget and Schedule — The budget and schedule for this subtask are given in Table 9-5. Capital costs for the fast reactor shield test facility are not included.

9.3. Physics Technology Cost Benefits

9.3.1. Introduction

The uncertainties associated with physics methods can have a significant effect on the design and the subsequent operation of an LMFBR. These methods uncertainties combine to produce uncertainties in the following important reactor parameters:

1. Initial criticality.
2. Power distribution.
3. Burnup distribution.
4. Breeding ratio.
5. Lifetime.
6. Shielding.
7. Safety coefficients.

This section will relate the uncertainties in calculational methods with the parameters listed above. In general, the fuel cost penalty associated with a given uncertainty in the selected parameter is given without regard for the source of the uncertainty. The individual calculational uncertainties are then discussed in relation to the selected parameter. Quantitative results are not readily obtainable in some cases. In these situations, qualitative effect will be discussed.

Uncertainties can be accommodated in two ways. If R&D has not resolved the uncertainties at the time the final design is fixed,

the designer can (1) fix the design on a nominal basis and accept the consequences if the uncertainty is unfavorable; or (2) modify the design to a new nominal position so that the least favorable results satisfy the design requirements. In general, the first alternative is potentially the more expensive approach.

9.3.2. Cost Benefit Basis

The B&W reference core²⁴ will serve as the basis for the cost benefit studies. Fuel cost penalties relative to the reference core fuel costs will be given.

In this study a number of ground rules are made regarding the accommodation of calculational uncertainties in the as-built core.

1. The number of fuel assemblies (i.e., no interchanging of core and blanket assemblies) and the active core length are fixed.
2. The reactor thermal power cannot be increased because of IHX, steam generator, and turbine limits.
3. The peak burnup cannot exceed 140,000 MWD/T.
4. The core lifetime in effective full power days cannot exceed the design value without a corresponding decrease in the maximum-to-average burnup.
5. Additional reactivity control cannot be provided.

The core is first assumed to be built on the basis of nominal values of the key parameters listed in section 9.3.1. If, at this point, it is discovered that these parameters are actually different from the anticipated values, changes in operational procedures consistent with the ground rules outlined above (e.g., reduce power or burnup) are specified and the cost penalty or benefit is determined.

Alternatively, the core can be modified physically and built so that even if the calculated parameters are low, the minimum values will still permit us to achieve design power level and cycle time. Since the design would be re-optimized on the basis of the required modifications, the exact penalty is difficult to assess. For the purposes of this comparison, no re-optimization is assumed.

9.3.3. Cost Benefits

9.3.3.1. Initial Criticality

Failure to accurately predict the k_{eff} of the reactor has a number of ramifications. If the true k_{eff} is too low, the reactor may not achieve criticality and almost certainly will have a reduced lifetime. If, on the other hand, the k_{eff} is at the upper limit of the uncertainty range, the available control rod worth may be insufficient.

The uncertainty in the k_{eff} results from two sources. First, there is the uncertainty in predicting the cold, clean k_{eff} of the reactor, which is attributable to uncertainties in cross sections and calculational models. This is treated separately, because conventional fast criticals can provide quite good benchmarks in this area. Temperature deficits (the change in k_{eff} from cold zero power to hot full power), such as core thermal expansion, Doppler, and sodium density, constitute the remaining source of uncertainty. At full power the average core temperature would be 950 F, and the average fuel temperature would be approximately 2500 F. Thus, the calculational methods used cannot be as readily benchmarked with criticals at room temperature. The estimated total current uncertainty from both sources is ± 0.03 in δk .

If the enrichment is set on the basis of the nominal calculated k_{eff} , the true k_{eff} of the reference core may fall in the range 0.98 to 1.04 because of the calculational uncertainties. Thus, the risk involved in not modifying the design, i.e., increasing the nominal enrichment, is considerable. In order to ensure that the k_{eff} is at least 1.01, the nominal k_{eff} becomes 1.04. The corresponding maximum k_{eff} , therefore, becomes 1.07. Unfortunately, the existing reactivity control system cannot control this excess reactivity. In fact, the ability of the control system to handle the 1.04 maximum k_{eff} associated with the nominal calculated k_{eff} is doubtful.

There is also an uncertainty associated with the calculation of the total control rod worth. The number and location of control rods and the absorber material cannot merely be selected on the basis of nominal calculated worths, since, for a number of obvious safety reasons, the total control rod worth in the as-built core cannot

be inadequate. Furthermore, the uncertainties in the core k_{eff} that result from uncertainties in cross sections, calculational methods, and manufacturing tolerances must also be accommodated. This is not a situation in which the performance of the as-built core can be downgraded if the uncertainties are unfavorable. Thus, a 3% uncertainty in k_{eff} is equivalent to a 43% uncertainty in total rod worth. If the considerable uncertainty associated with calculating the rod worth is also included, then accommodating the total uncertainty may require the provision of 60% extra rod worth.

The reactivity control system can be modified in two ways. The first involves changing the absorber material from tantalum to enriched B_4C . Even though the required worth could probably be achieved, the maximum worth of a control rod would exceed the ground rule limit of \$1.00 by a considerable margin, thus compromising safety. In addition the helium evolution associated with B_4C would require either frequent replacement of the control rods or a large gas plenum at the end of each B_4C pin, which would lengthen the control rod and deepen the reactor vessel.

The alternative is to provide additional control rods and drives, which would increase core diameter and capital costs. Furthermore, the additional drives and control rods themselves are expensive. Extensive design studies are required to generate quantitative cost penalties for these modifications.

In addition, reactor modifications to accommodate a k_{eff} that may be higher than the nominal value will cause additional capital expenditures, which will affect power costs for the life of the plant.

Figure 9-1 illustrates the fuel cost penalty associated with overloading to ensure a minimum k_{eff} of 1.01 as a function of the uncertainty in the nominal value of k_{eff} .

Deliberate overloading penalizes two fuel cost components. Since the fissile inventory is increased, the plutonium carrying charge increases; in addition, the increased fissile loading decreases the breeding ratio and hence the plutonium credit. Since the lifetime is burnup-limited, no credit can be taken for the additional lifetime that might otherwise be gained because of the higher nominal

k_{eff} . The fuel cost penalty resulting from accommodating the currently estimated $\pm 3\%$ in k_{eff} is 0.035 mill/kWh. The capital cost penalty for the increased reactivity control could also increase the power cost significantly.

The fuel cost penalty for failure to achieve the desired lifetime because the initial k_{eff} is too low is discussed in section 9.3.3.5.

9.3.3.2. Power Distribution

It is well known that the power distribution and energy deposition play an important role in the size of an LMFBR. In general the maximum linear heat rate is fixed by the fuel density and burnup rather than by the reactor environment. Thus, the power distribution peaking factor is very important in setting the average linear heat rate.

Factors affecting the power peaking factor include the incore power distribution, especially in the area of zone boundaries and local peak within assemblies; the core blanket power split, which varies through the cycle; and the deposition of the energy within the core. The first two factors are strongly dependent on the fuel management scheme used.

If the computed peaking factors are too low for the as-built core, the only alternative is to reduce the core power level until the maximum linear heat rate reaches an acceptable value; at this point, two alternatives are available. The design cycle time can be adhered to with a reduction in burnup, or the cycle time can be increased to produce the design burnup. The penalty for the former is approximately twice that of the latter (see Figure 9-2).

If the peaking factor uncertainty is accommodated in the as-built core, i. e., by adding extra fuel pins or increasing the active length, penalties for additional plutonium inventory and additional fabrication are incurred. The effects of an uncertainty in peaking factor for a given maximum linear heat rate and an uncertainty in the maximum linear heat rate itself are similar. The reader is referred to section 6.3.1 for a discussion of the cost penalties associated with accommodating uncertainties in the maximum linear heat rate.

The estimated uncertainties in power peaking factor and in the core-blanket power split are $\pm 10\%$ each. The corresponding fuel cost penalties for the as-built core are 0.10 mill/kWh for reduced power and design cycle time, and 0.04 mill/kWh for reduced power and design burnup. The capital and OM&I cost penalty resulting from the 10% reduction in power would be nearly 0.3 mill/kWh. Thus, the total power cost penalty could be as much as 0.4 mill/kWh.

The power cost penalty incurred by accommodating the uncertainty by an increase in the active core length or the number of pins is considerably less (0.02 mill/kWh), assuming that the design burnup of 100,000 MWd/T is achieved. If the cycle time is kept at the design value, the penalty from decreased burnup increases to approximately 0.13 mill/kWh.

9.3.3.3. Burnup Distribution

The maximum burnup experienced by any portion of the core loading determines the residence time of the entire batch of which that fuel is a part; the effect is similar to that of the peaking factor. Thus, if the maximum-to-average burnup ratio is higher than the calculated value, the average burnup will be reduced and the cycle time will decrease, or the plant will operate at a reduced capacity factor in order to maintain the design cycle time.

The uncertainty in the peak-to-average burnup is the result of calculational uncertainties involving the core power distribution and the core blanket power split and uncertainties in the details of the fuel management scheme and the control rod programming, which have been averaged over the residence time of the fuel. Thus, the behavior of these parameters throughout the cycle must be known.

The effect of the peak-to-average burnup uncertainty on fuel costs is shown in Figure 9-3. With the estimated uncertainty of 20%, the fuel cost penalty is 0.11 mill/kWh, assuming that the cycle time is reduced. If the cycle time is maintained while the capacity factor is reduced, the additional fuel cost penalty is 0.05 mill/kWh, and the capital and OM&I penalty is 0.021 mill/kWh.

Uncertainties in the allowable average burnup can be partially accommodated by a re-optimization of the design for a lower average burnup, i.e., reducing fabrication and reprocessing costs.

9.3.3.4. Breeding Ratio

An uncertainty in the breeding ratio has a significant effect on the economics of LMFBR operation. The breeding ratio also affects the core lifetime in that it is a measure of the ability of the core to replace the plutonium consumed in the course of power production. The lifetime effect will be discussed in the following section. Uncertainties in breeding ratio result primarily from cross-section uncertainties.

The plutonium credit fuel cost component is directly affected by the breeding ratio. More precisely, the plutonium credit is roughly proportional to the breeding gain, i.e., $(BR - 1)$. Thus, the B&W reference core with a BR of 1.36 would have a breeding gain penalty of 39% for a 10% reduction in BR. The plutonium credit would also be reduced by approximately the same percentage. If the lower breeding ratio is discovered after construction, nothing can be done to improve the situation.

The effect of a reduced breeding ratio on the fuel cost is shown in Figure 9-4. Clearly, an uncertainty of 10% could make the LMFBR significantly less attractive. In addition, the doubling time would far exceed the target range of 7 to 15 years.

The chief benefit to be gained from a reduction in the breeding ratio uncertainty would be that the design could be re-optimized in light of the improved data. Conceivably, the optimum pin size would be smaller for a lower breeding ratio, so that the inventory would be reduced.

9.3.3.5. Lifetime

Ultimately, uncertainties in most of the core parameters either directly or indirectly affect the core lifetime and cycle time. Some of these uncertainties have been mentioned earlier. For example, an uncertainty in initial k_{eff} or in breeding ratio could affect the lifetime. Similarly, an uncertainty in the burnup effects associated with the formation of higher isotopes and fission products can significantly affect the lifetime.

The as-built core will suffer extreme fuel cost penalties if the lifetime is shorter than anticipated. In addition, the

downtime for refueling could fall during a period of peak power demand. In any event, the plant schedule would be thrown off. Fuel cost penalties as a function of uncertainty in lifetime are presented in Figure 9-5. A 20% reduction in lifetime will increase fuel cost by 0.14 mill/kWh. Considering the small excess δk required for lifetime, this uncertainty is not unreasonable.

Accommodation of the uncertainty would most probably be accomplished by overloading the core to increase the initial k_{eff} . This effect is discussed in detail in section 9.3.3.1.

9.3.3.6. Shielding

Uncertainties in shielding calculational methods affect not only the direct cost of the shielding but, indirectly, the cost of the remainder of the nuclear island. The shielding thickness required to prevent activation of the intermediate sodium and the large components affects the diameter of the reactor vessel and hence the reactor building. Similarly, the radial biological shielding thickness also affects the size of the reactor building. The axial shielding required constitutes a large fraction of the load that must be supported by the cover structure and also influences the thickness.

According to the architect-engineer's capital cost estimates, between \$4 and \$5 million of the capital cost is associated with shielding. If the shielding thickness could be reduced, the direct shielding costs will, of course, be reduced correspondingly. However, the additional savings in the reactor building could also be considerable.

If uncertainties in shielding are accommodated in the design by making the shielding thicker, the capital cost will necessarily increase.

The release and deposition of fission products from vented and failed fuel significantly affects the functional requirements and hence the cost of the purification system; the selection of seals in the cover structure is also affected directly.

9.3.3.7. Safety Coefficients

Uncertainties in the reactivity coefficients (Doppler, sodium void, and core thermal expansion) can have a

significant effect on the economic performance of an LMFBR. If the values, including uncertainties, are within acceptable limits in the as-built core, obviously no penalty in degraded performance will result. If, as will probably be the case, the tolerances are unacceptable, the uncertainty must be accommodated in the design. For example, the use of BeO to improve the nominal Doppler and sodium void coefficients will impose a significant economic penalty. The use of spacers to reduce fuel movement is also expensive.

Figure 9-1. Effect of Uncertainty in k_{eff} on Fuel Cost

Figure 9-2. Effect of Power-Peaking Uncertainty on Fuel Cost

Figure 9-3. Effect of Uncertainty in Peak-to-Average Burnup Ratio on Fuel Cost

Figure 9-4. Effect of Breeding Ratio Uncertainty on Fuel Cost

Figure 9-5. Effect of Fuel Cost on Uncertainty in Cycle Time

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10. SAFETY

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10. SAFETY

10.1. Introduction

10.1.1. General

The proposed R&D program has several general objectives:

1. To provide a realistic analysis and understanding of the behavior of B&W's demonstration and reference LMFBR plant during accident conditions.
2. To ensure that the engineered safety features and the protection system are capable of preventing accidents or mitigating their consequences so that the B&W reference plant can be built and operated in a utility environment without undue risk to the health and safety of the public.
3. To aid in developing standards and codes for the safe design, siting, construction, and operation of commercial pool-type LMFBRs.

B&W's program includes a number of tasks, the successful performance of which will enhance reactor technology considerably. These tasks are as follows:

1. Identification of abnormal behavior and its causes.
2. An understanding of unsafe conditions.
3. Development and testing of analytical models and their incorporation into accident analysis methods.
4. Testing the protective system and the engineered safeguard features.
5. An understanding of the sources and magnitudes of potential releases of energy and fission products.

6. Development and testing of containments and engineered safety features, including any systems designed to protect them.

The work on this program, and the reporting thereof, is divided into seven sections. The first four parts incorporate the necessary work from LMFBR Program Plan Task area 10-2, Volume 10, "Safety" (WASH-1110). Task area 10-1 of the Program Plan is included with the plant design studies in section 2 of B&W's R&D program. Since several items from Task area 10-3 are covered in sections 2, 3, 4, or 6, these items were not included in B&W's section 10.2.5. Task 10-4.3, "Dissemination of Information," was deleted from Task area 10-4 because the development of a fast reactor handbook or textbook is not necessary for the B&W program.

Although the R&D Program described is specifically related to the B&W predecessor and reference plants, its scope is sufficiently general that it applies to the overall LMFBR Program Plan. However, it is not broad enough to cover the entire LMFBR Program Plan since a number of tasks have been reduced in scope.

10.1.2. Budget and Schedule

A summary of the overall budget and schedule is given in Table 10-1, which is taken directly from the detailed breakdown given for each major section in their descriptions. The schedule for the predecessor in 1976 and for the commercial plant in 1984 allowed the B&W R&D program pace to be slowed and gave more time for planning and better use of the experimental facilities. The total cost of the program including the construction of a safety test facility and the excursion test facility is about \$175 million. The cost without the new facilities was about \$65 million.

Table 10-1. Overall Safety R&D Budget

Task	Cost, \$ × 10 ³															
	Fiscal year															
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	Total
Liquid Na Superheat and Bubble Formation Dynamics																
1. Facilities	250	350	700	350	--	200	450	200	--	--	--	100	100	--	--	2,700
2. Superheat characterization	200	400	400	400	300	300	200	100	100	--	--	--	--	--	--	2,400
3. Void formation and two-phase flow	400	550	750	900	950	1,100	1,750	1,725	1400	1400	1600	1675	1675	1550	1450	18,875
4. Analytical methods and engineering techniques	75	75	125	135	135	160	210	185	170	170	195	195	195	110	110	2,245
Design Basis Analysis and Energy Release																
1. Facilities	450	700	250	300	750	350	--	--	--	--	150	150	--	--	--	3,100
2. Fuel pin behavior	200	400	400	500	500	400	400	150	100	100	--	--	--	--	--	3,150
3. Subassembly and core behavior	100	350	450	500	650	800	850	800	650	500	500	350	350	250	100	7,200
4. Analytical methods and engineering techniques	175	200	300	325	400	450	450	500	425	425	450	450	425	400	250	5,625
Behavior of Radioactive Materials																
1. Facilities	300	--	--	--	--	--	--	--	--	--	--	--	--	--	--	300
2. Experimental program	125	200	275	325	325	300	175	150	75	75	125	125	125	75	50	2,525
3. Analytical methods and engineering techniques	25	50	125	150	150	150	150	75	50	50	50	100	150	150	25	1,450
Na Coolant Safety																
1. Facilities	10	--	20	--	100	--	--	--	--	10	100	--	--	--	--	240
2. Experimental program	40	75	110	110	135	130	85	55	55	90	165	180	180	125	45	1,580
3. Analytical methods and engineering techniques	35	75	75	70	50	75	100	100	50	10	35	10	10	--	--	695
Engineered Safety Features and Containment	40	40	40	790	1,690	1,240	1,190	790	40	40	40	740	1390	1040	990	10,100
Special Safety Programs																
1. Codes, standards, and criteria	75	100	100	125	120	120	115	115	115	115	115	115	115	115	115	1,675
2. Analytical reliability methods	50	150	150	75	50	50	50	25	25	50	150	50	75	75	50	1,075
Facilities																
1. Safety test	--	200	400	2,300	10,300	10,000	10,000	2,000	2000	2000	2000	2000	2000	2000	1000	48,200
2. Steady-state facilities	500	1500	2000	1,500	2,000	3,000	3,000	2,000	2000	1000	--	--	--	--	--	18,500
3. Transient test	500	500	1200	1,000	1,000	1,000	1,000	1,000	1000	--	--	--	--	--	--	8,200
4. Excursion test	50	200	300	500	1,300	10,300	10,000	6,000	1000	1000	1000	1000	1000	1000	500	35,150
Total	3600	6115	8170	10,355	20,905	30,125	30,175	15,970	9255	7035	6675	7240	7790	6890	4685	174,985

10.2. R&D Requirements

Realistic safety analytical methods and techniques must be established and verified. The objective is to improve the technological base and data required to provide realism and confidence in the understanding and analysis of accident situations that may occur in B&W's demonstration and reference design plants. Accurate data must replace approximations. Current analytical methods rely on generally unproven models; hence, the margins of safety are not well known. Improved methods are required for evaluating, on the basis of statistical, probabilistic, and other techniques, the significance and potential occurrence of accidents, modes of failure and propagation, and the consequences of malfunction.

Basic data and experiments are needed to evaluate the progression of incidents that may occur in an operating plant and to follow through the plant the propagation of events that could lead to localized core damage, general core damage, plant damage, or breach of the containment barriers. The adequacy of the calculational methods to realistically describe accidents must be demonstrated so that confidence will be gained in the safe operation of demonstration and reference LMFBR reactors. This area of the R&D program covers those tasks which are not adequately covered in other areas of the program and which are required for an adequate safety analysis, including the effect of the engineered safety features incorporated in the core and the systems design for the B&W demonstration and reference plants.

10.2.1. Liquid Sodium Superheat and Bubble Formation Dynamics

10.2.1.1. General

This task includes the experimental and analytical studies of sodium boiling and two-phase flow needed to develop reliable thermal-hydraulic models for describing the void formation, coolant motion, and power feedback that occur during reactor accidents.

Objectives — The overall objectives of this task in the R&D Program for the B&W demonstration and reference plants is the basis for (1) understanding transient boiling and two-phase

hydrodynamics, (2) evaluating reliable thermal-hydraulic models for accident evaluation, and (3) developing these models and incorporating them into analytical methods for safety analyses.

Summary of Current Technology — The boiling of liquids was reviewed by Westwater in two articles published in 1956 and 1958.^{1,2} Since that time, literally hundreds of papers on boiling and related topics have appeared. Tong gives a good state-of-the-art on boiling heat transfer and two-phase flow in his book.³ Several authors have reviewed the subject of liquid-metal boiling heat transfer and two-phase flow.⁴⁻⁸ In general, only the surface has been scratched in understanding the phenomena of liquid-metal boiling heat transfer. Experimenters in the field of boiling heat transfer have concentrated on obtaining empirical or semi-empirical fits to data and only a small fraction of the work has been devoted to studies with liquid metals. The remaining discussion of the experimental program will be divided into two parts, and the analytical methods and engineering techniques are added as a third section.

1. Superheat Characterization: A considerable amount of basic liquid-metal superheat data has become available, but the data show large variations. The large superheat required to initiate boiling in liquid metals has caused concern in fast reactor safety analyses; this is evident from the new data and the great interest expressed at the Aix conference on fast reactor safety.^{9,10} Four papers dealt directly with new experimental data on superheat of potassium and sodium.¹¹⁻¹⁴ These data were particularly noteworthy because of their large variations which were sufficient to require semilog paper for graphical presentation, as shown in Figure 10-1. The variation ranged from 9 to 1480 F, and in previously published data it varied from 0 to 500 F, as in Figure 10-2. It is no wonder that designers are confused and hasten to recommend further R&D on the basic characterization of the processes. Idealized systems furnished much of the information, such as pool boiling, low, steady, or transient heat flux, and indirect as well as direct heating. Also, since the variables examined have been considerably removed from the actual reactor conditions, it is questionable whether the data could be used to determine the design requirements for LMFBRs.

It is apparent from the data that the problem of superheat is extremely complex and that the superheat phenomena depend on a large number of variables. Extrinsic variables that can affect superheat behavior include the heat flux, heating surface condition, fluid velocity, pressure, pressure-temperature history, aging, sodium purity, sodium gas content, heating method, and many others. Fauske¹⁵ presents the simplified superheat model developed by Marto and Rohsenow¹⁶ by considering the forces acting on the liquid vapor interface in a right circular cylindrical cavity.

$$\text{superheat} \approx \frac{T_s}{\rho_V h_{fg}} \frac{2\sigma \cos \theta}{R_c}$$

where

T_s = saturation temperature

ρ_V = vapor density

h_{fg} = latent heat of vaporization

σ = surface tension

R_c = cavity radius

θ = dynamic contact angle (see reference 16 for details)

Fauske then proceeds to discuss each of the extrinsic variables in light of the available data.

It appears that early experiments simulating reactor conditions (high conduction heating, geometric configurations, spacers, high coolant velocities, oxide and gas content, high irradiation fields, aging, etc.) should be carried in the R&D program, since experiments far removed from actual reactor conditions produce wide variations in superheat levels. A fair percentage of reactor designers believe that, for real reactor conditions, very little if any superheat will occur. In the R&D program particular attention must be given to the following parameters:

1. The conditions of the heating surfaces and the coolant in a large fast reactor, particularly after long periods of reactor performance.

2. High heat fluxes up to 2×10^6 Btu/hr-ft² from indirect heat sources.
3. Close mockup of reactor coolant geometry and velocity.

2. Subassembly Voiding and Two-Phase Flow Models: Implicit in studies of forced-convection boiling is an understanding of two-phase flow. Forced-convection boiling in itself is a complicated process, and liquid sodium superheat adds fog to this. It is almost impossible to write the usual differential equations of motion and energy for realistic models and establish the appropriate boundary conditions.

Even the very basic bubble formation models have not been accepted. Whether one bubble or many bubbles will control the expansion is still questionable. It is generally accepted that the pressures involved in void formation and growth will not be great enough to damage the structures, because they are restricted to the saturation pressure of the liquid at its interface with the bubble. However, it is very probable that a much higher pressure may arise when the vapor bubbles condense in the subcooled liquid surrounding the reactor. The problem of pressure pulses is complex. It is not clear whether inertia, heat transfer, or both, control bubble collapse. If both are controlling, the analytical solutions will be extremely complicated. It is possible that the reactor geometry may play an important role in the magnitude of the pressure pulses.

Several theoretical models for sodium expulsion from a subassembly have been described in the literature. Many of these were described at the Aix meeting. MacFarlane described calculations with a coupled code system that combines transient heat transfer, two-phase coolant hydrodynamics using annular flow model, and neutron kinetics.¹⁷ It was presented to display the development of accident analysis techniques, but the discussion following the paper was devoted almost exclusively to the two-phase hydrodynamics module.

A contrasting model, using sodium slug expulsion, was incorporated into a Karlsruhe sodium-voiding model, the BLOW computer code.¹⁸ The BLOW model was developed on the basis of analyses of experiments performed by Grass and his co-workers at Ispra¹² and was also related to results obtained by LeGonidee under CEA auspices.¹¹

A similar ejection mechanism was presented by the Belgians as part of a safety study to guide the design of an in-core sodium loop to test for fuel-element rupture. Again, heavy emphasis was placed on the Ispra data. Considerable concern was expressed in reviewing these models, since they were based on data obtained using direct heating, rather than indirect heating, as is the case in the LMFBRs.

Judd of the UKAEA was not sure whether one bubble or many bubbles control the expulsion.¹⁹ After examining both cases, he concluded that (1) the rate at which a reactor coolant channel is voided by boiling depends mainly on the degree of superheat, and (2) the number of bubbles participating in the expulsion has a small effect. The latter conclusion suggests that inertia, not heat transfer, is the controlling rate mechanism. Kosky,²⁰ also of the UKAEA, discussed similar problems of vapor growth and indicated good agreement with Judd's vapor-growth calculations. He also suggested that the burnout heat flux may be exceeded at the instant when the first vapor void is formed in low-pressure, liquid-metal boiling heat transfer.

Several possible rate-limiting processes for vapor growth were also presented at the Aix conference: mass and heat exchange at the liquid-vapor interface described partially by the kinetic theory,¹⁸ heat transfer and inertia controlled,^{19,20} or vaporization occurring according to thermodynamic equilibrium.^{17,21,22}

Calculational methods based on simply analytical models have provided an early starting point in the U.S. program. The General Electric VOID computer program²³ and Atomics International's BURP program²⁴ use the single-bubble model. Whereas VOID uses an empirical relationship of the internal bubble pressure as a function of the channel wall temperature, BURP assumes that liquid is evaporated only from the liquid-vapor interface of the bubble. AI also developed TRANSFUGE.²⁵ Here they assumed that the vapor generation was heat-transfer-limited. The TRANSFUGE-II code being developed from TRANSFUGE-I has (1) eliminated the numerical instability that arose at the initiation of boiling, (2) eliminated the fixed-vapor-density restraint, and (3) includes a variable two-phase heat-transfer coefficient. AI has also made modifications to the AIROS²² computer program by incorporating capabilities for

treating phase changes during transients, such as coolant voiding and fuel melting, based on very simple models.

It was pointed out at the Aix conference¹⁷⁻¹⁹ that the pressures involved in voiding a channel are not high enough to damage the structure if molten fuel dispersion into the sodium is not considered, since the pressures are limited to the saturation pressure corresponding to the initial superheat. However, much higher pressures arise when vapor bubbles condense in the subcooled liquids that may surround a bubble ejected from a subassembly. The English¹⁹ and the Germans¹⁸ have attempted to calculate bubble collapse pressures but have cautioned against the reliability of their calculations, since the problem is very complex. The Ispra investigators¹² reported recondensation pressure peaks as high as 800 psia, but no attempts were made to predict the results. In view of the possibility for mechanical damage that could be caused by vapor-bubble collapse in highly subcooled liquid sodium, it deserves considerable attention in the fast reactor R&D program.

3. Analytical Methods and Engineering Techniques: When the models for predicting the boiling in reactor subassemblies are available, the appropriate kinetics and stress computer models must also be available to complete the safety evaluation. The state of the art of fast reactor kinetics is described by McCarthy and Okrent,²⁶ who give an excellent review of the initiating parameters and the events that follow. The computer programs in use in 1963 are described by Kazi and Cherry.²⁷ Few new codes have been written since that time; however, several, e.g., AIROS and FORE, have been revised and updated. The level of sophistication for solving the neutronics problems has evolved as the need has grown, and better, bigger, and faster computing machines have become available. Several reactor manufacturers have used modified versions of the WIGLE computer codes^{28,29} for space-dependent calculations. However, work on the 1000-MWe LMFBR Follow-On Study has shown that the WIGLE codes do not have sufficient energy groups for space-dependent representation of the transients caused by an accident that results in the formation of voids. In general, it was found that four or more neutron energy groups are required to adequately describe the power transient during a voiding accident. It must also be recognized

that the local shift in reactor power may play an extremely important part in the propagation of voids.

Scope of Work — The importance of sodium boiling cannot be overemphasized, since it plays an important role in several phases of accident analysis. The problems start with a basic understanding of the sodium superheat characterization, single-channel expulsion and two-phase flow, and multi-rod geometry expulsion and two-phase flow. From these experimental programs will come the basic mathematical models for simulating subassembly voiding under realistic conditions. These models are required for inclusion in practical accident analysis codes to be used for evaluating the behavior of LMFBRs and for the analysis of initiating accidents.

1. Experimental programs aimed at an understanding of the processes involved in sodium boiling using indirect heating: These experiments must consider (1) surface conditions, (2) heat fluxes up to 2×10^6 Btu/hr-ft², (3) coolant velocity, (4) purity and gas content, (5) radiation, (6) pressure and pressure-temperature history, (7) aging, and (8) the geometry of the channel.

2. Experimental programs which closely simulate reactor conditions, with particular emphasis on (1) the conditions of the heating surfaces and the coolant in a fast reactor, particularly after long-term reactor operation; (2) heat fluxes up to 2×10^6 Btu/hr-ft² using indirect heating techniques; and (3) close mockup of reactor coolant geometry and velocity. If these experiments should indicate excessively high superheats, a means to prevent such an excess must be developed.

3. Develop the analytical models to be used by the design engineer to predict the reactor coolant boiling characteristics, and incorporate them into the transient analysis codes that will be used for safety evaluations.

10.2.1.2. Liquid Sodium Superheat Characterization

Objective — Characterize the process of sodium boiling—in particular, those processes involved in the initiation of boiling.

Summary of Current Technology — Considerable basic liquid-metal superheat data have become available, but these data show large variations, ranging from 9 to 1480 F. It is no wonder that the designers are confused and hasten to recommend further R&D on the characterization of the process.

Scope of Work — It is apparent from the superheat data available that even the experimenters are still in the dark in the study of liquid-metals superheat, and that the problem of sodium superheat is very complex, depending on a large number of parameters. Extrinsic variables that may affect liquid-sodium superheat behavior include the heat flux, heating surface condition, fluid velocity, pressure, pressure-temperature history, aging, sodium purity, sodium gas content, and the method of heating. It appears that early experiments simulating LMFBR conditions should be carried out, i.e., experiments incorporating high conduction heating, coolant configurations, spacers, high coolant velocities, oxide and gas content, high irradiation fields, aging, and the like. Particular attention must be placed on the following parameters:

1. The condition of the heating surfaces and the coolant in the LMFBR, particularly after long periods of reactor performance.
2. High heat fluxes up to 2×10^6 Btu/hr-ft² using an indirect heating source.
3. Close mockup of the reactor coolant geometry and velocity.

Relationship to Current or Planned Programs — Studies of the superheat phenomena are being carried out at ANL, AI, BNL, and ORNL. These programs are providing a large variety of superheat data and are not simulating the LMFBR conditions closely enough to permit direct use of the data obtained. The data collected will be the basis for voiding models in other segments of the safety program.

Major Facilities and/or Equipment Required — At least one loop using heaters rated up to 2×10^6 Btu/hr-ft² is required for out-of-reactor studies. At least one loop, capable of accepting real LMFBR fuel in the form of 7- or 13-pin subassemblies,

Table 10-2. Budget for Liquid Sodium Superheat and Bubble Formation Dynamics

		Cost, \$ × 10 ³															
		Fiscal year															
Task		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	Total
Facilities																	2,700
1. Single-channel																	
Heated		100	50	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Reactor		--	100	200	--	--	--	--	--	--	--	--	--	--	--	--	--
TREAT		--	100	150	--	--	--	--	--	--	--	--	--	--	--	--	--
2. Multi-channel																	
Heated		150	100	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Reactor		--	--	200	200	--	--	--	--	--	--	--	--	--	--	--	--
TREAT		--	--	150	150	--	--	--	--	--	--	--	--	--	--	--	--
3. Subassembly		--	--	--	--	--	200	200	--	--	--	--	--	--	--	--	--
4. Propagation		--	--	--	--	--	--	250	200	--	--	--	100	100	--	--	--
Superheat Characterization																	2,400
1. Heated loop		200	200	200	200	100	100	--	--	--	--	--	--	--	--	--	--
2. Reactor loop		--	200	200	200	200	200	200	100	100	--	--	--	--	--	--	--
Void Formation and Two-Phase Flow																	18,875
1. Void growth																	
Single-channel		200	200	100	100	100	100	50	50	--	--	--	--	--	--	--	--
Multi-channel		--	100	200	200	200	200	200	200	100	100	100	100	100	50	50	--
Subassembly		--	--	--	150	300	400	400	400	300	300	400	400	400	400	400	--
Propagation		--	--	--	--	--	--	400	400	400	400	400	500	600	600	600	--
2. Two-phase flow																	
Single-channel heated		200	200	200	200	100	100	100	100	100	100	100	100	--	--	--	--
Multi-channel heated		--	--	200	200	200	200	200	200	200	200	200	200	200	200	100	--
3. Model verification																	
Single-channel		--	--	--	--	--	50	50	25	25	25	25	--	--	--	--	--
Multi-channel		--	--	--	--	--	--	100	100	75	75	75	75	75	--	--	--
Subassembly		--	--	--	--	--	--	200	200	200	200	200	200	200	200	200	--
4. Bubble collapse		--	--	--	--	--	--	50	50	--	--	--	100	100	100	100	--
Analytical Methods and Engineering Techniques																	2,245
1. Boiling onset		25	25	25	25	25	25	50	25	--	--	--	--	--	--	--	--
2. Voiding hydrodynamics and heat transfer		50	50	50	50	50	50	100	100	100	100	100	100	100	50	50	--
3. Stress models		--	--	25	25	25	25	25	25	25	25	50	50	50	25	25	--
4. Kinetic feedback models		--	--	--	10	10	10	10	10	20	20	20	20	20	10	10	--
5. Void collapse		--	--	25	25	25	25	25	25	25	25	25	25	25	25	25	--
Total		925	1375	1975	1785	1385	1760	2610	2210	1670	1570	1795	1970	1970	1660	1560	26,220

in an existing reactor is required to study boiling initiation under reactor conditions.

Budget — A detailed breakdown of the budget for the program is included in Table 10-2. The estimated cost, including development of the required loops, is \$3.1 million, and they are scheduled for completion by 1976-1977.

10.2.1.3. Subassembly Voiding and Two-Phase Flow

Objective — The short-term objective is to determine the voiding characteristics of a subassembly under conditions that are anticipated in an LMFBR. The long-term objectives are devoted to an understanding of (1) sodium boiling and two-phase flow in a subassembly, and (2) pressure transients during bubble formation and collapse.

Summary of Current Technology — The experimental programs to date have not provided the design engineers with a consistent picture of the processes involved. The work has concentrated on the available superheat data reported in the literature using theoretical approaches. Even the theoreticians have not been able to agree on the very basic bubble formation models to be used. It is generally agreed that the pressures involved in void formation and growth will not be great enough to damage the structures; however, it is expected that high pressures may arise when a sodium vapor collapses in sub-cooled sodium.

Scope of Work — Experimental and theoretical studies of void formation, growth, and collapse will be required to provide the basis for the accident models that will be used to predict coolant behavior during extremely abnormal situations. The mechanism and rate of void growth control the expulsion rate of sodium and local pressure variations during accidents if a subassembly is involved in a power-to-coolant-flow mismatch. Two-phase flow regimes occurring during accidents that result in subassembly voiding are to be identified, and an understanding in quantitative terms should be developed. Experimental studies to determine the conditions under which coolant flow may

be limited by critical flow (choked flow) are included. Experimental studies of void propagation will be required to determine how void formation in one subassembly affects its neighbors.

In all these studies, the experiments should be carried out under actual LMFBR conditions, i.e., flow, geometry, linear heat rate, sodium purity, high irradiation fields, aging, and the like. Particular attention must be paid to the following parameters:

1. The condition of the heating surfaces and the coolant in the LMFBR.
2. High heat fluxes up to 2×10^6 Btu/hr-ft² using indirect heating sources.
3. Close mockup of reactor subassembly geometry.
4. Flow rates comparable to those in an LMFBR.

After each test run, the experimental test sections should be removed for inspection of the subassembly. Particular attention must be given to the condition of the fuel pins, the helical ribbon spacers, and the wrapper can. Methods for detecting (1) the onset condition for boiling and (2) the fact that voiding has occurred in the subassembly should be evaluated as the test program is carried to completion. The detectors should be external to the subassembly.

Relationship to Current or Planned Programs — The programs planned in 10.2.1.2 are related to the onset conditions for the program in this segment. The test facilities provided in that section will be utilized to continue the study in this segment. The data and models developed will be used in section 10.2.1.3. This section includes essentially all of the LMFBR Program Plan Task 10-2.3 and subtask 10-2.2.5, with the exception that the interaction of molten fuel with coolant appears in section 10.2.2 of the B&W R&D program.

Budget — A detailed breakdown of the budget is included in Table 10-2. The cost, including the subassembly and greater propagation loops, is estimated to be \$20.6 million.

10.2.1.4. Analytical Methods and Engineering Techniques

Objectives — Accurate analytical methods and engineering techniques must be developed as the data and models for superheat and subassembly voiding become available. These analytical methods must be in a form that will permit their use in codes to study the characteristics of the LMFBR following equipment failure, system malfunction, or local fuel failure.

Summary of Current Technology — No methods are now available for predicting the sodium superheat at onset for boiling. The spread in the data indicates that even the experimenters are confused as to how the parameters affect the liquid superheat required to initiate boiling.

Several simple models have been developed for subassembly voiding—VOID, BURP, BLOW, and TRANSFUGE. These simplified models are available today and must be modified and improved as better experimental information becomes available. Equally as important as the hydrodynamics and heat transfer during boiling is the ability to account for the reactor feedback effects, in particular those effects which govern the neutronics of the reactor. Two codes, FORE and AIROS, have been modified to incorporate simple void-type feedback. For instance, FORE incorporates an input reactivity-time-table which is initiated at a given fixed temperature. It is desirable to develop a space-time-dependent code that can be used to evaluate the power feedback from the local voiding events. In addition to the proper neutronics model, it will be necessary to incorporate the appropriate stress models to evaluate the extent of core damage.

In conclusion, the present capability for analyzing fast reactor accidents is inadequate. Tools to treat the major safety problems, sodium voiding from one or a few blocked subassemblies, propagation to other subassemblies, loss of subassembly integrity following a blockage, etc., is almost non-existent.

Scope of Work — Mathematical models will be developed for boiling and voiding for accidents initiated from malfunctions while operating at steady state, or from malfunctions initiated

during transients. These models will be included in practical analytical codes for the evaluation of initiating accidents.

1. Continue the development of simplified voiding models for use in the near-term LMFBR study programs.

2. Develop a computer model that combines transient heat transfer and two-phase coolant hydrodynamics in a tube bundle. The model must be in a form which permits its incorporation into a space-dependent neutronics code. The effects of voiding will require that the neutronics model carry four or more neutron energy groups to adequately describe the feedback to the neutron power during a voiding accident, even when using spectrum-weighted cross sections. The boiling model must then incorporate the proper weighting methods for the neutronics parameters in those regions where voiding is to be treated.

3. Develop a computer model for a pin-type analysis which will accept the boiling model described in item 2. A number of available models consider phase changes in the fuel. However, these will require improvements to account for fission gas release and volume change during a phase change. Here we are interested in the short-term transients, rather than the long-term swelling with lifetime.

4. Develop a stress model for evaluation of the condition of the fuel from the temperature profiles. The problem may be resolved by calculating the stresses from the pin temperature profile for comparison with the allowable stress while considering plastic flow, creep, fatigue, and cumulative damage. In the LMFBR reactor, the thermal stresses are high during steady state; this is evident from the high thermal gradients. The extremely large heat transfer coefficient from a wall to the sodium coolant aggravates the situation during transients, especially transients associated with boiling and two-phase flow. The surface layer of the materials may be so constrained by the remaining materials of the fuel pin that it cannot assume the expansion corresponding to its temperature.

Relationship to Current or Planned Programs — The programs planned in 10.2.1.3 provide the basic experimental input for the models being developed for sodium boiling and for

their verification. The reactor physics program will provide the proper space-time kinetics program into which the developed models will be incorporated, i.e., the basic space-time solution of the diffusion equations. This section includes work included in Task 10-2.1, 10-2.2, and subtask 10.2.3.3 of the LMFBR Program Plan.

Major Facilities and/or Equipment Required — No special major facilities or major equipment items are required.

Budget — A detailed breakdown of the budget is given in Table 10-2. The estimated total cost is \$2.2 million.

10.2.2. Design Basis Accident Analysis and Energy Release

10.2.2.1. General

The design basis accidents (DBAs) are a class of hypothesized, uncontrolled accidents that occur in an unusual sequence of component and system failures. The most severe consequences of DBAs form the basis for the design of the plant features and engineered safety features that are required to protect the public and the plant operators.

An example of an accident resulting from some unknown initiating accident is the following: (1) fuel and coolant temperature rise well beyond design levels, (2) fuel elements melt and release high-temperature fuel into the coolant channels, and (3) coolant is expelled from the core as a two-phase mixture generated by contact with the molten fuel. The expulsion from the core under the vapor pressure of the coolant will move some fuel, which would result in a reactivity loss of decrease; however, this may be overcompensated for in some cases by coolant movement and voiding of segments of the core. If it is overcompensated for by the coolant movement, then the reactor could go on for an extremely short period where the core would be disassembled under the vapor pressure of its contents.

Objectives — The objectives of the DBA analysis and energy release are to demonstrate that the plant can withstand any accident without undue risk to the public. This is one of the

basic findings required by the AEC before they will issue a provisional operating license. It is apparent that considerable R&D will be required to develop and verify the analytical tools to be used for analysis and to develop the necessary data to support the AEC's findings.

It will be necessary to develop and verify the analytical methods required to demonstrate a knowledge of the thresholds and mechanisms of fuel failure, prefailure fuel movements, fuel motion resulting from fuel failure, coolant motion for pressure relief, and destructive energy release or the partition of the energy in the relief mechanisms. It should be emphasized that analytical methods are normally used when evaluating the effects and determining the design requirement for engineered safety features that are imposed by a DBA.

It will be necessary to develop the pressure-density-energy relationships for the material components of the core. This is included in the materials segment of the program, section 7, and is indicated here because this task is the basis for the need.

Summary of Current Technology — It has almost become traditional in DBA analysis to calculate the nuclear excursion that results from superprompt critical reactivity insertions brought about by core meltdown and compaction.³⁵⁻⁴³ At the Aix conference, four papers⁴⁴⁻⁴⁷ were presented; this seems to indicate that it has also become traditional for fast reactor safety conferences. Core behavior leading to the explosion phase is not detailed in any of the analyses. If it is assumed that the nuclear excursion results from an initiating, less-severe accident involving the core, it will be necessary to examine the course of the postulated initiating accident, including the degree of coherence in fuel failure, the amount of fuel and coolant movement, and finally, the possibility of secondary collapse of the remaining fuel. If it is assumed that the nuclear excursion results from a severe localized accident, it will be necessary to consider the severity and speed of propagation of damage and the degree of coherence in secondary movements. It should be pointed out that for most of the reactors considered today, the resulting destructive energy releases calculated are about the equivalent of 500 pounds of TNT. It is interesting to note that the German paper at the Aix conference considered the molten fuel-sodium energy exchange that might occur following the traditional Bethe-

Tait accident. They noted that molten fuel remains after the nuclear excursion and that it can come into contact with large amounts of sodium coolant. Their calculation assumed that an adiabatic expansion of the sodium vapor bubble occurred down to 20 atm, the design pressure of the reactor cavity considered. The resulting mechanical energy release was about twice the mechanical energy release calculated for the nuclear excursion using an adiabatic expansion of the fuel materials.

It is very probable that the traditional re-assembly accident may not be possible in nature and that reasonable assurance can be demonstrated at a more realistic level.

The transient reactor test facility (TREAT) was designed to study the safety characteristics of fast reactor transients.⁴⁸⁻⁵⁶ It is a graphite-moderated, pulsed testing reactor in which relatively large specimens can be subjected to large integrated fluxes. The graphite-matrix in which 16- to 18-micron urania particles are suspended serves a double purpose: (1) it acts as a sizeable heat sink, and (2) it is the moderator which, upon heating, generates a large negative reactivity feedback by increasing the neutron leakage from the core. The basic approach in the TREAT fuels program is to separate the various phenomena involved in fuel meltdown. Burst durations for meltdown experiments can range from 150 msec to a few seconds. This time regime is suitable for the investigation of materials behavior, overpower conditions (such as uncontrolled rod withdrawal following a driver malfunction) fast flow blockage at power, and the like. However, TREAT is not appropriate for studying slower events like subassembly blockage during startup, slowly developing fuel faults, or slow penetration of subassembly walls following a localized event.

The TREAT studies on oxide fuels have largely been concerned with the material's behavior under accident conditions prior to gross disintegration of the pin. Some destructive tests have been run in inert gas in transparent capsules to establish fuel behavior by using high-speed color photography and fast-acting thermocouples in the cladding. EBR-II-sized, gas-bonded UO_2 pins with low burnups have been run. Some fuel swelling, distortion, and surface cracking were found in the post-experiment examination of the fuel. For up to 3 atom percent burnup, none of the tests showed evidence of

fragmentation, disintegration, or distortion of the fuel due to internally trapped fission gasses.⁵⁷

The conclusions differ considerably from those of Hanson and Field for their TREAT tests.^{58,59} Their interpretations and analyses led to these conclusions: (1) the release of fission gasses from mixed oxide fuels which are melted during a transient may be one of the primary mechanisms for cladding failure; (2) the release of fission gasses from the mixed oxide fuels which are melted during a transient provides a substantial driving force for the redistribution and dispersion of the fuel; and (3) it appears that the foaming of irradiated mixed oxide fuel does not occur even when a substantial portion of the fuel is melted.

The first real basic DBA information will come from TREAT experiments with the high-energy meltdowns of oxide fuels in sodium. The high-temperature sodium autoclave for TREAT is sized for a nine-pin cluster, and it has a piston above the sodium to simulate the inertial head of coolant above the core.⁶⁰ Since the TREAT studies on the oxide fuels have concentrated on the pin material's behavior under accident conditions prior to gross disintegration of the pin with some destructive tests run out of the sodium environment, high priority must be given to the study of the thermal interaction of the core component materials under reactor conditions. The movements of the fuel and coolant following pin failure and the pin failure rate will largely determine the destructive energy release and will probably determine whether it is possible for the core to be reassembled in the vessel into a critical geometry.

The Europeans have been concerned with the possibility that the violent injection of molten fuel into the sodium may propagate because of the explosion created by the violent boiling of the sodium. At the Aix conference it was indicated that rough calculations of the kinetic energy potentially available from the melting of a few tens of pins is in the range of the energy release associated with 0.5 to 2.0 pounds of TNT.⁶¹ Tests have also indicated that the threshold for propagation from one subassembly to its neighbors is between 0.5 and 2.0 pounds of TNT or its equivalent.

The final product from the experiments on core behavior under accident conditions is the derivation of the analytical tools, which describe the results as fully as possible in such a manner that they can then be used to predict the behavior of the LMFBR. The coupling is very poor between the computer programs used as analytical tools to evaluate the core conditions and the kinetics programs used to evaluate the power behavior. At the present time only point reactor models are used in the kinetic evaluation of the core's time behavior. In FORE-II, for instance, the effects of voiding of a subassembly must be input as a time dependent table initiated at a preselected initiating coolant temperature.⁶²⁻⁶⁴ The space-dependent effect of voiding on the local power shift may play an important role in the propagation of the void from one subassembly to the next, depending on the subassembly's location in the core. Even when using appropriately weighted cross sections, it appears that four or more neutron energy groups will be required to calculate the voiding feedback during a transient or accident in which the reactor is carried into the voiding condition. The computer codes used for sodium void formation are described in section 10.2.1. However, the boiling models presented in that section were not developed to consider the violent boiling associated with the injection of molten fuel into the sodium coolant. The final models selected will require verification tests on actual fast reactor subassemblies in a high-risk, fast-flux test facility. It seems certain that a firm demonstration of reactivity during subassembly meltdown will be required for the oxide fuel before the present concern over supercritical assembly or reassembly accidents can be alleviated completely. A considerable amount of reactivity was lost during the Fermi meltdown accident; however, Fermi uses metallic fuel.^{65,66}

Scope of Work — Experiments are needed to provide information for predicting events in an operating plant. More realistic models of the progression of events must be developed to reduce the uncertainties in margins of safety. The adequacy of the analytical methods and models must be demonstrated to ensure that the requirements developed for the core protection system, safety instrumentation, containment, and consequence-limiting safety systems are

reasonable, and that no undue risk is required for the design, siting, and erection of the LMFBR and its operation by a utility.

1. Experimental programs aimed at understanding the fuel pin behavior under power transients must continue to provide model information leading to fuel pin failure during unexpected abnormal power transients. These experiments must consider (1) fuel burnup, (2) gaseous fission product inventory (vented versus non-vented), (3) coolant-fuel interaction following failure, (4) surface heat fluxes, (5) molten fractions, and (6) time rate of power change.

2. Experimental programs aimed at an understanding of subassembly behavior following a meltdown within the subassembly caused by power transients or coolant starvation: These experiments must pay particular attention to (1) the ability of the subassembly can to prevent damage propagation in the form of further melting, (2) the movement of fuel and coolant for the prediction of reactor reactivity feedback effects, and (3) the resulting pressure transients. These experiments must simulate the LMFBR conditions closely with a close mockup of reactor geometry and coolant conditions.

3. Experimental p-V-e data to improve the equation-of-state representation for use in analyses of core disassembly caused by the vapor pressure of its components: Even slightly higher temperature data could result in a major reduction in the uncertainty of the estimated critical constants. The possibility of obtaining the equation of state directly through the use of explosive detonations that generate shock waves through the material of interest should be considered.

4. Development of the analytical models to be used by the reactor designers to determine the design requirements of the engineered safety features which are necessary to protect the public and to reduce the risk of plant operation to an acceptable level.

10.2.2.2. Fuel Pin Behavior Under Extreme Conditions

Objectives — First, to determine the behavior of the fuel pin during various transient conditions which carry the pin to destruction. These experiments will consider (1) fuel burnup,

(2) gaseous fission product inventory (vented versus non-vented), (3) prefailure fuel movement, (4) surface heat fluxes, (5) time rate of power change, (6) molten fractions at failure, and (7) coolant-fuel interaction following failure.

The second objective is to develop the analytical models to predict the behavior of the pin under failure conditions that result from accidents occurring in an unusual sequence of component and system failures.

Summary of Current Technology — The studies on oxide fuels have concentrated on the behavior of the pin under accident conditions prior to its gross disintegration. Some destructive tests have been run in inert atmospheres to permit visual observation. Some burnup samples have been studied, but the data released in the literature have not been sufficient to make clear-cut decisions, and there are often conflicts in viewpoints. Some information concerning the heat transfer and dynamics of the system as molten mixed oxide fuels are released from the fuel to the sodium coolant has been reported. However, insufficient attention was placed on the following: (1) the heat transfer characteristics of the dispersion, i.e., time constants, heat transfer coefficients, resulting fuel characteristic size, etc.; (2) the movement of the molten fuel and coolant throughout the accident; and (3) local pressures developed within the system and their rate of relief as a function of molten oxide fuel-to-sodium fraction changes.

Scope of Work — The current program should be expanded to include a large number of tests. As with most other processes in nature, the results may be somewhat statistical in nature and somewhat fuzzy unless the observation techniques are better defined. The experiments must concentrate on the behavior of the pin in the reactor environment. It is expected that the pin's environment will play a significant role in the pin failure mechanisms, since the heat transfer coefficient from the cladding to the coolant is extremely large. Particular attention must be given to the following parameters: (1) gaseous fission product inventory; (2) fuel burnup; (3) prefailure fuel movement inside the pin; (4) movement into the coolant; (5) surface heat fluxes, up to 5×10^6 Btu/hr-ft²; (6) time rate of power change; (7) molten

fraction of fuel at failure; (8) coolant-fuel interaction following failure; (9) the type of mixed oxide fuel; (10) cladding materials; (11) the bond between the fuel and cladding; and (12) fuel pin dimensions.

Relationship to Current or Planned Programs — The program proposed in this section is essentially an extension of the work proposed in 10.2.1. It will provide the experimental program required to develop the pin failure models that will be used in accident analysis methods and for subassembly behavior models.

This section includes the work proposed in subtasks 10-2.4.2, 10-2.4.3 and 10-2.3.4 and segments of Task 10-2.5 in the LMFBR Program Plan.

Major Facilities and/or Equipment Required — A reactor loop is needed for long-term, slowly increasing power transients. The TREAT loop is needed for intermediate- to short-term power transients; this loop can be expected to carry the bulk of the program. A PBF loop capable of shorter-period burst (less than TREAT, which is about 40 msec) is required.

Budget — A detailed breakdown of the budget is given in Table 10-3. The estimated cost, including the test loops, is about \$4.1 million.

10.2.2.3. Subassembly Behavior Under Extreme Conditions

Objective — The experimental program will concentrate on the basic understanding of fuel and coolant behavior and distribution following a meltdown within a subassembly.

Summary of Current Technology — Very little applicable data have been released from bundle tests. A number of investigators have looked at the problem theoretically, but experimental input has been minimal. The European designers seem to be more concerned about subassembly behavior than do the U.S. designers, who have tended to concentrate on the Bethe-Tait analyses for the DBA for the engineered safety features that are required to protect the public, without paying due consideration to the processes involved. The European tests indicate that the threshold for propagation of damage from

one subassembly to the next is in the neighborhood of 0.5 to 2 pounds of TNT or its equivalent, and this is the same range they are predicting for the interaction of molten fuel from a few tens of pins released to the contained sodium in a subassembly. It is apparent that the subassembly behavior under meltdown conditions is vital to the description of the core for accidents that involve overheating of a subassembly, such as the loss of coolant flow to a given subassembly. The Fermi reactor has encountered such an accident; however, in that case metallic fuels were being used, and no large sodium pressures were generated.

It can be said that the state of the art for fuel motion during meltdown is in its infancy, with most attention being given to the pin failure mechanisms discussed in section 10.2.2.2, rather than to the course of meltdown in a subassembly.

Scope of Work — This segment of the experimental program is designed to investigate the subassembly behavior following an accident that results in a power transient which causes fuel to fail by local melting, or an accident that results in local coolant starvation and causes subassembly melting. To improve reactor feedback analyses, the experimenters must pay particular attention to (1) the various effects, such as pressure, shock, heat transfer, and temperature; (2) the ability of the subassembly to prevent damage propagation in the form of further melting; and (3) the movement of the fuel and the coolant. It should be mentioned that in the Fermi meltdown, the reactor lost reactivity, even though there were no large pressure buildups in the molten subassemblies.

The evaluation of the resulting dynamics is essentially a continuation of the boiling and two-phase flow work in section 10.2.1. Experimental and theoretical studies of void formation, growth, and collapse are required to form a basis to build accident models that predict material behavior during partial core meltdown. The rate of void growth controls the expulsion rate, and local pressure variations, which, along with the shock phenomena, establish the damage that may lead to damage propagation. The bubble collapse, as it extends into the highly subcooled regions, should be investigated with extreme care, since the pressures generated by the collapse may exceed

those generated by the void formation and thus may result in more physical damage to the reactor vessel and the core.

In all the cases to be studied, the experiments must be carried out under conditions that exist in an LMFBR, including flow, geometry, linear heat rates, sodium purity, and high irradiation fields. After each test run the experimental test sections will be removed and examined to establish the condition of the test section and to estimate damage. The location of the fuel and its physical characteristics will also be established. The anomalous reactivity transient introduced into the driver core will be recorded as a function of time. In essence, we must establish on a firm basis the reactivity effects associated with a local incident and ensure that the engineered safety features incorporated into the core design will maintain the local nature of the event and confine the damage to the subassembly involved. Particular attention must be paid to the condition of the subassembly wrapper can and other structural components.

Since the thermodynamic properties of sodium are intimately involved in this study, some work on the physical properties of sodium is required. As was pointed out by Golden and Tokar,⁶⁷ the critical parameters (pressure, temperature, and specific volume) are not available because considerable uncertainty exists regarding reported values, and there are no experimental bases upon which a best choice could be made. It is desirable to extend the present p-V-T data up to between 4000 and 4500 F and to evaluate the critical constants to permit further extrapolations of the data. The priority of this work must be increased, since it will play a vital role in fuel meltdown accidents in the interaction between molten mixed oxide fuels and the sodium coolant.

Relationship to Current or Planned Programs — The programs planned in 10.2.2.2 and 10.2.1.2 are related to the onset conditions for the program in this section. They will provide the experimental theoretical input to 10.2.2.3 for model development and their incorporation into accident analysis methods.

The fuels and materials section (section 7) will provide p-V-E data to improve the equation-of-state representation of the oxide fuel for use in disassembly analyses of the core due to the

vapor pressure of the mixed oxide fuels. Even slightly higher temperature data could result in a major reduction in the uncertainty of the estimated critical constants. The possibility of obtaining equation-of-state data directly through the use of explosive detonations that generate a shock wave through the mixed oxide fuel should be considered.

This section provides the remaining experimental work for subtask 10-2.3.4 and Task 10-2.4 of the LMFBR Program Plan. It also includes much of the experimental work for Task 10-2.5 (in conjunction with section 10.2.6). It also covers the necessary experimental program required for Task 10-2.8 and 10-2.9, where additional information is needed.

Major Facilities and/or Equipment Required — One reactor loop for a complete subassembly is required; it must incorporate a means for removing the subassembly's remains intact to establish the final location of the fuel. The loop inlet and outlet must contain sufficient inlet and outlet strainers to ensure that significant amounts of fuel will not be carried into the sodium coolant conditioning equipment. A TREAT and PBF loop, similar to the reactor loop, is needed to establish transient power behavior effects. One multi-subassembly loop, as described above, is required to establish the propagation mechanisms or effects. This loop must be capable of accepting one whole and six partial subassemblies and will require a new test facility capable of both steady-state and transient operation.

Budget — A detailed breakdown of the budget is given in Table 10-3. The cost, including the required test loops, is estimated at \$9.4 million.

Table 10-3. Budget for Design Basis Accident Analysis and Energy Release

Task	Cost, \$ × 10 ³																Total
	Fiscal year																
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983		
Facilities																	
1. Single-pin loop																	
Reactor loop	150	150	--	--	--	--	--	--	--	--	--	--	--	--	--	300	
TREAT loop	150	150	--	--	--	--	--	--	--	--	--	--	--	--	--	300	
PBF loop	--	--	--	150	150	--	--	--	--	--	--	--	--	--	--	300	
2. Subassembly loop																	
Reactor loop	150	250	--	--	--	--	--	--	--	--	--	--	--	--	--	400	
TREAT loop	--	150	250	--	--	--	--	--	--	--	--	--	--	--	--	400	
PBF loop	--	--	--	150	250	--	--	--	--	--	--	--	--	--	--	400	
3. Multi-subassembly loop																	
STF loop	--	--	--	--	350	350	--	--	--	--	150	150	--	--	--	1,000	
Fuel Pin Behavior Under Extreme Conditions																	
Reactor loop	100	200	200	200	150	100	100	--	--	--	--	--	--	--	--	1,050	
TREAT loop	100	200	200	200	150	100	100	--	--	--	--	--	--	--	--	1,050	
PBF loop	--	--	--	100	200	200	200	150	100	100	--	--	--	--	--	1,050	
Subassembly Behavior Under Extreme Conditions																	
1. Subassembly																	
Reactor loop	100	250	200	200	200	200	200	150	150	100	100	--	--	--	--	1,850	
TREAT loop	--	100	250	200	200	200	200	200	150	100	100	--	--	--	--	1,700	
PBF loop	--	--	--	100	250	200	200	200	200	150	150	100	100	--	--	1,650	
2. Multi-subassembly																	
STF loop	--	--	--	--	--	200	250	250	150	150	150	250	250	250	100	2,000	
Analytical Methods and Engineering Techniques																	
1. Fuel pin failure model																	
Prefailure fuel movement	25	25	50	50	50	50	50	25	25	--	--	--	--	--	--	350	
Failure fuel movement	--	--	25	25	50	50	50	50	50	25	25	--	--	--	--	350	

Table 10-3. (Cont'd)

		Cost, \$ × 10 ³															
		Fiscal year															
Task		1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	Total
2.	Subassembly model																
	Fuel movement	--	--	--	25	50	75	75	75	25	25	25	50	75	75	25	600
	Failure propagation	--	--	--	--	25	50	75	75	25	25	25	25	50	50	25	450
3.	Core failure model																
	Subassembly-initiated malfunction																
	Fuel movement	--	--	--	--	--	--	25	25	25	25	25	25	--	--	--	150
	Molten fuel-coolant interaction and movement	--	--	--	--	--	--	--	25	50	50	50	50	50	50	25	350
	Failure propagation	--	--	--	--	--	--	--	--	25	25	50	50	50	50	25	275
	Energy partition	--	--	--	--	--	--	--	25	50	50	50	50	50	50	25	350
	Traditional core failure model																
	Prefailure description	25	50	50	50	50	50	50	50	25	--	--	--	--	--	--	400
	Failure description and energy generated	25	25	25	25	25	25	25	25	25	--	--	--	--	--	--	225
	Energy partition	50	50	75	75	75	75	50	50	25	--	--	--	--	--	--	525
	Fuel location following accident	50	50	75	75	75	75	50	50	25	--	--	--	--	--	--	525
	Energy and Na containment evaluation																
	In-vessel energy absorption	--	--	--	--	--	--	--	--	--	50	50	50	50	50	50	300
	Shock wave damage	--	--	--	--	--	--	--	--	--	25	25	25	25	25	25	150
	Sodium expulsion damage	--	--	--	--	--	--	--	--	--	50	50	50	25	25	25	225
	Pressure retention	--	--	--	--	--	--	--	--	--	25	25	50	50	25	25	200
	Fission product containment evaluation	--	--	--	--	--	--	--	25	50	50	50	25	--	--	--	200
	Total	925	1650	1400	1625	2300	2000	1700	1450	1175	1025	1100	950	775	650	350	19,075

10.2.2.4. Analytical Methods and Engineering Techniques

Objectives — The objective of this task is the development of analytical methods and engineering techniques which, when applied to an LMFBR, will predict its response to a series of reactor malfunctions and will determine the state of the reactor at all times. These methods must be capable of examining the effects of major accident up to and including the DBA. It should be noted that the experimental program upon which these methods depend is in its infancy and that several years of vigorous experimentation will be required before important results can be expected. At that point, a re-examination and re-orientation of the program will be required.

However, there are also some short range objectives which are required to establish some understanding of fuel motion in accident situations. These analytical methods must be somewhat conservative in nature and provide greater protection factors than are required to satisfy the guidelines established for public exposure. As a result of this kind of reasoning, the fast reactors designers are faced today with the development of engineered safety features to accommodate the energy released as a result of a superprompt critical excursion, even though there are no clear cut or feasible accidents that could initiate the reactivity insertions required for a significant energy release. The analysis of the traditional superprompt critical accident may impose unwarranted conservatism on the LMFBR, which must be capable of operating safely and competitively in a utility environment. Some effort will be expended to improve the calculational techniques as a better understanding of the physical parameters is obtained.

To summarize, the goals of this section are to provide methods and engineering techniques to predict any energy release associated with the dynamic characteristics of the reactor core.

Summary of Current Technology — There are no analytical models for evaluating the accidents associated with the dispersal of molten fuel into the sodium coolant. Only recently have the results of fairly crude calculations appeared in the literature, and these have indicated that for a subassembly the equivalent energy release is

about 0.5 to 2.0 pounds of TNT or the equivalent. This is also the estimated range for the propagation of damage across the wrapper can of a subassembly.

The analytical methods available for analyzing superprompt critical excursions are in relatively good shape. However, we must (1) develop realistic models to describe the reactivity input rate and the reactivity feedback effects throughout the accident; (2) improve the equation-of-state representation for mixed oxide fuels; and (3) develop realistic energy partition mechanisms, including the dispersal into sodium coolant; since there are essentially no models available for realistically evaluating reactivity input functions, and the equation of state requires tremendous extrapolations.

Scope of Work —

1. Initiate the study of subassembly meltdown using simplified models for the meltdown, energy transfer, and bubble formation. These models are required for the near-term LMFBR study programs to improve the understanding of the requirements for the engineered safety failures to be incorporated into the core.

2. Develop fuel meltdown models to be used in the major accident evaluations. These models must be compatible with the methods developed to study the dynamic response of the reactor system to equipment malfunctions or failures. The effects of fuel movement and core voiding may require that even those codes become more sophisticated in their space-dependent feedback evaluation.

3. Develop the energy release models to evaluate the destructive damage external to the core following a meltdown accident, including any nuclear excursion. These models must be capable of predicting the bubble growth and energy partition. They must take into consideration all of the barriers between the core and the designated energy barrier, such as the core vessel, the reflector, and the like. These barriers will absorb a considerable amount of energy during the pressure relief processes.

Relationship to Current and/or Planned Programs — This section will receive experimental input from many

segments of B&W's R&D program, including 10.2.1, 10.2.2.1, 10.2.2.2, and 10.2.2.3 from section 10 as well as data from other major sections (2, 3, 4, 6, and 8).

This section includes much of the methods development work included in the LMFBR Program Plan Tasks 10-2.2, 10-2.4, and 10-2.5. It incorporates most of the analytical methods required for the design basis analysis of the primary-pool concept LMFBR.

Major Facilities and/or Equipment Required — No special facilities or equipment are required.

Budget — A detailed breakdown of the budget is given in Table 10-3. The total cost of the methods development and their incorporation into the proper transient codes is estimated at \$5.6 million.

10.2.3. Behavior of Radioactive Materials

Traditionally, the fission and activation product leakage caused by an accident has been a major consideration in the analyses of potential accidents in all reactors. Several factors are unique to the sodium-cooled fast-breeder reactor: (1) the presence of sizeable quantities of plutonium; (2) radioactivity of the primary sodium; (3) the high fuel burnup; (4) the sodium combustibility; (5) the gettering properties of sodium for many of the fission products, including the iodines; and (6) the intentional release of fission products into the sodium through pin vent mechanisms.

This task in the R&D program is associated with the release of radioactive materials to the environment during normal operation and during and following accidents. To be considered are the fission products, the activation products, and the fuels. The program could be considered in two parts under each section: (1) gasses and fission products released through the pin vents, and (2) fission products released to the coolant when fuel is exposed to the coolant.

10.2.3.1. Objectives

The overall objectives of this task in the R&D program form the basis for (1) methods for the evaluation of the gaseous fission products released through pin vents and their transport

throughout the plant; (2) fission products and fuel materials release, transport, and deposition throughout the plant following a fuel failure up to the meltdown of one subassembly; and (3) the release, transport, deposition, and dispersal of fission products, activation products, and fuel material inside the reactor building following the design basis accident(s).

The basic objectives are, then, to develop the basic information and models for (1) the evaluation of the fission and activation products and distribution of fuels materials and fission and activation products within the reactor system during normal operation; (2) the distribution of fuel materials and fission and activation products following local fuel failure within the reactor system; and (3) the distribution of fuel materials and fission and activation products within the reactor building (containment) following a DBA. The first two objectives deal with the ability of the plant to cope with the radioactive material distributed through it and to maintain the discharges from it below acceptable levels. The ultimate goal of the third objective, when combined with section 10.2.4, is to provide reliable information and confident techniques for analytically predicting the accident consequences in a form that is useful for the design, analysis, and evaluation of engineered safety features and the development of basic engineering design criteria for the reactor and the plant.

10.2.3.2. Summary of Current Technology

Considerable effort has been expended by the AEC in performing studies relating to fission-product release and behavior in reactor accidents.⁶⁸ Until recently, the primary effort has concentrated on thermal water reactors.

A large effort is being carried out by ORNL in an extensive study of the release of fission products upon melting and vaporizing of a variety of fuel materials—both in-pile and out-of-pile.⁶⁹⁻⁸⁵ These simulations of possible reactor-accident conditions in water reactors are carried out in a small-scale containment mockup facility and in a pilot-plant-scale nuclear safety pilot plant. The information generated considers both the release of fission products from the fuel and their behavior in the containment systems.

Analytical models for fission-product transport have been proposed at ORNL^{79,86-89} and at BMI.^{75,90} Much engineering information has been accumulated to date, but the development of models and the understanding of processes is not complete enough for the analysis of accidents. Currently, a large effort is being devoted to large-scale proof tests in the LOFT program being carried out at NRTS,⁹¹ where a reactor core will be tested in an actual blowdown situation. The containment research installation at ORNL⁷¹ and the containment systems experiment at PNL are a part of the LOFT program to facilitate the study of fission product behavior in containment systems.

Part of the continuing program at ORNL is aimed at controlling the accidentally released fission products. A method for characterizing simple aerosols is used to test several absorber systems for removing very fine radioactive particles from air streams. This characterization method measures the distribution of radioactivity on the walls of an appropriate channel through which the aerosol-vapor system flows under controlled conditions. Similar engineering-scale experiments are being carried out or are planned for all of the facilities discussed above.

A number of parameters affect the fractional fission product release (release rate/birth rate) from UO_2 . The LOFT support work at ORNL will examine a number of variables such as fuel density, pin size, burnup, pin atmosphere, and the like.⁷⁸ The effect of fission density has received considerable attention already at ORNL. In their earlier work, ORNL's investigators found that an increase of fission density will result in a decrease in the fractional release rate if the UO_2 temperature remains constant over the fission densities of $(1.8 \text{ to } 9) \times 10^{12} \text{ f/cm}^3\text{-sec}$.⁹² This observation led them to postulate a defect-trap model, wherein the gas was released by a coupled diffusion-trapping process.^{93,94} Southier found that there was no change of fractional release rate in the range of $(1.8 \text{ to } 3.8) \times 10^{11} \text{ f/cm}^3\text{-sec}$.⁹⁵ Using single-crystal spheres of enriched UO_2 to obtain much higher fission densities, it was found that for fission densities starting at $9 \times 10^{13} \text{ f/cm}^3\text{-sec}$, and going to about $1.5 \times 10^{14} \text{ f/cm}^3\text{-sec}$, the fission-created traps become less effective and an increase in fission rate density increases the fractional release.^{96,97} There are no data for the

intervening parts of the curve, but the minimum fractional fission gas release for krypton was estimated to be about 2×10^{-4} at a fission rate density of $2 \times 10^{13} \text{f/cm}^3\text{-sec}$ at 1400 C. The LMFBR fuels will operate up to $6 \times 10^{13} \text{f/cm}^3\text{-sec}$. Characterizations of this type must be extended to other fission products, such as xenon, iodine, and the like, and to other pertinent parameters.

A research program at BNL is providing information on the physical and chemical state of fission products that might be released from a defective subassembly.⁹⁸⁻¹⁰⁴ The primary objective of this continuing program is to develop a fundamental understanding of the chemical reactions, mechanisms of release, and transport behavior of the fission products emanating from nuclear materials under conditions likely to be encountered in reactor accidents. The problems studied encompass those related to the safety of water-cooled, gas-cooled, and sodium-cooled reactors. They are studying the effects of temperature, burnup, and the surrounding chemical environment on the form of the fission products released from fuel materials. The physical properties of the fission products or associated compounds are evaluated.

More recently, studies directly related to the safety of sodium-cooled fast reactors have received primary emphasis at BNL. The chemical states of fission products in sodium are being determined from thermodynamic data, and the probable volatile species are being studied with a considerable amount of fission product-sodium equilibrium vaporization information being released. Measurements of the excess thermodynamic properties of infinitely dilute NaI solutions in sodium have been completed. Studies are currently in progress to measure both thermodynamic and transport properties of sodium solutions containing cesium, rubidium, strontium, NaI, barium, tellurium, and antimony. In these studies, BNL experimenters have also considered the formation and agglomeration of aerosols produced upon the high-temperature vaporization of fuels. The temperature of the materials being studied was rapidly raised above the melting point to simulate an accidental power excursion. Individual studies were made with PuO_2 and UO_2 in both argon and nitrogen gasses. In addition, they studied combinations of vaporized PuO_2 - UO_2 and sodium vaporized into moist air mixed with vaporizing UO_2 .

To evaluate the release and transport characteristics of fission products in a molten sodium environment, experimental and analytical studies are also being performed at Atomic International.^{105, 106} Injection methods have included fission-product simulants, equilibrium temperature studies using diffusion of fission products from pre-irradiated fuel, nuclear-transient melting of pre-irradiated fuel in a sodium environment, and melting by condenser discharge. The general objective of their work is to evaluate the ability of the sodium coolant to retain specific fission products released to the coolant stream. Specific objectives have been (1) to determine the ability of sodium to retain radioactive iodine as a function of such parameters as sodium temperature, elapsed time, sodium-to-fuel and sodium-to-gas volume ratios, geometric relationships, and conditions of the released iodine; and (2) to evaluate the ability of boiling sodium and sodium in the vapor phase to retain fission products. These experiments have demonstrated that sodium tends to retain fission products, and extensive plating and deposition on various surfaces was observed throughout the experiments. An examination of contaminated reactor sodium piping revealed some correlation of fission-product activity with carbonaceous impurities in the system.

At the Aix conference, Saroul¹⁰⁷ of CEA-Cadarache presented results obtained from experiments on the behavior of fission products in sodium and argon. The results indicated that the only fission product detected in the cover gas, other than the inert gasses, was iodine-131. The iodine was found in the vapor trap and in the asbestos filter. No iodine was detected in the activated carbon filter; thus, Saroul concluded that the iodine is present only as sodium iodide. The ratio of the measured specific activity per unit volume of iodine in the cover gas to iodine in the sodium ranged from 10^{-5} to 10^{-4} with concentration in the liquid as a variable.

Some insight into the problems associated with the release of fission products in sodium systems has been gained in the operating experiences with sodium-cooled reactors in the U.S. and in other countries. EBR-I, EBR-II, SRE, and Fermi have operated with sodium or NaK as coolant, and they have experienced fuel meltdowns in which fission products were released and presented a burden

on the safeguards systems. A significant amount of information has been and can be gained by examining these incidents.

10.2.3.3. Scope of Work

The LMFBFR with vented fuel pins will be faced with the release of fission products to the primary sodium during normal operation. A fundamental understanding of these products' behavior in the fuel, in the pin gas volume, and in the release mechanism is required to realistically describe the release mechanism with analytical models. Such a model must be capable of following the release during steady-state operation and normally anticipated transients. Even though the fission products may be gaseous at the time of their release to the sodium, their subsequent decay may lead to non-gaseous fission products which may accumulate at some point in the system and result in a continual buildup during the reactor plant lifetime. In summary, the experimental programs must develop a fundamental understanding of the fission product release mechanism from the pin, the chemical reactions and release processes from the coolant (including the transport throughout the plant) under conditions likely to be encountered in the LMFBFR during normal operation.

A similar experimental program is required to develop an understanding of the processes involved when the nuclear materials undergo conditions likely to be encountered during accidents that result in local fuel damage. In this case, the spectrum of fission products to be accounted for will increase significantly and will include many products that will react chemically with the sodium coolant and which are soluble to varying degrees in the sodium. The transport and plateout throughout the system will be much more complicated, but the time base will be much shorter. The relative distribution for this type of accident becomes important in evaluating the plant in accordance with the guidelines established in 10 CFR, Part 20.

The third experimental program is required for the more catastrophic accident, and again the time base will be shortened. This experimental program will include cases where fuel is melted and dispersed into the coolant as a liquid; here it may solidify in the form of a slurry in the liquid. Some fuel may be released to the gas space in

the form of an aerosol. The rate of precipitation of the solid material will determine the release mechanism to the reactor building, since the seals between the reactor building and the reactor vessel in the top plug are not expected to remain leak-tight after an accident of this type. The relative distribution for this kind of accident becomes important in the evaluation of the containment system requirements to satisfy the guidelines established in 10 CFR, Part 100.

We must develop the analytical methods to be used in the design of those systems and the engineered safety features associated with the handling of radioactive waste, including the fuel materials, activation products, and fission products. These methods and techniques will be used to determine the distribution of the activated material throughout the plant during normal operation with or without vented fuel, and for anticipated accidents which may result in fuel failure.

Analytical methods must be developed for use in the evaluation of accidents in which a considerable amount of fuel may be melted and released into the sodium coolant. These methods are concerned with the accidents that fall into the DBA category. For these cases, one may expect that the design leakage rate through the sodium or energy barrier cannot be maintained, and the release and transport of the radioactive materials within the reactor building must be considered. Thus, the release of fuel to the sodium, the release of radioactive material from the sodium to the gas space in the reactor vessel, and the release from the gas space to the reactor building must be considered. All of the engineered safety features associated with air conditioning and containment cleanup must be included. In summary, we must develop (1) the methods to predict the radioactive materials inventory and their distribution for the containment of the fission products and other radioactive materials for the B&W pool-type LMFBR, in which the energy and sodium barrier is separate from the fission product barrier; (2) the engineered safety features required to protect the integrity of the energy and sodium barrier, as well as the containment or fission product barrier; (3) the analytical tools required to evaluate the effects of the DBA on the plant internal environment; and (4) the effects of a DBA on the public with respect to the release of fission products to the environment and direct irradiation.

10.2.3.4. Relationship to Current or Planned Programs

Considerable experimental input will come from Tasks 10-2.1 and 10-2.2, and probably some from 10-2.4. These four tasks are all brought together in this one task to develop the overall safety requirements for the protection of the public and the operators.

This section completes the requirement of Tasks 10-2.4 and 10-2.7 of the LMFBR Program Plan needed for the primary pool concept.

10.2.3.5. Major Facilities and/or Equipment Required

A capsule test is required in which fuel can be melted down in sodium. This test must be capable of simulating the melting of fuel in one vessel and of measuring the activity released through an energy-sodium barrier into a second vessel.

10.2.3.6. Budget

A detailed breakdown of the budget is given in Table 10-4. The total cost is estimated to be \$4.3 million.

10.2.4. Sodium Coolant Safety

The sodium and NaK heat transport media require special consideration owing to their high degree of reactivity with many common materials. The hazards associated with the use of sodium and NaK are caused by this high chemical reactivity with many common materials or substances and with human tissue. The reaction products themselves pose a problem as irritants, they may be highly corrosive, and they may lead to fires and explosions unless properly controlled. When sodium and NaK are exposed to a neutron flux, as in the reactor core, they pose an additional hazard because of the induced radioactivity. Those responsible for the design, operation, and maintenance of systems and facilities must have a thorough knowledge and constant awareness of these characteristics and the potential hazard.

10.2.4.1. Objectives

The overall objective of this program is to predict the behavior of a system following equipment failure where

Table 10-4. Budget for the Behavior of Radioactive Materials

Task	Cost, \$ × 10 ³															Total
	Fiscal year															
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
Facilities																
1. Will use DBA loops	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
2. Capsule test																
Reactor	50	--	--	--	--	--	--	--	--	--	--	--	--	--	--	50
TREAT	50	--	--	--	--	--	--	--	--	--	--	--	--	--	--	50
PBF	50	--	--	--	--	--	--	--	--	--	--	--	--	--	--	50
STF	150	--	--	--	--	--	--	--	--	--	--	--	--	--	--	150
Experimental Program																
1. Normal operation with vented fuel																
F-P release from fuel	25	25	25	25	25	25	--	--	--	--	--	--	--	--	--	150
F-P release through vent	25	25	25	25	25	25	--	--	--	--	--	--	--	--	--	150
F-P transport and deposition	25	50	50	50	50	25	--	--	--	--	--	--	--	--	--	250
2. Failed fuel (local damage)																
F-P release	25	50	50	50	50	50	25	--	--	--	--	--	--	--	--	300
F-P transport and distribution	25	50	50	50	50	50	25	25	--	--	--	--	--	--	--	325
3. Molten fuel-Na interaction																
F-P dispersion (gaseous, halogens, etc.)	--	--	50	100	100	100	100	100	50	50	100	100	100	50	25	1025
Fuel dispersion	--	--	25	25	25	25	25	25	25	25	25	25	25	25	25	325
Analytical Methods and Engineering Techniques																
1. Develop model for F-P inventory during normal vented operation	--	--	25	50	50	50	50	25	--	--	--	--	50	50	--	350
2. Develop model for F-P dispersion during meltdown conditions	25	50	100	100	100	100	100	50	50	50	50	100	100	100	25	1100
Total	450	250	400	475	475	450	325	225	125	125	175	225	275	225	75	4275

(1) sodium or NaK is released to an oxygen-containing environment, such as the reactor building; (2) sodium or NaK is released to an oxygen- and moisture-containing environment; (3) water and/or steam are released into a sodium environment; and (4) other chemicals or materials are released into the sodium environment. For the sodium-oxygen-moisture reaction in the containment building the experiments must cover both spray- and pool-type reaction surfaces.

For the release of sodium or NaK to the containment we must develop (1) the knowledge required to predict and characterize accurately the various effects and material burdens imposed by any accident, i. e., we must be able to predict the pressures, temperatures, fission product, plutonium, and sodium distribution; and (2) techniques for the mitigation, control, and cleanup resulting from any accident. For the release of various materials to sodium, we must also develop (1) the knowledge to predict and characterize accurately the various effects, such as pressure, temperature, shock waves, etc.; and (2) techniques to prevent, mitigate, control, and/or clean up any effect of the release to the sodium.

10.2.4.2. Summary of Current Technology

The technology summary discussed here considers references 107 through 114 without particular reference to any given work. AI has continued work on the sodium fire codes to be used for predicting reactor building response to pool burning; SOFIRE-A, SOFIRE-B, and SOFIRE-C are available in preliminary form to organizations under AEC contract who have a need to know. These codes must be pulled together in a better form and verified to prove analytical methods for the examination of pool fires inside the reactor building and inside the reactor vessel. They must then be extended to cover (1) the release of the various radioactive components (such as fuel, fission products, and activation products) from the sodium into the local fire environment; (2) the transportation throughout the reactor vessel, buildings, etc.; and (3) the precipitation and/or settling processes involved.

In summary, early sodium fire investigation obtained widely varying results for pool burning rate and the surface release rates for pool burning, both of which appear to be very

sensitive to the geometry and air movement. It has been reported that shallow pools burn as nodules on an irregular oxide surface, whereas deeper, well insulated pools burn on the molten metal surface. The large sodium fire experiment by AI was conducted in a forced-draft hood arrangement. The air and oxide effluent from the hood are discharged through an air-cleaning system. The burnpot in the hood and the hood itself were instrumented with thermocouples. Using this system, they found a pool consumption rate of about 6 lbm/hr-ft² and a release rate of about 2 lbm/hr-ft². The remaining oxide remained in the burnpot, where it settled to the bottom; this buildup eventually penetrated the surface and formed a hard crust. In this large fire experiment, 67% of the sodium remained as residue in the pot; and 69% of the iodine tracer remained in the pot, 8% was released, and 23% was unaccounted for. It was postulated that this was lost during the dissolution process.

A second major category is the spray fire caused by mass ejection, where significant quantities of sodium are injected into an atmosphere containing oxygen. The early experimental work carried out by Humphreys has been challenged by theoreticians. As the theoretical models are improved, they move closer and closer to the measured closed-vessel response. However, the measured response is still considerably less than the calculated value. The differences will continue to be reduced as heat transfer and reaction rate kinetics are incorporated into the models.

In both pool and spray fires, the oxygen concentration in the gas environment near the sodium is a very important variable to understand. Another important variable is the surface temperature of the sodium, since evaporation of sodium from the surface will also play a major role. In the case of spray fires, the particle size and velocity will also be significant in determining the burning rate.

In any attempt to extrapolate the experimental results to large containment systems, it must be remembered that the rate of pressure drop in a large reactor building may be considerably slower than that in the experimental vessels owing to the smaller ratio of surface area to gas volume. The large heat capacity of the experimental apparatus contributes greatly to the rapid rate of system pressure reduction immediately after the establishment of peak pressure.

10.2.4.3. Scope of Work

1. Pool Burning Characterization: Data on the combustion or burning rate of sodium in air and in reduced oxygen atmospheres are required for methods development in order to develop the analytical procedures to predict the rate of energy release for any given fire size. An equally important quantity is the fraction of burned sodium that is released to the gas, or the sodium oxide smoke release, since it is one of the factors that determine the airborne radioactive sodium available for leakage from the building. Large-scale tests are required in controlled environments which are similar to conditions that may exist in the B&W pool-reactor concept, namely, pool burning in the reactor building and pool burning in the reactor vessel when plugs are removed from the cover structure. These experiments must consider mass release rate, burning rate, particle size, and aerosol transport as functions of (1) the location of the fire, (2) the gas temperature and composition in both the reactor building and reactor vessel, (3) geometry of the system, (4) moisture in the atmosphere, (5) sodium temperature, and (6) energy release and its feedback to burning rate.

2. Spray Burning Characterization: The sodium spray may result from any number of accidents, up to and including the DBA. Several factors which lead to more violent burning, such as increased surface and increased oxygen available, are no longer diffusion-limited. During the DBA a small quantity of sodium may penetrate the sodium and energy barrier, since some seals will be damaged. This sodium from the reactor vessel may contain significant concentrations of plutonium, uranium, activation products, and fission products. It has been postulated that a fine mist of sodium distributed in ambient air may react explosively with the oxygen and cause shock waves to impinge on the containment structure. Experimental data are required from scale model systems in which particular attention is paid to (1) initial system temperature, (2) oxygen concentration, (3) sodium spray particle size, (4) distance to impingement surfaces and system geometry, (5) sodium particle size (or stream size in the case of a jet), (6) aerosol settling rate, (7) moisture content in the atmosphere, and (8) energy release and its feedback to the burning rate.

3. Other Sodium Reactions: Inside the reactor building are used various materials, such as freon, lubricants, and the like, which are necessary to the operation of the plant. The reaction of these materials with sodium must be investigated to evaluate the energy release potentials and the effects on the reactor core reactivity if they are released to the primary system sodium. Thus, particular attention must be given to (1) the reaction rate, (2) the energy release rate, (3) the reaction products, (4) the transport of the reaction products, and (5) the effects on core physical composition, in particular the reactivity effects. The sodium-water reaction is considered in Sodium Technology, section 5, and in Components Testing, section 3. It is mentioned here for completeness.

4. Analytical Methods and Engineering Techniques: We must develop analytical methods and design techniques as required by a designer to predict the design requirements for the nuclear island. These methods must be capable of considering sodium fires which may (1) develop in the reactor vessel as a result of cover structure failure, (2) be generated at many locations in the reactor building, and (3) develop outside the reactor building as a result of intermediate loop failure and release of intermediate loop sodium. In all cases, one must consider the burning of sodium, from pool burning through jet burning to spray burning. These methods must provide the designer with (1) pressure changes, (2) temperature changes, (3) aerosol generation (including particle size distribution and settling rates), (4) vessel surface temperatures in the event of surface coating by a spray, and (5) the effects of engineered safety features designed to control or remove particulate materials.

10.2.4.4. Relationship to Current or Planned Programs

The experimental work and the analytical methods support the methods developed in 10.2.1, 10.2.2, and 10.2.3. These combined in 10.2.3 to form the basis for development of the overall safety requirements for the protection of the public and the operators. Some experimental input will be received from sections 3 and 5 of the B&W R&D program, i.e., for the reaction of sodium with water. This

section covers Task 10-2.6 and subtask 10-2.2.7 in the LMFBR Program Plan.

10.2.4.5. Major Facilities and/or Equipment Required

1. A scale model ($\geq 1/10$ scale) of the pool LMFBR concept for verification testing of analytical models developed for sodium fire analysis is required. The experiment must be flexible enough to permit the verification for fires in both the reactor vessel and reactor building.
2. Bench-scale test equipment for sodium-air.
3. Intermediate-scale test equipment for sodium-air.
4. Bench-scale test equipment for other materials.

10.2.4.6. Budget

A detailed breakdown of costs is given in Table 10-5. The total cost is estimated at \$2.5 million.

10.2.5. Engineered Safety Features and Containment

This section emphasizes the systems, components, and equipment associated with the mitigation and/or control of the consequences of a reactor accident. The three general goals are (1) to evaluate the need for specific engineered safety features, (2) to develop reliable engineered safety features, and (3) to test these engineered safety features to reasonably ensure their availability at the time when the need arises. Since the requirements of this section of B&W's R&D program are in agreement with Task area 10-3 of the LMFBR Program Plan (Volume 10, Safety), this section will not be broken down into separate subsections; however, a 1-2-3 breakdown will be used in the "Objectives" and "Current Technology" portions.

10.2.5.1. Objectives

1. The first objective is to investigate the various ways of preventing major catastrophic accidents in the reactor core and to assess the need for and the adequacy of the engineered safety features to be incorporated into the core design as protection systems and devices. Included are studies and tests on core design arrangements, startup and operating procedures, redundancy of shutdown

Table 10-5. Budget for Sodium Coolant Safety

Task	Cost, \$ × 10 ³															Total
	Fiscal year															
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
Facilities																
1. Bench-scale equipment	10	--	--	--	--	--	--	--	--	10	--	--	--	--	--	20
2. Intermediate-scale equipment	--	--	20	--	--	--	--	--	--	--	--	--	--	--	--	20
3. Scale model (≤1/10 scale)	--	--	--	--	100	--	--	--	--	--	100	--	--	--	--	200
Experimental Program																
1. Pool burning	15	25	25	25	25	20	10	10	10	20	20	10	10	10	--	235
2. Spray fires	25	50	50	50	50	50	25	25	25	50	50	50	50	20	--	570
3. Other Na reactions (Freon, pump lubricant, etc.)	--	--	10	10	10	10	--	--	--	--	20	20	20	20	--	120
4. Combination of pool and spray fires	--	--	25	25	50	50	50	20	20	20	50	50	50	25	20	455
5. Analytical methods verification scale model	--	--	--	--	--	--	--	--	--	--	25	50	50	50	25	200
Analytical Methods and Engineering Techniques																
1. Reactor building fires	10	15	15	10	25	50	50	50	25	--	--	--	--	--	--	250
2. Reactor vessel fires	25	50	50	50	25	25	50	50	25	--	25	--	--	--	--	375
3. Other chemical reactions	--	10	10	10	--	--	--	--	--	10	10	10	10	--	--	70
Total	85	150	205	180	285	205	185	155	105	110	300	190	190	125	45	2515

devices, inherent shutdown mechanisms, the need for backup shutdown devices, and emergency core cooling systems. Guidance in this work will come from section 2, Plant Safety Analysis Studies, Plant Design Studies, and Plant Instrumentation and Control Studies. This section must provide the coordination and review required to bring the analyses in section 2 and the components R&D programs in section 3 together and to fill in a few gaps. This objective is essentially the same as that of Task 10-3.1 of the LMFBR Program Plan.

2. The second objective is to develop sufficient data and information to permit the designers of containment and consequence-limiting systems (1) to evaluate potential accident damage to the fuel cladding, reactor vessel, biological shield, reactor cover structure, and other basic containment barriers and (2) to determine the adequacy of the engineered safety features incorporated to protect these energy or fission product barriers reliably. These objectives are essentially those of Task 10-3.2 of the LMFBR Program Plan.

3. The third objective is to develop specific instrumentation and systems requirements for special areas of concern for monitoring the power distribution in the reactor core to guard against power shifts which could lead to local meltdown and anticipating coolant boiling and fuel failure to avoid core meltdown. The functional requirements for plant safety instrumentation and core instrumentation will be developed in the plant design section (section 2) of the B&W R&D program, and further requirements for in-core instrumentation will be developed in the core design section (section 6). The development and out-of-pile testing of all instruments is the responsibility of the instrumentation and control section (section 4). The testing of the safety instruments in-pile will be performed in sections 10-2.1 through 10-2.4. The main objective is, then, to ensure that the proper coordination exists between the plant and core design sections and the instrumentation and control section, and to ensure that the proper in-pile testing is accomplished in other sections of this safety section.

10.2.5.2. Summary of Current Technology

1. The background information is provided in Task 13-3.1 in the LMFBR Program Plan (section 10) for the core

protective systems and devices. To this we should add additional comments. Since the studies of 1963, where safety-related requirements received strong considerations, economical considerations have been receiving more attention. For the latest round of 1000-MWe studies, it was shown that unwarranted requirements were placed on earlier designs, partly because of the plant design and arrangement, and that for the present plants, the core protection systems and other engineered safety features will protect the core.

2. The background information for the containments and consequence-limiting safety systems is provided in Task 10-3.2 of the LMFBR Program Plan (section 10). There are essentially two separate and distinct items which the barriers must take into consideration—energy and fission products; the B&W concept uses separate barriers for them. The energy barrier and sodium barrier were fixed by the cavity formed by the biological shield and the cover structure. However, following an accident it was not intended that the energy and sodium barrier should prevent the leakage of fission products, since the accident could damage the seals in that barrier.

3. The background information for the safety instrumentation is provided in section 4 of the B&W R&D program. There are two aspects in which the R&D for the B&W reference plants may differ from the R&D needs of other LMFBR plants. The B&W reference plant was designed to facilitate detector maintenance and replacement by protecting them from the sodium with wells and thimbles. Second, the B&W reference plant places some emphasis on digital computers for reactor protection. Digital computers have been used in this country in an open-loop fashion for control, in which case the reactor operator observing the computer output took control action. In a few cases outside the U.S., the loop between the computer and the reactor has been closed.

10.2.5.3. Scope of Work

1. The requirements for the core protection systems and devices will be developed in section 2 of the B&W R&D program in the plant safety analysis studies. Additional studies are required to determine the necessity for a backup shutdown system. If

these studies indicate that such a system is needed, then the devices which hold substantial promise as backup shutdown devices will be developed and tested in this section. The development of the emergency shutdown system is carried in section 3 of B&W's R&D program. For the emergency shutdown systems development, the safety section will actively coordinate the requirements from plant design studies and core studies to ensure that the emergency shutdown system requirements are adequate for the component development section to ensure that the development is orderly. For the emergency core cooling system development, the safety section will actively coordinate the requirement developed in the plant design and core studies to ensure that the emergency core cooling system requirements are adequate for the components section to assure that the system development is orderly. The backup cooling coil in the biological shield was incorporated, not as core protection, but as a means for protecting the sodium-energy barrier formed by the biological shield and cover structure.

2. The containment design schemes developed from the LMFBR design studies will be subjected to a detailed evaluation in the plant design safety studies, which are included in section 2 of B&W's R&D program. Any new and/or novel features of the containment structure which may not be amenable to analysis will require verification testing; however, this type of verification testing is conducted in components testing, section 3, of the B&W R&D program. The B&W primary pool reactor concept provides a backup for the emergency core cooling system heat exchanger located in the reactor vessel, since any component in the reactor vessel is subject to damage in the event of a large energy release in the core. These backup components are embedded in the biological shield for protection. Even though the heat transfer paths are well defined, model testing is considered to be necessary to ensure that the analytical methods used in the analyses of the overall heat transfer and transport from the fuel to the environment are correct and that the system will remove the decay heat generated in the reactor vessel by the core. This model testing should also consider any rearrangement of the reactor core and pool that occurs as a result of the energy release.

Table 10-6. Budget for Engineered Safety Features and Containment

Task	Cost, \$ × 10 ³															Total
	Fiscal year															
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
Core Protection Systems and Devices																
1. Shutdown system	8	3	8	8	8	8	8	8	8	8	8	8	8	8	8	120
2. Backup shutdown system																
Study	--	--	--	250	250	100	50	50	--	--	--	200	200	100	50	1,250
System development	--	--	--	--	500	700	700	700	--	--	--	--	250	500	500	3,850
3. Emergency cooling system	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	120
Containment																
1. Coordination	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	180
2. Backup cooler test	--	--	--	100	200	400	400	--	--	--	--	100	200	400	400	2,200
Instrumentation	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	180
Facilities																
1. Emergency shutdown	--	--	--	--	700	--	--	--	--	--	--	--	700	--	--	1,400
2. Backup cooler test	--	--	--	400	--	--	--	--	--	--	--	400	--	--	--	800
Total	40	40	40	790	1690	1240	1190	790	40	40	40	740	1390	1040	990	10,100

3. The functional requirements for the in-core instrumentation will be developed in the plant design safety studies and the core design studies (sections 2 and 6 of the B&W R&D program). The development and out-of-pile testing of all instruments is the responsibility of the instrumentation and control section and is included in section 4 of the B&W program. The in-pile testing of the safety instruments will be performed in sections 10-2.1 to 10-2.4 of the LMFBR Program Plan.

4. In all three areas discussed above, the safety portion of the B&W R&D program, section 10, will provide the coordination between the studies and design groups to ensure that component development, including instrumentation, is orderly and that the hardware provided meets the requirements placed upon them.

10.2.5.4. Relationship to Current or Planned Programs

This section of the B&W R&D program will provide the coordination between the other sections of the program to ensure that the engineered safety features, including the containment, are capable of providing reasonable assurance that the B&W reference plant can be built and operated in a utility environment without exposing the public to undue risk. It provides for the development and testing of those engineered safety features which may become desirable as a result of the extended plant design safety studies. It includes a study to determine whether a backup shutdown system is required as well as a proposed development pending the outcome of the study.

This section differs from the LMFBR Program Plan Task area 10-3 considerably. A large segment of the component development and testing was transferred to other sections of the B&W R&D program, with the coordination required to ensure an orderly development provided by this corresponding section. A study was included to determine the need for a backup shutdown system.

10.2.5.5. Budget

A detailed budget breakdown is given in Table 10-6. The backup emergency shutdown system is the one large

cost item. This section includes a study to determine the necessity for a backup emergency shutdown system and to provide guidance for its development if it is found necessary. Thus, the budget could range from \$2 to \$8 million.

10.2.6. Special Safety Programs

In this area, the target is a better understanding of the design requirement placed on the designer in the form of codes, standards, or criteria, which form the basis for the guidelines used by designers in their evaluation and design of plant components and features. It covers general safety criteria for plant, system, and core design, as well as the development of standards and codes for siting, containment, engineered safeguards, operating practices, and reliability.

Analytical methods must be developed for reliability and failure-analysis to identify potential accident sources. That is, the work is directed toward the use of probabilistic methods in reactor accident evaluation, safety analyses, and calculations of the reliability of reactor safety systems and engineered safety features.

10.2.6.1. Objectives

The major objectives are to study the safety-related design criteria and develop a set of regulatory design criteria to be used as guidelines for siting, design, construction, operation, and licensing; the criteria are applicable to the pool LMFBR commercial plants and demonstrators. It must also cover general safety criteria for plant, system, and core design.

The second major objective is to develop a basic theoretical program based on a sequential application of the conditional probability theorem to the probability of system and plant failure. Included is the development of probability methods for reliability analyses of safety systems and the entire nuclear steam supply. The collection, organization, and evaluation of statistical data for use in probabilistic analyses will be required before efficient use can be made by probabilistic evaluation of the plant, systems, components, and engineered safety features.

10.2.6.2. Summary of Current Technology

A review of current technology is included in the background information for the LMFBR Program Plan, Task 10-4.1, for codes, standards, and criteria. To this we should add the general design criteria which were produced by the LMFBR contractors in the Follow-On-Study. These criteria should be combined to form a good base on which to build the regulatory design criteria.

A technology review is also given in the LMFBR Program Plan background information, Task 10-4.2, for probabilistic evaluation. To this we should add that computer codes such as ARMM (automatic reliability mathematical model), which was developed by North American Aviation under contract to the U.S. Air Force, are available. It should be noted that ARMM has limitations that may be important in some system reliability analyses of engineered safety features, but it could form the basis for a good program. SAFTE-1 (system analysis by fault tree evaluation) has also been developed, using the Monte Carlo simulation. It views the system or fault tree to be studied as a statistical assembly of components, each characterized by an exponential failure distribution and a normal repair distribution.

10.2.6.3. Scope of Work

The scope of work is essentially the same as that provided in the LMFBR Program Plan Tasks 10-4.1 and 10-4.2. The magnitude of the work has been scaled down, since it is applicable to the B&W primary-pool reactor concept and it was felt that the plan had overestimated the magnitude of the program and was not giving proper consideration to the existing methods, codes, standards, and criteria.

10.2.6.4. Relationship to Current and/or Planned Programs

This section covers Tasks 10-4.1 and 10-4.2 in the LMFBR Program Plan. Task 10-4.3 has been omitted as not being applicable to the B&W R&D program, since it deals with the development of a fast reactor safety handbook and a fast reactor safety textbook.

Table 10-7. Budget for Special Programs and Facilities

Task	Cost, \$ × 10 ³															Total
	Fiscal year															
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
Codes, Standards, and Criteria Development																
1. Assess background	25	25	25	25	20	20	15	15	15	15	15	15	15	15	15	275
2. Support existing groundwork	--	25	25	50	50	50	50	50	50	50	50	50	50	50	50	650
3. Extend to undeveloped areas	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	375
4. Provide information and guidance	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	375
Analytical Reliability Methods																
1. Probabilistic methods	25	125	125	25	--	--	--	--	--	25	125	25	25	25	25	550
2. Component data	25	25	25	50	50	50	50	25	25	25	25	25	50	50	25	525
Total	125	250	250	200	170	170	165	140	140	165	265	165	190	190	165	2750

10.2.6.5. Budget

A detailed breakdown of costs is given in Table 10-7. The total for codes, standards, and criteria development is estimated at about \$1.7 million; for analytical reliability methods the cost is estimated at about \$1.1 million.

10.2.7. Facilities

Included in this task are the major facilities required to carry out the safety research programs. The test loops and experimental equipment associated with specific sections are carried in the first section of the B&W safety R&D program where they are needed; however, the driver reactors and support facilities are carried in this section. Only those facilities of lasting utility and/or high cost and which are in the funding category of line items in the Federal Budget are included.

10.2.7.1. Objectives

The objectives are (1) to review the kinds and number of reactor experiments which are required by the safety assessment of the LMFBR plants, (2) to review the existing facilities and those planned or under construction, (3) to establish the need for additional or modified facilities needed to carry the experimental program, and (4) to develop a conceptual design and a preliminary cost estimate for the additional or modified facilities. Depending on the outcome of the first objectives, the facilities would be designed, constructed, and operated to carry out the experimental program.

10.2.7.2. Review of Existing Facilities

A number of existing facilities are available in the transient analysis area and the steady-state area. These are reviewed in the LMFBR Program Plan in Task area 10-5. The suggested power burst facility and the safety test facility, however, should be upped in priority, since there is a need now for larger-scale tests than the existing facilities will handle. These large-scale tests will determine the core requirement for the DBA and will assure that reality, rather than tradition or emotion, will establish the needed engineered safety features.

Table 10-8. Budget for Facilities

Task	Cost, \$ × 10 ³															Total
	Fiscal year															
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	
Safety Test Facility																
1. Specification	--	200	100	--	--	--	--	--	--	--	--	--	--	--	--	300
2. Development	--	--	300	300	300	--	--	--	--	--	--	--	--	--	--	900
3. Construction	--	--	--	2000	10,000	10,000	10,000	--	--	--	--	--	--	--	--	32,000
4. Operation	--	--	--	--	--	--	--	2,000	2000	2000	2000	2000	2000	2000	1000	15,000
Steady-State Facility																
1. Thermal reactors	500	1500	2000	1000	1,000	1,000	1,000	--	--	--	--	--	--	--	--	8,000
2. Fast reactors	--	--	--	500	1,000	2,000	2,000	2,000	2000	1000	--	--	--	--	--	10,500
Transient Reactors																
1. TREAT	500	500	1200	500	500	500	500	500	500	--	--	--	--	--	--	5,200
2. SEFOR (PBF)	--	--	--	500	500	500	500	500	500	--	--	--	--	--	--	3,000
Excursion Test Facility																
1. Facility need and requirement	50	100	100	--	--	--	--	--	--	--	--	--	--	--	--	250
2. Specification	--	100	200	200	--	--	--	--	--	--	--	--	--	--	--	500
3. Development	--	--	--	300	300	300	--	--	--	--	--	--	--	--	--	900
4. Construction	--	--	--	--	1,000	10,000	10,000	5,000	--	--	--	--	--	--	--	26,000
5. Operation	--	--	--	--	--	--	--	1,000	1000	1000	1000	1000	1000	1000	500	7,500
Total	1050	2400	3900	5300	14,600	24,300	24,000	11,000	6000	4000	3000	3000	3000	3000	1500	110,050

The facilities required for safety studies are included as major facilities and/or equipment required in the various segments that comprise the experimental work. For instance, facilities are needed to study the failure of fuel clusters up to subassembly size and even larger for propagation studies; i. e., facilities are needed where subassemblies can be caused to fail deliberately by sustained overpower or coolant flow malfunctions. A power burst facility is needed for transient tests for periods shorter than 40 milliseconds, which is the lower limit for TREAT. Neither of these facilities is capable of studying multi-subassembly accidents; thus, larger facilities will be necessary. A larger facility would provide the proof testing required for the protection of an LMFBR and for the methods used to analyze accidents. Consideration should also be given to a facility to perform large integral proof test analyses to the BORAX-I and SPERT test performed in the water reactor safety program. The need for this final test facility will be established in 10.2.2., Design Basis Analysis and Energy Release.

10.2.7.3. Scope of Work

The scope of work is essentially the same as that given in the LMFBR Program Plan, Task area 10-5, as applied to the primary-pool concept being developed by B&W.

10.2.7.4. Relationship to Current and/or Planned Programs

This section covers Task area 10-5 in the LMFBR Program Plan.

10.2.7.5. Budget

The estimated budget breakdown is given in Table 10-8. The total is estimated at from \$87 to \$122 million. However, many of the facilities would be applicable to all the various concepts under consideration; thus, this total cost should cover much more than the B&W R&D program for the primary-pool reactor concept.

Figure 10-1. Recent Superheat Data, Presented at the Aix Conference (Scatter From 5 to ~ 800 C)¹⁵

Figure 10-2. Variation is Characteristic of Superheat Data (These measurements, older than those of Figure 10-1, spread from 1 to 500 F.)¹⁵

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