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Fission Energy Program of the U. S. Department of Energy

FY 1979

Assistant Secretary for Energy Technology
Office of Nuclear Energy Programs

June 1978

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Fission Energy Program of the U. S. Department of Energy

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Assistant Secretary for Energy Technology
Washington, D. C.

June 1978

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PREFACE

This description of the Fission Energy Program was prepared for the use of the Committee on Science and Technology, United States House of Representatives and represents the status of the program as of January 1978. It was prepared by the Office of Acting Program Director for Nuclear Energy, Department of Energy. The major portion of the document concerns civilian nuclear power development, the policy for which has been established by the National Energy Plan of April 1977, but it also includes descriptions of the space applications and naval reactor programs. Special nuclear materials production is not included because this area of the program is outside the cognizance of the Science and Technology Committee.

Chapter I derives the fission energy policy and program objectives from the National Energy Plan. Chapter II indicates the structure of the program, identifies management responsibilities in the DOE organization for the various program elements, and presents the overall program budgets. Chapters III through V describe the civilian fission power development, while Chapters VI and VII cover the space applications and naval reactor programs, respectively.

The Appendices to the document provide additional information on major facilities used in the development program, the roles of national and engineering laboratories in the pursuit of the work, and the international aspects of the fission program.

George W. Cunningham
Acting Program Director
for Nuclear Energy

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FISSION ENERGY PROGRAM SUMMARY

This document describes programs managed by the Program Director for Nuclear Energy, Department of Energy, and under the cognizance of the Committee on Science and Technology, United States House of Representatives. The major portion of the document is concerned with civilian nuclear power development, the policy for which has been established by the National Energy Plan of April 1977, but it also includes descriptions of the space applications and naval reactor programs. Special nuclear materials production, which is the responsibility of the Director, Nuclear Energy Programs, DOE, but is outside the Committee's cognizance, is not included.

I. THE NATIONAL ENERGY PLAN AND FISSION ENERGY POLICY

The National Energy Plan. The National Energy Plan (NEP)* sets forth three overriding energy objectives for the United States:

- "as an immediate objective that will become even more important in the future, to reduce dependence on foreign oil and vulnerability to supply interruptions;
- in the medium term, to keep United States imports sufficiently low to weather the period when world oil production approaches its capacity limitations; and
- in the long term, to have renewable and essentially inexhaustible sources of energy for sustained economic growth."

The NEP proposes to achieve these objectives through a combination of measures, including:

- Conservation--reduction of energy waste and improvement in fuel efficiency
- Rationalization of pricing of the scarcer fuels (i.e., oil and gas) in the direction of reflecting cost of replacement, thus:
 - Further encouraging conservation by consumers
 - Providing appropriate incentives to develop new sources of the scarcer fuels and increase domestic production

*Executive Office of the President, Energy Policy and Planning, "The National Energy Plan," p. ix (April 29, 1977).

THE NATIONAL ENERGY PLAN AND FISSION ENERGY POLICY

- Improving the economics of converting existing energy consumption and conversion facilities from the scarcer fuels to the more abundant coal where technologically feasible
- Encouragement of the construction of new energy consumption and conversion facilities to utilize coal and nuclear fission energy resources
- Development of renewable and essentially inexhaustible energy resource technologies, and appropriate energy conversion and storage methods, to enable these technologies to replace oil and gas on a commercial basis as these resources enter their final stages of depletion

It is in the last two of these major thrusts that nuclear fission energy has potential roles. In the near and intermediate term, coal and uranium must share the burden of electricity generation. National energy policy stresses an endorsement of the present generation of fission systems--the light-water reactor (LWR) operating in a once-through fuel-cycle mode. The National Energy Plan expresses a resolve to eliminate technical and institutional impediments to maturity of the LWR system. At the same time, demonstration and commercialization of advanced fission systems,* including the plutonium breeder reactor and reprocessing of spent fuel, have been deferred indefinitely pending a reassessment of all fission options. This deferral is based on a lessened sense of urgency for extending uranium ore supplies, and on a strategy that seeks technical as well as procedural solutions to the proliferation issue.

Near- and Intermediate-Term. Nuclear fission power generated by commercial light-water reactor plants now supplies nearly 13% of the Nation's electricity. New LWR plants are being brought into commercial operation at the rate of about 7 per year, and this rate, based on utility company schedules** for plants now on order, is expected to rise to 19 per year by 1984. After 1986, however, the projected rate of plant completions drops sharply, reflecting the lack of LWR plant orders in recent years.

It has been estimated that about 380 million kilowatts of nuclear-fission power-generation capacity will be required by the year 2000 to supplement the fossil fuels and other resources in satisfying United States energy demand, even with coal production tripled from its present rate. The completion of about 15 nuclear-fission plants per year from 1985 to 2000 is required if this goal is to be reached. This is illustrated in figure 1. The figure also shows clearly the decline in LWR orders since 1974. While orders in 1972-74 averaged 27 plants per year, the 1975-77 average was 3 plants per year. This decline is symptomatic of several complex underlying barriers.

*A system offering improved fuel utilization efficiency, compared to the PWR once-through system.

**United States Department of Energy, "United States Central Station Nuclear Electric Generating Units: Significant Milestones (Status as of October 1, 1977)," DOE/ET-0030/4(77), December 1, 1977 (updated to January 9, 1978).

Figure 1
Commercial Nuclear Power Plant Orders and Completions Between 1953 and 2000

Commercialization of LWRs has been underway for more than a decade. As with the introduction of any new technology, LWR deployment has suffered the effects of both institutional and technical impediments. A significant institutional impediment has been the long construction lead times. These are partially engendered by the licensing process, which tends to govern the 10-12 year construction period for nuclear power plants. The unencumbered construction time for a nuclear power plant is thought to be approximately 6-7 years.

The NEP recognizes and seeks solutions to these impediments in order to foster the maturing of the LWR as a fully commercial energy source. A number of steps are being taken to make LWRs a more attractive choice in the utilities' decisionmaking process:

- Demonstrate, on a scale appropriate to the quantities of radioactive wastes expected from enhanced LWR deployment, the safe management of such wastes, including both unprocessed spent fuel and the wastes that would result from fuel reprocessing should it become advantageous to exercise that option in the future

THE NATIONAL ENERGY PLAN AND FISSION ENERGY POLICY

- Develop and demonstrate methods of improving the productivity of LWR power plants
- Develop and demonstrate means of reducing costs and construction schedules through plant standardization and construction improvements
- Improve fuel utilization, i.e., the energy generated per unit of uranium mined, for conservation of resources
- Sponsor a licensing reform bill in Congress that would substantially reduce construction lead times by streamlining the licensing process

These steps constitute the near- and intermediate-term objectives of the Fission Energy Program administered by the Department of Energy (DOE) Assistant Secretary for Energy Technology and the Program Director for Nuclear Energy--the program described in this document.

Long-Term. The NEP recognizes that in the longer term the United States must place increasing reliance on renewable and essentially inexhaustible sources of energy. There are four basic categories of essentially inexhaustible energy resource technologies under active development at this time: solar, geothermal, fusion, and the fission breeder. Development and demonstration of the fission breeder has been based thus far on the classic uranium-plutonium fuel cycle, and the NEP calls for deferral of plutonium-based systems and a thorough examination of other systems that may offer a reduced potential for the proliferation of nuclear weapons.

The Fission Energy Program seeks advanced nuclear technologies that minimize the risk of proliferation, while deferring further demonstration efforts in technologies based on plutonium. A major system evaluation program is under way wherein a large number of alternative reactor concepts, alternative fuel cycles, and combinations thereof are being assessed both as to their proliferation resistance and as to their capability to extend the energy obtainable from the available uranium-resource base beyond that provided by the LWR using a once-through fuel cycle. Concurrently, base-technology work on advanced fission options is continuing, the major emphasis being directed toward reducing remaining uncertainties in such areas as safety, fuel performance, and component design.

Thus, the immediate goal of the advanced fission programs is to assemble all the information required to enable a decision as to the most advantageous approach to follow in utilizing fission energy for fulfilling the Nation's long-term energy needs consistent with minimizing proliferation risks. Based on the selected approach, appropriate options may then be pursued to develop either an essentially inexhaustible fission resource or a means for bridging the gap until other inexhaustible resource technologies can be commercially realized, depending on the needs as perceived at that time.

In summary, the fission energy policy as derived from the NEP is to:

- Enhance the potential for increased near-term commercial deployment of light-water reactors by improving LWR technology, reducing institutional impediments, demonstrating capability for safely managing radioactive wastes, and establishing a publicly acceptable plan for managing such wastes on a commercial scale
- While deferring commitment to technologies based on plutonium, all advanced fission options for their nonproliferation, economic, technical, and institutional merits.
- Maintain advanced fission systems in a research and development stage, but in a state of readiness should a decision be made in the future to demonstrate one or more of them

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II. PROGRAM STRUCTURE, MANAGEMENT AND BUDGET

The preceding chapter showed how the fission energy policy and overall program objectives are derived from the National Energy Plan. This chapter defines the program structure, summarizes its major elements, and identifies both the DOE organizational units responsible for their management and the DOE Budget and Reporting Structure categories under which they are funded. Finally, the overall budget for the program, by major element, is presented.

Subsequent chapters describe the individual program elements and their implementation.

Program Elements. The fission energy program is composed of product-oriented, manageable work elements. They are:

- Light Water Reactors

This element includes those activities needed to make LWRs a more attractive choice in the utilities' decision-making process. This work implements the National Energy Plan statement that "LWR technology with a once-through fuel cycle represents a current energy production capability that can play a significant part in the Nation's energy future, ease the demand on expansion of other fuel systems, and contribute to the reduction of United States petroleum imports."

- Advanced Fission Systems

The primary purpose of advanced fission systems is to improve fuel utilization. This program element includes system evaluations, with particular attention to nonproliferation aspects and development work on gas-cooled thermal reactors, water-cooled breeder reactors and fast breeder reactors. This implements the NEP call for seeking "a better approach to the next generation of nuclear power than is provided by plutonium recycle and the plutonium breeder." However, breeder technology work is being continued to maintain a viable option that can be deployed if it is needed and no better approach has been found.

- Fuel Cycle

This element covers all phases of the fuel cycle other than reactor irradiation--advanced isotope separation, spent fuel processing, spent fuel storage, and waste management. The element directly responds to the Presidential policy statements of April 7, 1977 that the United States will "accelerate our research into alternative nuclear fuel cycles which do not involve direct access of materials that can be used for nuclear weapons," and that "we will increase the United States capacity to produce nuclear fuels, enriched uranium in particular."

PROGRAM STRUCTURE, MANAGEMENT AND BUDGET

Fuel cycle work has been grouped under a separate element in this program structure, as opposed to associating cycles with their specific reactor concepts, because much of the fuel cycle work is generic to more than one reactor type. Nevertheless, in addressing fission power systems, entire systems incorporating both the reactors and their associated fuel cycles must be considered. Table 1 describes the interrelationships between the various reactor types and fuel cycle activities, in matrix form. The entries in the matrix represent the chapters in this document where the related program descriptions will be found.

Table 1. Interrelationships Among Reactor and Fuel Cycle Programs

REACTOR TYPE	REACTOR TECHNOLOGY	ISOTOPE ENRICHMENT	FUEL CYCLE TECHNOLOGY			
			FUEL PROCESSING		FUEL STORAGE	WASTE MANAGEMENT
			FABRICATION	REPROCESSING		
LWR	III	V B	VC 2	VC 2	VD	VE
GCTR	IV C	—	—	VC 4	—	VE
WCBR	IV D	—	VC 5	VC 5	—	VE
LMFBR	IV E 2	—	—	VC 3	—	VE
GCFR	IV E 3	—	—	—	—	VE

Other Fission Energy Program elements included in this document but not related to civilian power development and the National Energy Plan are:

- Space and Terrestrial Compact Power Systems
- Naval Reactors

Program Structure/Organizational Responsibility Matrix. Figure 2 is a matrix that identifies each of the major segments of work with the DOE organizational unit responsible for its management. The program structure as it is presented in subsequent chapters is shown along the top of the figure. The DOE organizational structure pertaining to the Fission Energy Program is shown along the left side of the figure. Responsibilities are indicated by circled intersections of the horizontal organization lines with the vertical program task lines.

Program Structure/Budget and Reporting Structure Matrix. Figure 3 is a matrix that identifies each of the major segments of work with the budget category under which the work is funded. The program structure as it is presented in subsequent chapters is shown along the top of the figure. The budget and reporting structure pertaining to the Fission Energy Program is shown along the left side of the figure. The circled intersections of the vertical program task lines with the horizontal budget category lines indicate where specific work packages are funded.

Budget. The funding for the Fission Energy Program for FY 1977 through FY 1979, subdivided by major program element, is shown in the following table. The totals are further subdivided to show additional detail in the subsequent chapters.

Figure 2
Relationship of Program Structure to Organizational Responsibility

Figure 3
Relationship of Program Structure to Budget and Reporting Structure
10

PROGRAM STRUCTURE, MANAGEMENT AND BUDGET

NUCLEAR ENERGY PROGRAM	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
PROGRAMS				
Light Water Reactors				
Operating	\$12,500	\$18,800	\$23,900	\$5,100
Equipment	0	0	0	0
Construction	0	0	0	0
Subtotal	12,500	18,800	23,900	5,100
Advanced Fission Systems				
Operating	636,943	465,640	415,901	(49,739)
Equipment	48,637	35,380	32,281	(3,099)
Construction	108,450	121,305	71,485	(49,820)
Subtotal	794,030	622,325	519,667	(102,658)
Fuel Cycle				
Operations	277,039	424,564	418,150	(6,414)
Equipment	21,633	35,967	27,700	(8,267)
Construction	113,900	170,962	208,750	37,788
Subtotal	412,572	631,493	654,600	23,107
Space and Terrestrial				
Compact Power Systems				
Operating	32,002	45,850	43,400	(2,450)
Equipment	3,565	4,680	4,700	20
Construction	0	0	0	0
Subtotal	35,567	50,530	48,100	(2,430)
Naval Reactors				
Operating	192,310	216,556	265,600	49,044
Equipment	6,000	15,150	22,000	6,850
Construction	18,200	15,700	3,000	(12,700)
Subtotal	216,510	247,406	290,600	43,194
TOTAL	\$1,471,179	\$1,570,554	\$1,536,867	\$ (33,687)

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III. LIGHT WATER REACTORS

A. PROGRAM OVERVIEW

Background. During the late 1940s, the AEC began an experimental power reactor program in which a number of experimental power reactors were built and operated in the 1950s. These early experimental plants were undertaken largely at AEC laboratories. The enactment of the Atomic Energy Act of 1954, which permitted private ownership of reactors and private use of nuclear fuel under leasing arrangements, ushered in the development of nuclear power for industrial applications. It also established provisions for declassifying data needed for the development of nuclear power.

The year 1957 was significant because it marked the initial operation of a 60,000 kWe prototype pressurized light-water reactor (PWR) power plant. This government-owned reactor facility, built at Shippingport, Pa., under a cooperative agreement with the Duquesne Light Co., was the Nation's first large-scale civilian nuclear power reactor.

In 1955, industry was invited to join with AEC in developing power reactors through the Cooperative Power Demonstration Reactor Program. Commercial nuclear power stations completely owned and operated by electric utilities followed soon thereafter; some of these were built under the joint demonstration program, others were privately financed. These included: the first large boiling-water reactor (BWR), Commonwealth Edison's Dresden Station (200 MWe) completed in 1960; Yankee Atomic Electric Company's Rowe unit (175 MWe) completed in 1961; and Consolidated Edison's first Indian Point unit (175 MWe) completed in 1962. Figure 4 shows a typical PWR nuclear steam system; figure 5 shows a typical BWR nuclear steam system; figure 6 shows a commercial-LWR nuclear fuel cycle in the United States with no reprocessing.

In the Cooperative Power Demonstration Reactor Program, the government contributed technical and financial assistance, particularly with regard to the fuel and the research and development (R&D) programs. These AEC initiatives, together with the larger R&D expenditures to develop PWR nuclear-propulsion systems for the U.S. Navy (which underlie much of development of LWR systems for civilian use), constitute an investment on the order of \$2 billion.

In following the AEC leadership, the nuclear industry also began contributing heavily to the development and commercialization of LWRs in the 1960s to the point where, in the United States by the end of 1977, 63 LWRs provided nearly 13 percent of the Nation's electrical energy. By 1985, an additional 75 nuclear plants already planned or under construction are scheduled to be in operation providing about 20 percent of the Nation's electrical power. The Department of Energy forecasts that by the year 2000 there will be an installed nuclear capacity of 380,000 MWe, consisting almost entirely of LWRs, compared to about 45,000 MWe installed nuclear capacity at present.

Figure 4
Typical Pressurized-Water Reactor (PWR) Nuclear Steam Supply System

Figure 5
Typical Boiling-Water Reactor (BWR) Nuclear Steam Supply System

Figure 6
The Commerical-LWR Present Fuel Cycle with Spent-fuel Storage

Despite the record of nuclear power to date, the nuclear industry is not without problems. In the last 3 years, projections of nuclear capacity have declined markedly because of construction delays, deferrals, and utility cancellations of projects. The reasons for the lack of orders are multiple, but the causes mentioned most include:

- Reassessments, during the recession caused by the 1973 oil embargo, of electricity-use growth projections made before the embargo
- Utility financing and economic considerations
- Uncertainty in Government policies regarding fuel reprocessing, enrichment capacity, and waste management
- Licensing delays, environmental approval, and public acceptance

In January 1975, ERDA established a task force to consider the need for government participation with industry in a cooperative technology-development program to improve light-water reactor productivity and reduce the capital costs and construction schedules of new nuclear plants. The task force found that government participation was appropriate in view of: (a) the national energy goals of minimizing oil and natural-gas consumption and reducing dependence on fuel imports and (b) uncertainties in the nuclear industry that discourage private-sector investments in technology development.

As a result of the task force's findings, in FY 1976 ERDA initiated the LWR technology-development program described here. In formulating the program, the priorities among numerous competing possibilities were set

LIGHT WATER REACTORS

based on the urgency of public need for the results, and the likelihood that the results would not be achieved without Federal assistance. In implementation, care has been taken to avoid displacing private-sector initiatives and interfering with private-sector competition, to elicit cost-sharing from all participants as evidence of mutual interest, and to ensure validation of results with in-plant demonstrations.

Objectives. This program's goals of conserving oil and natural gas by improving productivity and reducing new-plant costs and schedules reflect the Administration's policy toward LWRs as indicated in the National Energy Plan, which states that there will be a continued reliance on LWRs to offset this country's energy deficit. The policies and programs to be pursued will increase the willingness of utilities to invest in nuclear power plants.

In April 1977, an additional objective was added to the program--improve uranium utilization in LWRs (the total amount of energy produced per quantity of uranium mined). This objective was added on the basis of no fuel reprocessing in support of the Administration's change in the Nation's nuclear energy policy. Programs to improve the utilization of uranium fuel without fuel reprocessing were initiated in the summer of 1977 and are being continued.

Furthermore, the National Energy Plan, while recognizing that light-water reactors have a good safety record, cited additional actions that could be taken to improve safety. The President and members of Congress have also expressed the goal of further improving safety of these reactors. While some responsive regulatory actions have been initiated or are under way by the Nuclear Regulatory Commission, it is appropriate for the Department of Energy to conduct a complementary program to improve safety systems.

In summary, the principal objectives of this LWR technology development program are:

- To improve plant productivity of LWRs (measured by capacity and availability factors) now in operation or committed, through the development, demonstration, and widespread use of improved technologies. These improvements should be attainable by the mid to late 1980s.
- To reduce the construction costs and time required for LWRs to be constructed during the next 10 years.
- To improve the utilization of uranium fuel in existing LWRs without recycle, resulting in less consumption of uranium to produce an equivalent amount of electricity, and therefore to extend uranium resources.
- To improve the safety of LWRs.

Table 2 shows the projected benefits from this LWR development work for the first three objectives in the near term, up to 1985, and in the mid-term, up to 2000. The benefits of the fourth objective are not directly

measurable as resources conserved or dollars saved. Rather, benefits will be in terms of improved licensability.

Table 2. Projected Benefits of LWR Technology Development

	1980	1985	1990	1995	2000
Installed LWR capacity (MWe)	67,000	145,000	226,000	300,000	380,000
OBJECTIVE 1 IMPROVE PLANT PRODUCTIVITY					
Availability improvement (%)	4 0	6 0	6 5	8 0	8 5
Equivalent oil conserved (M bbl/yr)	43	116	235	385	570
Annual benefits (M \$)	500	1,300	2,700	4,400	8,000
OBJECTIVE 2 REDUCE CONSTRUCTION COSTS AND TIME					
Additional LWRs operating	0	0	15	30	45
Equivalent oil conserved (M bbl/yr)	0	0	180	360	540
Annual benefits (M \$)	0	0	2,100	4,700	6,300
OBJECTIVE 3 IMPROVE URANIUM UTILIZATION					
Uranium utilization improvement (%)	2	6	7	8	10
U ₃ O ₈ conserved (M lb/yr)	0 6	3 5	9 2	12 0	18 0
Annual benefits (M \$)	18	105	275	360	640

The objectives will be met by implementing the following subprograms:

- **Technology Base.** This subprogram concentrates on developing and demonstrating improved fuel technology, radiation control, and steam-generator cleaning methods.
- **Plant Design and Construction.** Two activities are being conducted in this subprogram: the demonstration of refueling and maintenance improvements and the development of technologies that will reduce construction costs and schedules.
- **Components.** Advanced failure detection and diagnostic methods will be demonstrated on operating plants. Basic mechanisms causing failures due to flow-induced vibration will be studied.
- **Safety.** Improvements in LWR safety will be developed through a program coordinated with the Nuclear Regulatory Commission's safety research program as well as with work being conducted by industry and other nations.

Strategy. The strategy of the program to ensure obtaining the near-term benefits includes the direct involvement of the electric-utility industry and its equipment and service suppliers. Therefore, work in the FY 1978 program, as in the past, focuses on participation and cost sharing by the user industry in these technology improvements. In particular, in-plant demonstrations will validate the results and encourage more rapid commercial acceptance of improvements than would otherwise occur.

LIGHT WATER REACTORS

At the end of FY 1977, 20 contracts worth \$102 million had been negotiated and signed. All were cost-sharing or no-fee contracts with DOE's share being \$44 million (or 43%) and industry's share being \$58 million. The work includes participation by nine major utilities that have a combined installed nuclear operating capacity of 16,200 MWe, all the LWR vendors and fuel suppliers, and other industry equipment and service suppliers.

The following table shows the funding levels by subprogram for the period FY 1977 through FY 1979.

LIGHT WATER REACTORS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Technology Base				
Operating	\$8,500	\$10,200	\$12,000	\$1,800
Equipment	0	0	0	0
Construction	0	0	0	0
Subtotal	8,500	10,200	12,000	1,800
Plant Design and Construction				
Operating	2,200	4,700	3,700	(1,000)
Equipment	0	0	0	0
Construction	0	0	0	0
Subtotal	2,200	4,700	3,700	(1,000)
Components				
Operating	1,800	3,900	4,200	300
Equipment	0	0	0	0
Construction	0	0	0	0
Subtotal	1,800	3,900	4,200	300
Safety				
Operating	0	0	4,000	4,000
Equipment	0	0	0	0
Construction	0	0	0	0
Subtotal	0	0	4,000	4,000
TOTAL	\$12,500	\$18,800	\$23,900	\$5,100

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B. TECHNOLOGY BASE

Three causes of unscheduled or extended forced outages are being addressed in this subprogram: premature fuel failure, high radiation levels in certain plant systems, and corrosion-induced damage in steam generators. Improvements in these areas will result in less plant downtime, with the obvious benefit being more continuous trouble-free operation, which translates into lower electricity cost to the consumer. The work concentrates on improving operation in plant areas that require significant work under normal operation, or which potentially would require excessive maintenance and repair if problems occur.

Objectives. The categories of this subprogram and their specific objectives are:

- Fuel Utilization
 - To reduce susceptibility of fuel to premature failure
 - To extend average burnup of fuel
- In-plant Radiation Control
 - To develop processes to chemically decontaminate the primary coolant system of a PWR or BWR
 - To develop processes to minimize and reduce contamination levels of operating reactors
- Steam Generator Cleaning
 - To develop a chemical process to clean the secondary side of steam generators

Strategy. The strategy for achieving these objectives is to conduct near-term demonstration projects using large commercial LWRs and to involve utilities, vendors, and equipment suppliers in cost-sharing roles. This approach ensures that the projects selected will have industrial support and commercial acceptance.

The following table shows the operating funding levels by category for the period FY 1977 through FY 1978.

TECHNOLOGY BASE	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Fuel Utilization	\$1,900	\$5,800	\$5,800	\$0
In plant Radiation Control	6,200	1,600	3,200	1,600
Steam Generator Cleaning	400	2,800	3,000	200
TOTAL	\$8,500	\$10,200	\$12,000	\$1,800

LIGHT WATER REACTORS

1.0 Fuel Utilization

One impact of the nonproliferation policy is the indefinite deferral of nuclear fuel reprocessing and recycle. Therefore, the objective of the fuel utilization category is to increase the amount of energy extracted from LWR fuels in current operating reactors. Programs will be conducted in two areas:

- Reducing LWR fuel susceptibility to premature fuel failure caused by a failure phenomenon known as pellet-cladding interaction (PCI). Alternative PCI-remedy fuel designs will be irradiated and demonstrated in the Big Rock Point (80 MWe BWR) and Quad Cities 1 (800 MWe BWR) reactors. One possible remedy is barrier fuel (Figure 7).

Figure 7
Barrier to Minimize Stress Corrosion of Cladding Caused
by Interaction of Pellet and Cladding

- Extending average burnup or exposure of LWR fuel in existing reactors by implementing improved fuel designs, management modes, and reactor core configurations. Studies and demonstration planning activities are being conducted in FY 1978; commercial demonstrations are to begin in FY 1979.

PROGRAM: LIGHT WATER REACTORS								
SUBPROGRAM: TECHNOLOGY BASE								
CATEGORY: FUEL UTILIZATION								
TASK	CY 1977		CY 1978		CY 1979		CY 1980	
	FY 1977	FY 1978	FY 1978	FY 1979	FY 1979	FY 1980	FY 1980	FY 1981
PCI-REMEDY FUEL DESIGNS	▲							
LARGE-SCALE DEMO AT QUAD CITIES			○					
IRRADIATION AT BIG ROCK POINT								▽

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2.0 In-plant Radiation Control

The difficulties in operating and maintaining LWRs due to radiation fields increase as plants age; as a consequence, personnel exposures increase, outage durations increase, and plant productivity declines. Therefore, the objective of this category is to improve productivity by reducing radiation fields in operating plants. Separate processes have been initiated to develop and demonstrate alternative decontamination techniques. These include:

- Development of a process in which concentrated chemicals are used to decontaminate thoroughly the primary coolant system of a BWR (after the fuel has been removed temporarily). A first-of-a-kind demonstration will occur using Dresden 1. Commonwealth Edison and Dow Chemical Co., are the principal participants.
- Development of a process to reduce and minimize contamination by adding dilute amounts of chemicals directly to the coolant system of a fully fueled reactor. It is planned to conduct this demonstration using Dresden 2 (800 MWe BWR).
- Development of a process to chemically decontaminate the primary coolant system of a PWR. A commercial demonstration using Indian Point 1 is being evaluated.
- Improved purification filters to reduce radiation levels in the primary coolant system will be designed and demonstrated.

PROGRAM LIGHT WATER REACTORS																															
SUBPROGRAM TECHNOLOGY BASE																															
CATEGORY IN-PLANT RADIATION CONTROL																															
TASK				CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
				FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
PWR																															
CONCENTRATED CHEMICAL DECONTAMINATION BWR																															
DILUTE CHEMICAL DECONTAMINATION BWR																															

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3.0 Steam Generator Cleaning

The objective of this category is to develop and demonstrate a process for chemically cleaning the secondary side of PWR steam generators, to alleviate the effects of a corrosion-induced damage mechanism known as denting. A commercial demonstration of the developed process will be conducted using Indian Point 2 (870 MWe PWR). Consolidated Edison and Halliburton Services are the principal participants.

LIGHT WATER REACTORS

PROGRAM LIGHT WATER REACTORS																												
SUBPROGRAM TECHNOLOGY BASE																												
CATEGORY STEAM GENERATOR CLEANING																												
TASK	CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
	FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
STUDY			▲																									
DEMONSTRATION					▼																							
EVALUATION									▼																			

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C. PLANT DESIGN AND CONSTRUCTION

Categories in this subprogram aim to improve plant productivity by reducing scheduled outages and to reduce construction costs and schedules. The categories of this subprogram and their specific objectives are:

- Refueling and Maintenance Outage Improvement
 - To develop procedures and equipment that reduce the duration of scheduled outages
- Construction Techniques and Systems
 - To evaluate new construction technologies so as to reduce plant cost and construction schedules

The strategy for achieving these objectives is similar to the approach described in Subprogram B.

The following table shows the operating funding levels by category for the period FY 1977 through FY 1979.

PLANT DESIGN AND CONSTRUCTION	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
CATEGORIES	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Refueling and Maintenance Outage Improvement	\$1,800	\$4,200	\$3,700	\$(500)
Construction Techniques and Systems	400	500	0	(500)
TOTAL	\$2,200	\$4,700	\$3,700	\$1,000

1.0 Refueling and Maintenance Outage Improvement

The objectives of this work are to develop and demonstrate refueling, maintenance, and inspection procedures and equipment that can reduce the duration of scheduled outages. The program is conducted under four contracts involving all four United States LWR vendors teamed with utilities as follows:

<u>Vendor</u>	<u>Utility</u>	<u>Plant</u>
Westinghouse	Commonwealth Edison	Zion

LIGHT WATER REACTORS

<u>Vendor</u>	<u>Utility</u>	<u>Plant</u>
Combustion Engineering	Omaha Public Power District	Ft. Calhoun
Babcock & Wilcox	Duke Power and Arkansas P&L	Oconee and Arkansas Nuclear 1
General Electric	Tennessee Valley Authority	Browns Ferry

Demonstrations of improved equipment designs and procedures using these plants or comparable units will begin in FY 1978. Examples include:

- Rapid-disconnect reactor-vessel studs
- "Wet eddy current" steam-generator tube inspection system
- Improved fuel removal equipment
- Computerized activity management system

PROGRAM: LIGHT WATER REACTORS																													
SUBPROGRAM: PLANT DESIGN AND CONSTRUCTION																													
CATEGORY: REFUELING AND MAINTENANCE OUTAGE IMPROVEMENT																													
TASK		CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
		FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
DEMONSTRATIONS						▲												▼											

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2.0 Construction Techniques and Systems

The objective of this work is to evaluate new construction technologies that can be applied to nuclear power plant construction projects. These include construction modularization, standardization, and data-base management. A decision on whether to commit large-scale demonstrations of these new technologies is scheduled for later in FY 1978.

PROGRAM: LIGHT WATER REACTORS																												
SUBPROGRAM: PLANT DESIGN AND CONSTRUCTION																												
CATEGORY: CONSTRUCTION TECHNIQUES AND SYSTEMS																												
TASK	CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
	FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
EVALUATIONS					▼																							
LARGE-SCALE DEMONSTRATIONS					○																							

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D. COMPONENTS

Component-related failures and malfunctions at nuclear power plants caused approximately 5% of the LWR plant outages in 1976. Therefore, the categories conducted in this subprogram are directed toward minimizing component-related failures. The categories and their specific objectives are:

- Advanced Diagnostic and Surveillance Methods
 - To develop techniques for detecting incipient failures
- Flow-induced Vibration (FIV)
 - To develop methods for predicting flow-induced vibration

The strategy for achieving these objectives is similar to the approach described in Section B.

The following table shows the operating funding levels by category for the FY 1977 to FY 1978 period.

COMPONENTS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Advanced Diagnostic and Surveillance Methods	\$600	\$600	\$800	\$200
Flow induced Vibration	1,200	3,300	3,400	100
TOTAL	\$1,800	\$3,900	\$4,200	\$300

1.0 Advanced Diagnostic and Surveillance Methods

In FY 1978 two types of monitoring equipment capable of advanced failure detection and diagnostics will be demonstrated at operating utility power plants. These forms of surveillance should be capable of detecting and localizing up to 50% of plant abnormalities in their early stages and thereby reduce the duration of any subsequent outage for inspection and maintenance:

- Acoustic-emission monitoring (AEM) systems will be demonstrated on Indian Point 1 (250 MWe PWR) and Peach Bottom 3 (1060 MWe BWR).

LIGHT WATER REACTORS

- Dynamic monitoring will be demonstrated on either LaSalle 1 (1070 MWe BWR) or Millstone 2 (820 MWe PWR).

Examples that will be demonstrated include:

- Main steam safety and relief valves
- Feedwater nozzles and spargers
- Auxiliary cooling pumps
- Large-diameter piping systems

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2.0 Flow-induced Vibration Technology

Analytical methods for predicting flow-induced vibration (FIV) of internal LWR systems and components are being developed to reduce lost productivity due to FIV-related fatigue failures of nuclear systems and components. Tests at a major high-flow test facility contributed by General Electric are a focal point of this program.

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E. SAFETY

The National Energy Plan, while recognizing that light-water reactors have a good safety record, cited additional actions that could be taken to improve safety. In addition, the President and members of Congress have expressed the goal of further improving the safety of these reactors. The Nuclear Regulatory Commission has responsive regulatory actions under way. The Department of Energy has a complementary effort for meeting these goals by emphasizing the development of improved safety systems.

This subprogram is structured around three major elements:

- Development of advanced concepts and new approaches to plant design, safety-system design, and the broadening of safety technology, such as an advanced emergency core-cooling systems (ECCS) and filtered, vented containment.
- Investigations of the man-machine interface to improve safety in control of normal operations and operator response during emergency situations, such as detection of incipient failures, early detection of malfunctions, and computer processing of control-room data for trend analysis and decisionmaking.
- Development and use of risk-based methods and studies to (a) identify sensitive systems, components, and processes, (b) reduce risk through design and operations changes, and (c) help influence licensing priorities, such as improving the application of nondestructive examination, providing input for the NRC update of WASH-1400 (Rasmussen report), and the analysis of failures to assure correction of generic problems.

In FY 1978, opportunities for improving LWR safety are being studied and preliminary work is being done on concepts for advanced safety systems and other safety improvements that have already been identified. In FY 1979, the second and third program elements above will begin to yield results, because their work generally affects procedures and computer software and can be implemented without the delays involved in changing hardware.

To assure focus on high-priority needs and to avoid duplication, this work is being coordinated with the Nuclear Regulatory Commission's confirmatory-research safety program, as well as with work conducted by the Electric Power Research Institute, the nuclear industry, and other nations.

LIGHT WATER REACTORS

PROGRAM LIGHT WATER REACTORS																															
SUBPROGRAM SAFETY																															
TASK				CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
				FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
PLANNING																															

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IV. ADVANCED FISSION SYSTEMS

A. PROGRAM OVERVIEW

An advanced fission system is defined as a reactor concept and associated fuel cycle that can achieve greater fuel utilization than can a light-water reactor (LWR) with a once-through fuel cycle (fuel utilization is defined as the amount of usable energy per unit quantity of natural uranium or thorium mined). The LWR with once-through fuel is the "reference" fission system in the United States --the system deployed now. There are 68 such power plants operating in the United States. Table 3 summarizes the characteristics of LWRs and other major types of power reactors of interest in the United States.

Table 3. Characteristics of Power Reactor Fission Systems

REACTOR TYPE, NEUTRON ENERGY	STATE OF DEVELOPMENT	EXISTING U S DEVELOPMENTAL REACTORS	COOLANT	FUEL CYCLE
LIGHT WATER REACTOR (LWR) Converter, thermal	Commercial	Many commercial	Water	U or U/Pu
HIGH TEMPERATURE GAS REACTOR (HTGR) Advanced converter, thermal	Semi commercial	Ft St Vrain	Helium	Th/U 235/U 233
CANADIAN DEUTERIUM URANIUM REACTOR (CANDU) Converter, thermal	Commercial	None	Heavy water	U
LIGHT WATER BREEDER REACTOR (LWBR) Breeder, thermal	Developmental	Shippingport	Water	Th/U 233
LIQUID METAL FAST BREEDER REACTOR (LMFBR) Breeder, fast	Developmental	EBR II & FFTF	Sodium	U/Pu
GAS COOLED FAST BREEDER REACTOR (GCFR) Breeder, fast	Developmental	None	Helium	U/Pu

In all power-reactor fission systems used today, the reactor is fueled by both fissile material (which neutrons can cause to fission, releasing energy) and fertile material (which neutrons can convert to fissile material). In the LWR reference fission system, the fissile material is uranium-235 and the fertile material is uranium-238, which is converted to fissile plutonium in the reactor. (A fuel cycle using thorium as the fertile material, which is converted to fissile uranium-233 in the reactor, is also being tested, but has not yet demonstrated commercial practicability.)

Fuel utilization depends largely on the rate at which fissile material is generated in the reactor in comparison with the rate at which it is consumed in fission in that reactor. The ratio of these rates is termed the conversion ratio of the system.

Among the advanced fission systems, there are two basic reactor types:

ADVANCED FISSION SYSTEMS

- Advanced converters, in which the conversion ratio is less than unity but greater than the approximately 0.6 ratio obtained in the reference LWR once-through system
- Breeders, in which the conversion ratio is greater than one, and thus more fissile material is generated than is consumed

Reactors also can be categorized as "thermal" reactors or "fast" reactors, designations that indicate the predominant speed of the fission-inducing neutrons with which the reactor is designed to operate. In the fission process, the neutrons that are released to continue the chain reaction are "born" at high velocities, with an average kinetic energy of about a million electron volts (1 Mev). These neutrons collide with the atoms of the reactor's coolant and structural materials, which slows the neutrons. Light atoms, especially hydrogen and carbon, are particularly effective in slowing neutrons; the lower limit of the attainable neutron energy, about 0.025 ev, is in thermal equilibrium with the materials in the reactor. Thus, thermal reactors are those in which the energy of the fission-inducing neutrons is of the order of magnitude of thermal equilibrium. LWRs and converter reactors have water or other light atoms in their cores, so they are thermal reactors. At the other end of the spectrum are the fast reactors, in which the average neutron energy is close to fission energies. The inherent nuclear properties of materials make breeding (i.e., conversion ratio greater than one) more readily achieved in fast reactors. But the design of fast breeder reactors is complex--only relatively high atomic-weight materials (e.g., sodium instead of water) can be used in their cores.

In LWRs and advanced converters, the additional fissile material generated from the fertile material is "burned" along with the fissile material initially charged, to extend the core life. Although the degree of extension depends on the conversion ratio, core life is limited not only by the amount of fissile material available, but also by the buildup of fission products, which tend to "poison" the chain reaction. Therefore, significantly improving fuel utilization requires reprocessing the spent fuel to remove fission products, then recycling the residual fissile material back into a reactor.

In a breeder reactor, the fissile material generated is more than sufficient to replace that initially charged and to keep the reactor fueled throughout the plant lifetime; excess fissile material produced can be withdrawn from the cycle and used to fuel new breeders or other types of reactors. Again, however, core life is limited by fission-product "poisoning," and the excess fissile material can be extracted for use in other reactors only by reprocessing the "spent" fuel.

History. The early commercialization of the LWR in the United States resulted from two major factors: (1) the availability of directly applicable technology spun off from the naval propulsion reactor program and (2) the existence of uranium enrichment facilities built for the nuclear weapons program. However, it was recognized even before LWR commercialization began that an advanced fission system such as the breeder would eventually be necessary if our nuclear fuel resources were to be fully exploited.

There are two principal types of advanced converter reactors now under consideration: the gas-cooled thermal reactor and the heavy-water-moderated natural-uranium-fueled reactor known as the CANDU (CANadian Deuterium-Uranium) reactor.

Gas-cooled reactors have had a long history abroad, their initial development in the UK and France culminating in the construction of more than 30 such low-temperature plants in Europe and Japan beginning in the late 1950s. They had low fuel conversion ratios and, although development continued to increase the coolant temperature and consequently the plant efficiency, their power costs generally have made them less favored than LWRs. In the United States, a small High Temperature Gas-cooled Reactor (HTGR) power plant was built in the late 1960s and operated more than 7 years (Peach Bottom No. 1), and an intermediate-scale (330 MWe) plant is now in the final stages of startup (Fort St. Vrain). There is hope that the economic performance and conversion ratios of future gas-cooled thermal-neutron power plants can be improved, and that their potential for high temperatures will make them particularly useful for industrial process heat applications.

Heavy-water-moderated natural-uranium-fueled reactors also have a long history, both as research and weapons-material-production reactors. Canada adapted the concept to power production, and since 1968 23 CANDU-type plants in the 200-700 MWe size range are operating or are under construction in Canada, India, Pakistan, Argentina, and South Korea. Until recently, little consideration has been given to deploying this type of reactor in the United States, largely because the United States has: (a) enrichment facilities, so it is not limited to using only natural-uranium-fueled reactors, and (b) questions as to licensability, economics, and institutional issues.

Interest in the Light Water Breeder Reactor (LWBR) concept, a thermal-neutron system, grew out of studies indicating that breeding (i.e., a conversion ratio greater than one) might be achieved in existing and future LWRs by using the thorium/uranium-233 fuel cycle, in which fissile uranium-233 is generated by neutron bombardment of fertile thorium. To test its breeding performance, an LWBR core has been operating in the LWR at the Shippingport Atomic Power Station since late 1977; after 3-4 years of operation, the core will be removed for analyses measuring its breeding ratio; those analyses are expected to be completed after 1984.

Development of liquid-metal-cooled fast breeder reactors (LMFBRs) began in the U.S. in the mid-1940s and progressed through the construction and successful operation of six small-scale liquid-metal-cooled reactor facilities. One of these, EBR-I, produced the world's first electricity generated by nuclear energy on December 20, 1951. Two (EBR-II and Fermi-1) demonstrated the delivery of power to a utility grid; the EBR-II facility also included in-plant recycle of fuel. Construction of a major liquid-metal-cooled reactor for performance testing fuels and materials, the Fast Flux Test Facility, is nearing completion. Another 12 LMFBR plants have been built and operated in the USSR, the UK, France, Japan, and West Germany, including three intermediate-scale (250-350 MWe) LMFBR demonstration plants in the USSR, France, and the UK. Another nine LMFBR-type plants are being built or planned. LMFBR development

ADVANCED FISSION SYSTEMS

now concentrates on technology for large components and safety of commercially economic power plants.

As a backup to the LMFBR system, the Gas-Cooled Fast Breeder Reactor (GCFR) is also being investigated. This reactor concept also stems from the long history of gas-cooled reactors abroad, but its design would differ considerably from the high-temperature gas-cooled thermal-neutron reactors. This is because the thermal-neutron systems use graphite-based ceramic fuels in which the carbon slows the neutrons, but the gas-cooled fast systems would use metal-clad LMFBR-type fuels and thus would require a considerably different thermodynamic cycle. Basic conceptual and safety studies are underway for this reactor type.

Program Redirection. The issue of proliferation of nuclear weapons was identified early in the history of nuclear power. Various institutional approaches have been taken toward its resolution, the most recent instrument being the Non-Proliferation Treaty of 1970, signed and ratified by 102 of the 154 nations of the world. The present United States policy announced by the President in April 1977 extends consideration of the proliferation issue to the seeking of technical as well as institutional solutions. As a result, the reprocessing of spent reactor fuel to obtain plutonium for recycling and the further demonstration and deployment of the fast breeder have been deferred indefinitely. The breeder program has been returned to a purely research and development mode to maintain this concept as a viable option for future exploitation if needed, and the entire advanced fission systems program has been keyed to the evaluation of reactor and fuel cycle options that may afford nonproliferation advantages.

Program Objectives. The primary objectives of the advanced fission systems development program are to:

- Assess all advanced fission options for their relative nonproliferation, economic, technical and institutional merits
- Develop the technology of the HTGR, both domestically and through international cooperation with foreign nations, and determine the potential for its large-scale deployment
- Determine the breeding characteristics of the LWBR
- Maintain and improve the technical and engineering basis for the fast breeder reactor as a viable option for future deployment if needed

Strategy. In pursuit of the first objective, two major system evaluation programs, one domestic and one international, have been established to assess the nonproliferation advantages and other characteristics of a wide variety of advanced reactor concepts and fuel cycles. Simultaneously, development is proceeding on HTGR fuels and systems, both for electricity and process steam applications, and the evaluation of LWBR breeding is proceeding. Fast breeder development continues, primarily centered around FFTF as a vehicle for testing reference and alternative fuel systems and gaining experience in operating

and maintaining large liquid metal systems, but also including basic work on the gas-cooled fast breeder alternative.

As illustrated in figure 8, this assessment and development effort is focused toward providing the necessary technical basis for an ultimate decision, within a few years, as to the future course of the fission option.

Figure 8
Advanced Fission Systems Program Strategy

ADVANCED FISSION SYSTEMS

The following table summarizes funding levels for the advanced fission systems program for FY 1977 through FY 1979.

ADVANCED FISSION SYSTEMS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Subprograms				
Systems Evaluation				
Operating*	\$207	\$7,265	\$2,309	(\$4,956)
Equipment	0	0	0	0
Construction	0	0	0	0
Subtotal	207	7,265	2,309	4,956
Gas Cooled Thermal Reactors				
Operating	15,900	31,000	28,500	(2,500)
Equipment	549	1,000	2,500	1,500
Construction	0	0	0	0
Subtotal	16,449	32,000	31,000	(1,000)
Water Cooled Breeder Reactors				
Operating	37,110	38,600	50,900	12,300
Equipment	2,350	3,100	3,100	0
Construction	9,500	0	9,000	9,000
Subtotal	48,960	41,700	63,000	21,300
Fast Breeder Reactors				
Operating	577,326	380,475	310,201	(70,274)
Equipment	45,738	31,280	26,681	(4,599)
Construction	98,950	121,305	62,485	(58,820)
Subtotal	722,014	533,060	399,367	(133,693)
Nuclear Power Studies				
Operating	6,400	8,300	23,991	15,691
Equipment	0	0	0	0
Construction	0	0	0	0
Subtotal	6,400	8,300	23,991	15,691
TOTAL	\$794,030	\$622,325	\$519,667	\$ (102,658)

* Excludes funding identified with DOE Office of Energy Research

B. SYSTEMS EVALUATION

1.0 Overview

In April 1977, President Carter announced a new national policy regarding nuclear power and its relationship to nuclear weapons proliferation that brought into question some of the basic planning assumptions regarding United States energy development.

The basic concern, leading to the Presidential policy statement, is that a nation possessing a civilian nuclear power system that uses separated plutonium, such as in certain spent-fuel reprocessing schemes, would have the material to build nuclear weapons. International and United States supply agreements and safeguards are inadequate to cope with this possibility on a very short time scale. There are, however, a number of technical and institutional barriers that might make the recovery of plutonium or other weapons-usable materials more difficult and time consuming.

Actions previously taken by the United States to control the spread of nuclear explosives capability include: banning export of sensitive facilities for enrichment or reprocessing of nuclear fuels; establishing strict controls on the use of United States-supplied nuclear materials; participating in the development and ratification of the Nonproliferation Treaty; instituting a strong safeguards program in the United States and aiding the implementation of the International Atomic Energy Agency (IAEA) Safeguards program.

To resolve this concern, two programs were set in motion: The Nonproliferation Alternative Systems Assessment Program (NASAP) and the International Nuclear Fuel Cycle Evaluation Program (INFCE). These programs provide the focus for the formulation and technical evaluation of alternative proliferation-resistant nuclear energy systems that could be deployed in the United States and foreign countries. NASAP is reevaluating nuclear energy systems; its emphasis is on nonproliferation characteristics and meeting international nonproliferation objectives. One of the terms of NASAP is that nonproliferation takes precedence over economic and commercial interests but that any nuclear systems selected must be competitive with other energy options if they are to enjoy widespread use. Thus tradeoffs among economics, resource utilization, safety, nonproliferation, and other factors must be carefully analyzed and considered, but the motivation for recommending a change in nuclear energy system development can only spring from its nonproliferation advantages.

NASAP will rank the alternative reactor types and fuel-cycle combinations as to their proliferation advantages with the expectation that a national fission energy development policy will be established within the next 2 or 3 years. The final recommendations of these studies will be in the form of strategies needed to allow international deployment of nuclear systems (reactor and fuel-cycle combinations, together with institutional constraints) providing acceptable resistance to proliferation.

Some nations are already building or planning to use nuclear systems that could allow access to directly usable weapons materials. They view such systems as a means for more efficiently using limited uranium resources, thus providing near- and long-term assurance of critical fuel supplies. Because of this active interest of other nations in plutonium-based systems, President Carter proposed the establishment of an International Nuclear Fuel Cycle Evaluation (INFCE) program to study alternatives that offer less proliferation risk.

The International Fuel Cycle Evaluation (INFCE) program was initiated by the United States under the direction of the Department of State (DOS). The more than 40 countries participating in INFCE will complete, by October 1979, technical and analytical studies of measures that could be taken at the national level and through international agreements to minimize nuclear-weapons proliferation. The Nonproliferation Alternative Systems Assessment Program (NASAP), which originates within the Department of Energy (DOE), will provide research and development recommendations to INFCE for the development and deployment of civilian nuclear power systems.

The following table shows the operating funding levels for NASAP for the FY 1977 to FY 1979 period. INFCE is an international program; as such, it is not funded by DOE.

SYSTEMS EVALUATION	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Nonproliferation Alternative Systems Assessment				
Nuclear Program	\$207	\$7,265	\$2,309	\$(4,956)
Office of Energy Research	1,080*	7,535*	4,000	(3,535)
Subtotal	1,287	14,800	6,309	(8,491)
International Nuclear Fuel Cycle Evaluation	0	0	0	0
TOTAL	\$1,287	\$14,800	\$6,309	\$(8,491)

* Funded under Nuclear Programs but shown under OER budget in FY 1979 Congressional submission to correspond with FY 1979 split of funds

2.0 Nonproliferation Alternative Systems Assessment Program (NASAP)

The overall goal of NASAP is to recommend to the Administration strategy options for implementing civilian nuclear power systems that, when deployed internationally, offer increased resistance to proliferation.

Objectives. In support of its goal, the principal objectives of NASAP are:

- To identify reactor and fuel-cycle combinations with high proliferation resistance and with potential for commercialization

- To identify institutional arrangements to enhance the proliferation resistance of selected systems
- To assess the ability of selected systems to gain international acceptance
- To provide technical support for the United States input to the INFCE
- To develop strategies to implement the most promising alternatives

Strategy. The strategy for achieving these objectives takes into consideration the following assumptions:

- A commitment to nuclear power with increasing nonproliferation constraints implies that a major change in the direction of presently planned nuclear power development programs may be required. The only justification for recommending such a change would be an increased proliferation resistance of proposed systems relative to existing systems.
- For any nuclear option to be viable, it must be competitive with other options and available when needed. Therefore, when considering proliferation resistance, the tradeoffs among technical feasibility, economics, resource utilization, and commercialization must be analyzed.
- It is difficult to specify nonproliferation constraints in concrete operational terms and to assess objectively the absolute amount of proliferation resistance of a system. Recourse must ultimately be made to achieving a consensus of technical, economic, and political viewpoints of relevant quantitative and qualitative issues. A practical methodology for generating the information necessary for evaluating the proliferation resistance of alternative systems in the context of commercial feasibility and international acceptability must be formulated.

Based on these assumptions, a three-part program strategy has been developed. In the first part, individual skills of private contractors, DOE program divisions and other public and private organizations provide basic data needed to characterize fully the nuclear systems and to support evaluation of the technical, political, and economic factors.

The second part of the program strategy consists of conducting an integrated assessment of alternative systems. All tradeoff analyses will be conducted using the same frames of reference and the same data base, i.e., the one created in part one of the strategy. Outside review will be obtained to place NASAP in perspective with evolving nuclear policy. The ultimate output of the integrated assessment will be strategy options for future nuclear policy.

The third part consists of providing necessary technical analyses and evaluations to develop and support United States participation in INFCE.

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It is expected that an important input from NASAP most likely will be the development of proposed proliferation-resistance criteria and assessment techniques. International input to NASAP will be provided through the INFCE communications channels established with other nations.

Structure. The NASAP program plan is structured into the following logical progression of steps:

- Select representative candidate systems for assessment. This process will allow for a workable number of systems to be presented for characterization without peremptorily dismissing any alternatives, while ensuring that all potential classes are included. Results of the sequential evaluative process may lead to increased scrutiny of additional systems within a particular class based on significant advantages in proliferation resistance.
- Develop a methodology for evaluating the key criteria, which consist of proliferation resistance, technological feasibility, economics, resource utilization, environmental and safety considerations, commercial feasibility, and international acceptability. This task will establish the basic approach to evaluation and will specify the data needed to perform the subsequent assessments.
- Characterize candidate systems in standard form to permit comparative evaluation. The technical and institutional factors of candidate systems will be defined in sufficient detail in accordance with the data requirements in order to permit valid assessments.
- Evaluate and assess the preliminary candidate systems. Under this task, the relative merits of each individual system with respect to each key assessment factor will be evaluated. Strong and weak points will be identified and other factors for each system will be summarized. The results of this task may require additional investigation of a selected system within a class, and will provide direct input to the integrated assessment.
- Perform an integrated assessment of all key factors of final candidate systems. This task represents the culminating effort of the program, providing the basis for the strategy option necessary to implement final recommendations.

The relationships of the above steps and the interfaces with the INFCE are detailed in figure 9.

Figure 9
Relationship of NASAP Steps and Interfaces with INFCE

Status. The selection of candidate systems has been completed. The development of methodologies for assessing proliferation resistance and other key factors and design guidelines will be completed by February 1978. All contracts have been let for characterizing preliminary reactor systems. Work plans are being received. Also work programs are being received for the evaluation of key factors of candidate systems. For the integrated assessment, a general approach, which includes quarterly updating of preliminary findings, has been defined.

The schedule is constrained by both INFCE support tasks and the level of effort required. All NASAP activities have been scheduled to achieve final recommendations by mid-1979 in order to provide timely input to the INFCE. Draft recommendations will be available in April 1979. Preliminary recommendations will be available as early as March 1978. Quarterly reports will provide interim recommendations.

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3.0 International Nuclear Fuel Cycle Evaluation (INFCE)

To ensure that international cooperation and participation is directed toward solving the problem of proliferation from the nuclear fuel cycle, the International Nuclear Fuel Cycle Evaluation (INFCE) Program was created. Representatives from interested countries approved the following statement of purpose at the INFCE organizing conference.

"The participants in the Organizing Conference of the International Nuclear Fuel Cycle Evaluation are conscious of the urgent need to meet the world's energy requirements and that nuclear energy for peaceful purposes should be made widely available to that end. They are also convinced that effective measures can and should be taken at the national level and through nuclear weapons without jeopardizing energy supplies or the development of nuclear energy for peaceful purposes."

Objective. The specific actions necessary to support the above purpose comprise the stated objective of the INFCE, which is to:

"Conduct technical and analytical studies of measures which can and should be taken at the national level and through international agreements to minimize the danger of the proliferation of nuclear weapons without jeopardizing energy supplies or the development of nuclear energy for peaceful purposes."

Strategy. The implementing strategy to be pursued by INFCE in meeting its objective consists of five points:

- All interested states and relevant international bodies may participate in the work and will have an equal opportunity to contribute.

- INFCE represents a technical analytical study, not a negotiation. Participants would not be committed to INFCE's results.
- Participation in the studies would not jeopardize participants' fuel-cycle policies or international cooperation, agreements, and contracts, provided that agreed-upon safeguard measures are applied.
- Special consideration will be given to the needs of developing countries.
- The evaluations will be conducted along lines set forth in "INFCE Technical and Economic Scope and Methods of Work" and the "INFCE Proposed Terms of Reference."

Structure. At the INFCE organizational meeting in October 1977, eight working groups were formed to deal with the areas of concern (Table 4). The technical tasks of each working group will be coordinated by a Technical Coordinating Committee comprised of the co-chairman of each working group. Comparative evaluations will be promoted by consistent guidelines established by the organizing conference.

Table 4. Working Groups for INFCE

WORKING GROUP	CO CHAIRMAN
Fuel resources and heavy water availability	Canada, Egypt, India
Enrichment availability	France, Federal Republic of Germany, Iran
Assurances of long term supply of technology, fuel, heavy water, and supplies for all net users	Australia, Philippines, Switzerland
Reprocessing, plutonium handling, and recycle	Japan, United Kingdom
Fast breeders	Belgium, Italy, USSR
Spent fuel and waste management	Argentina, Spain
Fuel conservation and fuel utilization in thermal reactors	Finland, Netherlands, Sweden
Advanced fuel cycle and reactor concepts	Republic of Korea, Romania, USA

Status. At the organizing meeting, the International Atomic Energy Agency agreed in principle to support INFCE by providing technical and secretariat assistance. Co-chairmen of the working groups were defined. As of December 1977, country membership of the working groups was established. Additionally, individuals representing the United States on the various working groups were being designated. One coordinating committee meeting of working group co-chairman has been held at which procedural issues were discussed. At the January 30, 1978 meeting of Working Group 8, economic and technical assessment ground rules will be issued.

The Department of State is the lead agency for United States participation in the INFCE. While no direct funding in DOE is provided for the INFCE, analytical studies conducted under the NASAP support the requirements of

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the INFCE. It is expected that NASAP's contributions will be most significant in the area of evaluating proliferation resistance and establishing an integrated assessment technique.

PROGRAM: ADVANCED FISSION SYSTEMS SUBPROGRAM: SYSTEMS EVALUATION CATEGORY: INTERNATIONAL NUCLEAR FUEL CYCLE EVALUATION (INFCE)																												
TASK	CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
	FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
ORGANIZING CONFERENCE				▼																								
WORKING GROUP ORGANIZATION				▽																								
TECHNICAL COORDINATING COMMITTEE MEETING				▽	▽																							
WORKING GROUP 8 MEETING				▽																								
INFCE PLENARY CONFERENCE								INITIAL REPORTS				FINAL REPORTS																
								▽				▽																

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C. GAS-COOLED THERMAL REACTORS

1.0 Overview

The Gas-Cooled Thermal Reactor (GCTR), referred to in the U.S. as the High Temperature Gas-Cooled Reactor (HTGR) and in Europe as the High Temperature Reactor (HTR), has evolved from more than 20 years of development.

The HTGR is an advanced fission converter, i.e., a reactor system that provides better uranium fuel utilization than the LWR but not as good utilization as the breeder.

The HTGR has the potential for operating on a variety of fuel cycles. As shown in figure 10, the major fuel-cycle options are: low-enriched uranium (LEU) throwaway or stowaway, medium-enriched uranium (MEU) and thorium* with throwaway/stowaway or recycle, and high-enriched uranium (HEU) and thorium with recycle. It appears that the most likely fuel-cycle candidate for future use in the HTGR is the MEU thorium cycle with recycle. But until the results of the Nonproliferation Alternative Systems Assessment Program (NASAP) are known, the reference cycle for the HTGR is the MEU/Thorium or LEU throwaway or stowaway fuel cycle. In the HEU diagram of figure 10, the "all fissile" and "all uranium" on the recycle branch are essentially identical because very little fissile plutonium is produced in the cycle.

Power Conversion Modes. There are four power conversion modes currently being considered for use with the HTGR:

- Conventional steam Rankine cycle to electricity
- Closed, direct, gas-turbine (Brayton) cycle to electricity
- Through heat exchanger(s) to high-temperature industrial process applications (coal conversion, hydrogen production, steelmaking, etc.)
- Combined cycle (combination of the third with either of the first two)

In the steam Rankine cycle, heat is transferred from the reactor core to helium, which then passes through a heat exchanger (steam generator) where the heat is transferred to water, which is converted to steam. The steam is then used in a conventional steam turbine-generator to produce electricity.

*When thorium is charged into a reactor along with fissile material, it is converted to uranium-233, which is fissile and can be consumed to produce energy additional to that obtained by fission of the originally charged fissile material.

Figure 10
Three HTGR Fuel Cycles

In the direct-cycle HTGR, heat is transferred from the reactor core to helium, which passes directly through a gas turbine to generate electricity. It is projected that an HTGR plant operating on a direct cycle will have lower capital cost than an HTGR operating on a steam cycle because it is estimated that 30 to 40% less concrete and steel will be required for the direct-cycle plant. The Brayton cycle has the additional characteristic of high reject-heat temperature, which can be used for:

- Economic dry cooling (no cooling water required)
- Bottoming cycle (increases efficiency to >45%)
- Low-temperature process heat

In the high-temperature process heat application, heat is transferred from the reactor core to helium, which then passes through a heat exchanger where this heat again is transferred, either directly or through an intermediate loop, to a high-temperature industrial process such as coal conversion, hydrogen production, steelmaking, etc.

In the combined-cycle mode, energy would be extracted through heat exchanger(s) at high temperatures for process applications, and the lower-temperature heat could then be used to produce electricity using either the conventional steam (Rankine) or direct gas-turbine (Brayton) cycle.

Markets. Gas-cooled thermal reactors have the potential of serving two distinct markets:

- Electricity production
- High-temperature (1000°C) industrial process heat

The share of the electricity production market the HTGR will capture will depend on its economics and how important its unique features are to the utility customers.

The gas-cooled reactor currently is the only reactor concept that has the capability to be used for high-temperature process applications such as coal conversion, hydrogen production, steelmaking, etc. For the sake of comparison, the outlet maximum temperature capability of the LWR is 500 to 600°F while that of the gas-cooled thermal reactor is 1800 to 2000°F. Germany has had an HTR (AVR) operating for several years with a helium exit temperature of 1740°F.

United States HTGR Development History. The Experimental Gas-Cooled Reactor (EGCR), a helium-cooled reactor having metal-clad fuel, was constructed at Oak Ridge during the early 1960s. It was never operated because the HTGR technology with its ceramic fuel cladding and higher temperature capability made obsolete the concept of gas-cooled converter reactors employing metal-clad fuels.

Commercial HTGR experience in the United States began with the Peach Bottom reactor, a 40 MWe plant, which was operated for over 7 years by the Philadelphia Electric Co. until operation was discontinued in 1974 due to the less favorable economics of small electric power plants. The Fort St. Vrain reactor, owned by Public Service of Colorado, is now undergoing startup testing and has recently operated at 68% of the design power level of 330 MWe. Several years ago, the General Atomic Co. had commitments, or near commitments, for larger central station power plants; however, all these orders were cancelled, and Fort St. Vrain remains the only United States HTGR plant in operation or under construction. DOE is working with utilities and reactor suppliers to determine the proper role of the HTGR system for steam Rankine-cycle power plant applications.

Objectives. The near-term objectives of the HTGR program are to:

- Determine the potential for commercializing the HTGR steam cycle, with a decision on Government support scheduled for mid-FY 1978
- Support key generic technology areas that apply to either HTGR steam cycle or advanced HTRs

- Explore the potential for international cooperation through the government-to-government gas cooled reactor "umbrella" agreement between United States, West Germany, France, and Switzerland

Strategy. The program strategy is to develop HTGR systems by providing United States utilities the opportunity to assume a leadership role in the program and, to a lesser extent, through a cooperative international program involving the United States, Germany, France, Switzerland, and possibly other countries.

DOE has established two basic requirements for continued government support of the HTGR steam-cycle system:

- The United States utility industry must take a leadership role and provide some funding.
- The United States reactor-supplier industry must accept the utility industry's leadership role and must be broadened beyond the General Atomic Co. in order to provide a stronger and more viable supplier industry.

Consistent with these requirements, a utility organization tentatively known as Gas Cooled Reactor Associates (GCRA) has formally organized. The purpose of this organization is to determine the plant design characteristics and type of development programs that best serve utility interests in gas-cooled thermal reactors, as well as to provide management support for the program. It is also expected that GCRA will provide direct utility funding to the HTGR program in FY 1978 and FY 1979.

Types of Gas-Cooled Thermal Reactors. There are two basic types of gas-cooled thermal reactors under development in the United States:

- High Temperature Gas-Cooled Reactor Steam Cycle (HTGR-SC), the concept General Atomic tried unsuccessfully to commercialize. It is being considered by U.S. utilities at present.
- Advanced High Temperature Reactors (HTR), including
 - HTGR Direct Cycle, an advanced HTGR system for electric power production that has potential for greater resource conservation and better economics than the HTGR steam cycle, but would require much more extensive development and is a longer term option
 - VHTR (Very High Temperature Reactor) for high-temperature process-heat applications such as coal conversion, hydrogen production, etc., which are not presently possible with the LWR or LMFBR

The following table shows the operating funding levels by category for the FY 1977 to FY 1979 period.

GAS-COOLED THERMAL REACTORS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
High Temperature Gas-cooled Reactor	\$12,300	\$25,000	\$22,500	\$(2,500)
Advanced High Temperature Reactor	3,600	6,000	6,000	0
TOTAL	\$15,900	\$31,000	\$28,500	\$(2,500)

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2.0 High Temperature Gas-Cooled Reactor

The steam-cycle HTGR is the most developed of the gas-cooled thermal reactor concepts and the one for which General Atomic Co. had 10 commitments, or near commitments, for large commercial electric power plants. Figure 11 shows the flow diagram for a typical HTGR steam-cycle plant.

Figure 11
Flow Diagram of Typical HTGR with Rankine Cycle

Some important characteristics of the HTGR are:

- Helium coolant
- Graphite moderator
- Ceramic fuel cladding
- Integrated into prestressed-concrete reactor vessel (PCRVR)
- Fuel cycle flexibility, advanced converter (conversion ratio 0.6 to 0.9)
- Theoretically could be a thermal breeder

The nuclear steam supply system (NSSS) is contained in a prestressed concrete reactor vessel (PCRVR); the major components of the system are the reactor core, circulators, and steam generator. High-pressure (700 psi) helium is moved through the system by means of the circulators where it picks up heat in the graphite-moderated ceramic-clad-fuel reactor core and then transfers this heat through the steam generator to water and steam. The steam side of the plant is a typical high-pressure high-temperature steam power conversion system not unlike those found in modern fossil-fired power plants.

Application Intentions and Improvements Sought. The application of the HTGR steam cycle is in the electric power utility industry. The potential improvements that could result from using this system are:

- Less uranium ore consumption, depending on which fuel cycle is employed

- Less environmental impact (water, wastes, etc.)
- Siting flexibility as a result of different safety characteristics and less environmental impact

It does not have a significant, if any, economic advantage relative to LWRs.

Objectives. The objectives are to:

- Determine the utility interest in the HTGR steam cycle and, as appropriate, develop government/industry arrangements for commercialization
- Proceed with the R&D in key technology areas

Strategy. The strategy involved in meeting the first objective is to give the utilities the opportunity to make the decision on the commercialization of the HTGR steam cycle by encouraging formation of the HTGR utility organization, Gas-Cooled Reactor Associates (GCRA). In this way DOE hopes to ensure that any government funds used to support the development of this system will be in support of a plant the utilities will purchase.

The strategy involved in meeting the second objective is to proceed with the necessary development in key technology areas to reduce the development risks and provide a sound basis for plant licensing.

Description of Work. Briefly, the work under the two broad categories involves:

- Utility Initiative:
 - Prepare preliminary program plan
 - Define design requirements so as to establish a standard optimized HTGR plant design
 - Provide site data for overall plant preliminary design
 - Establish utility benefits and risks of commercializing the HTGR steam cycle
 - Review the current HTGR-SC development program and recommend revisions as necessary
 - Establish supplier business arrangements
 - Review and assess fuel-cycle options
- HTGR Technology - design, analysis, and testing in the technical areas of:
 - Safety

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- Structural (PCRV, thermal barrier, reactor internals)
- Heat exchangers
- Materials
- Reactor core
- Fuels
- Graphite development
- Fission products
- Reactor operating experience (Fort St. Vrain, Peach Bottom)

Participants. The contractors involved in the HTGR steam-cycle program are: Gas-Cooled Reactor Associates, General Atomic Co. (GA), San Diego, Calif., and Oak Ridge National Laboratory (ORNL), Oak Ridge, Tenn.

Status. The status of the HTGR steam-cycle development in 1975 was that approximately 50% of the R&D and design work necessary for a large commercial plant was complete. As a result of the utilities reviewing their requirements for plant size, configuration, and type of HTGR fuel cycle, not much additional progress has been made relative to a specific HTGR steam-cycle plant design since 1975. The next steps involve the utilities firming up their HTGR steam-cycle design requirements (expected in FY 1978), and then Government and the private sector proceeding to implement those requirements in the development program.

The status of the utility initiative is that the Gas-Cooled Reactor Associates utility group was formed in February 1978 and their recommendations on the HTGR steam cycle were made to DOE in time for a mid-FY 1978 DOE management decision on future government support for the HTGR steam cycle.

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3.0 Advanced High Temperature Reactor

As indicated above, Advanced High Temperature Reactors presently include two basic concepts:

- Direct-cycle HTGR
- VHTR for process-heat applications

Direct-cycle HTGR. Figure 12 shows the flow diagram of the direct-cycle gas-turbine HTGR (GT-HTGR). Several important differences relative to the HTGR steam-cycle plant should be noted:

- Both the nuclear steam supply system (NSSS) and the power-conversion loop are contained within the prestressed-concrete reactor vessel (PCRV).
- Direct helium turbine power conversion (no steam loop)
- Recuperator added to increase efficiency

- High helium reject-heat temperature (434°F) leads to economical dry-cooling capability (LWR reject-heat temperature is typically 125 to 150°F).

Figure 12
Typical HTGR with Direct Brayton Cycle

VHTR for Process-heat Application. As shown in figure 13, this concept involves flowing helium through the reactor core by means of a circulator and then transferring the heat through a steam reformer or intermediate heat exchanger for use in a high-temperature process (i.e., H₂ production); then through a steam generator where steam is made for either electric power production or lower temperature process heat applications.

Applications and Improvements Sought, The HTGR direct cycle is for central-station electric power production. The improvements it offers, in addition to those offered by the HTGR steam cycle, are:

- Economic dry cooling
- Bottoming cycle with efficiencies > 45%
- Combined cycle using reject heat for low-temperature process applications
- 30 to 40% less steel and concrete relative to HTGR steam cycle and thus a significant capital cost advantage
- Additional reductions in uranium ore utilization and environmental impact due to higher thermal efficiency

The high-temperature process-heat application is one which, of all current

Figure 13
Very High Temperature Reactor (VHTR) for Producing Process Heat by Direct
Coupling or Intermediate-loop Coupling

nuclear fission reactors, only the gas-cooled thermal reactor can perform. It thereby opens up a new, and potentially very large, market for the application of nuclear power.

Objectives. The specific objectives of the work on the advanced high temperature reactors are to:

- Develop conceptual designs for both VHTR and GT-HTGR
- Conduct a high-temperature helium materials test program
- Conduct limited international cooperation with West Germany, France, and Switzerland under the gas-cooled reactor "umbrella" agreement

Strategy. In the event that the HTGR steam cycle cannot be commercialized, sufficient information should be developed on the advanced concepts in the form of conceptual design to permit the Government, utility and industry to determine what role these advanced HTR concepts should play in the long-term U.S. energy strategy. In the event the HTGR steam cycle is commercialized, these concepts can be thought of as long-range product improvements. The materials for these concepts are on the critical development path, so the high-temperature materials test program was initiated in FY 1977.

The international cooperation underway on advanced HTRs will be pursued in a manner that will not detract from the near-term emphasis on United States HTGR steam-cycle development, but will attempt to take advantage of the existing programs on advanced HTRs in other countries. International cooperation is expected to reduce the costs and technical risks associated with development of these reactor concepts. If the HTGR

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steam-cycle commercialization attempt is abandoned, the United States program could shift to focus on the longer-term development of advanced HTRs.

Description of Work. The specific work involves:

- Evaluation of candidate metal alloys for application in the VHTR and direct-cycle reactors, which entails determining:
 - Surface stability
 - Structural stability
 - Creep properties
 - Stress rupture properties
 - Fatigue properties
- Conceptual design and analysis of:
 - Alternate direct-cycle HTGR plant configurations
 - VHTR process-heat reactor
 - German pebble-bed reactors
- International cooperation with the FRG, France, and Switzerland on:
 - Fuels
 - Graphite
 - Materials
 - Safety
 - Systems
 - Components
 - Direct-cycle HTGR conceptual design

Participants. The contractors involved in this program are General Atomic Co., General Electric Co., Oak Ridge National Laboratory, and Los Alamos Scientific Laboratory.

Status. Conceptual design activities have been underway on both the direct-cycle HTGR and VHTR for process-heat applications for several years. A group of utilities has been providing some financial support to the General Atomic Co. for the direct-cycle HTGR and they are expected to continue in FY 1978. Work on the VHTR has involved the definition of conceptual designs by GE, Westinghouse, and General Atomic and an evaluation of these by Oak Ridge National Laboratory. The materials test program is being conducted by GE.

The gas-cooled reactor "umbrella" agreement between the United States and FRG was signed in Bonn, Germany, in February 1977. This agreement establishes the basis for any cooperative development undertaken. France and Switzerland signed this agreement in October 1977 and are expected to become actively involved. The nature of cooperation depends on the United States decisions on the HTGR steam cycle and could range from a pure technology exchange to an actual cooperative development program on a specific HTR concept that could provide each country the capability to build and deploy the particular HTR system.

A United States FRG Ad Hoc Committee comprised of government representatives has been meeting regularly (quarterly) to explore possibilities for cooperation. Subgroups have been established in the United States and FRG made up of industry, utility, and user representatives to advise their respective Ad Hoc Committee representatives.

There are now approximately thirty specific pieces of technical work that have been identified as candidates for potential cooperation. Several have been approved and are underway; the remaining are in various stages of preparation and negotiation. This approach was adopted to initiate some limited areas of cooperation while awaiting major HTGR/HTR programmatic decisions.

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D. WATER-COOLED BREEDER REACTORS

1.0 Overview

The purpose of this program is to improve significantly the utilization of nuclear fuel resources for the generation of electrical energy by water-cooled nuclear reactors.

In the early 1960s, work done at AEC laboratories under the direction of the Naval Reactors Division showed that it might be possible to develop a practical self-sustaining breeder cooled with ordinary (light) water and fueled with uranium-233 and thorium. Such a breeder would have the advantage of utilizing the well-established technology of light-water reactors, which is the basis of the present commercial nuclear power industry. The principal development necessary was the breeder core itself. This early work led to establishment of the Light Water Breeder Reactor (LWBR) project in December 1965.

Successful development of this breeder cycle would make available for power production about 50% of the energy potentially available in the thorium reserves, a source of energy that might be many times greater than known fossil-fuel reserves. This is the only known approach using light-water-cooled reactors that can make available for power production significantly more than 1 or 2% of the energy potentially available in the nuclear fuel reserves.

Objective. The immediate objective of the LWBR program is to confirm that breeding can be achieved in existing and future light-water-reactor systems using the thorium/uranium-233 fuel system.

Strategy. The work to achieve this goal involves:

- The operation and testing of the Shippingport Atomic Power Station
- The development, manufacture, and operation of a Light Water Breeder Reactor (LWBR) core
- The development and dissemination of technical information on the LWBR concept

An LWBR core has been developed, designed, fabricated, and installed and is now operating in the DOE-owned pressurized-water reactor plant of the Shippingport Atomic Power Station at Shippingport, Pa. It is expected that the core will be operated for 3 or 4 years. At the end of this period, the core will be removed and the spent fuel shipped to the Naval Reactors Expended Core Facility in Idaho for a detailed examination and determination of breeding performance. It is expected that the expended core will contain about one percent more fissionable material than the initial loading.

The Advanced Water Breeder Applications (AWBA) project was initiated in FY 1976 to develop and disseminate technical information that will assist United States industry in evaluating the LWBR concept and its commercial application. Over the next 5 to 10 years, the AWBA project will explore some of the problems that will be faced by industry in adapting technology developed in the LWBR program. The focus of this program will be on developing technical information and promptly disseminating it. In this way vendors, utilities, and other interested organizations will be able to determine the degree to which they desire to apply LWBR technology.

Participants. The Water-Cooled Breeder Reactor Program is carried out primarily at two government-owned laboratories: Bettis Atomic Power Laboratory, Pittsburgh, Pa., and Knolls Atomic Power Laboratory (KAPL), Schenectady, N.Y. These laboratories are operated for the Naval Reactors Division of DOE by the Westinghouse Electric Corp. and General Electric Co., respectively. Some aspects of the work, including procurement of developmental reactor-plant components, are handled by Westinghouse's Plant Apparatus Division, an engineering procurement organization, which works exclusively on Naval Reactors programs. Bettis Atomic Power Laboratory also manages the DOE's Expended Core Facility in Idaho, where most of the spent-core examination, including nondestructive proof of breeding, will be performed.

Proliferation Aspects. All types of breeders require reprocessing of expended cores and use of the recovered fuel material in subsequent cores. The problem of recovery of the fuel material is primarily related to the radioactive fission products. Recovery of uranium-233 and thorium from expended LWBR cores would be about as difficult as recovery of plutonium-239 and uranium from present light-water-reactor cores, since the quantities of fission products are essentially the same for uranium-233, plutonium-239, and uranium-235. Similar quantities of fission-product wastes would have to be permanently disposed of in either case.

The recycling of plutonium-239 must be performed in special enclosures, since plutonium is toxic and must be isolated from workers and from the environment. The recycling of uranium-233 recovered from spent LWBR cores would require remote handling behind heavy shielding because of radioactivity that is not removed with the fission-product wastes.

Plutonium-239, uranium-233, and highly enriched uranium-235 must be protected against theft, since each could be used to make nuclear weapons. The radioactivity associated with recycled uranium-233 might make such theft more hazardous and less difficult to detect.

Environmental and Safety Aspects. The types of reactor plants being considered in the Water-Cooled Breeder Reactor program are based on the proven light-water reactor technology that has been demonstrated to be practical in over 20 years of operating experience and that is well established today. The safety and environmental effects of LWBR reactors would be comparable to those for existing light-water-reactor plants.

The uranium-233 needed to fuel a light-water breeder will be made in prebreeder cores. The rate of buildup of uranium-233 in prebreeder cores

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will depend on the fuel strategy chosen; calculations indicate that enough uranium-233 for a commercial-size breeder core could be produced in 10 to 25 years of operation with prebreeder cores.

Once enough uranium-233 has been accumulated to put a light-water reactor on the breeding cycle, the increment of capacity represented by that reactor would be self-sustaining without further need for mining and enrichment of uranium. The only makeup nuclear fuel required would be small amounts of thorium. When a plant on the breeding cycle reaches the end of its useful life, the core removed from the plant would contain sufficient uranium-233 to start a replacement plant directly on the breeding cycle without the need for prebreeder cores.

The basic potential of the LWBR concept is to increase significantly the amount of useful energy that can be produced from our nuclear fuel reserves using water-cooled reactors. One measure of this potential is the reduction in the amount of uranium that must be mined to produce a given amount of useful energy.

Using current commercial light-water reactors, a plant with an electrical output of 1,000 megawatts (MWe) requires mining about 130 tons of uranium ore concentrate (U_3O_8) per year. Assuming such a plant lasts for 40 years, 5,200 tons of U_3O_8 would have to be mined (and enriched) to fuel the plant over its lifetime. An additional 5,200 tons of U_3O_8 would have to be mined to refuel a replacement plant over its 40-year lifetime, and so on.

Preliminary studies show that, using the LWBR concept, it may be necessary to mine only about 3,000 tons of U_3O_8 to make enough uranium-233 in prebreeder cores to fuel a 1,000 MWe breeder. Once enough uranium-233 is accumulated to put a light-water reactor on the breeding cycle, the increment of electrical capacity represented by that reactor would become self-sustaining without further mining of uranium. The only makeup fuel required would be small amounts of thorium. Thus, each increment of capacity using the LWBR concept instead of present types of commercial light-water reactors would greatly reduce the long-term uranium mining requirements.

The following table shows the operating funding levels by program category for the FY 1977 to FY 1979 period.

WATER COOLED BREEDER REACTORS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Shippingport Atomic Power Station	Data not available	Data not available	\$11,500	
Light Water Breeder Reactor			25,300	
Advanced Water Breeder Applications			14,100	
TOTAL	\$37,110	\$38,600	\$50,900	\$12,300

2.0 Shippingport Atomic Power Station

The Shippingport Atomic Power Station, located 25 miles northwest of Pittsburgh, Pa., was the first large-scale central-station nuclear power plant in the United States and the first plant of such size (initial operation, 60 MWe net) in the world operated solely to produce electric power. This project was started in 1953 by the United States Atomic Energy Commission (now Department of Energy (DOE)) to confirm the practical application of nuclear power for commercial purposes. Initial criticality of the first Shippingport reactor core was achieved on December 2, 1957. Operation of the Shippingport Plant with pressurized water reactor cores installed has provided much of the technology being used for design and operation of commercial central-station nuclear power plants now in use.

Objective. The objective of the Shippingport Atomic Power Station continues to be the advancement of basic technology of pressurized water reactors, including the development and testing of advanced reactor concepts. The operation of the Light Water Breeder Reactor Core in the Shippingport plant is an extension of the kind of development work that has been carried out at this facility since its initial operation in 1957.

Participants. The Shippingport Atomic Power Station, which is operated under the Water-Cooled Breeder Reactors Program, is a joint project of Naval Reactors Division of DOE and the Duquesne Light Co. Bettis Atomic Power Laboratory designed and developed the reactor portion of the Shippingport Station under the direction of the Division of Naval Reactors, and DOE owns the reactor-plant portion of the station. Duquesne Light Co. owns the Shippingport site and the conventional turbine-generator portion of the station and operates the reactor plant under DOE contract.

Status. During FY 1978, the Shippingport Atomic Power Station will continue in operation with the Light Water Breeder Reactor Core installed. In addition, disposal and examination of fuel from the previously installed PWR Core 2 will be initiated. Procurement of the pressurized water reactor core barrel disposal cask and preparations for cask loading procedures are in progress.

PROGRAM ADVANCED FISSION SYSTEMS																							
SUBPROGRAM WATER-COOLED BREEDER REACTORS																							
CATEGORY SHIPPINGPORT ATOMIC POWER STATION																							
TASK	CY 1977			CY 1978			CY 1979			CY 1980			CY 1981			CY 1982			CY 1983				
	FY 1977	FY 1977	FY 1977	FY 1978	FY 1978	FY 1978	FY 1979	FY 1979	FY 1979	FY 1980	FY 1980	FY 1980	FY 1981	FY 1981	FY 1981	FY 1982	FY 1982	FY 1982	FY 1983	FY 1983	FY 1983		
PLANT MODIFICATION			▼																				
CORE INSTALLATION	▲		▼																				
STARTUP AND ACCEPTANCE TESTING				▲	▼																		
POWER OPERATION				▲											▼								
POST OPERATION TESTING														▲	▼								
DEFUELING															▲				▼				
SHIPMENT OF SPENT FUEL															▲								
PRESSURIZED WATER REACTOR																							
DESIGN CERTIFICATION OF SHIPPING CONTAINER				▼																			
SHIPMENT OF SPENT FUEL AND CORE BARREL				▲							▼												

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3.0 Light Water Breeder Reactor

Objectives. Operation of the Light Water Breeder Reactor core at Shippingport is being conducted primarily to accomplish three objectives:

- To establish whether breeding can be achieved in a light-water nuclear power plant using a thorium/uranium-233 fuel system
- To confirm a practical way to obtain energy from thorium
- To show that a light-water breeder reactor core that utilizes thorium fuel can be installed in a pressurized-water nuclear power plant using the same type of pressure vessel, pumps, heat exchangers, piping systems, etc., as are used in commercial pressurized-water reactors (PWRs)

Strategy. The Light Water Breeder Reactor program includes development of the supporting technologies required for the successful operation of a breeder reactor, including:

- The development of a thorough understanding of the nuclear and materials characteristics of the thorium/uranium-233 fuel system in water reactors
- Analytical and experimental development supporting improved performance of Zircaloy-clad oxide fuel
- The engineering design of a practical movable-fuel control system that will accomplish all normal control functions without unproductive loss of neutrons

- The design of a reliable fuel-rod support system with minimum detrimental effect on neutron economy

Status. The Light Water Breeder Reactor (LWBR) core has been installed and is operating in the existing DOE-owned pressurized-water reactor plant at Shippingport, Pa. The startup and acceptance test program was completed and the plant was released for routine commercial operation on December 2, 1977. During FY 1978, operational inspections, tests, maintenance, and evaluation of the Shippingport plant LWBR will continue. The core thermal-hydraulic performance is being evaluated, documented, and periodically reviewed. Light Water Breeder Reactor core-protection system performance is being followed. Periodic tests are being conducted, including evaluations of performance predictions, a core-follow calculational effort, and a physics test program. Work to assess the full useful life of the Shippingport LWBR core will continue. Evaluations and documentation of subsequent operations of the Light Water Breeder Reactor will be provided. Additional support and evaluation of fuel, cladding, and structural samples are planned to determine the effect of irradiation on water-cooled reactors with the objective of improving fuel design parameters. Operation and test of the plant will provide valuable information on long-term performance of light-water reactors, systems, and components.

Planning and preparations for defueling and module disassembly of the spent Light Water Breeder Reactor core are in progress. The design and procurement of fuel removal equipment, module disassembly equipment, fuel storage racks, and fuel shipping equipment are being initiated. Prototypical equipment will be tested as the safety analyses are completed. The shipping container safety analysis report for packaging has been initiated.

After about 3 to 4 years of power operation, the Light Water Breeder Reactor core modules will be removed from the Shippingport reactor vessel and will be shipped to the Expended Core Facility (ECF) at the Naval Reactors Facility in Idaho for detailed core examination and determination of breeding performance. After the detailed spent-fuel examination, the spent fuel will be turned over to the DOE Division of Waste Management for storage. To support the core examination and determination of breeding performance, a construction project for extension of the Expended Core Facility (ECF) water pits, called the Breeding Nondestructive Assay Facility (BNAF) was approved as a part of the FY 1977 budget. Detailed design of the water-pit expansion at ECF for the BNAF is in progress and construction is scheduled to begin during FY 1978. Irradiation tests involving LWBR-type fuel and structural materials will continue.

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4.0 Advanced Water Breeder Applications

Objective. The Advanced Water Breeder Applications (AWBA) project was initiated in FY 1976 to develop and disseminate technical information that will assist United States industry in evaluating Light Water Breeder Reactor technology and applying it to their programs. The project explores the technical problems that will be faced by industry in adapting technology confirmed in the Light Water Breeder Reactor Program. The focus of this program is on first developing technical information and then promptly disseminating it.

To operate a thorium breeder reactor, it is necessary to have enough uranium-233 for the initial fuel loading. Because uranium-233 is not available in nature, this initial loading must be produced from thorium by some means such as the operation of a nuclear power plant. The LWBR core for Shippingport was built using uranium-233 produced in DOE production reactors at Savannah River and Hanford. This approach was chosen to directly prove whether breeding could be obtained in a light-water reactor using the thorium/uranium-233 system. Fuel for commercial LWBR breeders, however, would have to be produced in large commercial reactors fueled initially with thorium and the naturally occurring uranium-235 or with plutonium available from other reactor operations. These reactors to produce uranium-233 are designated "prebreeders."

Strategy. Information being developed under the AWBA program includes concepts for commercial-scale prebreeder cores that will produce uranium-233 for light-water breeder cores while producing power for customers, improvements for breeder cores based on the technology developed for the fabrication and operation of the Shippingport Light Water Breeder Reactor core, and other information and technology to aid in the evaluation of commercial application of the Light Water Breeder Reactor technology. Effort includes development of fuel-element concepts for producing uranium-233 in prebreeder cores that could be installed in existing commercial light-water reactors, as well as prebreeder cores for installation in new plants that are optimized for breeding. Fuel-element support grids that could reduce neutron losses and increase breeding potential for commercial applications are being evaluated. The direction of continuing efforts in later years will depend in large part on the results of current development work.

Status. During FY 1978, selected concepts for advanced water-cooled breeders and prebreeders are being evaluated in conceptual design studies, and development of fuel technology, fuel-fabrication technology, and reactor component concepts is continuing. The results of new developments are being documented and disseminated to industry.

Fuel technology is being advanced by work on fuel systems that optimize the use of fissile materials, by evaluating possible improvements in fuel element concepts, and by testing advanced fuels to establish their value and qualify procedures for design and analysis of fuel elements for both pre-breeders and breeder reactors. Additional irradiation tests are being conducted in the Advanced Test Reactor and in the NRX reactor at Chalk River, Ontario, Canada, including testing of a bundle of long rods containing developmental dual-region pellets. Interim and final postirradiation examinations in the Expanded Core Facility and Bettis hot laboratories provide data on the effects of irradiation, which are analyzed to support performance predictions.

PROGRAM: ADVANCED FISSION SYSTEMS													
SUBPROGRAM: WATER-COOLED BREEDER REACTORS													
CATEGORY: ADVANCED WATER BREEDER APPLICATIONS													
TASK	CY 1977		CY 1978		CY 1979		CY 1980		CY 1981		CY 1982		CY 1983
	FY 1977	FY 1978	FY 1978	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983					
CONCEPTUAL DESIGN OF PREBREEDER CONCEPT		▲											
CONCEPTUAL DESIGN OF IMPROVED PREBREEDER CONCEPT					△								
CONCEPTUAL DESIGN OF ADVANCED BREEDER CONCEPT													

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E. FAST BREEDER REACTORS

1.0 Overview

Incentives. The motivation for the breeder reactor is the prospect of extending the uranium resource base to a practically inexhaustible source of electrical energy. Without the breeder, exhaustion of uranium supply will ultimately foreclose the nuclear fission option. Introduction of the breeder would permit virtually complete utilization of fertile uranium and thorium resources for the generation of electrical power.

When viewed in isolation, the timing of the need for the breeder is governed principally by two factors:

- The magnitude of the uranium resource base
- The future nuclear-electric demand growth

The uranium resource constraint on the ability of the light water reactor to meet future nuclear electric load growth establishes the latest date when a breeder would be needed. There could, however, be commercial advantages in earlier breeder introduction.

History. The United States interest in breeder reactors dates back to the Manhattan Project in the 1940s when the possibility of breeding was first recognized by pioneers in the nuclear field. One of the earliest steps in this program was the construction of an experimental fast reactor called Clementine at the Los Alamos Scientific Laboratory. It was used from 1946 to 1953 to demonstrate the feasibility of operating a reactor with fast neutrons, plutonium fuel and a liquid metal coolant. The Experimental Breeder Reactor I (EBR-I) in Idaho, which produced the first nuclear-generated electric power in 1951, was built and operated by Argonne National Laboratory (ANL) through December 1963 to prove the feasibility of breeding and to demonstrate the technology of liquid metal coolants.

These developments led to the construction in the U.S of two fast reactors in the mid-1960s--the Experimental Breeder Reactor II (EBR-II) in Idaho and the Enrico Fermi Atomic Power Plant in Michigan. The EBR-II was built as a 20-MWe demonstration breeder facility and included the demonstration of onsite spent fuel reprocessing and remote refabrication incorporating both coprocessing and "spiking" (low decontamination factor) concepts for moderately enriched uranium fuel. EBR-II was later converted to a fuel test facility and is currently operating in that capacity. The Enrico Fermi plant was a utility-constructed commercial prototype reactor system. It experienced an accidental partial meltdown of its core in 1968, resumed operation in 1971 and was eventually terminated for economic reasons.

In 1962, the Atomic Energy Commission submitted to the President of the United States a report on the state of the Civilian Nuclear Power Program. This report stated:

"Thus, the future program should include the vigorous development and timely introduction of improved converters and especially of economic breeders; the latter are essential to long-range major use of nuclear energy."

The 1967 Supplement to the 1962 Report to the President set forth the changes that had taken place in the interim, and reported that the work on the Liquid Metal Fast Breeder Reactor (LMFBR) had been given priority status while the effort on alternative-coolant fast breeder reactors was being maintained at a constant level. Thus, much of the development effort on the breeder reactor in this country, as in foreign countries, has been focused on the liquid metal-cooled fast breeder. As an alternative, effort has initiated on the gas-cooled fast breeder reactor in 1963.

Strategy--Program Redirection. In April 1977 the Administration announced a change in the Nation's nuclear energy policy. Concern about the potential for the proliferation of nuclear weapons capability through the deployment of plutonium-based nuclear energy systems has emerged as the overriding issue in considering whether to proceed with the development, demonstration, and eventual deployment of breeder reactor energy systems. Plutonium recycle and the recommercialization of the fast breeder have been deferred indefinitely. Accordingly the fast breeder program has been redirected back to the R&D mode to maintain the breeder as a viable and accessible option for the future, if needed, but without plans or schedules for demonstration plant projects beyond the Fast Flux Test Facility.

Objectives. The Fast Breeder Reactor program is pursuing two breeder concepts: the Liquid Metal Fast Breeder Reactor (LMFBR) and the Gas Cooled Fast Breeder Reactor (GCFR). The interim goals of the breeder reactor program are:

- To maintain and improve technical and engineering bases for selection of a breeder system consistent with U.S. nonproliferation objectives and anticipated national electric energy requirements
- To provide technical support to the Administration's foreign policy initiatives on nuclear nonproliferation

The scope and timing of the fast breeder subprograms are determined by the need to produce information and resolve questions associated with meeting the interim goals of the program by late 1979 or 1980. By that time, the overall fission program studies, in particular, the Nonproliferation Alternative Systems Assessment Program (NASAP) and International Nuclear Fuel Cycle Evaluation Program (INFCE) (see Section IV-B), will have been completed and the National Uranium Resource Estimate (NURE) program will be well underway toward refining the estimates of uranium resources. These studies, along with a continuing breeder R&D program, will permit decisions on the future direction of fission power R&D. In the meantime, the fast breeder R&D program will center around the Fast Flux Test Facility (FFTF), with: (a) the testing of reference fuels and candidate fuels that might offer nonproliferation advantages, and (b) the testing of materials applicable to breeder reactors

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and other high-neutron-flux systems. The FFTF also serves as a baseline for scaling up systems and components.

The technology areas supporting the restructured breeder activity include: fuels, physics, materials and chemistry, reactor core components, engineered systems and components, and safety. Work in these areas has been redirected toward examination of alternative fuel cycles while providing a base for the resumption of demonstration and commercialization of the breeder, should a decision to do so be made in the future.

The following table contains a summary of the operating funding levels by breeder reactor type for FY 1977 through FY 1979.

FAST BREEDER REACTORS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Liquid Metal Fast Breeder Reactor	\$564,526	\$365,805	\$279,701	\$(86,104)
Gas Cooled Fast Breeder Reactor	12,800	14,670	30,500	15,830
TOTAL	\$577,326	\$380,475	\$310,201	\$(70,274)

2.0 Liquid Metal Fast Breeder Reactor

The priority type of fast breeder reactor has historically been the liquid-metal-cooled reactor. Its advantage is largely due to the excellent heat-transfer properties of sodium that enable the LMFBR to operate at high temperature without resort to pressurized systems. The practical result is a plant with high thermal efficiency: 40% for an LMFBR as compared with 32% for light-water reactors.

Figure 14 shows the major heat-transport systems in an LMFBR power plant. The flowing sodium removes the heat produced in the fissioning fuel in the reactor core, transfers the heat to a second sodium piping loop in the heat exchanger, then returns to the reactor. The heat in the secondary sodium loop passes through a steam generator where it turns water into steam that turns the turbine-generator.

The LMFBR can operate on a variety of fuel cycles, determined largely by the choice of fertile material and fissile material in the fuel. The fuel is in the form of pellets that are sealed in tubes to form the fuel elements. The fuel elements are grouped into assemblies for loading into the reactor core.

The LMFBR program, for management purposes, consists of projects and supporting technology development programs.

Figure 14
LMFBR Heat-transport System

The following table shows the operating funding levels by subprogram for the FY 1977 to FY 1979 period.

LIQUID METAL FAST BREEDER REACTOR	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Projects	\$312,955	\$113,792	\$94,599	\$(19,193)
Technology Base	189,893	184,652	18,474	(66,178)
Test Facilities*	61,678	67,361	66,628	(733)
TOTAL	\$564,526	\$365,805	\$279,701	\$(86,104)

*This item is for operating the facilities that conduct tests supporting LMFBR development

ADVANCED FISSION SYSTEMS

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2.1 Projects

This area element is currently comprised of three major activities:

- Fast Flux Test Facility (FFTF)
- Clinch River Breeder Reactor Plant (CRBRP)
- LMFBR Technology Integration Study

The reactor projects provide not only the central focus of the research and development efforts, but also technical information in support of the concept.

The following table shows the funding levels by project for the FY 1977 to FY 1979 period.

PROJECTS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
PROJECT	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Fast Flux Test Facility	\$75,355	\$80,792	\$66,199	\$(14,593)
Clinch River Breeder Reactor	237,600	33,000	13,400	(19,600)
Technology Integration Study	0	0	15,000	15,000
TOTAL	\$312,955	\$113,792	\$94,599	\$(19,193)

2.1.1 Fast Flux Test Facility

A high-priority part of the Fast Breeder Reactor Program is the completion of the Fast Flux Test Facility and its subsequent operation. FFTF will serve as an advanced fuels and materials irradiation facility for the Fast Breeder Reactor Program and as a baseline for scaling up systems and components. It has the capability to serve as an advanced fast-flux test bed for alternative fuels and materials experiments needed to continue DOE advanced reactor development programs.

The FFTF is a 400 MWt liquid metal (sodium) cooled fast-neutron-flux reactor designed specifically for irradiation testing of nuclear reactor fuels and materials for liquid metal fast breeder reactors (LMFBRs) (see Figure 15). It is being constructed at the Hanford Engineering Development Laboratory in Richland, Washington. The FFTF will play a key role in development and testing activities. It will provide a test bed for demonstrating and evaluating the performance of future LMFBR plant fuel assemblies and core designs at reference conditions:

- It will be used to test fuel elements up to and including failure under dynamic sodium-fuel conditions to establish ultimate capability and failure modes; and understanding of failure modes is essential to establish LMFBR core safety, reliability, performance, and lifetime.
- It will be used in tests to develop the advanced fuels (oxide, carbide, and nitride) and the advanced cladding and duct materials essential to attaining operational breeding ratios in the 1.25 to 1.3 range with fuel-doubling times of less than 15 years.
- It will serve as a prototype irradiation test bed for different fast reactor fuel and blanket materials considered in the non-proliferation fuel cycle studies.
- Although FFTF, as a testing reactor, was not designed to breed or to produce electricity, it has already provided and will continue to provide valuable information to follow-on LMFBR projects and base-technology programs in the areas of:

Figure 15
Fast Flux Test Facility (FFTF)

- Plant system and component design
 - Component fabrication
 - Prototype testing
 - Site construction
- In addition, FFTF will provide experience in the operation of a large LMFBR having coolant loops and components at temperatures and coolant flows typical of large LMFBR power plants.

For controlled and instrumented fast-flux conditions for testing fuel specimens, rods, subassemblies, and cladding, as well as reactor structural materials, eight instrumented test positions are being provided in the design of the initial FFTF cores; as presently envisioned, two of these eight positions are to be closed loops; six will be open loops.

In closed test loops, separate components inserted in the reactor core, the coolant, instrumentation, and heat transfer are completely separated from the main FFTF core permitting testing of fuels and materials over a wide range of temperatures in a controlled environment independent of the main reactor coolant system. The open-loop test positions, integral components of the reactor core, are cooled by the reactor primary coolant

system; they are for testing large quantities of candidate fuel pins and assemblies.

DOE's Hanford Engineering Development Laboratory is overall project manager and will be the operator when the facility is completed. The Westinghouse Advanced Reactor Division is reactor-plant designer and the plant architect-engineer and construction manager is Bechtel Corporation.

History. The FFTF schedule history is as follows:

- 1963: AEC authorized Hanford Laboratories to scope a Fast Test Reactor (FTR).
- 1966: Architect-engineering was authorized.
- 1967: Full project was authorized.
- 1969: Preliminary design of the Fast Flux Test Facility was initiated.
- 1970: Westinghouse was selected as AEC prime contractor to operate the Hanford Engineering Development Laboratory (HEDL) and manage FFTF project.
- 1972: AEC issued Environmental Impact Statement on LMFBR demonstration plant; FFTF containment vessel was successfully pressure tested; FFTF reactor vessel was received at site.
- 1974: The 122-ton reactor guard vessel, which will house the FFTF's reactor vessel and core, was installed in the containment.
- 1976: FFTF Final Safety Analysis Report was submitted to the Nuclear Regulatory Commission (NRC); FFTF Title-II design was completed and delivered to NRC.
- 1977: Pressure test in reactor Heat Transport System; prototype pump testing was completed; 90% of construction was completed.

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2.1.2 Clinch River Breeder Reactor

The National Energy Plan of April 29, 1977 proposed to cancel construction of the CRBR Project and all component construction, licensing, and commercialization efforts. Implementation of the plan is pending final resolution by the Administration and Congress.

Status. Project activities are currently continuing on a reasonable and prudent basis pending consideration of the project's future by the Administration and Congress. A summary of the status of project activities as of December 1977 is as follows:

- The engineering activities of research and development and design are approximately 60% complete.
- Approximately 75% of total equipment requirements, based on estimated cost, are on order; partial payments totalling approximately 25% of the total estimated requirements have been made.
- Approximately \$462 million had been invested through December 31, 1977, including approximately \$351 million of DOE funding and \$111 million of utilities and other participants funding.
- The Final Environmental Statement, the Site Suitability Report and Design Criteria issuance have been obtained from the Nuclear Regulatory Commission.
- Licensing proceedings necessary to securing the Limited Work Authorization were suspended on April 15, 1977.

The action required to implement the proposed revised objectives, i.e., cancel construction of the CRBR Project and all component construction, licensing and commercialization efforts, except for systems design, are described below:

- The construction activities and contract will be terminated.
- All contracts for equipment will be terminated or will be completed if that is in the best interest of the Government (e.g., lower cost).
- Licensing efforts will be terminated.
- To the extent the reactor manufacturers' and architect-engineers' services are not required to complete the systems design and termination activities, these contracts will be closed out.
- Financial, legal, and managerial activities will be conducted as required to settle claims, to negotiate open change orders, to inventory materials, work-in-progress, and complete equipment, and to complete other actions required as a result of the termination activities.
- Personnel not essential to conducting these termination activities or the design activities described below will no longer be supported by the project; personnel termination and relocation costs will be paid in accordance with contractual provisions.
- The completion of the design effort will cover such areas as: (1) identification and resolution of outstanding design issues on selected major systems, (2) evaluation of existing designs and the development of required changes to accommodate the most propitious alternate fuel systems, and (3) other specific design activities required to bring the CRBRP design to an orderly conclusion.
- Appropriate negotiations will be undertaken with the other project participants (PMC, TVA, and CE).

Implementation of the policy to cancel CRBRP except for systems design is pending final resolution by the Administration and Congress. In view of Congressional consideration of the proposed new objectives, project activities are continuing based on the project objectives as authorized prior to the Administration's redirection, although on a reasonable and prudent basis. Environmental hearings were suspended following the establishment of the Administration's proposed new objectives because there was no practical way to proceed with public hearings due to the uncertain future of the project.

The CRBR Project is presently authorized under Public Law 91-273 as amended. Under that law, Congress authorized the AEC (now DOE) to enter into a definitive cooperative arrangement for an LMFBR Cooperative Demonstration Plant and provide financial assistance. That cooperative arrangement, called the Quadripartite Contract, has been in effect since

1973 between DOE, Project Management Corporation (PMC), Tennessee Valley Authority (TVA), and Commonwealth Edison (CE).

The Federal role in the CRBR Project consists of both overall project management and major financial support. DOE financial support to the project is applied to all elements of the project. DOE was to provide all project funding in excess of the utilities' contributions of approximately \$260 million and the reactor manufacturers' contributions of approximately \$10 million.

The utilities were to provide both financial support and technical assistance to the CRBR Project. Some 753 public, private, cooperative, and municipal electric systems pledged contributions of approximately \$260-million to the project. These funds were to be used with the government funds to meet any project expense. The Commonwealth Edison Company (CE) and the TVA were the principal organizations that represented the utility interest through the PMC and the Breeder Reactor Corporation (BRC).

The CRBR Project is managed by the CRBR Project Office under DOE's Program Director for Nuclear Energy. The CRBR Project Office is staffed by DOE and utility personnel and is located in Oak Ridge, Tennessee, near the plant site. Contract administration is performed by the Project Office with some administrative support from the DOE Oak Ridge Operations Office. The utility financial contributions to the project and the utility personnel assigned to the project are administered by the Project Management Corporation under the Quadripartite Contract.

The three prime contractors performing the project work are Westinghouse as the lead reactor manufacturer, Burns and Roe as architect-engineer, and Stone and Webster as constructor. General Electric and Atomics International are major subcontractors to Westinghouse.

2.1.3 Technology Integration Study

With the Administration committed to a strong research and development program for advanced nuclear technologies, including base-program research on the liquid metal fast breeder, it is essential that there be a means for focusing and giving direction to the base-technology R&D program and to establish and integrate its outputs and schedules. Thus a reactor-plant conceptual design study, cognizant of new nonproliferation criteria, will be conducted to identify, define, and integrate the R&D needed in the base-technology programs to support the reactor-plant design.

The study, will--along with the Nonproliferation Alternative Systems Assessment Program (NASAP), the International Nuclear Fuel Cycle Evaluation (INFCE), and the National Uranium Resource Evaluation (NURE)--provide essential information to the fission energy development and design process in the next few years. In addition to the focusing aspects of the study, there will result a conceptual design for the development of an LMFBR that could be built and operated on an electric

utility system when the need for a breeder and United States nuclear energy policies are further defined. In the study, basic criteria for the systems will be identified in terms of requirements for:

- Safety and licensing
- Environmental acceptability
- Plant safeguards and controls
- Utility acceptance (cost, reliability, availability, and maintainability)
- Development and testing of components and systems

To achieve the objective of the study, the effort will involve the participation of the LMFBR community to produce information on:

- An integrated conceptual design of the plant
- Safety criteria
- Identification of R&D requirements
- Cost and benefits of using technology from other countries
- Capital-cost estimates
- Construction schedule estimates

The study will build on the foundation of past reactor-plant projects and design studies; emphasis will be on plant design simplicity and practicality, environmental acceptability, adequate breeder performance, reduced costs, and shortened plant design and construction schedules.

Objectives. The objective of this plant design study, which will be carried out by several reactor manufacturers and architect-engineers, is: (a) to provide an updated plant design with nonproliferation and safeguards characteristics to serve as a focus for structuring and pacing the Fast Breeder Reactor Program base-technology research and development program, and (b) to provide an advance in breeder reactor technology by integrating into the design specific component and systems design improvements that have evolved in the Clinch River Breeder Reactor Project and other LMFBR projects. A plant design will be provided in sufficient detail to permit a subsequent decision on a project to design, manufacture, construct, and operate this type of plant.

ADVANCED FISSION SYSTEMS

PROGRAM ADVANCED FISSION SYSTEMS SUBPROGRAM. FAST BREEDER REACTORS CATEGORY LIQUID METAL FAST BREEDER REACTOR SUBCATEGORY TECHNOLOGY INTEGRATION STUDY																															
TASK				CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
				FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
INITIATE STUDY								△																							
PHASE I SYSTEM CONCEPTUAL DESIGN									▽																						

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2.2 Technology Base

The breeder reactor base-technology program will continue so as to maintain the breeder power reactor option. These efforts will include studies on alternative fuel technologies so as to investigate, particularly, their nonproliferation characteristics.

Work on technology base involves the following areas:

- Reactor Core Design and Analysis
- Plant and Component Engineering
- Fuels and Absorber Systems
- Materials and Chemistry
- Safety

The following table shows the funding levels by task for the FY 1977 to FY 1979 period.

TECHNOLOGY BASE	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Reactor Core Design and Analysis	\$22,569	\$28,563	\$17,492	\$(11,071)
Plant and Component Engineering	66,091	52,786	32,330	(20,456)
Fuels and Absorber Systems	43,428	43,970	29,197	(14,773)
Materials and Chemistry	10,806	8,400	7,105	(1,295)
Safety	46,999	50,933	32,350	(18,583)
TOTAL	\$189,893	\$184,652	\$118,474	\$(66,178)

2.2.1 Reactor Core Design and Analysis

Work in this area supports the development and design of reactor systems through performance testing of reactor components and the development of design data. The work is carried out in five general work areas:

- Core Design and Analyses
- Reactor and Plant Instrumentation and Control
- FFTF Test-article Assembly
- Fuel Transportation and Protection
- Reactor Physics

With the first four of these work elements, new conceptual designs and design criteria, mechanics and assembly design methods, and assembly design and testing for advanced reactor cores are conducted, complemented by core-wide performance evaluation of Fast Flux Test Facility (FFTF) reference fuel assemblies. Instrumentation and control components and systems for FFTF and future FBRs are developed and tested, and shipping casks for fuel-assembly and test-fuel shipment are developed. Both the national laboratories and industrial research and development facilities are used on a coordinated and cooperative basis to bring maximum U.S. expertise to bear on problems, and to assure the availability of design options on which an informed United States industry can base the future development of the fast breeder reactor.

The reactor physics work includes critical experiments to verify the accuracy of core designs and measurements of nuclear data to permit accurate reactor design and fuel cycle evaluations.

Participants. For Core Design and Analyses, General Electric Co. (GE), Atomics International (AI), Westinghouse Electric Corp. (W), Combustion Engineering (CE), Argonne National Laboratory (ANL), Hanford Engineering Development Laboratory (HEDL), and Oak Ridge National Laboratory (ORNL) are conducting core design studies and developing proliferation-resistant conceptual designs using alternate fuel systems to provide a basis for programmatic decisions on core component and fuel and material development. Preconceptual designs will be completed by the end of 1978 and will serve as a basis for selecting the two most promising fuel systems for full conceptual design.

- Thermal-hydraulics codes in support of component and assembly design are being developed at Pacific Northwest Laboratory (PNL) and ANL; supporting mixing-test data are being supplied from test facilities at ANL and the Massachusetts Institute of Technology (MIT).
- Gridded fuel assemblies, which serve as a backup to the FFTF reference wire-wrapped assemblies, are being designed by GE and W for irradiation testing in FFTF. Blanket assemblies for

testing in FFTF are being developed by W for oxide fuel and by CE for carbide fuel. HEDL is designing and developing an advanced control assembly, which will extend control-assembly lifetime to 3 years, for testing in FFTF.

- HEDL, with computer-code support from GE and AI, is developing the methodology for evaluating core-wide performance of FFTF reference fuel.
- GE and ANL are also developing core-restraint design codes for designing optimized configurations for core internals. Design codes are checked experimentally in a full-core restraint facility at Westinghouse. Structural design criteria for use in reactor component design are being developed cooperatively under GE leadership.

For Reactor and Plant Instrumentation and Control, ORNL, HEDL, ANL, AI, and the Liquid Metal Engineering Center (LMEC) are developing and testing neutron-sensor systems and process instruments (flow, pressure, temperature, level sensors, and sodium leak detectors) in support of FFTF and in the longer term for large breeder reactor plants. HEDL is fabricating an ultrasonic under-sodium viewer for use at FFTF during fuel handling and maintenance and will extend this technology to large plants. With the ultrasonic viewer it is possible to "see" through opaque liquid sodium (figure 16).

Figure 16
Scanning Head of FFTF Under-sodium Viewer

For FFTF Test-article Assembly, HEDL is establishing fabrication facilities and processes for assembling instrumented and noninstrumented and fueled and nonfueled test assemblies for irradiation in FFTF. Twelve types of test articles will be inserted during startup tests beginning in early 1979; eight additional types will be inserted during the first full cycle of operation that begins in early 1980 (figure 17).

Figure 17
Typical FFTF Test Article

Du Pont, supported by HEDL, is developing and procuring prototype shipping containers for shipping fresh fuel to FFTF in 1983. HEDL is responsible for obtaining licensed truck-type casks for shipping irradiated test fuel pins or assemblies from FFTF by 1980 to interim offsite examination facilities. Sandia and Du Pont are developing two different designs of licensed prototype rail casks for offsite shipment of spent FFTF driver fuel (figure 18). GE, supported by HEDL, is also studying the preirradiation of fresh plutonium-based fuel to provide proliferation and diversion resistance. By briefly irradiating freshly fabricated fuel in a specially designed reactor, the fuel is made sufficiently radioactive that the handling and/or reprocessing of such preirradiated fuel requires special equipment comparable to that required for reprocessing of Light Water Reactor fuel.

For Reactor Physics, the program elements include integral experiments, nuclear data, nuclear design methods, and materials dosimetry.

Figure 18
Irradiated-fuel Cask on Rail Car

- Integral experiments involve critical experiments performed in very low power ("zero power") experimental critical reactor assemblies (figure 19) to obtain reference-design-related data such as fuel loading, control worth, power peaks, fuel conversion, and reactivity coefficients. These experiments are performed at ANL, which has the major laboratory capability for United States critical experiments supportive of FBRs. To ensure efficient technology transfer and real-time interaction with design activities in United States critical experiments supportive of FBRs. To ensure efficient technology transfer and real-time interaction with design activities in United States industrial organizations, the planning of experiments and the analyses of their results is shared by the major industrial and laboratory participants in the physics program.
- Nuclear data, such as energy-dependent neutron cross sections and radioactivity properties, are measured, primarily using particle accelerators at ANL and ORNL.
- Nuclear design methods involve using computer codes to analytically determine parameters such as those measured in the critical experiments as well as fuel-depletion information for all FBR reactor designs. Development of computer codes for nuclear design is carried out mainly within national laboratories with access to government-owned computers. Work conducted at ANL, LASL, and ORNL takes advantage of special expertise at each site and services to maintain a broad overall perspective.

Figure 19
Zero Power Plutonium Reactor (ZPPR)

- Materials-dosimetry work develops and maintains state-of-the-art radiation measurement techniques for accurately measuring the radiation doses of fuels and materials irradiation tests.

PROGRAM ADVANCED FISSION SYSTEMS SUBPROGRAM FAST BREEDER REACTORS CATEGORY LIQUID METAL FAST BREEDER REACTOR SUBCATEGORY REACTOR CORE DESIGN AND ANALYSIS																							
TASK		CY 1977			CY 1978			CY 1979			CY 1980			CY 1981			CY 1982			CY 1983			
		FY 1977			FY 1978			FY 1979			FY 1980			FY 1981			FY 1982			FY 1983			
CORE DESIGN																							
FABRICATION OF REFERENCE URANIUM OXIDE AND THORIUM OXIDE FUEL RODS								▽															
PRECONCEPTUAL PROLIFERATION RESISTANT CORE DESIGN STUDIES UTILIZING ALTERNATIVE FUELS						▽																	
DEVELOPMENT OF ANALYSIS METHODS FOR DETERMINING SEISMIC RESPONSE OF CORE INTERNALS										▽													
REACTOR AND PLANT INSTRUMENTATION AND CONTROL																							
SODIUM TESTS ON FFTF UNDER SODIUM VIEWING SYSTEM				▽																			
HIGH TEMPERATURE NEUTRON DETECTION TECHNOLOGY AVAILABLE									▽														
FUEL TRANSPORTATION AND PROTECTION																							
FFTF TEST FUEL CASK DESIGN						▽																	
FFTF TEST FUEL CASK DELIVERED										▽													
REACTOR PHYSICS																							
SENSITIVITY ANALYSIS OF MEASUREMENTS REQUIRED FOR NUCLEAR DESIGN OF ALTERNATE FUEL SYSTEMS						▽																	
MEASUREMENTS OF PRINCIPAL THORIUM AND U 233 RELATED DATA NEEDS										▽													

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2.2.2 Plant and Component Engineering

Liquid-metal systems and components are being developed to the point where they have a demonstrated capability to meet the safety, reliability, economic, and performance requirements of breeder reactor plants. The work is organized in three elements:

- Systems Improvements, which simplify and improve the design, reliability, and constructability of the principal nuclear steam supply systems
- Component Development, which develops and proof-tests models of major components before they are committed as plant equipment
- Materials and Structures Engineering, which defines component structural adequacy and determines the effect of fabrication methods and manufacturing processes on materials properties

Major emphasis is on steam generators and main coolant pumps.

In Systems Improvement, the principal liquid-metal systems that experience indicates would be most fruitful for further development have been identified and evaluated as to the potential benefits, the development

required to achieve the benefits, and the cost and risk involved in developing major large sodium components that meet system requirements. The engineering evaluations and analyses are conducted by reactor manufacturers that have the predominant liquid-metal systems experience so as to take advantage of that experience and previous trade-off studies and evaluations.

Atomics International is evaluating:

- Methods for alleviating primary-coolant temperature limitations
- Methods for mitigating the effects of plant thermal transients
- Innovative improvements in the reactor-cover system
- Improved means of decay-heat removal
- The comparative attributes of alternative steam generator designs and steam-cycle conditions

General Electric Co. is evaluating:

- Improvements to the maintenance and inspection capabilities of the major heat-transport systems
- The development of improved fuel storage and shipping concepts
- The establishment of systems requirements for a large steam generator under saturated cycle conditions

Westinghouse Electric Corp. is evaluating:

- Improvements to the heat-transport and steam generator systems
- Improvements to the auxiliary liquid-metal systems and associated support systems
- Systems requirements for large sodium steam generators and primary-system and intermediate-system coolant pumps under superheated cycle conditions

In Component Development, to provide the technology for reliable and economic equipment, models of critical components applicable to future breeder reactors are designed, fabricated, and proof-tested, and reliable operation and economic feasibility are demonstrated. Experience with demonstration breeder reactor plants in this country (Fermi) and abroad (PFR, Phenix, BN-350) has shown that the heat-transport components have been the major source of plant operational deficiencies, so the development emphasis is on long-lead, high-risk, and unproven but essential components such as steam generators and main coolant pumps. Component vendors design and fabricate model components and perform the concept-dependent supporting development. Tests and generic-type development are conducted by the national laboratories.

ADVANCED FISSION SYSTEMS

The most important item to be developed is the steam generator because no reliable large steam generator has been demonstrated in the United States or abroad. The planned steam generator development in the near term includes:

- Preliminary design of the model steam generators
- Thermal-performance studies and analyses
- Development of a chemical leak-detection method
- Preparations for a second series of large-leak tests

Participants include Argonne National Laboratory (ANL), Atomics International (AI), Babcock & Wilcox Co., General Electric Co., and Westinghouse Electric Cor. A portion of the Atomics International Hockey Stick Steam Generator is shown in figure 20.

Figure 20
Hockey-stick Steam Generator

The near-term plans for pump development include:

- Completing the preliminary design of model primary-system and intermediate-system pumps by AI, Byron Jackson, and Westinghouse
- Sodium tests at the Liquid Metal Engineering Center (LMEC) of the subscale inducer pump developed by AI
- Testing at LMEC of the intermediate-size inducer pump developed by AI and Hanford Engineering Development Laboratory (HEDL)

- The design of a double-suction primary-system pump by Byron Jackson

Figure 21 shows a FFTF liquid-metal pump under inspection.

Figure 21
FFTF Liquid-metal Pump

In the near-term development of other components:

- AI will determine the effects of sodium carryover on fuel handling devices.
- Foster Wheeler, Combustion Engineering, and Westinghouse will complete a computer code for predicting the performance of a straight-tube intermediate heat exchanger.
- ANL will continue to improve the understanding of flow-induced vibration.
- ANL will continue to develop flow distribution and mixing technology and to develop advanced mixing components.
- Westinghouse will continue to develop the technology of natural circulation.

ADVANCED FISSION SYSTEMS

In Materials and Structures Engineering, structural and seismic design methods and criteria are being developed and codified to establish the technology needed to demonstrate structural adequacy and to assure safe and reliable operation of components on a practicable and verified basis. Manufacturing processes will be developed for economically producing plant components such as large-diameter pipe and fittings, flexible pipe joints, and welded pipe joints, and authoritative materials-properties data for use by designers will be obtained for steam generator materials. The Nuclear Systems Materials Handbook will be updated and expanded. For these tasks:

- Oak Ridge National Laboratory develops high-temperature structural design methods and criteria and the technology of large-diameter pipe and fittings and dissimilar-metal welded joints.
- Atomic International develops flexible pipe joints and high-temperature piping design methods.
- The Westinghouse Advanced Reactors Division addresses the validation and application of high-temperature structural design methods and criteria.
- General Electric is qualifying steam generator materials and developing dissimilar-metal welded pipe joints.
- Agbabian Associates and HEDL are developing seismic analysis methods and criteria.
- HEDL is managing the Nuclear Systems Materials Handbook.

PROGRAM ADVANCED FISSION SYSTEMS SUBPROGRAM FAST BREEDER REACTORS CATEGORY LIQUID METAL FAST BREEDER REACTOR SUBCATEGORY PLANT AND COMPONENT ENGINEERING																					
TASK	CY 1977			CY 1978			CY 1979			CY 1980			CY 1981			CY 1982			CY 1983		
	FY 1977	FY 1977	FY 1977	FY 1978	FY 1978	FY 1978	FY 1979	FY 1979	FY 1979	FY 1980	FY 1980	FY 1980	FY 1981	FY 1981	FY 1981	FY 1982	FY 1982	FY 1982	FY 1983	FY 1983	
STEAM GENERATOR																					
LARGE SODIUM/WATER REACTION TEST OF MODULAR STEAM GENERATOR					▽																
PUMP DEVELOPMENT																					
PHASE II PROPOSALS FOR LIQUID METAL PUMP DEVELOPMENT					● 2																
2000 HOUR SODIUM PERFORMANCE OF SUBSCALE INDUCER PUMP					▽																
MATERIALS AND STRUCTURES																					
ELEVATED TEMPERATURE TENSILE TESTS AND RELAXATION TESTS ON COMMERCIAL HEAT TREATED ALLOY 718 MATERIAL					▽																

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2.2.3 Fuels and Absorber Systems

The work in this part of the program identifies and develops breeder reactor fuel, blanket and control absorber elements, and demonstrates their performance capabilities under steady-state and design-transient conditions. The work includes fuel element fabrication processes, development of analytical models, and verification of results through various levels of irradiation testing.

The development work is organized into three elements:

- Reference fuels, absorber and materials
- Advanced and alternate fuels and materials
- Fuel fabrication processes

Objectives. In the development of reference fuels, absorbers, cladding and duct materials, the major objectives are to: (a) obtain statistically reliable fuel and absorber pin steady-state and off-normal transient performance data to allow the reliable operation of the FFTF, (b) determine the effects of irradiation on reference-fuel-pin-cladding and subassembly-duct materials, (c) determine the mode and consequences of cladding breach and the advisability of continuing element operation beyond cladding breach, and (d) define the parameters limiting performance of fuel and absorber components as a basis for innovative improvements and design changes.

In the development of advanced and alternative fuel, cladding, and duct materials, the major objectives are to: (a) determine the steady-state and transient irradiation performance, endurance capability, and cladding breach modes of alternative breeder fuels and fuel pins with advanced cladding and duct materials so as to form a basis for the development of alternative fuel cycles, and (b) provide, by late FY 1979, an adequate data base (from EBR-II, TREAT, and fuel properties studies) for selection of a carbide or nitride fuel concept for detailed design and development.

In the development of fuel fabrication processes, the major objectives are to: (a) develop fabrication technology for fuel manufacture in support of the domestic and international initiatives on the use of proliferation-resistant alternate breeder fuels, and (b) provide an assured source of fuel for FFTF.

Strategy. The strategy of this reference materials works, in summary, is to complete the on-going near prototypic testing in EBR-II (figure 22) by late 1980 while preparing to use FFTF in early 1980 as a prototypic test facility for fuel and control elements. The transient test reactor (TREAT) will continue to be used during the remainder of the EBR-II phase and into the FFTF phase. Activities include: (1) engineering design, fabrication, irradiation, examination, performance analysis, and correlation of EBR-II, FFTF, and TREAT test results for FFTF reference and improved fuel and absorber designs; (2) tests on nonfueled cladding/duct material in EBR-II and FFTF; and (3) ex-reactor material-property determinations to support the irradiation testing and analysis.

Figure 22
Top of EBR-II Reactor

The strategy of the advanced and alternative materials work, in summary, is to design and fabricate fuel pins and cladding and duct materials for irradiation tests to evaluate performance in EBR-II, TREAT, and FFTF, and to analyze the results of data provided from hot cell examinations. Irradiations in EBR-II will continue through the end of 1978, after which new irradiation tests will be prepared for FFTF. FFTF tests and related TREAT tests will focus on those candidates with the best demonstrated performance.

The strategy of this fuel-fabrication process development work is to design, develop, test, and evaluate fuel-fabrication equipment and processes in existing facilities on a unit or limited-integrated process operation. As fuel-system decisions become clarified, this development work will increasingly concentrate on remote operation and remote maintenance for alternate fuel.

The reference materials work is broken down into three activities:

- The development of reference fuel and blanket elements:
 - Steady-state tests in EBR-II and power transient tests in TREAT have demonstrated satisfactory performance of the reference fuel design (see figures 23-25); 2000 reference mixed oxide fuel pins have been irradiated in EBR-II and 33

Figure 23
FFTF Fuel System

transient tests have been conducted in TREAT. Burnups achieved on reference type fuel pins in EBR-II have exceeded the design goal burnup for FFTF; 12 cladding breaches have occurred in a benign fashion with no effect on neighboring pins.

- FFTF tests of reference fuel in a full-size system under actual FBR conditions will confirm and refine the performance correlations developed from the EBR-II and TREAT tests; surveillance will provide extensive fuel performance data; and tests of the ability of the fuel to continue operation following a breach of the cladding will be conducted in a Closed Loop In-Reactor Assembly (CLIRA).
- Performance codes are being developed to improve the understanding of and the ability to predict thermal/hydraulic and structural behavior (for both steady-state and transient conditions) of fuel, blanket, and absorber elements and cladding/duct materials. Verification and calibration with FFTF test data will allow these codes to be used for predicting and evaluating the performance of entire reactor systems.
- The postirradiation examination of EBR-II test-size fuel

Figure 24
FFTF Driver Fuel Assembly

Figure 25
FFTF Driver Fuel Pin

pins is being conducted at Hanford Engineering Development Laboratory (HEDL), Los Alamos Scientific Laboratory (LASL), and Argonne National Laboratory-West (ANL-W). To support the tests in FFTF, a new Fuels and Materials Examination Facility (FMEF) is being built to handle full-size fuel elements and assemblies.

- The characterization of cladding and duct materials:
 - EBR-II irradiation testing of 20% cold worked Type 316 stainless steel indicates satisfactory behavior characteristics for the early operation of FFTF, but not for higher-exposure cores or for advanced proliferation resistant fuels. The primary properties of concern have been shown to be swelling, irradiation creep, and stress rupture. FFTF tests will be required to confirm EBR-II results under prototypic conditions and to further characterize these properties. Plans call for tests to be accomplished with a materials test assembly in FFTF's first fuel loading and with another such assembly in the second loading.
 - Additional issues regarding the reference cladding and duct materials that will be addressed in tests in FFTF include: (1) resolution of the swelling-irradiation creep interaction issue, (2) confirmation of EBR-II results in FFTF, (3) effects of temperature changes on swelling and mechanical

properties, (4) chemistry and metallurgical effects on 20% cold worked Type 316 stainless steel, and (5) development of mechanical damage accumulation rules.

- The development of absorber control pins:
 - This includes irradiation tests of prototypic boron-carbide pellets and pins, chemical and physical property characterization, performance code development, and control rod fabrication development.
 - EBR-II tests of reference FFTF absorber pins and advanced, vented concepts have achieved burnup levels equivalent to two years of FFTF operation; the data have been incorporated in performance prediction codes.
 - Analysis and test results indicate that economically attractive advanced control pins can be developed, and work toward that goal is underway.

The advanced and alternative materials work is broken down into two activities:

- The development of advanced and alternative fuel and blanket pins:
 - Eleven uranium-plutonium carbide and nitride subassemblies have been fabricated and are being irradiation tested in EBR-II. The maximum burnups attained are approximately equal to the near-term goal for FFTF driver fuel for both sodium bonded and helium bonded pins. Six more carbide/nitride subassemblies will be inserted in EBR-II in FY 1978. Four power transient irradiations of the same types of carbide/nitride fuel elements have been completed in TREAT. Three alternative oxide tests specimens are being designed and fabricated for insertion in EBR-II in 1978. Plans are underway to transient test selected pins in TREAT after their irradiation in EBR-II.
 - Plans are proceeding for: (1) completing design and fabrication of two to four equivalent full-size alternative-fuel oxide subassemblies for test in FFTF, (2) completing design and fabrication of two subassemblies partially fueled with advanced carbide/nitride pins for test in FFTF, (3) continuing irradiation of the initial alternative fuel tests in EBR-II to over one-third of the near-term FFTF goal burnup, and (4) designing and fabricating equivalent full-size carbide and oxide assemblies for test in FFTF. Design of special FFTF test hardware to accommodate the special needs of these fuels will be complete in 1978. Plans are also underway to transient test these FFTF-irradiated fuels in TREAT.
 - The codes being developed and calibrated to describe the performance of the reference fuel system will be applicable

to the advanced and alternative fuels systems, with appropriate consideration for the differing properties of the advanced and alternative fuels. Performance correlations will undoubtedly differ from those established for the reference system, but the basic codes are expected to be similar.

- The development of advanced cladding and duct alloys:
 - By means of irradiations of nonfueled specimens in EBR-II and simulation tests in charged particle bombardment facilities, a large number of initial candidate alloys have been screened and 10 alloys are being investigated intensively.
 - Several prime candidate advanced alloys will be selected in March 1978 and thoroughly tested in FFTF materials-test-assembly irradiations and in a wide variety of fuel-elements and duct-component tests with different advanced and alternative types of fuel. Based on the results of these tests, primary alloys will be selected as appropriate for candidate advanced and alternative driver- and blanket-fuel applications in the 1982-1983 period, at which time the test program will be redirected to provide a comprehensive design data base for the primary alloys.

The fuel fabrication process work is divided into two activities:

- The development of fabrication equipment and processes:
 - The equipment development effort will concentrate on fabrication and inspection equipment, close-coupled analysis, wet scrap recycle, waste management, and safeguards. Fabrication process equipment development is divided into three major areas: ceramic line, pin line, and assembly line.
 - The scope of development areas will vary depending on the fuel form and degree of remote operations required. For particulate fuels, major efforts will be devoted to moving the current lab scale operations into the prototype unit operation demonstration stage as rapidly as practicable as part of a coordinated program with the LWR Program. The program for other fuel forms would require an initial evaluation of applicability of the process (e.g., injection casting for metal fuels) to remote, high throughput operations.
 - The breeder fuel fabrication development efforts will be conducted on a broad front through 1980 to encompass work on candidate fuel types and forms. However, in light of the extensive data base on FBR pellet fuel performance, major work will be on pellet fuels with increasing emphasis on particulate and metal fuels as they are able to contribute to nonproliferation objectives and to minimizing the

problems associated with remote fabrication. The NASAP and INFCE evaluations, as they progress, will assist in focusing on the fuel system(s) for which fabrication development is to be carried out.

- The HPFL is an approved FY 1977 construction project (77-4-c). The HPFL is a pilot-scale fuels development laboratory with supporting facilities for developing and demonstrating integrated fuel manufacturing processes, equipment, and systems for handling materials in a safe, reliable, and economic manner. It was scheduled for operation in the 1982-83 period. The HPFL is currently zero funded for FY 1979 and if not reinstated the Title II design in progress will be terminated in an orderly manner so as to permit reinstatement of the project with minimum perturbation.

● Providing fuel for FFTF

- FBR fuel will continue to be procured from industrial fabricators so long as economic sources are available. Procurements will be planned so as to match industrial production capacities to plant usage and to provide sufficient spares to accommodate abnormal demands or delays in supply.

PROGRAM: ADVANCED FISSION SYSTEMS SUBPROGRAM: FAST BREEDER REACTORS CATEGORY: LIQUID METAL FAST BREEDER REACTOR SUBCATEGORY: FUELS AND ABSORBER SYSTEMS																							
TASK	CY 1977			CY 1978			CY 1979			CY 1980			CY 1981			CY 1982			CY 1983				
	FY 1977	FY 1977	FY 1977	FY 1978	FY 1978	FY 1978	FY 1979	FY 1979	FY 1979	FY 1980	FY 1980	FY 1980	FY 1981	FY 1981	FY 1981	FY 1982	FY 1982	FY 1982	FY 1983	FY 1983	FY 1983		
REFERENCE FUEL ABSORBERS AND CLADDING/DUCT DEVELOPMENT																							
FIRST OXIDE FUEL ELEMENT WITH ADVANCED CLADDING AT GOAL EXPOSURE (EBR-II)						▽																	
DRIVER FUEL AT GOAL EXPOSURE (FFTF)														▽									
ADVANCED/ALTERNATIVE FUEL AND CLADDING/DUCT DEVELOPMENT																							
SHIP LAST TEST ASSEMBLY TO EBR II FOR SCREENING OF ADVANCED FUEL DESIGNS		▼																					
DESIGN OF FIRST OXIDE AND CARBIDE ALTERNATIVE FUEL TEST FOR EBR II		▲																					
SELECTION OF SIX CANDIDATE ADVANCED CLADDING AND DUCT ALLOYS					▽																		
IRRADIATION OF PHASE I ADVANCED CARBIDE/NITRIDE ASSEMBLIES IN EBR II							▽																
FUEL FABRICATION PROCESSES																							
ROTARY PRESS DELIVERED						▽																	
PROCUREMENT OF FFTF CORE 5									△														

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2.2.4 Materials and Chemistry

Materials and chemistry technologies are being developed to support breeder reactor development. Although directed essentially at technology development outside the reactor, the work also provides support for the development of alternate fuel technologies. The materials task includes establishing operational limits on current component materials and the developing advanced structural alloys, hardfacing alloys, and cover-gas seal materials. The chemistry task develops methods for monitoring and controlling both radioactive and nonradioactive impurities in sodium and the reactor cover gas. Also included are the process developments needed to support such maintenance operations as sodium removal, radioactivity decontamination, and component requalification after sodium removal and decontamination.

An adequate materials and chemistry base technology, which is required to achieve safe, economical, and reliable design and operation of FBRs, provides plant designers the data needed for design and safety analysis. It also provides development information that can be used to select optimum materials and operational parameters for advanced designs. As the FBR option matures, this development work will provide improved materials and chemistry technology beyond the current state-of-the-art, thereby helping to improve the overall economics of the FBR option.

Objectives. The specific objectives of this work are to:

- Provide the operational limits of materials for near-term FBR applications
- Develop alternate structural alloys for more economic, safe, and reliable FBRs and determine the compatibility of advanced fuel-cladding alloys with sodium
- Develop nondestructive in-service inspection methods
- Establish decontamination and requalification methods for FBR components
- Provide methods to monitor, reduce, and control coolant impurities, including radionuclides

Strategy. The strategy of the materials program is to emphasize development required for near-term FBRs, primarily by working directly with plant designers to define the immediate needs and schedule; this approach has proven highly profitable in that deficiencies in design have been identified and corrected before plant startup. Concurrently, the work provides the technology base needed for upcoming reactor experiments in FFTF and for understanding the experimental results and operational characteristics of EBR-II. The technology is also developed for materials for use with advanced fuels and plant designs including alternate structural and fuel-cladding alloys, improved hardfacing materials, and high-temperature cover-gas seals.

The strategy for the chemistry program is similar in that it aims primarily at the near-term coolant monitoring and impurity control requirements of FFTF, followed by substantial emphasis on component decontamination and requalification needs of FFTF in the near future. Improvements in impurity monitoring and control instruments and components are also being developed.

This work is organized in six major elements, four of which concern primarily materials and two involve chemistry:

Materials

- Mechanical properties
- Fabrication and inspection
- Hardfacing and seal materials
- Advanced clad compatibility and structural alloy development

Chemistry

- Radioactivity control
- Sodium removal and decontamination

Mechanical Properties. Basic test data are obtained for such material properties as creep life, fatigue life, fracture toughness, cumulative damage relationships, and environmental (coolant/irradiation) effects on properties. The materials of interest are the present day reference steels.

Fabrication and Inspection. Improved fabrication methods (e.g., tube-to-tubesheet and bimetallic welds) and inspection methods for more use before and after reactor plant startup are developed. Major emphasis is on the fabrication and inspection needs for reference plant materials; increasing emphasis will be on advancing the current state-of-the-art in these areas.

Hardfacing and Seal Materials. The operational limits of hardfaced components now in place at FFTF and in the design stage for other FBRs are established. Work is starting on improving hardfacing-alloy applications for optimizing component performance and reliability. Elastomer seals are being developed with lifetimes exceeding 5 years at temperatures above 150°C.

Advanced Clad Compatibility and Structural Alloy Development. An improved structural alloy is being developed with greater economic, safety, and reliability characteristics. Alloy composition is being optimized. The stability of the alloy to the environmental and thermal conditions in FBRs is being studied. Near-term development will concentrate on codification, inspection, and fabrication. The work on advanced cladding alloys is limited to studies of sodium corrosion and mechanical properties in sodium.

Radioactivity Control. On-line traps are being developed to reduce

radionuclide levels in FBR coolant systems. The effort is not limited to radioactive specie control, and currently includes efforts to improve sodium-system cleanup devices and on-line instruments.

Sodium Removal and Decontamination. Suitable cleanup methods are being established for component maintenance and failed-fuel shipping. Procedures are being developed for cleaning components and for chemical decontamination methods for removing deposited radionuclides from component surfaces with minimum deleterious effects on the subsequent performance of these components. These developments are of immediate use for EBR-II and of near-term use for FFTF.

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2.2.5 Safety

To attain and demonstrate the required degree of safety for fast breeder reactors without unacceptable economic penalties, a sound frame of technical reference is being developed for assessing the risk to public health and safety from postulated accidents with the full spectrum of breeder reactor designs and fuel cycles being considered. Analyses and experiments are providing the broad technology base with which designers can: (a) combine features to obtain requisite levels of safety while

minimizing impacts on plant cost, performance, and schedule, and (b) demonstrate that the probability of a major accident is negligible and that minor accidents will not escalate into major accidents.

Thus, this development work aims to provide a technology base fully responsive to safety considerations in the design, evaluation, licensing, public acceptance, and economic optimization of breeder reactor systems for electrical power generation.

That goal is being achieved by developing safety technology within four integrated lines of assurance (LOAs) for preventing release of significant amounts of radioactivity to the environment: (see figure 26.)

Figure 26
Safety Lines of Assurance (LOA)

- First LOA

Prevent accidents, so that fuel melting will not occur. Plant features that prevent accidents include a design that is inherently stable under normal operation, has a high tolerance for abnormal operation and component malfunction, and has highly reliable protective devices to prevent core damage as the result of plant system and component failures. These features combine to produce a plant with an extremely low probability of accident initiation.

- Second LOA

Limit core damage, so that whole core disruption does not occur even in the unlikely event that a severe accident begins. Thus, significant amounts of radioactivity could not be released if the accident progression is terminated before the whole core is disrupted.

- Third LOA

Control accident progression, so that secondary containment

does not fail even if whole core disruption occurs. Significant amounts of radioactivity would not be released to the environment unless large amounts of accident products leak through the containment boundary. Such leakage could result if the containment is breached (a) structurally by a strong pressure generating event or (b) by being melted through because of inadequately accommodated core debris.

- Fourth LOA

Assure availability of attenuating mechanisms, both inherent and engineered, for the radiological consequences which can be postulated for low probability failures in the ability to control the accident progression. Thus, LOA 4 protects the public by attenuating the consequences of postulated accidents severe enough to breach LOA 3.

This line-of-assurance approach provides a risk-assessment framework for making rational decisions for evaluating R&D choices based on cost-benefit analysis, and ensures optimal use of program resources by channeling R&D in directions that promise the greatest payoff for reducing overall risk.

Breeder reactor safety technology is being developed by analytical and experimental work at national laboratories, industrial contractors, and universities. The analysis development improves the basis and means for reliably analyzing the course of events in postulated accidents and the estimated consequences of such accidents. This improvement is being accomplished by: (1) developing the understanding of the basic phenomena needed for analytical modeling of postulated accidents, (2) accumulating basic data on these phenomena so that the analytical models will have the appropriate realism or conservatism over the desired range of application, (3) integrating these data and models into complete analytical descriptions of the accidents so that the quantities and parameters important to a review of a safety can be accurately determined, (4) establishing requirements for large-scale experiments to test the adequacy of the analytical models for predicting accident sequences and consequences, and (5) applying the analytical models to the analyses of generic questions.

The breeder reactor safety analytical technology data base will be used by designers to demonstrate that the probability of a major accident is negligible and that minor accidents do not escalate into major accidents.

The analytical modeling and computer code development work in FY 1978 is being redirected to resolve the safety issues of alternate fuel technologies and advanced converters. The development of breeder reactor risk assessment methods will be continued; preliminary estimates of the environmental effects of using alternate fuels in breeder reactors will be completed.

Experimental work coupled to the analytical work contributes to the overall safety program by verifying the analytical tools and allowing the development of a fundamental understanding of the key phenomena controlling the progression of breeder reactor accidents. The experiments

ADVANCED FISSION SYSTEMS

use the Sodium Loop Safety Facility (SLSF), the Transient Reactor Test Facility (TREAT) and the Thermal Hydraulic Out-of-Reactor Safety Facility (THORS).

PROGRAM ADVANCED FISSION SYSTEMS SUBPROGRAM FAST BREEDER REACTORS CATEGORY LIQUID METAL FAST BREEDER REACTOR SUBCATEGORY SAFETY																												
TASK	CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
	FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
SAFETY BENCHMARK TEST (PINEX 2) AT TREAT					▽																							
ISSUE SAREF PROGRAM GUIDANCE						▽																						
SLSF SODIUM VOIDING AND FUEL DISPOSAL AND SRM TEST IN ETR						▽																						
PRELIMINARY ESTIMATES OF ENVIRON MENTAL IMPACTS OF ALTERNATE FUEL CYCLES							▽																					

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3.0 Gas-Cooled Fast Breeder Reactor

Another breeder concept, historically being developed on a lower priority than the LMFBR, is the gas-cooled fast breeder reactor. DOE development of the GCFR, initiated in 1963 under the United States Atomic Energy Commission (AEC), is one of three interrelated programs being carried out on the GCFR. In addition to the DOE GCFR development program are: (1) a United States industry program being supported by General Atomic Co. and the Helium Breeder Associates (HBA), which represents utility companies that comprise 35% of the electrical generating capacity in the United States and (2) an international program being carried out primarily with the Federal Republic of Germany (FRG) and Switzerland under the United States/FRG Umbrella Agreement. Cumulative program support from all sources (domestic and foreign) has amounted to approximately \$100 million.

The GCFR uses the inert, nonradioactive, and transparent gas helium under a pressure of about 1,300 pounds per square inch (8.96 MPa), to remove heat from the reactor core and to produce high-temperature steam for generating electrical power (figure 27).

By building on the present gas reactor technology, the GCFR is spared many of the development needs which would otherwise have to be addressed. The GCFR is at an earlier stage of development than the LMFBR and can use much of the existing breeder reactor technology as a basis for the GCFR fuels and materials development program. Attractive potential characteristics of the GCFR include a high breeding ratio, low capital costs, low maintenance costs, and the ability to incorporate alternate or advanced fuels.

Objectives. The DOE GCFR program is in agreement with the

Figure 27
Gas-Cooled Fast Breeder Reactor (GCFR) Systems

objectives of the National Energy Plan to further evaluate alternate breeder reactor concepts. Specific objectives for the DOE GCFR program are:

- To conduct research and development on the GCFR to provide for a viable alternate to the LMFBR which will be consistent with the developing United States nonproliferation policy
- To provide GCFR technology and other needed support on a schedule consistent with an internationally developed cooperative program which is led and financially participated in by private industry

PROGRAM ADVANCED FISSION SYSTEMS								
SUBPROGRAM FAST BREEDER REACTORS								
CATEGORY GAS-COOLED FAST BREEDER REACTOR								
TASK	CY 1977	CY 1978	CY 1979	CY 1980	CY 1981	CY 1982	CY 1983	
	FY 1977	FY 1978	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983	
POLP DESIGN AND LICENSING								
PSAR PREPARED		▽						
PRELIMINARY DESIGN			▽					
FUELS AND MATERIALS								
CFTL CONSTRUCTION		△				▽		
PHYSICS AND SHIELDING								
TOWER SHIELDING TESTS		△						
ZPR CRITICALS			△					
COMPONENTS								
CIRCULATOR TEST FACILITY			△					
SAFETY								
DMFT TESTS	▲							

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3.1 Plant Design Study

In FY 1978, priority will be put on initiating the GCFR Program Definition and Licensing Phase (PDLF) -- an outgrowth of the utility proposal to ERDA in June 1976. During this 5 year program, the design of a 300 MWe demonstration plant will be prepared, which would serve to guide and focus all research and development on systems, components, fuel, and safety. Also included will be preparation of portions of a preliminary safety analysis report, which will contribute knowledge for deciding whether a GCFR demonstration plant should be built. The overall program strategy is to continue developing the GCFR on a schedule which would permit the operation of the first GCFR developmental plant by about 1990, if needed, and consequently the first large GCFR prototype (also a developmental plant) could be operational by about the turn of the century. This schedule is predicated on the development of a breeder reactor and fuel cycle consistent with national nonproliferation objectives, and a government-industry consensus on the need for the nuclear breeder. This will be done in cooperation with the Helium Breeder Associates, Germany, Switzerland, and other countries yet to become involved in the GCFR program.

3.2 Technology Base

The technical program focuses on those features of the GCFR that require an extension of existing technology, such as in the components area. The R&D program includes engineering and safety technology in fuels, materials, physics, shielding, and components. Major reactor-component development in the High Temperature Gas Reactor (HTGR) program on steam generators, circulators, and prestressed-concrete reactor vessels provides data useful for the GCFR. Information obtained for prior breeder reactor fuels program from fuel irradiations and stainless steel cladding tests provides a data base for input to the GCFR fuel element design.

Fuels and Materials. This work element provides the analytical, out-of-pile and in-pile experimental development program necessary to establish the basis for the design of the GCFR fuel, control and radial blanket assemblies, and pressure equalization system. Although the major operational parameters of the GCFR fuel rods are similar to those used for the LMFBR fuel rods, the use of pressure-equalized and vented fuel rods, ribbed cladding, fuel rod spacer grids, and a gas coolant requires a comprehensive development program to extend the LMFBR fuels and materials technology to the conditions and design features which are specific to the GCFR. For FY 1978, major milestones include initiation of construction of the thermal-hydraulics Core Flow Test Loop (CFTL) at ORNL, the F-5 experiment will be irradiated in EBR-II and the post irradiation examination of previous tests in the GB series and F series at ANL will be completed. Mechanical testing of the core elements will continue at GA. The international cooperative program includes a 12 pin in-pile test presently being irradiated in the BR-2 Reactor in Mol, Belgium.

Physics and Shielding. The purpose of the shielding work is to validate nuclear radiation shielding design methods and to reduce the uncertainties in design related nuclear techniques. The purpose of the physics work is to confirm the core inventory, enrichment, reactivity, and

breeding calculations. Both work areas will provide accurate specifications for licensing procedures, margins on core inventory, and permit more precise power, life, and breeding calculations for the core. The reactor shielding is used to assure the integrity of vital internal structural components of the reactor, particularly for the grid plate which supports the core and blanket.

The adequacy of nuclear design methods and data is being evaluated by direct experimental comparison with GCFR critical experiments in the Zero Power Reactor (ZPR) at ANL. Planning has begun for a GCFR Engineering Mock-up Critical (EMC) at ANL in FY 1979. The grid plate shield experiment in the tower shielding facility was initiated at ORNL in October 1977, and a radial shield experiment is also under design with additional experiment work planned for FY 1979.

Components. The objective of this work element is to develop the GCFR Nuclear Steam Supply System (NSSS) and its associated auxiliary system components. The entire primary system is contained within a multi-cavity Prestressed Concrete Reactor Vessel (PCRV) similar in design and construction to the HTGR PCRV structures tested and operated in Europe and for the Fort St. Vrain HTGR. The GCFR design pressure, however, is significantly higher than these plants and its PCRV is a multi-cavity, rather than the more commonly used single cavity design. Major NSSS component development has been carried out at a relatively low level to date due to the need to resolve feasibility questions in the other program elements. Several components, including the PCRV, main helium circulators, and steam generators represent long lead or early installation items. For FY 1978, PCRV closure tests will continue, the conceptual design for a helium circulator test facility will be initiated, and the preliminary design of the Core Auxiliary Coolant System will be completed.

Safety. The GCFR program element will develop the safety analysis methods and the experimental information required for licensing and further utilization of the GCFR concept. Specifically, information is being obtained for assessments of the probabilities and consequence of various postulated accidents in the GCFR; to evaluate the post-accident fuel containment capability of the GCFR; to provide a technical basis for the integration of safety and reliability targets into the GCFR engineering effort; and to verify the fuel and cladding behavior during accidents which may lead to core damage.

An out-of-pile, direct electrical heating test program is underway at LASL to investigate the behavior of simulated GCFR fuel assemblies during Loss of Flow (LOF) and Depressurization (DBDA) accident conditions. These tests will include full-size fuel assemblies. During FY 1978, Title I design work was started on the Gas Reactor In-Pile Safety Test (GRIST-2) loop which is intended to transient test GCFR fuel assemblies at high reactivity rates in the TREAT-Upgrade facility at INEL.

Both analytical and experimental work will continue at ANL to investigate the consequences of various core disruptive accidents. This will include transient reactivity effects, fuel compaction and/or dispersion mechanisms, and the thermal-chemical interactions among molten core materials and the various shielding, insulation, and PCRV materials. Work will continue at GA to integrate the results of safety work at LASL and ANL into the GCFR plant design and the PSAR effort.

F. NUCLEAR POWER STUDIES

1.0 Nuclear Energy Assessments

Conceptual design and applications studies assess nuclear energy concepts, including associated fuel support facilities and services for the production of energy for both electrical power and for nonelectric-energy-consuming processes (e.g., process heat). The studies and evaluations support management planning in evolving the objectives of the nuclear energy programs. Activities include investigations of the applicability of nuclear power systems to a wide variety of energy needs (such as process heat), conceptual designs, and feasibility studies of low-temperature and waste-heat utilization systems for various applications, and assessments of technologies for improving heat dissipation systems to conserve water resources and increase the flexibility of nuclear power plant siting. Additionally, design and applications studies of the energy center concept as a nuclear power plant siting option are being undertaken. Associated with all nuclear energy concept evaluations is the assessment of their environmental acceptability and means of mitigating undesirable environmental impacts.

2.0 Alternative Advanced Reactor Systems

The Nonproliferation Alternative Systems Assessment Program (NASAP) will evaluate advanced and alternative nuclear reactors and fuel cycles that have potential advantages for extending the utilization of uranium resources while minimizing the potential for unauthorized use of weapons-grade material. Although the NASAP study is still under way, several promising concepts have been identified. Preliminary plans are being formulated for an experimental program to evaluate such concepts further. The concepts include Spectra-Shift-Controlled Reactors (SSCR), Heavy-Water Reactors (HWR) either alone or in tandem with light-water reactors, and High-Temperature Gas Reactors (HTGR) on a Low-Enriched Uranium (LEU) cycle. The experiments will initiate work on these or other advanced concepts not currently being developed but which the NASAP studies indicate are promising.

V. FUEL CYCLE

A. PROGRAM OVERVIEW

Two policy decisions by President Carter in 1977 have had a major impact on the DOE fuel cycle program.

- On April 7, 1977, the nonproliferation policy was announced, which deferred indefinitely the reprocessing of spent fuel from commercial reactors.
- On October 18, 1977, the policy that the Federal Government would take title to spent commercial fuel was announced.

The fuel cycle is depicted with reprocessing in Figure 28 and without reprocessing in Figure 29. The new policy engenders increased emphasis in the areas of conservation, more efficient separation of uranium-235 from natural uranium, the search for reprocessing fuel cycles that can meet nonproliferation criteria, and long-term management of spent fuel. At the same time there is diminished need for effort in the areas of conventional fuel reprocessing and the disposal of wastes from such reprocessing. These considerations have been incorporated in the principal objectives of the fuel cycle program, which are to:

- Conserve uranium resources by the development of separation technologies that will permit economic production of the equivalent of natural uranium from the tails of current enrichment facilities and from the existing tails stockpile
- Develop safe, protected spent fuel processing methods that will enhance the contribution of the nuclear fission power option with acceptable proliferation risk
- Develop and demonstrate safe, protected processes and facilities for packaging, storage, and disposal of radioactive wastes.

The subprograms of the fuel cycle program are:

- Advanced Isotope Separation
- Fuel Processing
- Spent Fuel Storage
- Waste Management

Advanced Isotope Separation. The magnitude of the contribution of light water reactor technology to the Nation's energy requirements is dependent on the availability of U-235 fuel. Because the new nuclear energy policy defers indefinitely the reprocessing of commercial spent fuel

Figure 28
Fuel Cycle with Fuel Reprocessing

in order to support the nonproliferation objective, other methods to increase the availability of U-235 are being developed.

One potential source of U-235 which requires no mining is the tails from the enrichment plants. The economic recovery of this fuel is a principal purpose of the advanced isotope separation program. Three advanced separation technologies--two laser techniques and a plasma technique--are under development. Work through the preprototype should be completed by the end of FY 1980 at which time an evaluation will be made to support a decision as to which process will be selected for further development.

Fuel Processing. As a result of the new policy, the program supporting the nuclear fuel cycle that had been directed toward the commercialization of fuel reprocessing and fuel recycle has now been reoriented toward the development of a full range of safe, protected, alternative fuel cycle technologies. Fuel cycle alternatives under consideration would use various combinations of uranium, plutonium, thorium, and fission products. The possible recovery and reuse of fissile values from these fuels with minimum proliferation risk would involve the unique application of familiar separations technologies and/or the introduction of new technologies.

A preliminary evaluation of alternatives is scheduled for completion by

Figure 29
Fuel Cycle without Fuel Reprocessing

the end of FY 1979. Subject to interim decision, it is planned to complete by about 1984 the construction of a component test facility that will be used to obtain data needed for the final design of fuel cycle facilities. The program includes work aimed at the development of technology that could accommodate a wide selection of possible fast breeder reactor fuel cycles.

Spent Fuel Storage. The indefinite deferral of reprocessing of spent nuclear fuel presents the operators of nuclear power plants with the problem of how to store such fuel after their on-site storage basins are full. To alleviate this problem, DOE is developing a program under which it will accept, take title to, and store spent fuel from United States reactors and from a limited number of foreign reactors consistent with national energy goals and nonproliferation objectives. A one-time fee based on full cost recovery will be paid to the United States Government for this service.

Current plans call for the first geologic storage facility for spent fuel to be available by 1985. In the interim, DOE approved away-from-reactor (AFR) water storage basins will be used to store spent fuel in a retrievable mode. DOE will seek to obtain such interim services from private industry; however, if such services are not reasonably available, DOE will provide the storage facilities.

Waste Management. The continued operation of both commercial reactors and defense-related reactors and associated processing facilities

FUEL CYCLE

dictate that technology be developed for the confinement and safe disposition of the radioactive waste that are an inevitable byproduct of such activities. Work is underway leading to the solution of the problems associated with the management of high- and low-levels of solid, liquid, and gaseous radioactive waste, including the development of processing, transportation, and storage technologies.

Defense waste originates from the operation of nuclear materials production reactors, naval propulsion reactors, and R&D reactors. Except for minor quantities from R&D, medical, and industrial applications, the origin of commercial radioactive waste is nuclear power stations, which now supply nearly 13% of the Nation's electricity. Prior to the nonproliferation and spent fuel policy announcements, DOE's commercial waste management program was based on the assumption that spent commercial fuel would be reprocessed for the recovery of uranium and plutonium and that the separated fission products would be delivered to the Federal Government in the form of solidified high-level waste. DOE's waste management program has now been redirected to:

- Demonstrate the concept of geological isolation of commercial spent fuel and provide sites for such activity
- Provide for the increased spent fuel storage capacity required by the nonproliferation policy
- Develop and demonstrate technologies for safe packaging radioactive waste from different fuel cycles, for initial handling, interim storage, processing, and permanent isolation

The key goal of the commercial waste management program is to have in operation by the late 1980s a deep geologic repository together with the technical support base for the retrievable storage of spent commercial fuel, and the disposal of commercial high-level and transuranium contaminated wastes. For defense wastes, the goal is to complete long-term disposition of the backlog of high-level wastes by about the end of the century.

The following table summarizes the operating, equipment, and construction funding levels for the fuel cycle program, by subprograms, for FY 1977 through FY 1979.

PROGRAM OVERVIEW

FUEL CYCLE	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
SUBPROGRAMS	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Advanced Isotope Separation				
Operating	\$39,720	\$43,617	\$46,200	\$2,583
Equipment	6,996	8,485	8,000	(485)
Construction	<u>0</u>	<u>0</u>	<u>0</u>	<u>0</u>
Subtotal	46,716	52,102	54,200	2,098
Fuel Processing				
Operating	54,700	92,400	48,850	(43,550)
Equipment	4,300	9,000	2,000	(7,000)
Construction	<u>0</u>	<u>3,000</u>	<u>6,750</u>	<u>3,750</u>
Subtotal	59,000	104,400	57,600	(46,800)
Spent Fuel Storage				
Operating	0	5,000	6,000	1,000
Equipment	0	0	0	0
Construction	<u>0</u>	<u>0</u>	<u>0</u>	<u>0</u>
Subtotal	0	5,000	6,000	1,000
Waste Management *				
Operating	182,619	283,547	317,100	33,553
Equipment	10,337	18,482	17,700	(782)
Construction	<u>113,900</u>	<u>167,962</u>	<u>202,000</u>	<u>34,038</u>
Subtotal	306,856	469,991	536,800	66,809
TOTAL	\$412,572	\$631,493	\$654,600	\$23,107

* Includes defense-waste management, which is funded under Atomic Energy Defense Mission (is not an HS&T program).

B. ADVANCED ISOTOPE SEPARATION

1.0 Overview

History of Isotope Separation. The fuel for the current generation of light water reactors (LWRs) operating in the United States is uranium enriched to about 3% of the U-235 isotope. Currently, all United States uranium enrichment is accomplished by gaseous diffusion technology. The gaseous diffusion process was developed in the Manhattan Project during World War II. Three gaseous diffusion plants were constructed over a period of time from 1943 to 1956. These plants originally were constructed to produce material for nuclear weapons, but now only about 5% of the capacity is used for all United States Government programs; the remainder of the capacity is used to produce low-enriched uranium for nuclear power reactors in the United States and abroad. Underway are two programs to increase the capacity of the diffusion plants. The Cascade Improvement Program (CIP) and the Cascade Upgrading Program (CUP) will increase the separative capacity of the three plants from 17.2 million separative work units (SWU) per year to about 27.3 million SWU per year. These improvements will be completed in the early 1980s.

During the Manhattan Project, centrifuge technology was evaluated, but gaseous diffusion was judged to be more attractive. Advances in materials and technology for machinery during the 1950s increased the potential for centrifuge technology. Investigation of centrifuge technology during the 1960s led to long-term testing of advanced centrifuges in the early 1970s. The President announced on April 20, 1977, that the next increment of uranium enrichment capacity built by the United States Government would use gas centrifuge technology.

Selective laser-induced chemistry emerged in the United States in 1971 at the Los Alamos Scientific Laboratory. Since then, the United States program to develop advanced isotope separation technology has grown to an annual program of about \$50 million.

Introduction. The extent to which light water reactor (LWR) technology can contribute to the Nation's future energy requirements is dependent on uranium fuel availability. One potential source of U-235 other than mining new natural uranium is the depleted uranium "tails" from conventional gaseous diffusion or planned centrifuge enrichment plants. Therefore, the major objectives of the advanced isotope separation program are to:

- Conserve uranium resources by developing separation technologies that will permit economic production of the equivalent of natural uranium from the tails of the enrichment process
- Evaluate the nonproliferation implications of alternative novel enrichment technologies
- Apply advanced isotope separation technologies to other areas of

the nuclear fuel cycle, such as the separation of nonfissile isotopes

The advanced isotope separation program consists of two categories of effort:

- Advanced uranium isotope separation technologies
- Applications and assessments of advanced separation technologies

The approach of the advanced separation techniques effort involves the development of three methods through the preprototype phase two laser techniques and plasma technique. The preprototype devices will demonstrate in conjunction with additional off-line experimentation, the components and scientific principles that would be utilized in a later engineering demonstration facility. This phase will be completed at the end of FY 1980, at which time a comprehensive economic evaluation of the techniques will be made in support of a decision as to which will be selected for development and construction of the facility.

The approach of the proliferation and advanced applications effort involves the assessment of promising technologies and their proliferation implications.

The following table summarizes the operating funding levels by category for FY 1977 through FY 1979.

ADVANCED ISOTOPE SEPARATION	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Advanced Isotope Separation Techniques	\$38,385	\$41,137	\$43,550	\$2,413
Applications and Assessments of AIST	1,335	2,480	2,650	170
TOTAL	\$39,720	\$43,617	\$46,200	\$2,583

2.0 Advanced Uranium Isotope Separation Techniques

Natural uranium is composed of several isotopes. In terms of fuel value, the most important are U-235 at 0.711% and U-238 at over 99%. Before natural uranium can be used as fuel in current commercial nuclear power plants in the United States, it has to be enriched, i.e., the concentration of the fissionable isotope, U-235, has to be increased from its normal concentration of 0.711% to nominally 3% for LWRs. The residue from the enrichment process, depleted tails, still contains on the order of 0.25% U-235. If the uranium in the tails can be recovered and "enriched" to natural uranium assay, 0.7%, it can then be used as feed for conventional enrichment operations.

FUEL CYCLE

The major objective of DOE's advanced uranium isotope separation techniques program is to produce normal uranium assay (0.711% U-235) from depleted tails of conventional enrichment plants at costs below natural feed, i.e., 50-75% below the present market price of \$43/pound of U_3O_8 . Meeting this objective would increase the effective economics of the uranium resource base by 25-30%.

Three processes are currently being pursued -- two laser techniques (atomic vapor and molecular) and a plasma technique. The primary responsibility for each technology and location of work are indicated in the following:

PROCESS	CONTRACTOR/LOCATION	FORM OF ENERGY ABSORBED
Atomic Vapor	Lawrence Livermore Laboratory (LLL), Livermore, Calif	Visible
Molecular	Los Alamos Scientific Laboratory (LASL), Los Alamos, N M	Infrared Ultra violet
Plasma	TRW, Inc Redondo Beach, Calif	Radiofrequency

In addition, DOE's Gaseous Diffusion and Y-12 production plants in Oak Ridge, Tennessee, provide support in the areas of source and collection development, uranium metal handling, process physics and spectroscopy, process equipment development, conceptual engineering, and economic evaluations.

The strategy of this program is to pursue the three processes in parallel through the basic research stage. At the start of FY 1978, laboratory-scale devices required to address the scientific issues of the three technologies were available. This work will continue until the end of FY 1980, when a comprehensive economic evaluation and assessment of the risk associated with large-scale implementation will be made. This will result in a determination of which process, if any, warrants further development. This would be followed by an engineering demonstration to permit an accurate economic assessment of the process.

The accomplishments of the advanced isotope separation program are classified. However, it can be stated that conceptual engineering cost studies continue to indicate that these technologies have great promise in meeting the program objective.

In general, the expected results of the R&D activities are:

- FY 1978 Process Definition. Major science efforts with significant impact on the process economics will be resolved.
- FY 1979 Component Development. Activities will be focused on lasers, magnets, collectors, extractors, optics, and feed delivery systems to be installed in preprototype systems.
- FY 1980 System Integration. Efforts will be concentrated on demonstrating technologies in an integrated large-scale (pre-prototype) system. Major issues of engineering scaling for all parts of the process will be identified.

ADVANCED ISOTOPE SEPARATION—TECHNIQUES

The following table summarizes the operating funding levels by process for FY 1977 through FY 1979.

ADVANCED URANIUM ISOTOPE SEPARATION TECHNOLOGY	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Atomic Vapor Process	\$10,441	\$13,395	\$13,220	\$ (175)
Molecular Process	20,089	17,350	18,625	1,275
Plasma Process	6,351	9,235	10,080	845
Supporting Technology	1,504	1,157	1,625	468
TOTAL	\$38,385	\$41,137	\$43,550	\$2,413

2.1 Atomic Vapor Process

The atomic vapor process takes advantage of the small isotopic differences in the electronic spectra of uranium atoms. The process requires the controlled vaporization of uranium atoms followed by selective excitation and ionization of uranium-235 using tunable lasers in the visible or ultraviolet regions of the spectrum. The resulting plasma of uranium-235 ions can then be removed from the uranium vapor using electromagnetic methods. The large-scale generation of vapor under required process conditions, material handling of molten uranium, the demonstration of a scalable uranium ion extractor, and the development of full-scale lasers suitable for plant operation are under intensive investigation. Figure 30 illustrates the atomic vapor process.

The major areas of required development efforts for the atomic process are:

- Laser Development:
 - Increase the average power output per laser
 - Increase the number of times the laser can operate per unit time
- Material Development:
 - Increase the lifetime of material exposed to liquid uranium and uranium vapor
- Extractor Development:
 - Develop an extractor technique to operate at the densities required for a production module
- Science:
 - Establish the value of the physical parameters required to define a process

Figure 30
Atomic-Vapor Separation Process

An important component of the process development program is the construction and evaluation of an advanced uranium vapor generation source. This source is expected to be operational, providing preliminary data on uranium vaporization, by the end of FY 1978.

2.2 Molecular Process

In the molecular process, UF_6 is expanded through a nozzle to reduce the temperature and simplify the spectrum of the gas. The UF_6 is then irradiated by laser radiation of a particular frequency and the ^{235}U is selectively photo-dissociated to UF_5 (a solid) which can be separated from gaseous UF_6 . This process has intrinsically high throughputs and uses existing materials handling technologies. However, the high power lasers at the specific wavelengths required are not yet available. Figure 31 illustrates the molecular process.

The major areas of required development effort for the molecular process are:

- Laser Development:
 - Increase the average power output per laser
 - Increase the number of times the laser can operate per unit time
- Science:
 - Increase the basic understanding of the interaction of the

Figure 31
Molecular Separation Process

laser power and the process gas

Status. In the early phases of laser development, efforts were concentrated on developing a laser system which could be used to demonstrate uranium enrichment. Previously developed systems had neither the proper spectral characteristics nor the potential for scaling to the power needed for large-scale operation. Efforts during FY 1978 will be concentrated on the engineering of laser systems which have the potential of being scaled to larger power levels at the required wavelengths. Fabrication of an enlarged gas flow facility will be completed and in operation early in FY 1979.

2.3 Plasma Process

The plasma process is based on ion cyclotron resonance. The method involves generating a stable plasma in a magnetic field thereby establishing the ion cyclotron frequencies of the different isotopes. The frequency of an RF field is then matched to the ion cyclotron frequency of the U-235 where resonance is achieved and energy added to the U-235 ion. This results in an increase of the cyclotron orbit of the U-235. This difference in orbits is used to separate U-235 from U-238. Figure 32 illustrates the plasma process.

Figure 32
Plasma Separation Process

The major areas of required development effort for the plasma processes are:

- Plasma Density:
 - Increase the operating plasma density
- Extraction:
 - Increase the data base on extraction and establish scaling laws
- Science:
 - Increase the knowledge and establish the upper limits on the plasma process

One of the main scientific concerns regarding the plasma process has been the development of a stable plasma at densities that will allow for economically handling the throughput required in a production plant. Theoretically, as plasma densities increase, collision processes are established that can interfere with the process. The program is structured to evaluate the density-throughput relationship as soon as possible.

2.4 Supporting Technology

This area of effort is comprised of two separation program activities: (1) Research common to both laser technologies and (2) scoping conceptual engineering and cost analysis for the three separation technologies.

Unsolicited proposals are reviewed and funded from universities and private firms in the areas of laser isotope separation (LIS) where a programmatic need exists. This effort represents about 50% of the funds for FY 1977 and about 30% of the FY 1979 efforts.

For the scoping conceptual engineering and cost analysis, detailed cost models for the atomic vapor and molecular processes were developed during FY 1977; during FY 1978 a detailed model will be developed for the plasma process. As more information is developed for the three processes, the cost models are updated and sensitivity analyses on selected process variables are performed. These analyses highlight the relative importance of the process parameters and serve to direct the R&D programs by identifying the variables that have high leverage in the process costs.

3.0 Applications and Assessments of Advanced Separation Technologies

The objectives of the applications and assessments of advanced separations technologies are to:

- Maintain DOE on the forefront of understanding uranium separation technology, evaluate new uranium enrichment technologies, and accurately assess their proliferation implications
- Apply advanced separation technologies to the separation of non-fissile isotopes of interest to the nuclear fuel cycle, e.g., deuterium

Advances in science and technology have opened new vistas for economical isotope separation using processes previously thought to be impractical, if not impossible. In addition, the development of laser isotope separations technology affords a new and powerful technology that can be applied to the solution of problems in other areas of the nuclear fuel cycle. Research and development on new concepts are designed to ensure that our Nation maintains its leadership in isotope enrichment technologies and exploits the use of these technologies to the fullest potential in all aspects of the nuclear fuel cycle.

Experiments are being performed to validate the theory of chemical exchange (the French process) and chemi-ionization to develop the scaling laws required to evaluate the proliferation implications of these technologies.

Since the cost of deuterium represents approximately 20% of the cost of a heavy water reactor, its costs will play an important role in the economic viability of using heavy water reactors in the United States. Several promising technologies that have the potential for significant reductions in the cost of producing heavy water are being studied to develop accurate cost projections.

C. FUEL PROCESSING

1.0 Overview

Background. On April 7, 1977, President Carter announced a new nuclear energy policy, which deferred indefinitely the reprocessing of commercial spent fuel in order to allow time to investigate more proliferation-resistant alternatives. The program in support of the nuclear fuel cycle was previously directed toward the commercialization of fuel reprocessing and plutonium recycle. As the result of the new policy, the program has been reoriented to the investigation of a full range of alternative fuel cycle technologies. Fuel cycle alternatives under consideration involve a variety of fuel materials containing combinations of uranium, plutonium, and other actinides (radioactive heavy elements produced by irradiation), thorium and fission products. The possible recovery and reuse of fissile values from these fuels with minimum proliferation risk would involve the unique application of familiar separations technologies and/or the introduction of new technologies.

Objectives. The major objective of the spent fuel processing program is to develop safe, protected processes which will enhance the contribution of the nuclear fission power option to the Nation's energy economy through greater utilization of nuclear fuel resources.

The specific objectives of the program are:

- Develop alternative fuel cycle technologies and design bases that will recover and reuse fuel values with minimum proliferation risk. (This will facilitate a preliminary fuel cycle evaluation by the end of FY 1979.)
- Provide support to the Nonproliferation Alternative Systems Assessment Program (NASAP) and International Nuclear Fuel Cycle Evaluation (INFCE) studies.

Strategy. The strategy of the program is to develop technologies and design bases for the recovery and reuse of fuel values contained in spent reactor fuels by means of processing methods resistant to proliferation of nuclear weapons, and to develop and improve the safeguards, operability, and maintainability of fuel processing systems for selected cycles.

To facilitate the achievement of the objectives, the program is divided into five categories of effort:

- Alternative Fuel Cycle Technologies
- Advanced Fuel Recycle
- High Temperature Gas-cooled Reactor Fuel Recycle
- Thorium Fuel Recycle

- Private Processing Facilities

The following table summarizes the operating funding levels by category for FY 1977 through FY 1979.

FUEL PROCESSING	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Alternative Fuel Cycles	\$27,900	\$30,900	\$16,000	\$(14,900)
Advanced Fuel Recycle	13,500	27,500	16,000	(11,500)
High Temperature Gas Cooled Reactor Fuel Recycle	13,300	12,000	0	(12,000)
Thorium Fuel Recycle	0	8,000	16,850	8,850
Private Processing Facilities	0	14,000	0	(14,000)
TOTAL	\$54,700	\$92,400	\$48,850	\$(43,550)

2.0 Alternative Fuel Cycle Technologies

Background. Spent fuels from light water reactors (LWRs) are being discharged at an increasing rate both in the United States and abroad. Residual fissile isotopes in these fuels are a potential source of materials that could provide additional fuel for the production of power. Conventional processing technology results in the production of a pure stream of plutonium in a form that could be utilized for the manufacture of weapons if diverted for that purpose. Processing technology is required that will minimize the utility and increase the difficulty of diversion of nuclear materials for weapons purposes, while at the same time allowing utilization of these materials for power production. The processes to be investigated in this program will avoid the production of clean, separated plutonium at any step in the fuel cycle, and will maximize the difficulty for diversion of the material for weapons purposes.

Objectives. The objective of this program is to provide the technology base needed to evaluate selected alternative systems for the disposition of spent nuclear fuel and to determine an acceptable fuel cycle consistent with the President's nonproliferation policy. The program will support NASAP and INFCE studies and facilitate preliminary fuel cycle evaluation by the end of FY 1979. The recovery and reuse of fissile values from various fuels could involve some unique applications of current technologies. This work will be carried out by:

- Broad evaluation and scoping studies to identify feasible alternative nuclear fuel cycles
- Research and development activities to establish the feasibility of the selected fuel cycles
- Design studies to help identify necessary development areas and safeguards and nonproliferation aspects

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- Development and testing of key systems, equipment, and components

The program will use a coprocessing fuel cycle as a reference case. The program includes both reprocessing and refabrication technology development. Planned activities include the following:

- A preliminary Design Integration Study (DIS) Report will be provided in September 1978. This report will be based on the original reference flowsheet of separated plutonium and uranium and will include an assessment of safeguards as well as preliminary cost estimates for both fuel processing and refabrication. The report will be used as a benchmark to assess the current reference of coprocessing and other alternatives.
- Fuel storage density experiments will be conducted to establish critical configurations of fuel-rod clusters by mocking up LWR-type fuel elements in close-proximity water storage.
- Coprocessing is the basis for the current reference flowsheets for both fuel processing and refabrication. In these flowsheets, uranium and plutonium are separated and decontaminated together so there is never direct access to clean separated plutonium. The preliminary scope of work will be defined by the first quarter of FY 1979 and a final report will be issued in the first quarter of FY 1981.
- The study, research and development activities are scheduled so that the final Design Integration Study Report will be completed by November 1980. This report will provide the basis for comparing alternative fuel cycles from a technical standpoint and will provide cost estimates for fuel cycle facilities.
- The early development of conceptual process flowsheets will lead to an assessment of alternative fuel fabrication technologies by April 1978. The program will provide technological bases for a decision in FY 1981 on proceeding with the design and construction of a production refabrication facility based on coprocessed pellet fuel.
- To verify the research and development findings, it is planned to test critical components and systems in a non-radioactive environment in a Component Development Facility at some future date.

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3.0 Advanced Fuel Recycle

Background. Future potential for the fast breeder reactor will be highly dependent upon the ability to recycle fuel under conditions that meet realistic nonproliferation criteria. Currently identified applications include its use in an internationally controlled nuclear energy center either with a modified uranium-plutonium fuel cycle, or as a plutonium burner/uranium-233 producer, or as a uranium-233 breeder. Accordingly, the program has been broadened to provide technical information necessary for evaluation against these criteria.

Objective. The objective of the advanced fuel recycle program is to develop the technology to accommodate a wide selection of possible fast reactor fuel cycles.

Status. Since one option under consideration is the use of a plutonium fueled core, the base technology under development for core fuel recycle is being continued but redirected to emphasize coprocessing and partial decontamination. In addition, several activities will be initiated aimed at integration of thorium based core fuel concepts into the current technology.

Fast reactor fuels of United States designs are generically similar to light water reactor fuels and can be reprocessed using the same base technology, i.e., the shear-leach head-end followed by solvent extraction. There are,

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however, significant differences in such things as fuel cladding, fuel-geometry, fissile content, decay heat and radiation level, which create special problems in applying directly the technology base that exists and is being further developed for thermal reactor alternative fuel cycles. Although solutions to the technology needs that are unique to reprocessing fast reactor fuels will be obtained by building from the LWR technology base, there are many instances where new equipment concepts must be developed and additional process steps provided.

Alternative process evaluation with potential nonproliferation characteristics will be undertaken as a part of the advanced recycle program. The fast reactor fuel cycle system has the potential for nonproliferation advantages and long-lived waste product burnup in the reactor. The tolerance of the fast breeder reactor for impurities could allow the operation of alternative processes in which some of the highly radioactive components of irradiated fuel are not removed during fuel reconstitution.

The need to test process and equipment concepts using irradiated fuels will continue regardless of the specific fast reactor alternative selected. Conceptual design studies for the Hot Experimental Facility (HEF) will be broadened in FY 1978 to provide the added capability required for a variety of fast reactor fuels and alternative processes. Complete conceptual design will be available by the end of FY 1979 to support preliminary design in FY 1981. This pilot scale facility, which will handle fully irradiated fuel, will have the additional capability of testing thorium based fuels. Over the past several years, the development work has been conducted on a small laboratory and bench scale aimed at defining spent fuel characteristics and process and equipment developments to accommodate those characteristics specific to fast breeder fuels. Work has been concentrated on the head-end systems. A full-scale fuel shear, originally used in LWR fuel reprocessing work, has been modified for shearing fast breeder fuel, and tests have been run. An engineering scale fuel dissolver has been fabricated and is being installed for testing with unirradiated mockup fuel. Off-gas control systems, for the control and retention of krypton, iodine and tritium, have been investigated on a pilot scale (krypton) and on a smaller bench scale (tritium and iodine). The other systems, including feed preparation, solvent extraction and waste management have been investigated largely at the small hot cell to bench development scale. Testing of these systems on an engineering scale will commence in the near future. This work has permitted preliminary evaluation of generic process equipment concepts that will apply to most fast reactor fuel cycle alternatives.

Additional effort in the advanced fuel recycle program includes the following:

- Analytical support to the LWBR proof-of-breeding program will be provided.
- Fuel processing technology based on pyrochemical and other nonaqueous methods will be evaluated. (Preliminary flowsheets for these methods will be defined in FY 1978 to serve as the basis for further laboratory-scale testing.)

- Final design of an Integrated Equipment Test (IET) facility, which will be a non-radioactive component and system test, will be initiated in FY 1978 and construction will begin in FY 1979. (The facility is scheduled to be operational in FY 1981, and will demonstrate process alternatives in engineering - size components.)
- Technology and systems to identify and control radioactive effluents during FBR fuel fabrication will be developed.

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4.0 High Temperature Gas-Cooled Reactor Fuel Recycle

Background. HTGRs utilize the thorium/U-233 fuel cycle, and can operate in a nonproliferation mode using medium enrichment uranium-235 and denatured recycled U-233. HTGRs are also the only fission energy systems that have the potential of providing very high temperature heat, and so appear promising for such applications as gas turbines and process heat systems. Much of the potential of the HTGR is based upon the use of the thorium/U-233 fuel cycle, which depends upon the availability of complex technology for chemical reprocessing, refabrication, waste disposal, and other allied fuel cycle operations. Essential portions of this technology have not been demonstrated on an adequate basis to support the design of a large-scale facility. Therefore, government support and industry/government facilities are needed to develop the technology in order to keep pace with the anticipated needs of the HTGR industry.

Objectives. The HTGR fuel recycle program objective is to develop the technology needed to recover and recycle U-233 from spent HTGR fuel,

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with particular attention to the fuel cycles involving the use of medium enriched uranium (MEU) and denatured uranium fuel (U-233 with added U-238) to minimize the potential for nuclear weapons proliferation. The strategy is to develop processes and equipment through successive stages (laboratory scales, cold engineering, hot engineering, and cold prototype) to enable design, construction, and operation of a demonstration facility using production scale equipment.

The major elements of this program are:

- Reprocessing development
- Refabrication process development
- Waste pretreatment development
- Conceptual design modifications to scheduled facilities required by change in the reference fuel from HEU (high enrichment uranium) to proliferation resistant MEU (medium enriched uranium)
- Alternative fuel cycle assessments
- Conceptual design studies for an HTGR recycle reference facility

Status. The near-term activities of the HTGR recycle program are associated with the evaluation of nonproliferation alternatives and with the prerequisite development of a more proliferation-resistant technology in support of the demonstration facility design. The assessment of non-proliferation reprocessing options focuses on evaluating the fuel cycle aspects of gas-cooled reactor concepts with reduced proliferation potential. In the reprocessing development area, engineering-scale component development to demonstrate reference flowsheet feasibility continues. An integrated cold head end pilot line was installed and initial operation began in July 1976. Refabrication components and systems are also at the stage of engineering-scale testing and under design and procurement for prototype-scale testing. Operation of offgas treatment components is proceeding, including continued successful operation of an engineering-scale cryogenic facility for removal of krypton-85 from burner offgases. New fuel material combinations will be subjected to irradiation testing.

The R&D work also includes a Hot Engineering Test Facility (HETF) for engineering-scale components to resolve technical issues involving fission product transport and radiation effects. Procurement is expected to begin in FY 1981. Conceptual design of pilot recycle facilities was completed during 1975. During 1978, efforts will concentrate on definition of design requirements for an HTGR Recycle Reference Facility (HRRF) to provide a technically sound baseline upon which to base the elements of the R&D effort. Included in the design is the provision for use of MEU and denatured fuel.

In order to make maximum use of international activities in HTGR recycle, DOE is maintaining cooperation with the Federal Republic of

Germany Ministry of Research and Technology. An umbrella agreement was signed on February 11, 1977, providing for cooperation in the field of gas-cooled reactor research and development. Work packages have been defined and agreed upon as part of the FY 1978 program and should result in a meaningful exchange of technology.

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5.0 Thorium Fuel Recycle

Background. Table 5 summarizes previous experience with thorium/U-233 reactors and concepts.

One of the unique advantages of the thorium/U-233 fuel cycle is its adaptability to recycle operations using "denatured" uranium (mixtures of U-233 and U-238). From a proliferation standpoint, "denatured fuels" are on a par with fuels slightly enriched with U-235. Thus, U-233 generated in or bred from thorium can serve as an alternative fuel to low-enriched U-235 fuels. It also has some advantage in the effective use of uranium resources. Economic application of a thorium economy implies establishment of recycle capability, currently unavailable in the commercial sector, strongly dependent upon government initiatives for implementation.

Table 5. Experience with Thorium/U-233-fueled Reactors and Concepts

REACTOR OR CONCEPT	LOCATION OR SPONSOR	EXPERIENCE
AETR	Los Angeles, Calif	First use of thorium in a critical experiment, 1961
AVR	West Germany	15 MWe helium cooled "pebble bed" reactor, operated with fuel pebbles containing microspheres of thorium uranium carbide in graphite
BORAX IV	INEL, Idaho	Thorium oxide was included in BWR core experiments
CANDU	Canada	D ₂ O moderated reactor concept adaptable to thorium fuel cycle
Dragon	U K	20 MWt helium cooled reactor using uranium fuel and thorium fertile material, operated beginning in 1964
Elk River	Elk River, Minn	BWR operated with thorium oxide in core
Fort St Vrain	Nr Denver, Colo	330 MWe HTGR thorium converter, is in power ascension stage
HWOCR	U S	D ₂ O moderated, organic cooled thorium reactor concept
Indian Point No 1	Buchanan, N Y	Commercial PWR that used pellets of uranium thorium in first core, 1962
LSBR	U S	Core concept using thorium core seeded with chunks of U 235, plans to build were dropped in mid 1960s in favor of LWBR development
LWBR	Nr Pittsburgh, Pa	Seed blanket core similar to LSBR built and operating in Shippingport Atomic Power Station
MSRE	ORNL, Tenn	Molten Salt Reactor Experiment was first reactor fueled exclusively with U 233, 1968
MSBR	ORNL, Tenn	Molten Salt Breeder Reactor concept would have core surrounded by molten thorium fluoride blanket
Peach Bottom No 1	Peach Bottom, Pa	40 MWe gas cooled reactor fueled by carbon coated uranium thorium carbide microspheres operated 1966-1974
THTR	Euratom	Concept for a larger reactor based on AVR, being considered for future construction

Objectives. The primary objective of the program is to provide the fuel processing data required to allow effective utilization of the extensive thorium reserves in the United States and abroad within the guidelines of the President's nonproliferation policy. The thorium fuel cycle technologies program is incremental to the alternative fuel cycle technologies program in that both programs share common problem areas and in many instances share common research and development goals. Therefore, research and development activities will be structured to provide data for both programs, where appropriate. The program will be carried out by:

- Broad evaluation and scoping studies to identify the feasible thorium/uranium recycle strategies
- Research and development activities to establish the feasibility of the selected strategies
- A design integration study to help identify necessary development areas and safeguards and nonproliferation aspects

- Development and testing of key systems, equipment, and components

Status. The thorium fuel recycle program has been expanded to assess, evaluate and initiate R&D on the use of thorium-based fuels in thermal reactors. It includes the development of both reprocessing and refabrication technology. Planned activities include definition of reference recycle processes for the major reactor concepts of interest, and conceptual design studies to identify R&D needs and to provide a basis for continued development should future decisions call for such development. R&D activities will include small and laboratory-scale and engineering-scale development and test programs for thorium recycle flowsheets and components. Because of similarities in processes and components, the alternative fuel cycle and thorium fuel recycle programs will share in use of facilities and components, and will mutually build upon the technologies developed.

Prior work, conducted under a much smaller effort, has indicated that the use of thorium in thermal reactors could show an advantage in a proliferation-resistant fuel cycle and in resource utilization. Preliminary planning for the expanded effort was initiated in FY 1977. More detailed planning was begun in FY 1978 and preparations were made for starting evaluations studies, conceptual designs and development work on pellet and particle thorium-based fuels.

The initial focus of the effort will be on the evaluation of thorium/uranium fuel cycle alternatives, which will identify the resource utilization and nonproliferation features. Upon completion of these evaluations a reference fuel cycle will be selected. The processing technology developed within this effort will be sufficiently flexible to accommodate a variety of fuel types (e.g., oxide or metal).

Initial scoping studies will be completed in May 1978. The basic data to support preliminary technical data summaries for the design integration study will be completed in July 1978. Study, research and development activities are scheduled so that a Design Integration Study Report will be completed by August 1981.

Conceptual design of a fuel refabrication facility will be initiated in FY 1978. This effort will provide conceptual process flowsheets in early FY 1979. Cold laboratory development of remote fabrication will also be started in FY 1978. A pilot plant facility to demonstrate remote fuel fabrication processes is planned to be operational by FY 1983.

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6.0 Private Processing Facilities

Background. The three privately owned United States spent-fuel reprocessing plants, Nuclear Fuel Services, Inc. (NFS), Allied General Nuclear Services (AGNS), and General Electric (GE), are currently inoperative. The NFS plant operated about six years but withdrew from the reprocessing business. AGNS and GE have never operated. FY 1978 legislation is expected to require DOE to develop options for resolving the NFS and AGNS situations consistent with national needs.

D. SPENT FUEL STORAGE

On October 18, 1977, the Department of Energy announced its policy to accept and take title to spent fuel from United States nuclear reactors and a limited quantity from reactors located in other countries. The spent fuel program will provide the necessary storage facilities and supporting activities to carry out this commitment.

President Carter's order to defer indefinitely the reprocessing of spent fuel in the United States and his urging that other nations join in this deferral requires retrievable storage of spent fuel for an indefinite period of time pending a decision either to recover and reuse that fuel or to dispose of it permanently. The various options for management of spent fuel are shown in figure 33.

Figure 33
U.S. Spent-Fuel Options

Objective. The objective of the spent fuel storage program is to provide for the safe storage of United States and some foreign spent fuel consistent with national nuclear energy goals and United States nonproliferation objectives.

This program should remove the uncertainty faced by utilities with respect to the disposition of spent fuel discharged from reactors, and it

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should reduce foreign pressures to reprocess spent fuel because of full storage basins.

Strategy. A one-time fee will be established and paid to the United States Government for storage and final disposal of each fuel element. The fee will be based on full cost recovery to the United States Government. If the fuel is eventually retrieved and reprocessed, an appropriate adjustment to the fee will be made after retrieval.

Current plans are for the first geologic storage facility for spent fuel to be available by the end of 1985. This facility will provide retrievable storage and will also be suitable for permanent isolation of radioactive waste. The facility is being developed under DOE's waste management program.

In the interim, DOE-approved away-from-reactor (AFR) water storage basins could be used to store spent fuel in a retrievable mode. DOE will seek to obtain interim storage services from private industry. If private services are not reasonably available, DOE will provide storage facilities. Interim storage should be available to accept spent fuel routinely in 1983.

Prior to 1983, DOE will consider accepting spent fuel in cases where reactors do not otherwise have sufficient storage capacity.

Implementation. The following activities are planned for implementing this program:

- Spent fuel storage services in large AFRs provided by private companies will be sought through a request for expressions of interest.
- An estimate of the quantity of United States and foreign spent fuel that is likely to be transferred to DOE for storage will be made.
- A determination will be made as to whether additional DOE storage capability in new or existing modified Government facilities will be needed.
- A request for proposals will be issued calling for private interim storage services, if interest has been shown by industry as a result of DOE's previous request for expressions of interest.
- A methodology will be established for determining the one time storage and disposal fee.
- Early in 1978, comprehensive legislation will be submitted by DOE to provide the legal authority for DOE to carry out this program.
- A draft generic environmental impact statement on the policy and program will be issued.
- The necessary mechanisms for the transfer of spent fuel from

interim storage facilities to a geologic facility for retrievable spent fuel storage will be established.

The storage of spent fuel in DOE approved facilities, the establishment of multinational and international fuel storage facilities, and technical assistance to international studies of fuel storage technology are efforts in direct support of United States nonproliferation initiatives.

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E. WASTE MANAGEMENT

1.0 Overview

Radioactive wastes are inevitable byproducts of the operation of nuclear reactors. During reactor operation, radioactive nuclides are generated as fission products from the fissioning of nuclear fuel, as transuranium (TRU) elements from the capture of neutrons in the nuclear fuel, and as induced activity from the capture of neutrons in fuel cladding, reactor structures, and reactor coolant. Radioactive waste management consists of the planning, execution, and surveillance of essential functions in the control of radioactive waste.

DOE's radioactive waste management program has the following three categories of activity:

- Management of Defense Waste
- Management of Commercial Waste
- Transportation

Defense Waste. As shown in figure 34, defense waste originates from the operation of nuclear materials production reactors, naval propulsion reactors, and government-owned research and testing reactors. Spent fuels from these reactors are reprocessed to recover uranium and nuclear products such as plutonium and tritium, which are used in nuclear weapons, and californium, which is a radiation source in a variety of applications. In reprocessing, over 99 percent of the fission products are accumulated in the high-level liquid waste. Finishing and weapons production operations involving plutonium are the source of most of the transuranic (TRU) waste.

DOE is responsible for the defense nuclear materials production program and related activities including waste management. The radioactive wastes are stored at a limited number of isolated sites, principally Savannah River, South Carolina; Richland, Washington; Idaho National Engineering Laboratory; Oak Ridge National Laboratory; and Los Alamos Scientific Laboratory.

The broad objectives of DOE's defense waste management program are to:

- Provide for the confinement of radioactive wastes to meet and maintain requirements of applicable laws and regulations
- Develop and demonstrate, at the earliest possible date, the technology for safe disposition of nuclear waste, to assure public confidence
- Provide, at the earliest practical date, long-term management

Figure 34

“Defense” Waste Originates from Materials-production, Naval-propulsion, and R&D Reactors

practices for DOE radioactive waste inventories that place minimal reliance on surveillance and maintenance

The strategy of the defense waste management program is structured to meet the needs for the safe management of the existing substantial inventories of defense radioactive wastes as well as wastes that may be generated in the future.

Commercial Waste. DOE's role in the management of commercial radioactive waste is to accept and provide long-term management for commercial radioactive waste for which the Federal Government is responsible, and to provide the technology for converting the waste to forms acceptable for long-term isolation. Except for minor quantities from R&D, and medical and industrial applications, the origin of the radioactivity in commercial radioactive waste is the operation of the nuclear power stations which now supply about 10 percent of the Nation's electricity.

Until recently, DOE's commercial waste management program was based on the assumption that spent commercial fuel would be reprocessed so that uranium and plutonium could be recovered and returned to the nuclear fuel cycle and that the separated fission products could be delivered to the Federal Government in the form of solidified high-level waste. However, on April 7, 1977, President Carter announced the nonproliferation policy, which defers reprocessing of commercial spent fuel indefinitely. The question of the near-term management of spent fuel was answered on October 18, 1977, with President Carter's announcement that the Federal Government would accept and take title to spent commercial fuel. DOE's commercial waste program has been redirected accordingly.

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The broad objectives of the commercial waste management program are to:

- Demonstrate the geological isolation of commercial spent fuel and to provide appropriate sites for such isolation
- Develop and demonstrate technologies to provide safe forms and packaging of radioactive wastes from different fuel cycles for initial handling, interim storage, processing, and permanent isolation consistent with Federal regulations and Presidential policy
- Develop the required spent fuel storage technology in order to meet the requirements of our nonproliferation policy

The strategy of the commercial waste management program is to conduct, as appropriate, geological studies and exploration, research, development, demonstration design and construction, and technical, economic, and environmental evaluations necessary to achieve the program objectives. The commercial waste management program will provide the capability for processing and disposing of high-level waste in the form of both reprocessed waste and spent fuel elements since a decision may be made to discard spent fuel and not reprocess it.

Transportation. Previously, the solution of problems related to the shipment of radioactive waste has largely been left to the private sector, which operates under the regulations of the Department of Transportation and the Nuclear Regulatory Commission. However, DOE's waste management strategy may require the shipment, beginning in the mid-1980s, of substantial quantities of radioactive waste to central repositories. Therefore, timely availability of the necessary transportation facilities and equipment is essential to the success of the waste management program. Accordingly, to assure the necessary coordination of transportation related activities, a transportation program has been established.

Integrated plans and programs are now in an early stage of formulation.

The following table summarizes the operating funding levels by category for FY 1977 through FY 1979.

WASTE MANAGEMENT	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Management of Defense Waste *	\$99,889	\$125,047	\$165,000	\$39,953
Management of Commercial Radioactive Waste	82,730	158,500	152,100	(6,400)
TOTAL	\$182,619	\$283,547	\$317,100	\$ 33,553

*Not an HS&T program.

2.0 Management of Defense Waste

Since 1943, radioactive wastes have been generated in the nuclear programs that produce and utilize nuclear materials for national defense and other federally-sponsored applications. Current waste inventories and projections through 1985 are given in Table 6.

Table 6. DOE Defense Waste Inventories

	JAN. 1, 1977	DEC. 1985
HIGH-LEVEL LIQUID WASTE (Millions of Gallons)		
Liquid in Tanks	34.9	15.3
Salt Cake and Sludge in Tanks (at Savannah River and Richland)	38.6	53.7
Calcine in Bins (Idaho)	0.4	1.1
Total	73.9	70.1
ORNL INTERMEDIATE-LEVEL LIQUID WASTE (Millions of Gallons)		
Liquid in Tanks	0.2	0.1
Sludge in Tanks	0.4	0-
Liquid Mixed with Cement Disposed of as Grout in Shale Formation Underlying Site	1.1	2.5
Total	1.7	2.6
LOW-LEVEL & TRU SOLID WASTE (Millions of Cubic Feet)		
Non-TRU in Land Burial Sites (17 Sites)	29.2	40.45
TRU in Land Burial Sites (6 Sites)	13.0	13.0
TRU Retrievably Stored (6 Sites)	1.5	3.75
Total	43.7	57.2

High-level wastes are generated and stored at Savannah River, Richland and Idaho. Most of the high-level waste at Savannah River and Richland is a byproduct of the reprocessing of production reactor fuel to recover uranium and plutonium for defense purposes. The high-level waste at Idaho is a byproduct of the reprocessing of irradiated research and naval reactor fuels. The aggregate inventory of high-level waste contains approximately 99 percent of the fission products and 35 percent of the transuranics in the total inventory of defense waste.

The specific objectives of the defense waste program are to:

- Complete the development and demonstration of improved systems for treatment of transuranic (TRU) waste by end of 1980
- Have available by 1983 the technology, methods, procedures for additional collecting, immobilizing and storing of certain volatile and gaseous radioactive wastes should it be required that defense program facilities meet recent Federal emission control regulations (1977 Clean Air Act - PL 95-95 and 1976 EPA Standards on Radiation Protection of Nuclear Power Operations)
- Have in operation by mid-1985 a facility to demonstrate the technical and operational principles for permanent geological disposal of defense TRU wastes

FUEL CYCLE

- Complete by the end of 1986 the construction of facilities and the introduction of advanced operational practices for improved interim management of high-level waste
- Complete by the end of 1986 the disposal by hydrofracturing of the backlog of ORNL intermediate-level waste and the decommissioning of unneeded intermediate-level waste management facilities
- Complete by the end of 1987 the upgrading of shallow land burial operations and the stabilization of shallow land burial sites to meet long-term management criteria established for confinement of radio-activity with minimal maintenance and surveillance requirements
- Complete by about the end of the century the long-term disposition of the backlog of defense high-level wastes.

The defense waste program subcategories of effort are:

- Long-term management of high-level waste
- Management of low-level solid waste
- Management of transuranium contaminated (TRU) solid wastes
- Management of ORNL intermediate-level liquid waste
- Management of airborne waste
- Development of Waste Isolation Pilot Plant (WIPP)

The following table summarizes the operating funding levels by sub-category for FY 1977 through FY 1979.

MANAGEMENT OF DEFENSE WASTE *	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Interim Management of High-Level Waste	\$61,532	\$63,580	\$78,385	\$14,805
Long-Term Management of High-Level Waste	12,923	20,629	35,984	15,355
Management of Low-Level Solid Waste	11,717	18,851	24,751	5,900
Management of Transuranium-Contaminated Solid Waste	4,117	5,289	2,550	(2,739)
Management of ORNL Intermediate-Level Liquid Waste	57	220	300	80
Management of Airborne Waste	3,143	4,770	2,530	(2,240)
Waste Isolation Pilot Plant	6,400	11,708	20,500	8,792
TOTAL	\$99,889	\$125,047	\$165,000	\$39,953

*Not an HS&T program.

2.1 Interim Management of High-Level Waste

"High-level waste" is a generic term for a variety of aqueous wastes from fuel reprocessing and their solidified derivatives, e.g., acidic aqueous waste, alkaline aqueous waste, calcine, crystallized salts, insoluble precipitates, salts of cesium and strontium extracts, and coating waste from chemical decladding of fuel elements. The latter comprises most of the total volume of the inventory of defense high-level waste but contains comparatively little radioactivity.

The strategy of this program is to provide interim storage of high-level waste in a way that is compatible with alternatives under consideration for long-term management and that minimizes the potential for release to the environment, specifically: (1) to reduce the quantity of waste stored as liquid to the extent possible by evaporation at Savannah River and Richland and by calcination at Idaho; and (2) to limit operationally required storage of liquid to high integrity tanks of advanced design.

The current activities are:

- Improved Containment:
 - Double-shell waste tanks are being constructed to meet needs for replacement of old waste tanks and additional storage capacity at Savannah River and Richland.
 - Existing waste tanks are being evaluated to determine their acceptability for continued use at Savannah River and Richland.
 - Improvements are being made to existing interim storage facilities at Savannah River, Richland, and Idaho.
 - Stainless steel underground bins are being constructed to meet needs for additional storage capacity for calcined waste at Idaho.
- Waste Processing and Solidification:
 - Excess water is being evaporated from waste at Savannah River and Richland to reduce the volume requiring storage and to convert liquid to immobile salt cake to reduce the potential for releases to the environment; the waste backlog at Richland has been evaporated.
 - New waste evaporation facilities are being constructed at Savannah River to replace existing facilities.
 - Improved technology is being developed and demonstrated at Savannah River for removing waste from old storage tanks.
 - Cesium-137 and strontium-90 are being separated from high heat generating waste and encapsulated separately at

FUEL CYCLE

Richland, to allow storage of salt cake in tanks not equipped with cooling coils; waste backlog has been processed.

- Wells are being installed in the salt cake in storage tanks at Richland to remove interstitial liquor for further processing.
- The waste calcining facility at Idaho is being operated to convert liquid waste to solids.
- A new waste calciner is being constructed at Idaho to replace the existing calciner which is reaching the end of its serviceable life.
- Process development to improve waste handling is a continuing activity at Savannah River, Richland, and Idaho.

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2.2 Long-Term Management of High-Level Waste

The strategy of this program is to:

- Evaluate on a site-by-site basis, taking into account variations in waste and site characteristics, the alternatives for the long-term disposition of the high-level wastes that would be practical to implement by about the end of the century

- Select and implement the preferred alternative or alternatives, giving due consideration to public acceptability, safety and economics

Three basic alternatives appear reasonable for implementation by the end of this century, namely:

- (1) Retrieve waste from interim facilities, process as required to produce a stable solid form, package in canisters and store indefinitely in surface facilities designed to minimize future maintenance and surveillance
- (2) Retrieve, process, and package essentially as in Alternative (1) and dispose of waste permanently in a geologic repository
- (3) Continue interim practices indefinitely with evolutionary improvements

A comprehensive R&D program supporting each of the three alternatives is underway to develop the technical information needed in the selection process. Important corollary strategic considerations are:

- Alternatives (1) and (3), which would defer permanent disposal while retaining the option, are consistent with:
 - Allowing time for additional decay of radioactivity to reduce the risk and expense of permanent disposal operations
 - Allowing time for development of advanced disposal concepts
- While not integral and essential parts of the long-term program, federally sponsored work is continuing on advanced concepts such as space disposal, seabed disposal and transmutation. Developments occurring in these areas are assessed periodically.
- The three alternatives are not mutually exclusive. Given the variety of high-level wastes and the range of site characteristics and interim storage conditions, different alternatives may be selected for different sites.

The current activities are:

- Technology Development:
 - The technology base necessary for the evaluation of alternatives is being established. Savannah River has the lead for waste vitrification.
 - Technology is being developed and demonstrated at each of the three sites for the removal of waste from interim storage facilities.

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- Alternative Selection
 - Technical Defense Waste Documents describing and assessing the practical alternatives have been issued for each of the three sites.
 - Programmatic Environmental Impact Statements covering the practical alternatives are being prepared for each of the three sites.
- Facility Construction:
 - Conceptual design studies are underway on facilities needed at each of the three sites to implement the practical alternatives.
 - Engineering studies and tests of equipment for removal of waste from interim storage facilities are being conducted at each of the three sites.

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2.3 Management of Low-Level Solid Waste

"Low-level solid waste" is a byproduct of special nuclear materials production, utilization, and R&D activities. Typical examples of solid low-level waste are discarded equipment and metals, filters from the cleanup of gaseous wastes, ion exchange resins from the cleanup of liquid wastes, liquid wastes that have been converted to solid form by techniques such as mixing with cement to make concrete, and miscellaneous trash such as paper, rags, glassware, and protective clothing. (Low-level liquid waste is decontaminated and released under controlled conditions.)

The strategy of this program is to:

- Continue shallow land burial of low-level solid wastes
- Develop a thorough technical understanding of potential mechanisms and rates for radioactive releases from shallow land burial sites, and apply this knowledge through criteria, procedures and standards to burial ground sitings, operation, and long-term stabilization

The disposition of low-level solid waste by shallow land burial is an established practice. However, the understanding of potential migration mechanisms and rates for fission products and transuranics in various burial ground environments is largely empirical. The strategy for the long-term management of low-level solid waste is burial in a location and in a manner that requires little maintenance and surveillance to assure that established health and safety standards for radioactive releases are maintained.

"Stabilization" refers generically to steps taken to limit the effect of potential mechanisms for radioactive releases from burial grounds to the biosphere. The stabilization must remain effective in assuring that health and safety standards are met and maintained while the radioactivity of the waste is decaying to insignificant levels. Effective burial ground stabilization for several hundred years may be required.

The current activities are:

- Disposal Technology:
 - A multi-laboratory program was started in FY 1977 to improve the scientific basis for disposal methods and develop disposal criteria by the end of FY 1980. Work is in progress on establishing a technical data file, site selection procedures, site operating practices, waste preparation requirements and a systems analysis method to optimize procedures.
 - Preparations are being made to conduct at least two small-scale demonstrations of the shallow land burial technology being developed. One demonstration is to be in an arid environment and the other in a relatively humid environment.
- Disposal Operations:
 - Shallow land burial operations are continuing on a routine basis.
 - An assessment has been made of the adequacy of existing DOE shallow land burial sites and a report on the findings is being prepared for issuance early in FY 1978.
 - Based on the findings of the assessment of shallow land

burial sites, some priorities have been established and improvements are being made.

- Preliminary plans are being developed for implementing the initial DOE shallow land burial criteria scheduled to be issued in mid-FY 1978. It is expected that significant changes will be required in present practices at some sites.
- Burial Ground Minimum Surveillance:
 - At each disposal site, characterization is underway of the geohydrological environment and the migration behavior of the wastes contained. Based on the results of these studies the stabilization requirements of each burial site will be established.
 - Burial ground stabilization technology is being developed and some field tests are being conducted.

PROGRAM: FUEL CYCLE SUBPROGRAM: WASTE MANAGEMENT CATEGORY: MANAGEMENT OF DEFENSE WASTE SUBCATEGORY: MANAGEMENT OF LOW-LEVEL SOLID WASTE																					
TASK	CY 1977			CY 1978			CY 1979			CY 1980			CY 1981			CY 1982			CY 1983		
	FY 1977			FY 1978			FY 1979			FY 1980			FY 1981			FY 1982			FY 1983		
DISPOSAL TECHNOLOGY						FIELD DEMONSTRATION					FINAL CRITERIA					DEMONSTRATIONS					
DISPOSAL OPERATIONS					INITIAL CRITERIA						SITE COMPLIANCE					SITE UPGRADING					
BURIAL GROUND MINIMUM SURVEILLANCE											STABILIZATION TECHNIQUES						STABILIZATION				
														PROGRAM PLAN							

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2.4 Management of Transuranium-Contaminated Solid Waste

"Transuranic (TRU) solid waste" is a special case of low-level solid waste. The distinction is the presence of long-lived transuranic radionuclides of high specific radiotoxicity at levels necessitating special consideration. Plutonium-239, the radionuclide of principal concern, has a half-life of 24,000 years. The routine disposal of TRU solid waste at DOE sites by shallow land burial was stopped in 1970. Since that time the waste has been stored in a way that permits ready retrieval for a period of 20 years to allow time for decisions to be made on its permanent disposal.

The strategy for this program is to:

- Develop technology for treating TRU waste to meet the range of possible restrictive criteria for geologic disposal
- Provide facilities and equipment for treatment, packaging and

transport of TRU waste to a Federal geologic repository

The strategy deals with the disposition of three categories of TRU waste, namely: (1) previously buried waste, (2) retrievably stored waste, and (3) future wastes.

Some previously buried waste may require exhumation as a prerequisite to the stabilization program. Each location where TRU waste or TRU-contaminated soil is buried will be evaluated individually. Since most burial grounds will probably become part of dedicated areas at DOE sites, it is not expected that extensive exhumation will be necessary to maintain health and safety standards. The Waste Isolation Pilot Plant is being developed in a southeastern New Mexico bedded salt formation to provide the capability for permanent geological disposal of defense TRU waste. This plant is discussed in a later section of this document.

The current activities are:

- TRU Waste Technology Development:
 - Methods being developed include volume reduction systems such as incineration, smelting and electropolishing to remove contamination from metals, and immobilization of waste in various matrixes such as polymers, concrete, metals and glass.
- TRU Waste Treatment Facility Construction and Operation:
 - Studies are being conducted to determine the facilities required for processing and packaging the retrievably-stored TRU waste for geologic disposal. Conceptual designs will be prepared and evaluated for specific sites and central facilities.
- TRU Retrievable Storage Operations:
 - Routine retrievable storage of TRU waste is continuing.
 - A comprehensive plan is being prepared for retrieval and treatment as necessary for permanent disposal in a geologic repository.
- TRU Waste Exhumation Operations:
 - Studies are being conducted to analyze the long-term environmental behavior of currently buried waste and to evaluate the alternatives for both exhumation and stabilization of the waste where buried. Experimental exhumation projects are underway to determine methods for exhumation and the associated risk.

2.5 Management of ORNL Intermediate-Level Liquid Waste

"Intermediate-level liquid waste" is the traditional term used at Oak Ridge National Laboratory (ORNL) for aqueous waste produced in their R&D activities. To the extent such wastes are produced at other sites, they are either concentrated and managed as high-level waste or decontaminated and the residues managed as solid low-level waste.

Hydrofracturing, with injection of a liquid waste/cement grout under high pressure, has been used to dispose of this waste in shale formations underlying the ORNL site. It is both the most economical and the lowest risk alternative for achieving the long-term waste management goals. The strategy of this program is to continue this practice.

The current activities are:

- Hydrofracture Operations:
 - Periodic injections of intermediate-level waste/cement grout continue at a frequency of one or two per year.

- An Environmental Impact Statement has been issued for management of ORNL intermediate-level waste.

- **Waste Handling System:**

A second waste evaporator, stainless steel tanks and double containment transfer lines are being installed at ORNL to collect, reduce the volume and store the intermediate-level liquid wastes pending disposal by hydrofracture. These items are replacements for existing handling systems that have exceeded their design life.

- **New Disposal Facility:**

The present hydrofracture disposal facility is an old demonstration facility. Preparatory activities are underway for construction to begin in FY 1978 of a new hydrofracture disposal facility, which will be able to handle the waste generated for the next 15 years as well as the sludge that has been accumulated in the interim storage tanks.

- **Decommissioning:**

After the new hydrofracture facility becomes available and liquid waste has been disposed of, the sludges will be disposed of and the old facility will be decommissioned. Experiments are underway to determine the composition of the sludges in the waste tanks, methods to remove them and the best means to inject them using the hydrofracture process.

PROGRAM: FUEL CYCLE														
SUBPROGRAM: WASTE MANAGEMENT														
CATEGORY: MANAGEMENT OF DEFENSE WASTE														
SUBCATEGORY: MANAGEMENT OF ORNL INTERMEDIATE-LEVEL LIQUID WASTE														
TASK	CY 1977			CY 1978		CY 1979		CY 1980		CY 1981		CY 1982		CY 1983
	FY 1977		FY 1978		FY 1979		FY 1980		FY 1981		FY 1982		FY 1983	
ORNL ILW														
NEW HYDROFRACTURE FACILITY CONSTRUCTION					△				▽					

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2.6 Management of Airborne Waste

The strategy of this program is to increase the service life and reliability of filters; and to develop improved technology for additional collection, fixation, and long-term management of gaseous radionuclides in anticipation of regulatory actions that would require further reductions in airborne radioactive releases from defense program facilities.

Available technology and practices are adequate to meet current health and safety standards. The airborne waste program is aimed

FUEL CYCLE

primarily at cost-effective improvements, quality assurance, and new capability in areas where more restrictive standards seem likely to apply in the future.

The current activities are:

- Filter Service Life Extension:
 - To reduce cost and waste, iodine absorbents are being developed that have substantially longer service life and can be regenerated.
 - More durable filters are being developed for service in off-gas streams from radioactive processes that emit acid vapors and/or high temperature gases.
 - Several prefilter concepts are being developed for capturing radioactive particulates near their source. These concepts will extend the service life of HEPA filters and thus reduce the costs for their replacement and disposal.
- Gas Monitoring:
 - Improved monitoring systems are being demonstrated for specific long lived airborne radioactive constituents in the offgas of waste processes, particularly iodine-129, carbon-14 and tritium.
 - Improved particle measurement technology is being developed for high temperature applications.
- Gaseous Waste Treatment:
 - Technology is being developed to remove tritium from air and water, to immobilize it and to store it safely.
 - The state of the art for limiting emissions of iodine-129, krypton-85, and carbon-14 is being assessed. Preliminary studies are being made on the adaptation of advanced technology for use at DOE sites.
- Filter Quality Assurance:
 - The operation of three filter test stations for testing of HEPA filters prior to use in DOE plutonium facility filtration systems is continuing.
 - Improved filter test methods are being developed for DOE radioactive iodine and particle removal filtration systems. This work also supports the development of filtration standards, regulations, and test procedures.

PROGRAM FUEL CYCLE																						
SUBPROGRAM WASTE MANAGEMENT																						
CATEGORY MANAGEMENT OF DEFENSE WASTE																						
SUBCATEGORY MANAGEMENT OF AIRBORNE WASTE																						
TASK	CY 1977			CY 1978			CY 1979			CY 1980			CY 1981			CY 1982			CY 1983			
	FY 1977	FY 1978	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983	FY 1977	FY 1978	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983	FY 1977	FY 1978	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983	
FILTER LIFE EXTENSION																						
NEW IODINE ABSORBENT FIELD TESTS																						
ELECTRONIC PREFILTER																						
ACID RESISTANT																						
IODINE RETENTION																						
HIGH TEMPERATURE																						
FILTER QUALITY ASSURANCE																						
FILTER MATERIALS TESTING PROCEDURES																						
IN PLACE FILTER TEST METHOD																						
AIRBORNE WASTE MANAGEMENT																						
FIELD TESTING MONITORS																						
GASEOUS WASTE TREATMENT																						
KRYPTON WASTE FORM EVALUATION																						
DEMONSTRATION OF KRYPTON SEPARATION TECHNOLOGY																						
TRITIUM IMMOBILIZATION DEMO																						

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2.7 Development of Waste Isolation Pilot Plant

The strategy of this program is to:

- Continue the Waste Isolation Pilot Plant (WIPP) project to build a geologic disposal facility in a southeastern New Mexico bedded salt formation
- Request NRC to license the WIPP
- WIPP will demonstrate the permanent disposal of defense TRU waste and provide test capability for experiments with various forms of high-level waste in an actual repository environment.

The current activities are:

- Site Evaluation:
 - Geoscientific and environmental investigations are being conducted to characterize and evaluate the suitability of the WIPP site.
 - Completed the evaluation of potash resources associated with WIPP.

FUEL CYCLE

- Land Protection and Acquisition:
 - The commercial development of oil, gas, and potash resources within the site is currently being restricted.
 - Plans are being developed for action (legislative or administrative) to effect land withdrawal and purchase of mineral leases following the approval of the site.
- Facility Design and Construction:
 - Bechtel has been selected as the A-E for WIPP. Title I engineering is scheduled to begin in February 1978, based on the conceptual design and design criteria completed in FY 1977. Construction is scheduled to begin in FY 1980 and to be completed in FY 1984.
- Waste Acceptance Criteria Development:
 - Work is continuing on the development of DOE TRU waste form and packaging performance criteria for waste to be disposed of at WIPP. These criteria, which are to be completed in FY 1979, will be based on requirements for both short-term operational safety and long-term geologic integrity.

PROGRAM: FUEL CYCLE														
SUBPROGRAM: WASTE MANAGEMENT														
CATEGORY: MANAGEMENT OF DEFENSE WASTE														
SUBCATEGORY: WASTE ISOLATION PILOT PLANT														
TASK	CY 1977		CY 1978		CY 1979		CY 1980		CY 1981		CY 1982		CY 1983	
	FY 1977	FY 1978	FY 1978	FY 1979	FY 1979	FY 1980	FY 1980	FY 1981	FY 1981	FY 1982	FY 1982	FY 1983	FY 1983	
REPOSITORY DEVELOPMENT & OPERATIONS (WIPP)			⬡	OPERATING CONTRACTOR				⚠	CONSTRUCTION					

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3.0 Management of Commercial Radioactive Waste

The commercial waste management program has been modified in FY 1978 in accordance with President Carter's policy deferring reprocessing of commercial spent fuel. In the terminal storage program an operating repository in salt by 1985 remains the primary goal. However, first consideration will be given to spent fuel elements for storage or disposal rather than solidified wastes from fuel recycle reprocessing. In the area of waste processing technology, activities are directed toward:

- Identification and characterization of wastes
- Development and demonstration of waste forming and packaging processes

- Design, construction and operation of waste handling system demonstrations

In addition, the program includes R&D support for the solidification and long-term disposition of the existing approximately 600,000 gallons of commercial high-level waste stored at the Nuclear Fuel Services facility in West Valley, New York.

The specific objectives of DOE's commercial radioactive waste program are to:

- Identify and qualify a site and construct an NRC-licensed repository with capability for retrievable storage of commercial spent fuel and the disposal of high-level and transuranium contaminated waste
- Develop and demonstrate technologies to provide safe forms and packaging of radioactive wastes from different fuel cycles for initial handling, interim storage, processing, and permanent isolation
- Provide for the increased spent fuel storage requirements resulting from our nonproliferation nuclear power policy
- Issue the draft Generic Environmental Impact Statement for commercial radioactive waste management by mid-1978
- Complete the demonstration of the vitrification of high-level waste by the end of 1979
- Demonstrate technology for incineration of transuranium-contaminated waste by the end of 1978.

The commercial radioactive waste program subcategories of effort are:

- National waste terminal storage repositories, including packaging and buffer storage capabilities
- Waste processing techniques
- Waste vitrification demonstration
- Supporting studies and evaluations

The following table summarizes the operating funding levels by subcategory for FY 1977 through FY 1979.

MANAGEMENT OF COMMERCIAL RADIOACTIVE WASTE	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Terminal Storage Research and Development	\$43,303	\$117,000	\$125,500	\$8,500
Waste Immobilization Research and Development	20,658	22,000	21,100	(900)
Waste Vitrification Demonstration	9,420	16,500	0	(16,500)
Supporting Studies and Evaluations	9,349	3,000	5,500	2,500
TOTAL	\$82,730	\$158,500	\$152,100	\$(6,400)

3.1 Terminal Storage Research and Development

Deep geologic repository concepts are being developed in alternative geologic formations for the disposal of radioactive waste with the primary emphasis on spent nuclear fuel as the waste form. Two potential repository sites are to be identified in 1979. These two sites are expected to be in salt formations since there is an extensive scientific and engineering foundation for these as a result of federally-sponsored work going on since the late 1950s. Geologic salt areas that appear particularly promising technically are the Permian Basin, the Salina Salt Basin, the interior salt domes of the Gulf Coast, and the Paradox Basin. The location of these areas is shown in figure 35. The first repository site is scheduled for final selection by the end of FY 1979, with initial operation to begin in 1985.

General geologic, hydrologic, and engineering studies will continue in granite, basalt, shale, clay, etc., to develop understanding of their suitability. The DOE reservations at Richland and Nevada have formations of shale, granite and basalt of particular interest that may be particularly well suited for retrievable storage. Evaluation of the geology of these DOE reservations has been accelerated and specific information on their potential suitability as terminal waste repositories should be available in about two years.

The current activities are:

- Geologic Studies:
 - Work continues to determine the fundamental suitability of a number of rock types for geologic waste repositories. The work includes the review of available information and, for promising areas, detailed studies such as aerial surveys, geophysical surveys, and exploration drilling.
 - Project managers have been selected to coordinate and oversee the rock salt geologic studies, namely: Law Engineering Testing Company for the interior Gulf Coast

Figure 35
Salt Basins Being Investigated for Terminal Storage of Radioactive Wastes

salt domes, Stone and Webster for the Salina Basin, and Woodward-Clyde for the Paradox Basin in Utah. The Texas Bureau of Economic Geology guides the work on the Permian Basin in Texas, but with a somewhat limited scope as compared to the contractors designated for the other formations.

- Technical Support Studies:
 - Generic studies continue in areas such as heat transfer and thermal analysis, waste/rock interactions, rock mechanics, bore-hole plugging and radionuclide migration.
- Engineering Studies:
 - Preparations are being made for in-situ experiments to determine the effects of thermal heat and radiation on host rock formations and mine stability.
- Spent Fuel Handling:
 - Packaging and backup surface storage capability and technology are being developed. Analyses are being made of passive cooling concepts (sealed cask and dry well), with a concept being selected for conceptual design during the FY 1978-79 period.

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- System Studies:
 - Studies on high-level waste canister envelopes and spent fuel canisters are scheduled to be completed in FY 1978.
 - Studies related to waste transportation will continue.
 - Economic studies will continue related to development of storage/disposal charges to utilities.
- Regulatory Affairs:
 - A comprehensive program continues to assure that all regulatory, statutory, environmental and safety requirements are met on a timely basis.
 - Contractors have been selected to coordinate and oversee regulatory activities in the areas of the rock salt formations of interest, namely: Nuclear Utility Services (NUS) for the Salina Basin and the Permian Basin and Bechtel for the interior Gulf Coast Domes and the Paradox Basin. NUS will perform broad overall planning and definition of regional programs.

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3.2 Waste Immobilization Research and Development

The strategy of this program is to assess the waste management requirements of alternative proliferation-resistant fuel cycles and to develop technology applicable to the safe management of wastes that would be generated by these fuel cycles, to the extent necessary to support their evaluation.

The current activities are:

- Processing Liquid Waste:
 - Investigation of ultrafiltration and bacteriological/biological processes for concentrating waste as alternatives to non-high-level waste evaporation is continuing. Concreting is being studied as a method for solidifying the resulting sludges.
 - The effects of alternative fuel cycles on high-level waste forms are being evaluated.
- Processing of Non-high-level Solid Waste:
 - Work is continuing on the demonstration of the incineration of combustible TRU waste. This activity is scheduled to be completed in 1978.
 - Studies continue on smelting as a method for reducing the volume of spent fuel cladding hulls.
 - The use of electropolishing for decontaminating metal surfaces is being tested on a pilot scale. The tests are expected to be completed in 1978.
- Processing of Gaseous Waste:
 - Technology is being developed for the solidification of krypton.
 - Studies continue on the fixation and packaging of carbon-14 and iodine-129.

PROGRAM: FUEL CYCLE														
SUBPROGRAM: WASTE MANAGEMENT														
CATEGORY: MANAGEMENT OF COMMERCIAL RADIOACTIVE WASTE														
SUBCATEGORY: WASTE IMMOBILIZATION RESEARCH AND DEVELOPMENT														
TASK	CY 1977		CY 1978		CY 1979		CY 1980		CY 1981		CY 1982		CY 1983	
	FY 1977	FY 1978	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983							
CONTROLLED AIR INCINERATION TESTS		▲												
NON-RADIOACTIVE CERAMIC MELTER VITRIFICATION DEMONSTRATION		▽												

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3.3 Waste Vitrification Demonstration

The strategy of this program is to confirm solidification and packaging technology by preparing full-scale canisters of vitrified waste.

In the initial demonstration, pilot plant facilities at Richland will be

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used to vitrify and encapsulate the waste products.

The current activities are:

- Design, Construction and Modification of Facilities:
 - The modification of existing facilities, construction of additional facilities needed, and installation of equipment are underway.
 - Plans are being prepared for nonradioactive tests in the latter part of 1978 and for operations at full levels of radioactivity to begin before the end of the year.

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3.4 Supporting Studies and Evaluations

The strategy for this program is to maintain a continuing effort of waste-management-related studies on alternative systems and health, safety and environmental concerns, as a mechanism for early identification of issues and approaches for dealing with them.

The current activities are:

- Health, Safety, and Environmental Studies:
 - Work is nearing completion on the Generic Environmental Impact Statement on commercial radioactive waste management. It is expected that the draft EIS will be issued early in 1978.

PROGRAM: FUEL CYCLE SUBPROGRAM: WASTE MANAGEMENT CATEGORY: MANAGEMENT OF COMMERCIAL RADIOACTIVE WASTE SUBCATEGORY: SUPPORTING STUDIES AND EVALUATIONS																															
TASK				CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
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VI. SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS

A. PROGRAM OVERVIEW

The unique characteristics of nuclear-powered electric generators--compact size, light weight, and long life--enable operation of the sensing, analytical, and communication systems of spacecraft, satellites, and other remotely located devices for long time periods without relying on external sources of energy. Thus the Space and Terrestrial Compact Power Systems Program, by its technical initiatives, supports the national security as well as the civilian scientific exploration and exploitation of space. By applying the technology developed for space applications, the program also fosters beneficial terrestrial uses of reactor byproducts and of special nuclear materials recoverable from reactor wastes.

Objectives. The principal objectives of the Space and Terrestrial Compact Power Systems Program are to:

- Deliver qualified nuclear-electric systems to other Federal agencies for both outerplanetary space missions and earth-orbiting satellites
- Provide power units for scientific experiments, telemetry, navigation, communication, or surveillance purposes
- Develop analogous isotope-powered "nuclear batteries" for terrestrial deployment at remote or undersea sites
- Investigate various isotopes and their utilization using the skills developed for the handling and applications of plutonium-238

During the last 20 years, the availability and development of the plutonium-238 isotope has been stimulated by its successful application as an energy source in systems for space nuclear power.

Strategy. These basic objectives of the program are being met by implementing subprograms in the area of:

- Space applications
- Technology base and special projects

By means of research and development, the program delivers environmentally acceptable, operationally safe, and technically qualified nuclear energy systems to Federal user-agencies for earth-orbital and outerplanetary space missions as well as for terrestrial applications.

For national-security missions, the program meets Department of Defense requests for improved isotope and reactor power-systems technology for future defense operations and communications, for which a

SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS

compact and reliable power supply is a principal factor in determining mission viability.

Improved power supplies are also essential for the National Aeronautics and Space Administration's unmanned planetary explorations to obtain scientific data on the nature and origin of our solar system and universe. In 1976 nuclear electric generators provided the power for the first scientific exploration of the surface of Mars. The outerplanetary missions to the planet Saturn and beyond launched in August and September 1977, are the beginning of a program of scientific exploration of the remote regions of the solar system planned by NASA for the coming decades. These missions require higher-powered spacecraft with reduced weight and increased endurance and reliability at reduced cost.

The directions the program takes are determined both by user-agency product requests and by initiatives indicated by user trends and technology developments. Figure 36 shows the general approach for developing systems for flight or terrestrial purposes. Before a major application effort starts, cost/benefit analyses are conducted when the development risks and potential benefits can be meaningfully identified in advance; these analyses include mission and commercial requirements, environmental, health, and safety requirements, and specific needs (such as reliability, longevity, and survivability), and consider nonnuclear as well as nuclear alternatives. Program work is undertaken only after appropriate interagency formal agreements. Liaison between DOE, NASA, and other user agencies is pursued at several management levels through the medium of coordinating boards with both technical and administrative membership. These interagency coordinating groups provide the information upon which the direction of the advanced research and technology activities is focused.

Participants. Mission-system development work is usually performed by industrial contractors. Fuel and safety research and development, special metals technology, capsule hardware fabrication, and heat-source assembly are done in government-owned contractor-operated laboratories and plants (such as Los Alamos Scientific Laboratory, Oak Ridge National Laboratory, Savannah River Laboratory, and the Mound Facility) because those operations depend on the nuclear facilities, equipment, and expertise maintained for nuclear fuel and weapons programs. Activities performed by industrial or institutional contractors include:

- Quality assurance and reliability monitoring
- Isotope materials and capsule development
- Environmental-impact and safety evaluation
- Advanced studies on thermoelectric materials, mission-requirements projections, and technical consultation

As the technology and support programs advance and are integrated into a system technology, user hardware requirements are specified and the system development and fabrication activities begin. The appropriate environmental assessments and impact statements, preliminary and final

Figure 36
Logic Flow of Space and Terrestrial Compact Power Systems Program

safety analysis reports, safety testing, evaluations and fuel fabrication and assembly are done concurrently with the system development. Upon completion and acceptance testing, the end product is delivered to the user for operational use.

Management. Program planning, guidance, and policy formulation are largely pursued at DOE headquarters; contractor administration is shared by headquarters and the DOE field offices at Albuquerque, Chicago, Oak Ridge, San Francisco and Savannah River. Plans being reviewed would place more of this responsibility in the field offices.

The first-of-a-kind modular heat sources of the future are proposed to be funded by DOE; the costs of flight units are planned to be reimbursed to DOE by the user agencies requesting them.

The following table summarizes the funding levels by subprograms for the FY 1977 to FY 1979 period.

SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS

SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
SUBPROGRAMS				
Space Applications				
Operating	\$20,680	\$31,008	\$39,400	\$8,392
Equipment	2,020	3,400	3,800	400
Construction	0	0	0	0
Subtotal	22,700	34,408	43,200	8,792
Technology Base and Special Projects				
Operating	11,322	14,842	4,000	(10,842)
Equipment	1,545	1,280	900	(380)
Construction	0	0	0	0
Subtotal	12,867	16,122	4,900	(11,222)
TOTAL	\$35,567	\$50,530	\$48,100	\$(2,430)

B. SPACE APPLICATIONS

1.0 Overview

The Space Applications subprogram supports other Federal agencies by providing progressively improved nuclear power sources to meet mission objectives. Development work reduces weight, enhances power levels, improves reliability and mission lifetime, and reduces cost to the user.

Objectives. The objectives of this subprogram are to:

- Support the power-source requirements of the NASA outer-planetary exploration program or other planned missions
- Support the power-source requirements of the DOD in its space and/or terrestrial activities, including communication satellites, surveillance systems and navigation aids
- Extend the range and potential applicability of nuclear power sources to meet future requirements

Strategy. The strategy of the Space Applications subprogram is to attain the objectives by:

- Providing safe, reliable, and long-lived nuclear-electric systems to space-mission agencies
- Using fully the accumulated experience in radioisotopes, thermoelectric elements, and specialized metals and alloys
- Developing, designing, analyzing, and testing materials and components to meet mission-performance requirements
- Maintaining rigorous standards of quality control and environmental safety
- Comparing actual with predicted performance so as to guide improved design
- Developing technology directed toward increased power, improved conversion efficiency and safety, and reduced cost
- Studying alternate power sources, such as reactors, to anticipate potential high-power-demand applications

The chronology of successful missions in Table 7 illustrates the adaptability of nuclear power to a variety of orbital and space applications.

Work in the Space Applications subprogram is conducted under the categories of:

SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS

Table 7. Space Missions that Have Used Nuclear Electric Generators

MISSIONS	LAUNCH	POWER SOURCE	POWER, WATTS ELECTRIC	MISSION PURPOSE
Transit 4A, 4B	1961	RTG*	27	Navigation satellite
Transit 5BN 1, 2	1963	RTG	25	Navigation satellite
Snapshot	1965	Reactor	500	Flight test
Nimbus III	1969	RTG	28	Weather satellite
Apollo 12	1969	RTG	74	Lunar science expts
Apollo 14, 15	1971	RTG	74	Lunar science expts
Apollo 16, 17	1972	RTG	74	Lunar science expts
Pioneer 10	1972	RTG	80	Jupiter exploration
Transit	1972	RTG	36	Navigation satellite
Pioneer 11	1973	RTG	80	Jupiter/Saturn exploration
Viking I & II	1975	RTG	85	Mars lander exploration
LES 8/9 #	1976	MHW° RTG	310	Communications satellite
Voyager I & II	1977	MHW° RTG	475	Jupiter/Saturn/Uranus

*Radioisotope Thermoelectric Generator #Lincoln Experimental Satellites

°Multi Hundred Watt

- Flight Systems, which designs, develops, qualifies, and fabricates flight-type generators
- Power Systems, which tests and evaluates materials and components for proposed flight systems and provides analysis, quality assurance, and reliability support functions
- Isotope Fuels, which prepares plutonium-238 fuel forms and capsule hardware, delivers fuel assemblies, and performs final assembly of the radioisotope heat source
- Space Power System Safety, which includes assessments to ensure safe and environmentally acceptable operations leading to flight approvals
- Decommissioning and Decontamination, which will restore previously utilized fueling facilities to a condition suitable for other uses

The subprogram is organized in this manner to assure explicit attention and appropriate management of each of the major sectors of activity required to develop and deliver high-performance systems.

The following table shows the operating funding levels by category for the FY 1977 through FY 1979 period.

SPACE APPLICATIONS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
CATEGORIES	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Flight Systems	\$2,275	\$7,979	\$17,881	\$9,902
Power Systems	7,538	7,106	4,101	(3,005)
Isotope Fuels	8,652	9,923	11,301	1,378
Space Power System Safety	2,215	3,000	3,117	117
Decommissioning and Decontamination	—	3,000	3,000	0
TOTAL	\$20,680	\$31,008	\$39,400	\$8,392

2.0 Flight Systems

Work in this category includes designing, developing, and fabricating nuclear-powered electric generators for specific current missions of the Department of Defense (DOD) and the National Aeronautics and Space Administration (NASA). That requires establishing and maintaining a flexible plan and strategy to accommodate user-agency requirements:

- By developing modular nuclear electric power generators on a first-of-a-kind basis
- By supplying flight units on a reimbursable basis in accordance with interagency agreements

The work includes:

- Reducing and analyzing the telemetry data on the performance of the Viking I and II, Lincoln Experimental Satellites (LES) 8/9, and the Voyager I and II radioisotope thermoelectric generators (RTG) flight units so as to guide design improvements
- Designing, developing, fabricating, and testing the components of the static Selenide Isotope Generator (SIG) for the NASA Jupiter Orbiter/Probe (JOP) mission in December 1981
- Designing, developing, fabricating, and testing the components of the dynamic isotope generator system for DOD's Space Test Program (STP)

Selenide Isotope Generator. At the beginning of its mission, the SIG (figure 37) is expected to deliver 260 watts of electricity at 29 volts d.c. with a converter efficiency of 10%. Power/weight goals are 3 watts/lb at a nominal weight of 88 lbs. The mission requires one flight generator of this type for each of two spacecraft, plus a spare for contingent replacement. The generator is approximately 22 inches high and 47 inches in diameter (at the finned radiator). The Multi-Hundred Watt Heat Source, already well established by its performance record in the LES 8/9 and Voyager

- 1 RADIOISOTOPE HEAT SOURCE (MHW)
- 2 SIDE THERMAL INSULATION
- 3 HIGH TEMPERATURE LOAD BEARING END THERMAL INSULATION
- 4 LOW TEMPERATURE LOAD BEARING END THERMAL INSULATION
- 5 INSULATION SUPPORT SLEEVE
- 6 T/E MODULE SUPPORT FRAME
- 7 T/E MODULE AND COLD END HARDWARE
- 8 HOUSING SHELL STRUCTURE
- 9 MOUNTING LUGS
- 10 END COVER WITH SYSTEM VENT
- 11 SYSTEM VENT DEVICE
- 12 END COVER WITH ELECTRICAL POWER OUTPUT CONNECTOR
- 13 ELECTRICAL POWER OUTPUT CONNECTOR
- 14 OUTGASSING PINCH OFF TUBE
- 15 AUXILIARY COOLING SYSTEM CIRCUIT
- 16 HEAT PIPES
- 17 HEAT PIPE RESERVOIRS
- 18 FIN STRUCTURE
- 19 FIN ATTACHMENT STIFFENING GUSSETS

Figure 37
Selenide Isotope Generator (SIG)

missions, will provide the energy for the selenide thermoelectric system, which will operate with the hot junction at about 900°C and the cold junction at 150°C .

Prime contractors for the SIG system are Teledyne Energy Systems, Timonium, Md., and Minnesota Mining and Manufacturing Co., St. Paul, Minn.

Dynamic-cycle Systems. Two dynamic isotope power generators are being evaluated as competitors for the ultimate system application:

- The Brayton Isotope Power System (BIPS) receives radioisotopic decay heat from a standard Multi-Hundred Watt Heat Source and converts this to electrical energy by the Brayton thermodynamic cycle, operating with helium gas and delivering its energy to a high-speed turbine-electric alternator.
- The Kilowatt Isotope Power System (KIPS) operates in an

analogous manner, but employs the Rankine cycle with Dowtherm A (diphenyl/ diphenyl ether) as the working fluid. Work is extracted from the hot vapor to drive the turbine-electric alternator.

Both units are expected to deliver approximately 1,300 watts of electric power at 28 volts d.c. Figures 38 and 39 show these systems.

Participants. The BIPS contractor team includes the Garrett AiResearch Corp. at Phoenix, Arizona and Los Angeles; the Space Division of General Electric Co., King of Prussia Park, Pa., and Thermo-Electron Corp., Waltham, Mass. The KIPS program is contracted to Sundstrand Energy Systems Co., Rockford, Ill., with Teledyne Energy Systems, Timonium, Md., as subcontractor. Several DOE laboratories provide technical assistance; the NASA Lewis Research Center at Cleveland provides substantial support for the BIPS system.

Data Reduction. The data-reduction work mentioned above includes receipt and evaluation of data transmitted from the respective spacecraft to mission centers, comparison with predictions of mission performance, and assessment for data applicable to similar future systems.

Figure 38
Brayton Isotope Power System (BIPS)

Figure 39
Kilowatt Isotope Power System (KIPS)

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3.0 Power Systems

The development work in this category provides the technological basis for meeting near-term needs, especially for missions in the early 1980s, that will require radioisotopic generator systems with low unit costs and improved performance potential.

NASA requirements for outerplanetary exploration in 1981 can be achieved with higher-performance static systems. For these, Radioisotope Thermoelectric Generators (RTGs) using selenide thermoelectric materials are being developed to achieve efficiencies of greater than 10% over a 5-year life.

The Department of Defense has identified a need for a dynamic conversion system with high power output (1.3 kilowatts electric, 18-25% efficient) for satellite and surveillance networks in the early 1980s. For this, parallel development of Brayton (by AiResearch Corp.) and organic Rankine (by Sundstrand Corp.) technology programs are being completed, after which one of the two systems will be selected for final system development for use in the United States Air Force space test program using the NASA shuttle in January 1983.

SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS

The work includes developing technology and providing support for flight-system development by:

- Performance-testing the ground demonstration system for the static Selenide Isotope Generator (SIG) for the NASA Jupiter Orbiter/Probe (JOP) mission in December 1981
- Performance-testing the ground demonstration system for the dynamic isotope generator system for DOD's Space Test Program (STP) for potential application in the Global Positioning Satellite System
- Developing the technology for the improved Multi-Hundred Watt (MHW) heat source

Ground Demonstration. The Ground Demonstration System (GDS) for the SIG project comprises a selenide-converter subsystem, designed to yield 100 Watts(e) under flight conditions, coupled with an electric heat source as stand-in for the eventual radioisotope heat source. This assembly will be tested early in 1978 at the Teledyne laboratories, with the assistance of the 3M Co. Staff. In addition, smaller selenide thermoelectric modules will be evaluated parametrically to determine conditions under which higher conversion efficiencies may be achieved.

Ground Demonstration Systems for the BIPS and KIPS programs are scheduled for test during the spring of 1978 at the respective contractor sites.

MHW Technology. The Multi-Hundred Watt heat source, mentioned previously in connection with the LES 8/9, Voyager, SIG, and dynamic systems, is inherently a safe and reliable system (figure 40). However, improvements are planned with: (1) a new "homogeneous" fuel sphere researched by LASL and being developed in the Plutonium Fuel Facility at the Savannah River Plant, (2) an improved iridium capsule alloy developed at ORNL and fabricated at the Mound Facility, (3) improved graphite structural supports designed by General Electric's Space Division, and (4) carbon/carbon composites designed to replace the bulk polycrystalline graphite used in earlier systems.

The graphite-composite work is being done jointly by Teledyne Energy Systems, Fairchild Space and Electronics Co., and the Applied Physics Laboratory of Johns Hopkins University. Analytical and testing methods are being devised and specimen materials closely simulating flight system components are being prepared to determine whether available composite materials might offer performance and safety advantages. Laboratory work is being done at LASL, the Battelle Columbus Laboratories, and ORNL.

Figure 40
Multi-Hundred Watt Heat Source (MHW)

PROGRAM: SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS								
SUBPROGRAM: SPACE APPLICATIONS								
CATEGORY: POWER SYSTEMS								
TASK	CY 1977		CY 1978		CY 1979		CY 1980	
	FY 1977	FY 1978	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983	FY 1984
SELENIDE ISOTOPE GENERATOR FOR NASA JOP								
GDS PERFORMANCE TESTING		▲ ▼						
DYNAMIC ISOTOPE GENERATOR FOR DOD STP								
ASSEMBLY—ORGANIC RANKINE GDS (KIPS)	▲ ▼							
ASSEMBLY—BRAYTON SYSTEM GDS (BIPS)	▲ ▼							
GDS PERFORMANCE TESTING		▲ ▼						
FLIGHT DESIGN OF IMPROVED MHW HEAT SOURCE	▲ ▼							

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4.0 Isotope Fuels

Work in this category:

- Develops, processes, and produces isotope-fuel forms
- Develops capsule materials and metallurgical technology
- Encapsulates and assembles the heat sources

for the nuclear generator systems used in the Space Applications Program.

Fuel-form production and first-phase encapsulation activities are being transferred from the Mound Facility to the new Plutonium Fuel Fabrication (PuFF) facility at the Savannah River Plant so as to limit further the already very low probability of dispersion of plutonium-238 into an inhabited area, by moving such operations to areas with low population density. During FY 1978, the PuFF facility is beginning to startup operations to produce fuel forms in time for the DOD space test program (STP) of the dynamic generator and for the NASA Pioneer Jupiter Orbiter/Probe launches in 1981 and 1982 and other future systems.

Fuel-form. Fuel-form process development aims to improve performance and safety in all heat-source applications. Research at the LASL plutonium-238 laboratories is translated by the Savannah River Laboratory into a production-process flowsheet, which is tested further in a preproduction mode in the cell line of the PuFF Facility in the Savannah River Plant. Thus studies with small pellets at LASL precede the manufacture of 252 gm 100 watt (thermal) MHW spheres at SRP for the heat sources destined for the static and dynamic systems. The resulting ceramic fuel form is more resistant to fragmentation and fines generation, and thus will be safer than those in earlier heat sources.

Capsules. All plutonium-238 fuel forms are encapsulated within one or more metallic containers to preclude release of the isotope during launch or orbital-reentry accidents. The hazards of explosion, high-velocity travel, high temperatures during atmospheric reentry, and possible impact are counteracted by protective materials, such as special tantalum or platinum-rhodium alloys, iridium, etc., together with Hastelloy or graphite outer-cladding layers. Process controls ensure quality welds, control grain growth, adequate impact strength, and resistance to corrosion or oxidation in the event of a return to earth.

Development of improved platinum-rhodium or iridium alloys is pursued through the preproduction and production phases. Sheet and blanks prepared at ORNL are formed into hemispheres at the Mound Facility. When completed and fully characterized, matched sets of these hemispheres are sent to the PuFF facility at SRP for encapsulation of the fuel spheres formed there; each sphere contains a pair of sintered iridium filters or vents to permit helium escape while retaining Pu-238 fines. Each encapsulated sphere is returned to Mound for enclosure within a composite-graphite impact shell, which becomes the assembly unit for the Multi-Hundred Watt heat source.

Heat-Source Assembly. The assembly of the heat source for a typical 2400 watt (thermal) MHW radioisotope heat source involves placing three groups of eight spheres each into special graphite retainer rings, each with a central bolt structure that holds the group of eight spheres together. The three "eight-packs" are loaded into a graphite aeroshell and the enclosure is locked into position. This heat-source assembly is sent in a sealed environmental-protection container to the systems contractor, who inserts the heat source into the thermoelectric or dynamic converter for a particular mission.

PROGRAM SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS																											
SUBPROGRAM SPACE APPLICATIONS																											
CATEGORY ISOTOPE FUELS																											
TASK	CY 1977			CY 1978			CY 1979			CY 1980			CY 1981			CY 1982			CY 1983								
	FY 1977	FY 1977	FY 1977	FY 1978	FY 1978	FY 1978	FY 1979	FY 1979	FY 1979	FY 1980	FY 1980	FY 1980	FY 1981	FY 1981	FY 1981	FY 1982	FY 1982	FY 1982	FY 1983	FY 1983	FY 1983	FY 1983	FY 1983	FY 1983	FY 1983	FY 1983	FY 1983
FUEL PRODUCTION AT PUFF				△																							
QUALIFICATION HEAT SOURCES AVAILABLE																											
DYNAMIC SYSTEM (STP)							△	▽																			
STATIC SYSTEM (JOP)									△	▽																	
FLIGHT SYSTEM HEAT SOURCES AVAILABLE																											
DYNAMIC SYSTEM (STP)													△	▽													
STATIC SYSTEM (JOP)														△	▽												

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5.0 Space Power System Safety

Work in this category entails:

- Supporting the safety requirements of the flight and terrestrial applications
- Conducting safety research and development
- Conducting environmental, analytical, and experimental programs

It includes:

- Testing mission hardware to evaluate, for example, the resistance of a graphitic aeroshell to blast overpressure
- Evaluating the shelf-life aspects of mission hardware to determine, for example, the limitation of storage time to preclude excessive helium pressure in unvented capsules, or the long-term compatibility of an alloy when exposed to Pu-238 fuel at operating temperature for extended times

SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS

- Reviewing the postmortem analyses of fuel hardware to determine whether the fuel form and capsule hardware from the current manufacturing processes conform to the expected levels of impact resistance, fuel deformation and fines development
- Participating in interagency reviews that support the work of the Interagency Nuclear Safety Review Panel (INSRP)
- Monitoring continuously contractor safety programs to ensure compliance with safety design specifications and all launch support activities

In FY 1978, the main emphasis is on the static and dynamic flight systems and the Stirling terrestrial system. Laboratory tests continue to determine how nuclear fuel forms are affected by the environment to evaluate the critical parameters for performing safety analysis. Similarly, a complete analysis of radioisotope inventories launched into space is maintained and provided to DOE management for environmental-impact assessment.

Participants. LASL and Sandia Laboratories test mission hardware with analytical support from the Applied Physics Laboratory of Johns Hopkins University and Fairchild Space and Electronics Co. Interagency reviews include contributions to environmental-impact studies performed by LASL, the Naval Sea System Command, the Naval Undersea Center, NOAA, Nuclear Utilities Services Corp., and others. These efforts precede the final safety evaluation reports and solicitation of Presidential approval for launch of the mission system.

PROGRAM: SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS							
SUBPROGRAM: SPACE APPLICATIONS							
CATEGORY: SPACE POWER SYSTEM SAFETY							
TASK	CY 1977	CY 1978	CY 1979	CY 1980	CY 1981	CY 1982	CY 1983
	FY 1977	FY 1978	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983
SELENIDE (STATIC) PROGRAM							
SAFETY REVIEW		INSRP RECOMMENDATION					
IMPROVED MHW HEAT SOURCE							
PRELIMINARY SAFETY REVIEW							
DYNAMIC SYSTEM FLIGHT							
CONCEPT SAFETY REVIEW (BIPS & KIPS)		INSRP RECOMMENDATION					
SPACE TEST PROGRAM							
PRELIMINARY SAFETY ANALYSIS REPORT (PSAR)							
FINAL SAFETY ANALYSIS REPORT (FSAR)							
JUPITER ORBITER PROBE							
PSAR							
FSAR							

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6.0 Decommissioning and Decontamination

After the plutonium-238 fuel-form production and encapsulation operation is transferred from the Mound Laboratory to the Savannah River Plant in FY 1978, the Mound Facility will be decommissioned and decontaminated. This will involve cleaning up 300 lineal feet of glove-box lines and decontaminating the fuel-sphere production line. Health-physics monitoring will be intensive and precautions will be taken to ensure the safety of personnel and the return of the building to conditions suitable for nonradioactive uses.

C. TECHNOLOGY BASE AND SPECIAL PROJECTS

1.0 Overview

This subprogram develops advanced isotope fuels and improves alloys, graphites, and thermoelectric materials in support of both space and terrestrial applications of nuclear materials. Terrestrial applications that are technically feasible and have programmatic or economic potential are developed through the prototype or demonstration phase; this work is analogous to that in the Space Applications subprogram.

Objectives. The specific objectives of this subprogram are:

- Advanced development of a modular Pu-238 heat source with improved performance and safety and reduced cost, for application to static- or dynamic-conversion systems for missions in 1983 and beyond
- Corollary advanced development of improved thermoelectric materials with conversion efficiencies in the 11-14% range and improved compatibility in heat-source systems
- Application of the gamma radiation and decay heat available from the cesium-137 isotope to thermoradiation of sewage sludge so as to sterilize the product and demonstrate its utility as fertilizer, cattle feed, etc.
- General studies on the beneficial utilization of isotopes recoverable from reactor waste streams
- Completion of design, development, fabrication and fueling of the prototype Stirling Isotope Power System (SIPS) for the Navy

Strategy. The strategy of this subprogram is to attain these objectives by using the resources of the national laboratories and selected contractors where the necessary skills are available.

Alternative heat-source module designs are evaluated for power density and fabricability; the associated capsule-materials forming technology is augmented where necessary to support the selected prototype fuel form. Compatibility and impact tests are conducted to select high-performance trial designs; analyses evaluate potential mission performance.

Candidate thermoelectric materials are tested under simulated service conditions to select compositions showing potential improvements in conversion efficiency at normal or elevated operating temperatures.

A cesium-137 sewage-sludge irradiation pilot plant will be activated; plans will proceed toward a facility and program to demonstrate application of the thermoradiation principle and technology to composted sewage

sludge. Experiments will continue to determine the product's applicability as agricultural fertilizer and animal feed.

Improved methods of forming and containing the Cs-137 source material are being explored. Additional applications of krypton-85 will be evaluated in nondestructive testing methods.

A modified Multi-Hundred Watt heat source and a Philips Stirling-cycle engine system will be coupled to generate 1000 watts of electric power for the Navy.

Categories. Work in this subprogram is conducted under the categories of:

- Technology Base, which includes materials and components, is mainly related to a new-generation heat source for service in applications 5 years in the future.
- Terrestrial Isotope Applications, which includes the beneficial uses of isotopes in sewage-sludge thermoradiation and other areas, as well as the Stirling Isotope Power System for the Navy.
- Special-purpose Reactor and Energy Projects, including zirconium hydride reactor technology and a dry-cooling-tower project. Work on the former has been terminated because of an unfavorable economic assessment for potential applications; the prospects of the dry-cooling-tower project are being evaluated.
- Systems Safety work primarily concerns the SIPS program, and includes criteria and guidance relative to emergency cooling systems and safety analysis reports.

The development of free-piston Stirling-engine technology is being transferred to the Fossil Energy Program of DOE, as is the evaluation of the potential of improving thermionic power-plant efficiencies.

The work in this subprogram encompasses technologies of interest and value to the various elements of the overall program of the Division of Advanced Systems and Materials Production, most of whose activities concern reactors of several types and nuclear-energy assessments; the balance includes space applications and advanced isotope separation technology. Thus, reactor operations, their products, byproducts, and wastes, and the reprocessing and recovery of specific isotopes for potential beneficial utilization, are of common interest.

The projects in the Technology Base and Special Projects subprogram have the following relationships:

- Technology development of isotope fuels, capsule metals, and graphite components relates to plutonium-238 systems for space and terrestrial applications.
- Thermionic conversion and dry cooling towers relates to improved efficiency in the operation of nuclear- or fossil-fueled

SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS

power stations.

- Applications of cesium-137 (derived from reactor waste) to sewage-sludge thermoradiation relate to the relief of municipal-waste disposal and savings in natural-gas fuel.
- Development of the technologies of krypton-85 and strontium-90 relates to commercial applications in light sources and remote power units.

The following table shows the operating funding levels by category for the FY 1977 through FY 1979 period.

TECHNOLOGY DEVELOPMENT AND SPECIAL PROJECTS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Technology Base	\$6,176	\$7,736	\$1,492	\$(6,244)
Terrestrial Isotope Applications	3,387	5,123	2,314	(2,809)
Special purpose Reactor and Energy Projects	1,559	1,378	0	(1,378)
Systems Safety	200	605	194	(411)
TOTAL	\$11,322	\$14,842	\$4,000	\$(10,842)

2.0 Technology Base

This category established the fuels, materials, and components technology base needed for developing improved isotopic and other advanced nuclear power systems. The work focuses on advanced technology for longer-lifetime isotopic-fuel containment and the development of modular general-purpose heat source use of isotope power supplies in space and special terrestrial applications in the mid-1980s and beyond.

The work involves developing new technology with emphasis on safety, economy, and reliability for:

- The General Purpose Heat Source (GPHS) for the Solar Polar Mission in 1983 and others beyond
- Reactor conceptual studies for space power system in the 10-100 kilowatt range

General Purpose Heat Source. The General Purpose Heat Source project is expected to yield a Pu-238-fueled module with inherently improved performance, reliability, and safety, and with the added feature of adaptability to a variety of power systems, either static or dynamic. Preliminary studies dealt with modules of one to three 25 watt (thermal)

capsules; recent assessments indicate that a module of four 62.5 watt units may offer advantages in terms of performance, stability, and simple multiplication. Thus, four modules, each comprising four capsules at 62.5 watts thermal output, when coupled with a static converter of 10% efficiency, would yield 100 watts of electricity.

The GPHS project is evaluating alternate fuel-form designs and capsule materials, with a view toward achieving a dense ceramic structure, improved resistance to impact, and low fission generation when encapsulated in iridium or platinum-rhodium alloys. During 1978 a prototype design will be selected and forming studies on capsule metals will be completed.

Development of the GPHS is stimulated by a diversity of missions, such as the NASA Solar Polar flight, which can be powered by nuclear electric systems. It is expected that in the future such a unit would be competitive with solar power and would offer the intrinsic advantages of long life, reliability, relatively light weight and maneuverability.

LASL, ORNL, and the Mound Facility are cooperating in the GPHS work; Battelle Columbus Laboratory is contributing to performance analysis and testing.

Conceptual Studies. The demand for nuclear electric power aboard spacecraft or satellites has increased dramatically from the 2.7 watts on the first Transit to 475 watts on each Voyager. Systems contemplated by the Department of Defense for navigation and surveillance might require 1-2 kilowatts; a space-based radar system could use 10-100 kilowatts. Dynamic systems and space reactors have the potential to be developed to meet these potential power-level requirements.

Figure 41 shows the regimes of possible space power systems for DOD missions. Figure 42 shows a typical reactor system with a Brayton-cycle dynamic converter.

Figure 41
Regimes of Possible Space Power Systems Applicable to DOD Missions

Figure 42
Typical Reactor System With a Brayton-cycle Dynamic Converter

Various reactor concepts are being studied to enable a decision on selecting or developing a design potentially adaptable to provide the power requirements of missions such as the NASA Space Construction Base or the DoD Space-Based Radar program. Space reactor power system requirements are being defined for system conceptual designs. Critical design verification experiments are planned in 1979 to determine the heat-transfer characteristics between the fuel core, heat pipe, radiator, and other system components.

PROGRAM: SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS																		
SUBPROGRAM: TECHNOLOGY BASE AND SPECIAL PROJECTS																		
CATEGORY: TECHNOLOGY																		
TASK	CY 1977			CY 1978			CY 1979			CY 1980			CY 1981		CY 1982		CY 1983	
	FY 1977			FY 1978			FY 1979			FY 1980			FY 1981			FY 1982		FY 1983
GENERAL PURPOSE HEAT SOURCE (GPHS)																		
SELECT GPHS PROTOTYPE DESIGN																		
SELECT GPHS FINAL DESIGN																		

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3.0 Terrestrial Isotope Applications

Work in this category involves the development of isotope-powered nuclear batteries for remote or undersea applications and technology for economic application of the isotopic byproducts in nuclear-reactor wastes.

The work involves:

- Fabricating and testing a 1 kilowatt(e) prototype Stirling-cycle generator for a Navy demonstration program

- Developing technology for using cesium-137 in the irradiation of municipal composted sewage sludge to reduce pathogenic contamination to acceptable levels for safe agricultural use (fertilizer and animal feed)
- Determining factors that could influence other beneficial and economic uses of reactor byproducts on a commercial basis

Battery. For the Navy nuclear battery, fabrication of both the converter and isotope heat source is being completed during FY 1978. Final system tests will begin early in FY 1979, with delivery planned for April 1979.

The system contractor for the Stirling Isotope Power System (SIPS) program is the Space Division of the General Electric Co. in King of Prussia Park, Pa. The Stirling engine system is being designed and built by the Philips Laboratories, Briarcliff Manor, N.Y. Heat-source hardware fabrication and assembly are being done by the Mound Facility, Monsanto Research Corp., Miamisburg, Ohio.

Sludge Irradiation. For sludge irradiation with cesium-137, a cost-benefit study has concluded that irradiation has the potential for being an economic sludge-treatment alternative. A pilot irradiator will be built during FY 1978. Approval of a demonstration plant at Beltsville, Md., using Washington, D.C., composted sludge from the Blue Plains plant is expected in 1978. The data from this study, plus the operating experience on the pilot irradiator and the expected approval of the demonstration plant, will form the basis of a major decision point in 1978 on the future course of the beneficial-use program. The sludge-irradiation work is centered at Sandia Laboratories. The Washington, D. C., metropolitan-area sanitation districts will be involved in the sludge-irradiator demonstration plant along with the Environmental Protection Agency and the Maryland State Environmental Services Organization.

Other Beneficial Uses of Radioisotopes. The balance of the program developing beneficial uses of isotopes is centered at the Battelle Columbus Laboratory, Pacific Northwest Laboratory, and Oak Ridge National Laboratory:

- The projected selling prices of selected separated isotopes are being studied for use in the byproduct-utilization program
- Efforts are underway to identify and develop a less-soluble form of cesium-137, which is critical for any wet-sludge treatment options
- Consideration is being given to:
 - Krypton-85 use in self-luminous lights or nondestructive testing
 - Platinum-family metals as potential alternatives to importing that precious metal

SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS

- Strontium-90 in remote power-system applications

PROGRAM SPACE AND TERRESTRIAL COMPACT POWER SYSTEMS																												
SUBPROGRAM TECHNOLOGY BASE AND SPECIAL PROJECTS																												
CATEGORY TERRESTRIAL ISOTOPE APPLICATIONS																												
TASK	CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
	FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
PUBLISH ⁹⁰ SrF ₂ DATA SHEETS					▼																							
DECISION ON SLUDGE IRRADIATION					◻																							
DELIVER SIPS TO USER										▽																		

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4.0 Special-Purpose Reactor and Energy Projects

Work in this category has involved the development of zirconium-hydride reactor technology and a dry-cooling-tower project. The former has been terminated because of an unfavorable economic assessment for potential applications. The prospects of the dry cooling tower are being evaluated.

5.0 Systems Safety

Systems safety activities are largely devoted to the Stirling Isotope Power System program in preparation for the Interagency Nuclear Safety Review Panel for development of the system in FY 1979.

VII. NAVAL REACTORS

A. PROGRAM OVERVIEW

The Naval Nuclear Propulsion Program is a joint Department of Energy (DOE) and Navy effort that designs and develops nuclear propulsion plants for installation in ships of various sizes. Within DOE, Naval Reactors is responsible for the design, development, construction, testing, operation, and maintenance of nuclear power plants and reactors for Naval vessels and their land prototypes. The developmental work on the design of advanced reactors and improvements to existing ones is based on over 25 years of research and development. Today's technology base is, to a very large extent, an outgrowth of earlier developments. In addition, operational data from existing Naval nuclear vessels and the land-based prototypes are analyzed continually and the results are factored into current developmental efforts.

Objectives. The objectives of this program include design and development of:

- Naval nuclear propulsion plants and reactors having high reliability
- Reactor plants of maximum simplicity
- Long-life cores

The ultimate objective is to develop a reactor plant and cores for each application that will last the lifetime of the ship.

Subprograms. The overall program is divided into three principal subprograms:

- Submarine Propulsion Reactors
- Surface Ship Propulsion Reactors
- Supporting Research and Development

The program is further subdivided into a number of projects for the design and follow of specific reactors. The work in these projects is highly interrelated--work conducted in one project often depends on or contributes to efforts in another.

The following table shows the funding by subprogram for the FY 1977 to FY 1979 period.

NAVAL REACTORS

NAVAL REACTORS	BUDGET AUTHORITY (DOLLARS IN THOUSANDS)			
	ACTUAL FY 1977	ESTIMATE FY 1978	ESTIMATE FY 1979	INCREASE (DECREASE)
Submarine Propulsion Reactors				
Operating	\$134,630	\$148,937	\$177,210	\$ 28,273
Equipment	4,305	10,530	14,635	4,105
Subtotal	<u>138,935</u>	<u>159,467</u>	<u>191,845</u>	<u>32,378</u>
Surface Ship Propulsion Reactors				
Operating	43,385	51,481	71,360	19,879
Equipment	1,280	3,530	5,680	2,150
Subtotal	<u>44,665</u>	<u>55,011</u>	<u>77,040</u>	<u>22,029</u>
Supporting Research and Development				
Operating	14,295	16,138	17,030	892
Equipment	415	1,090	1,685	595
Subtotal	<u>14,710</u>	<u>17,228</u>	<u>18,715</u>	<u>1,487</u>
Construction	18,200*	15,700	3,000	(12,700)
TOTAL	\$216,510 *	\$247,406	\$290,600	\$43,194

* Includes \$15,600 additional request for S8G prototype

B. SUBMARINE PROPULSION REACTORS

The objectives and strategy of this work are to design, develop, and test improved reactor cores and nuclear propulsion plants for a wide variety of submarine applications to meet the needs of the Navy, and to provide support for refuelings, overhauls, and maintenance of the operating submarine fleet.

TRIDENT-class Submarines. This effort provides for the design, development, testing, and support of a propulsion plant for installation in the TRIDENT strategic ballistic-missile submarine. The TRIDENT S8G design builds on advanced submarine technology and represents a major reactor plant advancement. The S8G prototype is being built to confirm the performance of the plant for the TRIDENT-class submarine.

During FY 1978, manufacture and installation of components and construction of the S8G prototype will continue. The S8G prototype core will be delivered and installed. Design and engineering support for the shipboard plant design and for plant components and core procurement will continue. System checkout in the lead ship, OHIO (SSBN 726), will commence.

LOS ANGELES-class Submarines. This effort provides for the design, development, and application of a high-shaft-horsepower reactor plant for use in the LOS ANGELES (SSN 688) class high-speed attack submarine. The application of an advanced long-life core to the S6G plant design is being developed.

Through FY 1977, 31 ships of this class have been contracted for construction. Engineering and physics design, testing, and construction follow of LOS ANGELES-class submarines will continue during FY 1978. Cores will be installed in SSN 695, 696, 697, 698, and 699; the SSN 693, 694, 695, and 696 will be delivered to the Navy.

Advanced Submarine Nuclear Propulsion Plant (ASNPP). This effort provides for the design and development of the ASNPP for application to future classes of higher-speed attack submarines. The ASNPP reactor and plant will provide significantly higher performance than any submarine propulsion plant now in use or under development.

This work includes designing and procuring a prototype of the ASNPP reactor for installation in an existing prototype plant. The manufacture of the fuel cluster for the prototype core and major government-furnished equipment will be completed in FY 1978.

Modifications and Additions to Reactor Facilities (MARF). This effort provides for the design, development, operation, testing, and evaluation of an advanced reactor.

Initial design and acceptance testing of the S7G prototype core, which was installed in MARF in May 1976, was largely completed by December 1977.

Advanced Fuel, Poison and Reactor Systems. This effort provides for the development of advanced fuel and poison systems to improve the reliability, performance, and lifetime of Naval reactor plants.

Prototype Operations. Five land-based prototype nuclear-propulsion plants are operated to test and evaluate reactor and propulsion plant designs, and to train and qualify personnel for operating and maintaining Naval nuclear propulsion plants. Prototype operations provide: (a) qualified nuclear-propulsion plant operators to support the needs of the fleet, (b) test data and experience that help improve reactor and plant designs, (c) information that results in improved reactor-core and plant-systems designs and improved casualty and operating procedures, and (d) valuable data on reactor fuel materials and systems and the effect of long-term exposure of reactor and reactor-plant materials and components to an actual operating environment.

During FY 1978, the SLW, SLC, MARF, and S5G prototypes and their instrumented cores will continue full-time operation, periodic tests will be performed and the results will be analyzed, plant maintenance will be conducted as required, and crew training will continue.

Operational Support for Fleet Submarines. This program provides engineering and physics support for 110 operating fleet submarines and the NR-1 Deep Submergence Research Vehicle. Design engineering and physics support for the operating submarine fleet and for the Deep Submergence Research Vehicle will continue in FY 1978. A total of approximately 21 overhauls and 26 refuelings and subsequent startup and acceptance tests of the replacement cores will be supported. Engineering support will continue for the procurement of replacement cores for the operating fleet.

SUBMARINE PROPULSION REACTORS

PROGRAM NAVAL REACTORS									
SUBPROGRAM SUBMARINE PROPULSION REACTORS									
TASK	CY 1977	CY 1978	CY 1979	CY 1980	CY 1981	CY 1982	CY 1983		
	FY 1977	FY 1978	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983		
TRIDENT (S8G SUBMARINE PROGRAM) PROTOTYPE CONSTRUCTION SUPPORT	CONSTRUCTION FUELING TESTING AND OPERATION								
ACCEPTANCE PRESSURE TEST CONTAINMENT	▼								
INITIATE FUELING		▲							
SAFETY ANALYSIS		▲							
SSN 688 LOS ANGELES (S6G) LEAD SHIP SUPPORT	SHIP DELIVERY TESTING AND OPERATIONS								
LOS ANGELES CLASS (S6G) FOLLOW SHIP SUPPORT (CORE INSTALLATIONS INITIAL CRITICALITIES SHIP DELIVERIES	CONSTRUCTION FUELING TESTING AND OPERATION - 30 SHIPS								
ASNPP PARTIAL PROTOTYPE POWER UNIT									
CORE MANUFACTURE AND ASSEMBLY									
PREPRODUCTION PROGRAMS	▼								
FUEL AND POISON ELEMENT MANUFACTURE	▼								
FUEL CLUSTER MANUFACTURE	▲		▼						
POWER UNIT ASSEMBLY		▲	▼						
MODIFICATIONS AND ADDITIONS TO REACTOR FACILITIES (MARF)	TESTING OPERATIONS TRAINING AND MAINTENANCE WITH S7G CORE								
ADVANCED SYSTEMS	INITIAL TESTING OF ADVANCED TEST CORE TESTING OPERATIONS TRAINING AND MAINTENANCE WITH ATC CORE								
PROTOTYPE OPERATIONS	TESTING OPERATIONS TRAINING AND MAINTENANCE WITH S1C 1F CORE								
S1C	TESTING OPERATIONS TRAINING AND MAINTENANCE WITH SSW-4 CORE								
S1W	TESTING OPERATIONS TRAINING AND MAINTENANCE WITH S5G CORE								
S5G	TESTING OPERATIONS TRAINING AND MAINTENANCE WITH S3G CORE								
S3G	TESTING OPERATIONS TRAINING AND MAINTENANCE WITH S3G CORE								
OPERATING SUBMARINE SUPPORT (ENGINEERING SUPPORT FOR SCHEDULED REFUELINGS OVERHAULS AND AVAILABILITIES)	TESTING OPERATIONS AND REFUELING - 110 SHIPS AND REACTOR PLANTS								

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C. SURFACE-SHIP PROPULSION REACTORS

The objectives of this work are to design, develop, and test improved reactor cores and nuclear propulsion plants for various sizes and types of Navy surface ships, and to provide technical support for refuelings of the operating surface-ship fleet.

NIMITZ-class Aircraft Carriers. This area of effort is directed toward the design, development, test, evaluation, and operational support of a high-power long-life propulsion plant for use in the Navy's NIMITZ-class two-reactor aircraft carriers. The first ship of this class, the USS NIMITZ (CVN 68), was commissioned in May 1975; the second, the USS DWIGHT D. EISENHOWER (CVN 69), was commissioned in October 1977. The third ship of the NIMITZ class, CARL VINSON (CVN 70), is under construction. A prototype of the A4W/A1G reactor is installed and operating in the A1W-B prototype plant.

Engineering and physics design and construction follow and test of NIMITZ-class aircraft carriers will continue in FY 1978. Operations of CVN 68 and CVN 69 will be monitored to ensure that all equipment and systems are performing in accordance with specifications and that operating instructions are correct and workable. The CVN 69 will undergo postshakedown availability in FY 1978. Development of a refueling method for NIMITZ-class aircraft carriers, as well as for the A4W prototype, will continue in 1978. The A4W prototype reactor in the A1W-B plant will continue planned operation to support reactor development and training requirements.

Nuclear Powered Cruisers. This work provides for the design, development, test, evaluation, and improvement of two-reactor nuclear-propulsion plants for installation in nuclear-powered cruisers. Construction and test support for the new VIRGINIA-class guided-missile dual-reactor cruisers are included. Four cruisers of this class have been authorized; three have been completed and one is under construction. In addition, the Navy is proceeding with a new class of CGNs that will use the same basic propulsion plant as the VIRGINIA. Design improvements based on ongoing research, development, and testing will be incorporated in the design. These new ships will have the D2W long-life core installed during construction.

Major emphasis in this area is devoted to the design, development, and testing of a longer-life advanced reactor core (D2W) for application in the guided missile cruisers (CGNs). A prototype of this D2W core is installed and operating in a prototype plant.

During FY 1978, design and construction follow and test of VIRGINIA-class cruisers will continue. Initial operations of USS VIRGINIA (CGN 38), USS TEXAS (CGN 39), and USS MISSISSIPPI (CGN 40) will be monitored. Support for the propulsion-plant development for the new class of CGN cruisers will be provided. Engineering follow will be provided for procurement of D2W shipboard cores.

Prototype Operations. Two land-based prototype nuclear propulsion plants are operated under the Surface-Ship Propulsion Reactors program.

Operation of these instrumented reactors and reactor plants provides for: (1) the test, evaluation, and improvement of reactor-plant systems and components and reactor fuel materials and systems, (2) the development of reactor-plant operating, casualty, and maintenance procedures, and (3) the training and qualification of personnel for assignment to nuclear ships.

The AIW and D1G prototypes will continue full-time operation during FY 1978. Periodic tests will be performed and the results will be analyzed, plant maintenance will be conducted as required, and crew training will continue. Preparations for refueling the AIW plant will continue.

Operational Support for Surface Ships. This area of effort provides design, engineering, and physics support for 10 operating fleet surface ships and the AIW-3 and D1G-2 cores used in them.

Design, engineering, and physics support for the operating surface-ship fleet will continue during FY 1978. Engineering support for the procurement of replacement cores for the operating fleet will continue.

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D. SUPPORTING RESEARCH AND DEVELOPMENT

Supporting research and development is provided in three areas: examination and analysis of expended cores at the Expended Core Facility (ECF), operation of a High Temperature Critical Test Facility (HTTF) for methods reactor physics information, and development of advanced-concept obtaining and techniques that can be used in the Naval Reactors programs.

Expended Core Facility. The ECF program has the objective of providing for the examination and analysis of expended cores and irradiation-test specimens to determine their nuclear, physical, mechanical, and chemical properties. ECF, which is located at the Naval Reactors Facility in Idaho, consists of a series of deep water pits and hot cells with associated handling gear and remotely operated disassembly and inspection tooling. ECF is also used for preparing fissionable materials from all spent Naval cores for shipment to appropriate uranium recovery facilities and for servicing and controlling of shipping containers for this function.

The ECF program will continue to support the ongoing Advanced Test Reactor test program in FY 1978. Examinations of expended prototype cores and assemblies will be continued.

High Temperature Test Facility. The HTTF program provides for the operation, modification, and maintenance of a high-temperature pressurized-water system used to obtain reactor-physics information on critical reactor assemblies for development of power cores having longer life, high power density, and improved nuclear-fuel utilization. Associated general development required for extending the life of Naval reactor cores is also supported.

During FY 1978, work to improve the cross-section sets will continue. Development of benchmark critical experiments in support of design and methods development will continue.

Naval Reactor Design Studies. These studies develop advanced concepts, develop and qualify advanced technical, engineering design, manufacturing, and inspection methods and techniques, and assist in their application to the Naval Reactors program.

SUPPORTING RESEARCH AND DEVELOPMENT

PROGRAM NAVAL REACTORS																												
SUBPROGRAM SUPPORTING RESEARCH AND DEVELOPMENT																												
TASK	CY 1977				CY 1978				CY 1979				CY 1980				CY 1981				CY 1982				CY 1983			
	FY 1977				FY 1978				FY 1979				FY 1980				FY 1981				FY 1982				FY 1983			
EXPENDED CORE FACILITY PROGRAMS																												
EXAMINATION OF TEST REACTOR SAMPLES																												
SPENT CORE EXAMINATION AND DISPOSAL																												
NAVAL CORE PRIORITY EXAMINATIONS																												
D1 PROTOTYPE CORE																												
A4W RTA FUEL ASSEMBLY																												
S3G 3 PROTOTYPE CORE																												
HIGH TEMPERATURE TEST FACILITY																												
ASNPP BEGINNING OF LIFE PARTIAL PROTOTYPE MOCKUP TESTING IN SUPPORT OF CORE DESIGNS AND METHODS DEVELOPMENT																												
NAVAL REACTOR DESIGN STUDIES																												

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APPENDIX A

MAJOR RESEARCH AND DEVELOPMENT FACILITIES

The major R&D facilities used in the program are located at both government-owned and privately owned installations. Most of the equipment itself is government-owned and operated at government expense, although some is privately owned and operated. To provide a perspective, Table 8 briefly describes a sample of these R&D facilities under seven major categories:

1. Irradiation Test Facilities
2. Component Test Facilities
3. Fuel-cycle Test Facilities
4. Safety Test Facilities
5. Physics Test Facilities
6. Operational Support Facilities
7. Materials Examination Facilities

Table 8. Major Research and Development Facilities of Fission Energy Program

Name (letter designation) Location (operator), Nature of facility	Purpose and Program. (Ref. for addl. information)
<hr/>	
1. <u>IRRADIATION TEST FACILITIES</u>	
Advanced Test Reactor (ATR), INEL, Idaho (ANC), nuclear reactor	Irradiation of fuels and materials specimens for Naval Reactors program. ("INEL," p. 14, 1977)
Engineering Test Reactor (ETR) INEL, Idaho (ANC), nuclear reactor	Safety studies for LWR and LMFBR programs, including irradiation of SLSF experi- ments ("INEL," p. 13, 1977)
Experimental Breeder Reactor No. 2 (EBR-II), INEL, Idaho (ANL) nuclear reactor	Tests of fuel, materials, in-reactor instrumentation, and sodium components and systems for LMFBR program. (ERDA-68, p. 181, 12/75)
Fast Flux Test Facility (FFTF), Richland, Wash. (HEDL), nuclear reactor	Tests of fuel, materials, and sodium components and systems for LMFBR program. (ERDA-68, p. 96, 12/75)

APPENDIX A—MAJOR RESEARCH AND DEVELOPMENT FACILITIES

High Flux Isotope Reactor (HFIR),
Oak Ridge, Tenn. (ORNL),
nuclear reactor

Irradiation testing of
fuels and graphite for
HTGR program. (ORNL-
3572, Vol. 1a, Rev. 2, 5/68;
Vol. 1b, 6/68)

Oak Ridge Research Reactor (ORR),
Oak Ridge, Tenn. (ORNL),
nuclear reactor

Irradiation testing of fuel
and graphite for HTGR
program. (ORNL-4196, Vol.
1, 1/68)

Transient Reactor Test Facility (TREAT),
INEL, Idaho (ANL),
nuclear reactor

Transient overpower and
under-cooling tests on en-
capsulated fuel elements and
packaged loops for LMFBR
safety program ("INEL," p.24,
1977; ERDA-68, p.208, 12/75)

2. COMPONENT TEST FACILITIES

• Air Flow Test Facility,
San Diego, Calif. (GA),
flow dynamics test rig

Flow tests of full-size and
scale-model components for
HTGR program ("Capabilities
and Facilities for Research,
Development, Engineering,
and Testing," GA Co., 12/76)

Component Handling and Cleaning Facility
(CHCF),
Santa Susana, Calif. (LMEC),
support facility

Used for handling, cleaning,
and inspecting large test
components for LMFBRs,
including large pumps for
SPTF. (ERDA-68, p. 56,
12/75)

Heat Transfer Loop No. 1 (HTL-1),
Richland, Wash. (HEDL),
test loop

Thermal shock and life tests
of LMFBR components in
sodium. To be deactivated
9/78. (HEDL-TME-72-145,
Rev. 2, 4/75)

High Temperature Sodium Facility
(HTSF),
Richland, Wash. (HEDL),
test facility

Test of LMFBR-system large
components in sodium and air.
To be deactivated in 6/78.
(HEDL-TME-72-145, Rev. 2,
p. 2-5, 4/75)

Hot Gas Test Facility,
San Diego, Calif. (GA),
test loop

Test of helium circulators
under simulated reactor
operating conditions for
HTGR program.

APPENDIX A—MAJOR RESEARCH AND DEVELOPMENT FACILITIES

Hydromechanical Facilities, Richland, Wash. (HEDL), test loop	Hydraulic and vibration tests for verifying design of LMFBR components. To be deactivated in 9/78. (Rev. 2, p. 4-1, 4/75)
Large Leak Test Rig (LLTR), Santa Susana, Calif. (LMEC), test loop	Study effects of large water and steam leaks into sodium; verification of methods to predict effects from such leaks. Being modified by 10/78 to accom- modate prototypical-size steam generators. (ERDA-68, p. 59, 12/75)
Pilot Line Fuel Fabrication Facility, Richland, Wash. (HEDL), fuel fabrication, inspection, and storage facility	Fabrication of LMFBR fuel; inspection of vendor-made pins; storage of completed assemblies. (HEDL-TME-72-145, Rev. 2, p. 5-1, 4/75)
Prestressed Concrete Reactor Vessel (PCRV) Model, Oak Ridge, Tenn. (ORNL), 1/12-scale vessel	Test of effects of penetra- tions on strength and mode of failure of internally loaded vessels for HTGR program. (ORNL-5274, 11/77)
PCRV Structural Test Equipment, Oak Ridge, Tenn. (ORNL), concrete properties test rig	Automatic acoustic-emission monitoring of concrete properties under various loadings for HTGR program. (ORNL-5274, 11/77)
PCRV Test Facility, San Diego, Calif. (GA), concrete stress test rig	Demonstration of PCRV stress modeling techniques for HTGR program. (GA-7150, 11/4/66)
Prestressing Tendon Test Bed, San Diego, Calif. (GA) PCRV tendon and anchor test rig	Demonstration of strand and concrete behavior under prestress conditions in PCRVs for HTGR program. (GA-P-901-59 Sec. 2)
Prototype Application Loop (PAL), Richland, Wash. (HEDL), test loop	Test of small components for monitoring impurities in sodium and cover gas. To be deactivated in 9/78. (HEDL- TME-72-145, Rev. 2, 4/75)
Seismic Test Facility, Los Angeles, Calif. (Approved Engineering Test Laboratories, AETL), two-axis shaker test rig	Horizontal excitation for sinusoidal and time-history earthquake-type studies for HTGR-core seismic program. Currently inactive.

APPENDIX A—MAJOR RESEARCH AND DEVELOPMENT FACILITIES

Seismic Test Facility, Los Angeles, Calif. (Wyle Laboratories), 1/5-scale fuel-array core model	Uniaxial horizontal excitation for sinusoidal and time-history earthquake-type studies for HTGR-core seismic program. Currently inactive.
Small Components Evaluation Loop (SCEL), Richland, Wash. (HEDL), test loop	Corrosion, erosion, crack propagation, and hydraulic studies of structural materials in sodium; proof-of-concept studies for small reactor components. To be deactivated in 9/78. (HEDL-TME-72-145, Rev. 2, 4/75)
Small Components Test Loop (SCTL), Santa Susana, Calif. (LMEC), test loop	Tests of components in liquid metals up to 1200F and 225 psig, including transients. (ERDA-68, p. 62, 12/75)
Sodium Components Test Installation (SCTI), Santa Susana, Calif. (LMEC), test loops	Test of large heat exchangers, including sodium steam generators. Being upgraded from 35 to 70 MWt by 12/78. (ERDA-68, p. 64, 12/75)
Sodium Pump Test Facility (SPTF), Santa Susana, Calif. (LMEC), test loops	Test of large sodium pumps and other components during steady-state or simulated reactor thermal-transient operation. (ERDA-68, p. 66, 12/75)
Source Term Control Loop-1 (SCTL-1), Richland, Wash. (HEDL), test loop	Determination of effect of oxygen on source rate for corrosion products in flowing sodium systems. To be deactivated 9/78. (HEDL-TME-72-145, Rev. 2, 4/75)
Source Term Control Loops Nos. 2, 3, 4 (SCTL-2, -3, -4), Richland, Wash. (HEDL), test loops	Determination of coolant impurity effects on source rates and effectiveness of trap devices; determination of corrosion rates of fuel cladding alloys. To be deactivated 9/78. (HEDL-TME-72-145, Rev. 2, 4/75)
Spin Pit Test Facility San Diego, Calif. (GA), test rig	Test of helium-circulator rotors and discs for HTGR program. ("Capabilities and Facilities for Research, Development,

Static Sodium Test Facilities, Santa Susana, Calif. (LMEC), test loops	Engineering, and Testing," GA Co., 12/76)
Thermal Transient Facility (TTF), Santa Susana, Calif. (LMEC), test loops	Test of mechanical devices, pump seals, friction in sodium; sodium storage and purification. (ERDA-68, p. 68, 12/75)
Test Tower Facility, San Diego, Calif. (GA), test rig	Test of sodium-system components in hot nitrogen under thermal transients postulated for reactor plant operation. (ERDA-68, p. 73, 12/75)
Transient Test Loop, Richland, Wash. (HEDL), test loop	Test of long components (control-rod drives and fuel handling equipment) for HTGR program.
Wire Winding Test Facility, San Diego, Calif. (GA), concrete test rig with wire winding machine	Thermal shock and life tests of engineered components in sodium. To be deactivated 9/78. (HEDL-TME-72-145, Rev. 2, 4/75)
	Demonstration of prestressing concrete structures by circum- ferential wire wrapping for HTGR program. ("Capabilities and facilities for Research, Development, Engineering, and Testing," GA Co., 12/76)

3. FUEL-CYCLE TEST FACILITIES

Controlled Air Incinerator, Los Alamos, N.M. (LASL), incinerator	Development and demonstration of incineration processes for solid combustible TRU waste. ("TRU Waste Technology Assessment Task Group Report," p. 35, 12/77)
Hot Cell, San Diego, Calif. (GA), hot cell	Disassembly of irradiation capsules and metallographic examination of HTGR fuel materials ("Capabilities and Facilities for Research, Development, Engineering, and Testing," GA Co., 12/76)

APPENDIX A—MAJOR RESEARCH AND DEVELOPMENT FACILITIES

Hot Experimental Facility (HEF), Oak Ridge, Tenn. (ORNL), chemical engineering laboratory	Test of process and equipment concepts for advanced fuel recycle using irradiated fuel. Being designed. ("Interim Design Report," 10/77)
Hot Fuel Examination Facility (HFEF), INEL, Idaho (ANL), two hot-cell complexes	Postirradiation handling and examination of LMFBR fuels and materials. (ERDA-68, p. 205, 12/75)
Hot Laboratory, San Diego, Calif. (GA), hot engineering laboratory	Thermochemical experiments (stability, fission-product release, annealing) on irradiated HTGR fuel samples. ("Capabilities and Facilities for Research, Development, Engineering, and Testing," GA Co., 12/76)
Integrated Equipment Test Facility (IET), Oak Ridge, Tenn. (ORNL), fuel-processing test facility	Demonstration of integrated "cold" operation of process- stream equipment for advanced-fuel recycle. Being developed. ("Conceptual Design Report," 11/77)
Irradiated Microsphere Gamma Analysis System (IMGA), Oak Ridge, Tenn. (ORNL), analytical instrument	Measurement of failed particle fractions in irradiated HTGR fuel specimens; unique handling system enables examination of thousands of particles under computer control. (ORNL-5274, 11/77)
Materials Development Laboratory, Richland, Wash. (PNL), fuel-fabrication test equipment	Development of fabrication technology and analytical methods for pellet fuels containing thorium for refabrication and AFCT/TFCT programs. ("Program Plan, Fuels Refabrication and Development Program," 11/9/77)
Remote Fuels Refabrication Engineering Pilot Plant (RFREPP), Richland, Wash. (PNL), fuel fabrication and refabrication line	Demonstration of remote fabrication of pellet of packed-particle fuels, as well as a production refabrication facility for AFCT/TFCT programs. Being designed. ("Program Plan, Fuels Refabrication and Development Program," 11/9/77)

4. SAFETY TEST FACILITIES

Large Leak Test Rig (LLTR)	See Component Test Facilities.
Out-of-Pile Expulsion and Reentry Apparatus (OPERA), Argonne, Ill. (ANL), sodium loop	Experiments on LMFBR coolant behavior following postulated coolant transients. (ERDA-68, p. 20, 12/75)
Out-of-Sodium Test Laboratory	See Component Test Facilities.
Outlet Plenum Mixing Test Facility, Argonne, Ill. (ANL), water loop	Study simulated temperature and flow fields in an LMFBR plenum under steady-state or transient conditions. (ANL-LEO-TR-76-2, pp. 26-27, 9/76)
Sodium Loop Safety Facility (SLSF), INEL, Idaho (ANL), packaged sodium in-reactor loop	Performance of LMFBR safety experiments in controlled thermal and hydraulic environments. ("SLSF," 9/75; ERDA-68, p. 175, 12/75)
Static Sodium Test Facilities	See Component Test Facilities.
Thermal Transient Facility (TTF)	See Component Test Facilities.
Transient Reactor Test Facility (TREAT)	See Irradiation Test Facilities.

5. PHYSICS TEST FACILITIES

Argonne Fast Source Reactor (AFSR), INEL, Idaho (ANL), low-power nuclear reactor	Provision of thermal and fast neutrons for calibration purposes and experimental development. (ANL-LEO-TR-76-2, p. 44, 9/76)
Argonne Thermal Source Reactor (ATSR), Argonne, Ill. (ANL), low-power nuclear reactor	Provision of thermal neutrons for detector calibration and experimental development. (ANL-LEO-TR-76-2, p. 9, 9/76)
Automatic Gamma-Ray Scanning System, Oak Ridge, Tenn. (ORNL), analytical instrument	Determines fission-product release from HTGR fuel elements by measuring fission-product inventories in all components of selected elements. (ORNL-5274, 11/77)

APPENDIX A—MAJOR RESEARCH AND DEVELOPMENT FACILITIES

Fast Neutron Generator, Argonne, Ill. (ANL), ion accelerator	Determination of fast-neutron cross sections and microscopic nuclear properties of fissile, fertile, structural, and other materials for LMFBR program. (ERDA-68, p. 13, 12/75)
Oak Ridge Electron Linear Accelerator (ORELA), Oak Ridge, Tenn. (ORNL), high-current electron accelerator (LINAC)	Measurement of cross sections of fissile, structural, coolant, and shielding materials for LMFBR program. (ERDA-68, p. 161, 12/75)
Plutonium Critical Mass Laboratory (PCML), Richland, Wash. (PNL), critical assemblies	Measurement of criticality of Pu and Pu/U systems for various shapes and concentrations of fuel. (ERDA-68, p. 144, 12/75)
Tower Shielding Facility (TSF), Oak Ridge, Tenn. (ORNL), movable nuclear reactor	Obtaining integral reactor shielding design data and verifying shielding design parameters. (ERDA-68, p. 166, 12/75)
Zero Power Plutonium Reactor (ZPPR), INEL, Idaho (ANL), split-table-type critical facility	Obtaining integral reactor physics parameters from fast-spectrum critical assemblies. (ERDA-68, p. 216, 12/75)
Zero Power Reactors Nos. 6 & 9 (ZPR-6, -9), Argonne, Ill. (ANL), split-table-type critical facilities	Obtaining reactor-physics integral parameters for variety of fast-reactor cores. (ERDA-68, p. 28, 12/75)

6. OPERATIONAL SUPPORT FACILITIES

Alkali Metal Cleaning Facility (AMCF), Richland, Wash. (HEDL), cleanup system	Cleaning sodium-wetted CRCTA and FFTF components and dry testing; sodium removal development. (ERDA-68, p. 89, 12/75)
Carbon Equilibrium Loops Nos. 1 & 2 (CEL -1, -2), Madison, Pa. (WARD), sodium loops	Obtaining experimental information and analytical relationships on rate and direction of carbon and nitrogen transfer. (ERDA-68, p. 225, 12/75)

Chemical and Metallurgical Laboratories, Santa Susana, Calif. (LMEC), laboratories	Chemical and metallurgical analyses and services for LMFBR Program.
Chemistry and Analysis Laboratory Facilities, Richland, Wash. (HEDL)	Chemical and microstructural analysis for LMFBR program. (HEDL-TME-72-145, Rev. 2, p. 10-1, 4/75)
Component Handling and Cleaning Facility (CHCF)	See Component Test Facilities
Controlled Atmosphere Chamber, Santa Susana, Calif. (LMEC), "cold" test cell	Evaluation of sodium-to-gas leak detection methods. (ERDA-68, p. 58, 12/75)
Electron Microprobe, San Diego, Calif. (GA), analytical instrument	Elemental identification of very small regions of samples in HTGR program. ("Capabilities and Facilities for Research, Development, Engineering, and Testing," GA Co., 12/76)
Electron Microscopy Facilities, San Diego, Calif. (GA), metallurgical laboratories	Structural examination of materials in HTGR program. ("Capabilities and Facilities for Research, Development, Engineering, and Testing," GA, Co., 12/76)
Gamma Spectrometry Facility, Oak Ridge, Tenn. (ORNL), gamma counter	Analysis of fission-gas release rates from HTGR fuels during irradiation (ORNL-5274, 11/77)
High Level Chemical Development Facility (Bldg. 4507) Oak Ridge, Tenn. (ORNL), hot cells	Chemical development; solvent extraction; 100-ml-scale head-end processing studies. (ERDA-68, p. 156, 12/75)
High-Radiation Level Examination Laboratory (HRLEL), Oak Ridge, Tenn. (ORNL), hot cells	Examination of HTGR fuel specimens after irradiation in ORR, HFIR, Peach Bottom, and other reactor facilities. (ORNL-3479, 11/63)
High Temperature Sodium Test Facility (HTSF)	See Component Test Facilities.
High Voltage Electron Microscope (HVM) Richland, Wash. (HEDL), high-voltage generator and accelerator	Materials studies in support of FFTF and LMFBR using penetration power of high-

	energy electrons. (ERDA-68, p. 109, 12/75)
Instrumentation and Standards Laboratory, Santa Susana, Calif. (LMEC), laboratory	Maintenance, calibration, and standardization of instruments for LMEC facilities traceable to NBS.
Interim Examination and Maintenance Training Facility (IEMTF), Richland, Wash. (HEDL), mockup of FFTF IEM cell	Evaluation and performance improvement of equipment designed for fuel handling, disassembly, inspection, and reassembly; remote maintenance of reactor components. (ERDA-68, p. 114, 12/75)
Interstitial Transfer Facility (ITS), Madison, Pa. (WARD), sodium loop	Investigation of the transfer phenomena of interstitial alloying elements in a dynamic polythermal system; LMFBR program. (ERDA-68, p. 234, 12/75)
Materials Tests Loops Nos. 1 & 2 (MTL-1, -2), Madison, Pa. (WARD), sodium loops	Obtaining data on chemical, physical, and mechanical properties of coatings and other wear-resistant surface for FFTF applications. (ERDA-68, p. 235, 12/75)
Mechanical Properties Systems Nos. 1-3 (MPS-1, -2, -3), Madison, Pa. (WARD), sodium loops	Development of creep, creep rupture, and fatigue data on LMFBR materials in a prototypic sodium environment. (ERDA-68, p. 237, 12/75)
Mound Facility, Miamisburg, Ohio (MLM), production line in hot cells	Development and fabrication of capsule hardware and assembly of heat sources for space applications.
Oak Ridge Reactor Data Acquisition System (ORRDACS), Oak Ridge, Tenn. (ORNL), computer	Monitoring and controlling HTGR graphite creep experiments. (ORNL-5274, 11/77)
Out-of-Sodium Test Laboratory	See Component Test Facilities
Sodium Cleaning Facility (NACF), Madison, Pa. (WARD) outdoor mobile rig	Removal and disposal of residual sodium from various-size articles. (ERDA-68, p. 246, 12/75)

Sodium Purification Loop (SPL), Richland, Wash. (HEDL), sodium loop	Source of small amounts of purified sodium; glove-box facility. (ERDA-68, p. 131, 12/75)
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Sodium Removal Development Apparatus Nos. 1 & 2 (SRDA-1, -2), Richland, Wash. (HEDL), vessels/loop rig	Development of processes and procedures for removal of sodium from articles by various methods. (ERDA-68, p. 132, 12/75)
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Transient Reactor Test Facility (TREAT)	See Irradiation Test Facilities
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7. MATERIALS EXAMINATION FACILITIES

Alpha-Gamma Hot Cell Facility (AGHCF), Argonne, Ill. (ANL), hot cells & glove boxes	Examination of experimental fuel elements; assembly/ disassembly of irradiation vehicles; postirradiation fuels and materials experiments; mechanical-property testing. (ANL-LEO-TR-76-2, 9/76)
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CMR Bldg. (Wing 9) Hot Cell Facility and DP-West Hot Cells, Los Alamos, N.M. (LASL), hot cells	Diagnostic examination of irradiated fuel pins for FBR program. (ERDA-68, p. 221, 12/75)
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Environmental Fatigue Apparatus, Oak Ridge, Tenn. (ORNL), load cycling rig	Test of Candidate HTGR structural materials in He containing impurities simulating reactor operating conditions. (ORNL-5274, 11/77)
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High Temperature Reactor & Materials Testing Laboratory, Schenectady, N. Y. (GE), metallurgical laboratory	Evaluation of effects of coolant and temperature on candidate HTGR alloys for process heat and DCHT applications. Con- struction to be completed 5/78. (COO-2975-3, -10, -13, -17, 1/77, 4/77, 7/77, 11/77)
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Materials Testing Facilities Oak Ridge, Tenn. (ORNL), metallurgical laboratory	Performance of creep tests in air and vacuum and fatigue testing in air for HTGR pro- gram. (ORNL-5274, 11/77)
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Neutron Radiography Facility (NRF), Richland, Wash. (HEDL), reactor irradiation facility	Radiographic examination of FFTF fuel pins. (ERDA-68, p. 119, 12/75)
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Radiometallurgy Laboratory (RML),
Richland, Wash. (HEDL),
hot metallurgical laboratory

Destructive examination of
irradiated LMFBR fuels and
materials. (ERDA-68, p. 124,
12/75)

Shielded Materials Facility,
Richland, Wash. (HEDL),
hot cell

Nondestructive examination
and testing of irradiated
LMFBR fuels and structural
materials. (ERDA-68, p. 127,
12/75)

Special Environment Radiometallurgy
Facility (SERF),
Richland, Wash. (HEDL),
hot cells

Visual and metallurgical
examination of reactor
fuels and structural
materials in a dry nitro-
gen atmosphere. (ERDA-68,
p. 138, 12/75)

APPENDIX B

NATIONAL AND ENGINEERING LABORATORY ROLES

The national and engineering laboratories traditionally have played an important role in achieving goals in the fission energy program. They have demonstrated the ability to achieve solutions to specific R&D problems affecting the technical and economic feasibility and the timetable of major nuclear energy activities. In addition to the performance of in-house research and development at each installation, their personnel also have functioned in liaison and supervisory capacities for the management of subcontracts with universities, other laboratories, and industrial organizations for the performance of supporting research and development work. The laboratories are staffed with highly skilled scientists, engineers, and technicians with a long history of accomplishments in research, development, and demonstration in all aspects of nuclear energy. Consequently, new ideas, data, and processes for better utilization of fission energy resources evolve from these laboratories.

Table 9 indicates the Fission Energy Program elements in which each laboratory is working. The brief descriptions that follow provide more information on the work being conducted on each program at each laboratory. Figure 43 shows where the laboratories are located.

Table 9. National Laboratory Participation in Fission Energy Programs

CHAPTER LABORATORY	LWR	EVALU- ATIONS	GCTR	WCBR	LMFBR	GCFR	ADV ISO SEP	FUEL		WASTE MGT.	SPACE APPL	NAVAL REACTORS
								PROC.	STORAGE			
								V-C	V-D			
ANL		X			X		X	X		X		
BAPL				X								X
BNL		X								X		
HEDL		X			X					X		
INEL					X					X		X
KAPL				X								X
LASL		X	X				X			X	X	
LLL		X					X			X	X	
LMEC					X							
Mound							X			X	X	
ORNL		X	X			X		X		X	X	
PNL		X						X		X	X	
RF										X		
RHO										X		
Sandia										X	X	
SRL				X				X	X	X	X

Figure 43
Locations of National and Engineering Laboratories

ARGONNE NATIONAL LABORATORY (ANL) Argonne, Ill., operated by the University of Chicago.

Systems Evaluation (NASAP, INFCE). ANL is providing the Office of Fuel Cycle Evaluation technical studies concerning: (a) the scoping of accelerator breeder design and (b) core design and system analyses of selected reactor systems. To support the system characterizations under NASAP, ANL is also assisting private contractors in preparing the descriptions of safety aspects of reactor systems. It is also providing information services to a world energy model used to assess the economics and resource utilization of proposed systems.

LMFBR. The major breeder reactor activity at ANL is the safety program. ANL is the center for the breeder-reactor safety activities that focus on providing a technology base responsive to the safety consideration in design evaluation and licensing of breeder reactor systems. Argonne conducts the reactor core physics work for the breeder program and supports engineered components, materials and chemistry, and reactor design and analysis. ANL is responsible for the operation of a number of safety and physics test facilities at the Idaho National Engineering Laboratory, including the Transient Reactor Test (TREAT), Sodium Loop Safety Facility (SLSF), and the Zero Power Plutonium Reactor (ZPPR). Argonne also operates the EBR-II and its irradiation tests at INEL.

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Advanced Isotope Separation (AIS). Argonne performs research on the chemi-ionization process for uranium enrichment. Its work aims primarily at assessing the proliferation aspects of the technology.

Fuel Cycle Process Development. ANL has the lead role for non-aqueous processing development. Pyrometallurgical and pyrochemical processes are the main methods being studied.

Waste Management. Argonne participates in the waste management R&D program in the following areas: (1) technology for converting high-level waste to forms suitable for permanent disposal, (2) criteria for converting the cladding-hull waste from fuel reprocessing to a form suitable for permanent disposal, and (3) studies of the potential for the migration of transuranium elements through various soils and geologic media.

BETTIS ATOMIC POWER LABORATORY (BAPL) Pittsburgh, Pa., operated by Westinghouse Electric Corp.

Water-cooled Breeder Reactors. Bettis performs research and development for water-cooled breeder reactor projects. The Light Water Breeder Reactor (LWBR) program is a part of the Water-cooled Breeder Reactor project. The LWBR core installed in the Shippingport Atomic Power Station commenced fullpower operation on December 2, 1977.

Naval Reactors. BAPL performs research and development for naval nuclear-propulsion plants. The Naval Reactors Facility (NRF) at Idaho National Engineering Laboratory operates three prototype naval reactor plants and a facility for examination of expended naval reactor cores.

BROOKHAVEN NATIONAL LABORATORY (BNL) Upton, L.I., N.Y., operated by Associated Universities of New York.

Systems Evaluation (NASAP, INFCE). BNL is providing NASAP a conceptual design and cost estimate for a once-through electronuclear power producer, which is one of the many systems being reviewed for proliferation resistance under that program.

Waste Management. Brookhaven's contribution to the waste-management program consists of R&D on the removal of tritium from effluents and its conversion to a solid form suitable for permanent disposal.

HANFORD ENGINEERING DEVELOPMENT LABORATORY (HEDL) Richland, Wash., operated by Westinghouse Electric Corp.

Systems Evaluation (NASAP, INFCE). HEDL is assessing the resource use and economics of deploying proposed proliferation-resistant systems under the NASAP.

LMFBR. The central point of the breeder reactor fuel development program is the Fast Flux Test Facility (FFTF) at HEDL. In addition to FFTF, whose construction was 90% complete in November 1977, the fuel development program at HEDL will be supported by the new Fuels and

APPENDIX B—NATIONAL AND ENGINEERING LABORATORY ROLES

Materials Examination Facility to be constructed by 1984. Other significant contributions to the breeder program by HEDL are reactor component development (including core control rods, instrumentation, and fuel transportation containers), site and facility safety, safeguards, and security capabilities.

Waste Management. HEDL's role in the waste-management program is R&D on processes: (1) for solidifying liquid wastes other than high-level waste, and (2) for treating transuranium-contaminated solid waste by acid digestion to reduce its volume. HEDL also operates facilities for testing high-efficiency filters for removing particulates from gaseous effluents.

IDAHO NATIONAL ENGINEERING LABORATORY (INEL) Idaho Falls, Idaho, operated by Edgerton, Germeshausen, and Grier Corp.

LMFBR. INEL is the site of the Experimental Breeder Reactor II (EBR-II) that is used for fuel and materials tests, the Zero Power Plutonium Reactor (ZPPR) for physics experiments, the Sodium Loop Safety Facility (SLSF) for use in the Engineering Test Reactor (ETR), and the Transient Reactor Test (TREAT). INEL operates the ETR-SLSF Loop and provides engineering support for all the facilities.

Waste Management. INEL activities in the waste-management program include the development of techniques and the evaluation of the impact of recovering previously buried TRU-contaminated waste, the investigation of alternatives for the long-term management of the buried TRU waste, the development of technology for the conversion of Idaho high-level waste to a form suitable for permanent disposal, and the selection of an alternative for implementing the long-term high-level waste management program.

Naval Reactors. The Advanced Test Reactor at INEL performs irradiation tests and experiments and provides support services to the Naval Reactors Facility located on the INEL site.

KNOLLS ATOMIC POWER LABORATORY (KAPL) Schenectady, N.Y., operated by General Electric Co.

Naval Reactors. KAPL performs research and development for Naval nuclear propulsion plants and, to a minor extent, for the Water Cooled Breeder Reactor project. The TRIDENT prototypes reactor plant located at the Kresselring Site of the Laboratory will become operational in FY 1978.

LAWRENCE LIVERMORE LABORATORY (LLL) Livermore, Calif., operated by the University of California.

Systems Evaluation (NASAP, INFCE). As part of the development of proliferation-resistance assessment criteria, LLL is assessing the usability for weapons of fuel-cycle material under NASAP.

Advanced Isotope Separation (AIS). LLL has the lead responsibility for the research and development work on the atomic-vapor laser isotope-separation program. During FY 1978, LLL will resolve the science aspects having a significant impact on the projected cost of the atomic-vapor process.

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Waste Management. LLL's activity in the waste-management program consists of R&D to improve the performance and reduce the cost of particulate filtration systems.

LIQUID METALS ENGINEERING CENTER (LMEC) Canoga Park, Calif., operated by Rockwell International Corp.

LMFBR. LMEC is the primary United States facility for nonnuclear testing of large sodium power plant components, such as pumps, steam generators, and intermediate heat exchangers. LMEC also performs the design and construction management for the large, specialized facilities required for such testing. Testing and development work is performed on a broad range of sodium components and instrumentation required in liquid metal breeder reactors.

LOS ALAMOS SCIENTIFIC LABORATORY (LASL) Los Alamos, N.M., operated by the University of California.

Systems Evaluation (NASAP, INFCE). For NASAP, LASL is performing scoping studies to define accelerator breeder designs to allow evaluation of the proliferation resistance and viability of such a system.

GCTR (HTGR, HTR). The LASL program supports the DOE planning of the U.S./FRG cooperative program for the international development of an advanced HTR concept. The work, which involves international development of an advanced HTR concept, entails: (1) IHX designs, (2) development of computer calculation models for comparative analyses of prismatic versus pebble-bed core designs, and (3) evaluation of LASL facilities for use as a High Temperature Helium Test Facility.

Advanced Isotope Separation (AIS). LASL has the lead responsibility for the research and development work on the molecular laser isotope-separation program. During FY 1978, LASL will resolve the science aspects having a significant impact on the projected cost of the molecular process.

Waste Management. LASL's role in the technical program for reducing the volume of transuranium-contaminated waste consists of testing incinerators and other volume-reduction equipment under conditions representative of plant operations. LASL also is developing techniques for the long-term evaluation of TRU-contaminated waste burial grounds. In addition, LASL's waste-management program includes R&D on methods for testing high-efficiency particulate filters and improved methods for the siting, operation, and long-term stabilization of shallow land burial grounds for the disposal of low-level waste.

Space Applications. LASL is doing plutonium-238 fuel-form research, capsule-hardware process development, heat-source design and evaluation, environmental studies on Pu-238 fuel under typical terrestrial exposure conditions, and support for the Interagency Nuclear Safety Review Panel activities.

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MOUND LABORATORY Miamisburg, Ohio, operated by Monsanto Research Corp.

Advanced Isotope Separation (AIS). Mound Laboratory is investigating a chemical exchange process for uranium enrichment. Efforts are primarily aimed at assessing the proliferation aspects of the technology.

Waste Management. Mound is developing ultrafiltration technology for application in decontaminating nonhigh-level liquid waste and is doing R&D on volume-reduction systems for transuranium-contaminated waste.

Space Applications. Mound Laboratory is developing and fabricating capsule hardware and is assembling heat sources.

OAK RIDGE NATIONAL LABORATORY (ORNL) Oak Ridge, Tenn., operated by Union Carbide Corp.

Systems Evaluation (NASAP, INFCE). ORNL is providing technical overview and coordination of the various studies conducted under NASAP to the Office of Fuel Cycle Evaluation, which provides the program management for NASAP. As such, ORNL is involved with collecting, reviewing for technical accuracy, and assembling in a consistent package, the results of analytical studies and assessments undertaken by the many organizations participating in NASAP. ORNL is also performing some of the comparative assessments required of the program, and is assisting in the development of final program recommendations.

GCTR (HTGR, HTR). ORNL is the lead contractor for the HTGR program. It provides major support through several R&D programs that generate fuels, materials, and components technology, and through reactor design evaluations. Significant continuing programs are underway in fuel development and irradiation testing, fission-product behavior studies, structural materials evaluations, graphite irradiation effects, prestressed-concrete reactor vessel (PCRVR) analyses, and pebble-bed-reactor systems assessment. The major program at ORNL continues to be the development of coated-particle fuels.

GCFR. ORNL's major activities in support of the GCFR breeder reactor are in the areas of high-temperature material-property data and design methods, component and procedure standards, and support for unique Gas Cooled Fast Breeder (GCFR) activities. ORNL will construct and operate the Core Flow Test Loop (CFTL) for the thermal-hydraulic testing of GCFR fuel assemblies and will test scale models of the GCFR prestressed-concrete reactor vessel. ORNL also provides reactor shielding information through the operation of the Tower Shielding Facility (TSF).

Fuel Cycle Process Development. ORNL has the lead role in the Advanced Fuel Recycle Program. It also works under subcontract in the Alternative Fuel Cycle Technology (AFCT) and Thorium Fuel Cycle Technology (TFCT) programs.

Waste Management. ORNL broadly participates in the waste-management program, specifically in: (1) studies for volume-reduction, packaging, and disposal of high-level, ORNL intermediate-level, TRU-contaminated and tritium-contaminated wastes by various techniques, and (2) fuel-cycle analysis to determine quantities and characteristics of waste generated under various technology assumptions.

Space Applications. ORNL performs research and development in precious metals (principally iridium and platinum) and alloys, fabrication of specialized heat-source components, and studies advanced insulating materials and their behavior under space environments.

PACIFIC NORTHWEST LABORATORY (PNL) Richland, Wash., operated by Battelle Memorial Institute.

Systems Evaluation (NASAP, INFCE). PNL is assessing the environmental aspects of a set of alternative reactor systems for NASAP. Environmental acceptability is one of several key factors determining the viability of proliferation-resistant systems developed by the NASAP.

Fuel Cycle Process Development. PNL has the lead role in the Refabrication Development Program, which is a major part of Alternative Fuel Cycle Technology and Thorium Fuel Cycle Technology Programs (AFCT/TFCT). It is responsible for developing processes for fabricating pellet and particle fuels containing uranium, plutonium, or thorium.

Waste Management. PNL broadly participates in the waste-management program with activities in the following areas: (1) vitrifying high-level waste, (2) development of waste management concepts and methods, monitoring risk assessment procedures, and preparation of GEISSs, and (3) implementation of international program support plans.

Space Applications. PNL provides support in studies and data publication relative to strontium-90 isotopic material for terrestrial applications.

ROCKWELL HANFORD OPERATIONS (RHO) Richland, Wash., operated by Rockwell International Corp.

Waste Management. RHO waste-management activities include the development of technology for conversion of Richland high-level waste into a form suitable for permanent disposal, the selection of an alternative for implementing the long-term high-level waste management program, and the investigation of alternatives for the long-term management of Hanford contaminated soils and sediments.

ROCKY FLATS (RF) Golden, Colo., operated by Rockwell International Corp.

Waste Management. RF's primary R&D activity in the waste-management program consists of investigating the fluidized-bed incinerator concept for TRU waste treatment.

APPENDIX B—NATIONAL AND ENGINEERING LABORATORY ROLES

SANDIA LABORATORIES Albuquerque, N.M., operated by Western Electric Co.

Waste Management. Sandia is the principal contractor in the Waste Isolation Pilot Plant program. Technical activity focuses on the selection of a specific site. Sandia's role includes exploratory drilling and the highly detailed studies necessary for site confirmation. In addition, Sandia participates in the gaseous-waste management program with the responsibility for evaluating alternative long-term storage facility concepts for krypton-85.

Space Applications. Sandia provides quality-assurance overview and on-site representatives at selected contractor locations; it also assists in isotope-separation studies and sludge thermoradiation experiments.

SAVANNAH RIVER LABORATORY (SRL) Aiken, S.C., operated by E. I. Du Pont de Nemours and Co.

Fuel Cycle Process Development. SRL is the lead organization for the Design Integration Study and the reprocessing portion of the AFCT/TFCT programs.

Spent Fuel Storage. SRL is preparing the environmental impact statement for the implementation of the spent-fuel storage policy. In addition, R&D is being conducted on means of further increasing spent-fuel storage capacity at existing basins. Alternative away-from-reactor storage concepts are being evaluated.

Waste Management. SRL's activities in the waste-management program consist of developing technology for the conversion of Savannah River's high-level waste to a form suitable for permanent disposal, and studies related to minimizing potential radionuclide migration from shallow land burial grounds that are in a humid environment.

Space Applications. Process development at SRL is coordinated with research in that area at LASL and provides the basis for plant and equipment design and operation to produce plutonium-238 fuel for radioisotope heat sources.

Water-Cooled Breeder Reactor. SRL will supply heavy water (deuterium oxide) to the Water-Cooled Breeder Reactor program.

APPENDIX C

FISSION ENERGY DEVELOPMENT IN OTHER COUNTRIES

All the technologically advanced countries, as well as some of the developing nations, are experiencing a severe energy crisis. Many lack adequate indigenous energy sources or the capital required to purchase the needed fuel in the world market at current and likely future prices. Experience with fission-energy power systems has shown that electricity from nuclear plants is less costly than that from conventional sources and is environmentally acceptable. This is especially true in those countries that do not have the fossil-fuel resources the United States has. Consequently, the technologically advanced countries and some of the developing countries have turned to nuclear power as a major source of energy.

This appendix provides a perspective on nuclear power development in other countries. Table 10 indicates the areas discussed in this appendix; the United States is included for perspective. Then the work being done in each technology area by each country is described briefly.

Table 10. Types of Fission Energy Development by Countries

COUNTRY	GCTR	LMFBR	FUEL CYCLE			
			ADV. ISO. SEPARATION	SPENT FUEL		WASTE MANAGEMENT
				PROCESSING	STORAGE	
Argentina						X
Belgium		X		X		X
Canada						X
France	X	X	X	X		X
India						X
Iran						X
Italy	X	X				X
Japan	X	X				X
Netherlands		X				
Spain	X					
Switzerland	X					X
Un. of S. Africa			X			
USSR		X				X
United Kingdom	X	X		X		X
United States	X	X	X	X	X	X
West Germany	X	X	X	X	X	X

SYSTEMS EVALUATION

See Part B of Chapter IV.

GAS-COOLED THERMAL REACTOR (GCTR)

France. The first French gas-cooled power reactors, designed for both electricity and plutonium production, were of the Magnox type cooled by CO_2 , fueled by metallic uranium, and moderated by graphite. The 40 MWe G2 and G3 reactors at Marcoule were started in 1959-60 and demonstrated on-load refueling and the use of prestressed-concrete reactor vessels (PCRVs). France has six additional Magnox plants in operation and one gas-cooled heavy-water-moderated reactor (GCHWR), for a total of 2392 MWe of gas-cooled nuclear power plants. The largest of these plants is the St. Laurent Unit 2, generating 571 MWe. France has maintained a limited development effort on gas-cooled thermal reactors, estimated to be about \$12 million in FY 1977. France is also a signatory to the gas-cooled-reactor umbrella agreement of February 1977 as amended.

United Kingdom. In 1956-57, the 35 MWe Calder Hall reactors were started in England. They were of the Magnox type, CO_2 cooled, fueled by natural metallic uranium and moderated by graphite. They were contained in steel pressure vessels and had off-load refueling. These early reactors had a low thermal efficiency ($\sim 20\%$) but operated with a high load factor. A total of 26 commercial Magnox systems were added to the British grid between 1956 and 1971, for a total Magnox capacity of 5,206 MWe. The largest of these plants are the twin 590 MWe Wylfa units.

The main drawback of the natural-uranium Magnox reactors is their high capital cost that results from their low power density and low coolant exit temperature. This led to the use of stainless-steel cladding and slightly enriched UO_2 fuel elements in the Advanced Gas-cooled Reactors (AGRs). A 33 MWe prototype AGR began operation at Windscale in 1963 and ten large units are in operation or on order at present. The only helium-cooled High Temperature Gas-cooled Reactor (HTGR) to operate in the UK was the 20 MWt Dragon reactor, which ran successfully from 1965 to 1976.

West Germany (FRG). Construction began on the first commercial gas-cooled reactor plant in Germany in 1960. This was the 15 MWe graphite-moderated helium-cooled AVR system. The AVR first went critical in 1967 and is now the only operating pebble-bed reactor in the world. A second High Temperature Reactor (HTR) demonstration plant, the 300 MWe Thorium High Temperature Reactor, is under construction and is scheduled for operation around 1980. The German government has active development programs on HTRs for both direct-cycle gas-turbine systems and process-heat applications. Recently the German government has selected the direct cycle as the primary HTR system to be developed because it is a longer-range program with potentially more significant incentives than the HTGR steam cycle. Approximately \$77 million was devoted to HTR development in Germany in FY 1977. Germany is also signatory to the gas-cooled-reactor umbrella agreement along with the United States, France, and Switzerland.

Japan. Japan has had an interest in gas-cooled reactors for both electricity and industrial-heat applications. In 1965 the Tokai-Mura plant began operation 70 miles northeast of Tokyo. It is a 157 MWe graphite-moderated CO_2 -cooled natural-uranium reactor. For industrial applications, particularly nuclear steelmaking, the Japanese government has a limited development effort (\$10 million in FY 77) on high-temperature helium-cooled reactors.

has a limited development effort (\$10 million in FY 77) on high-temperature helium-cooled reactors.

Switzerland. Swiss experience with the gas-cooled reactor began in 1967 with the startup of the Lucens experimental reactor, a 7 MWe gas-cooled heavy-water-moderated plant. The reactor is cooled by CO₂ and fueled by natural uranium. Development continues on gas-cooled thermal reactors at a level of about \$7 million in FY 1977. Switzerland is a signatory to the GCR umbrella agreement.

Italy. The Latina reactor began operation in 1963, 35 miles south of Rome. The reactor is a 210 MWe CO₂-cooled graphite-moderated system fueled by natural uranium.

Spain. The Vandellus reactor began operation in 1972. It is a 500 MWe Magnox plant.

United States. Extensive development work on high-temperature gas-cooled reactors has been conducted in the United States over the past 20 years. The High Temperature Gas Reactor (HTGR) base-technology program began in the 1950s and has continued to date. The Nuclear Engine for Rocket Vehicle Application (NERVA) program also began in the 1950s and was conducted until the early 1960s when that program was terminated after having built and operated some 18 open-cycle nuclear reactors designed to heat hydrogen propellant to 2,200°C for short periods of time (1 hour). The Experimental Gas-Cooled Reactor (EGCR) was a metal-clad helium-cooled reactor project constructed at Oak Ridge, Tenn., but EGCR was not operated because HTGR technology with its higher temperature incentives overtook and made obsolete the concept of gas-cooled converter reactors employing metal-clad fuels.

Commercial gas-cooled reactor experience in the United States began with the Peach Bottom reactor, a 40 MWe plant, which was operated for more than 7 years by the Philadelphia Electric Co. until operation was discontinued in 1974 due to the less favorable economics of small electric power plants. The Fort St. Vrain reactor, a 330 MWe HTGR, is undergoing startup testing. Several years ago, General Atomic Co. had commitments, or near commitments, for 10 larger central-station power plants. For other than technical reasons, however, all these orders were subsequently canceled, so Fort St. Vrain remains the only United States HTGR plant in operation or under construction.

France. The French fast-reactor program, which has been conducted by the Commissariat à l'Energie Atomique (CEA), emphasizes the construction and operation of a series of fast reactors of increasing size. The development effort is strong in steam generators, physics, and fuels and materials. France has built and operated two LMFBRs, the 37 MWt Rapsodie irradiation-test reactor and the 252 MWe Phoenix demonstration plant, and plans to build the 1,200 MWe Super-Phoenix plant.

United Kingdom. The United Kingdom Atomic Energy Authority (UKAEA) began studies of fast reactors in the early 1950s. Initially, the

APPENDIX C—FISSION ENERGY DEVELOPMENT IN OTHER COUNTRIES

studies concentrated on examining in detail the basic physics of the system and confirmed that breeding new fuel was practicable. An intensive design and construction program was started in the mid-1950s. The U. K. has built and operated the 14 MWe Dounreay Fast Reactor and the 250 MWe Prototype Fast Reactor, and plans to build the 1,300 MWe Commercial Fast Reactor.

USSR. The USSR's LMFBR program is under the purview of the State Committee for the Utilization of Atomic Energy, except that the design, construction, and operation of the world's largest LMFBR to date, the 600 MWe BN-600, is under the Ministry for Power and Electrification. The USSR has built and operated the 100 Watt BR-1, the 100kW BR-2 (which was reworked to form the 5 MW BR-5, then the 10 MW BR-10), the 12 MWe BOR-60, and the 350 MWe BN-350, and is building the 600 MWe BN-600.

West Germany. The program of the Federal Republic of Germany is similar to the United States program in many ways--establishing a body of technology, proof-testing components, and building a succession of increasingly larger plants to establish industrial capability. Major test facilities are located at Bensberg and Karlsruhe. Licensing requirements appear to be as rigorous as in the United States. KNK-I, a 19 MWe sodium-cooled thermal-neutron test reactor, was the initial "stepping stone" in the West German fast breeder reactor program. It has been converted to a fast-flux test reactor, KNK-II.

West Germany, a highly industrialized country with few energy resources, is the major participant in the DeBeNeLux program, a cooperative program with Belgium and the Netherlands, with Luxembourg as a minor participant. It also has a tripartite program, through its utilities, with French and Italian utilities. West Germany and its partners are building the 300 MWe SNR-300 demonstration plant.

Japan. The Japanese LMFBR program, which began in the early 1960s, consists of broad technology development and component prooftesting along with construction of a progression of LMFBR plants. The experimental 50 MW Joyo reactor has been built and operated and the 300 MWe Monju prototype demonstration plant is under construction. A 1500 MWe commercial plant is planned.

Netherlands. The Dutch LMFBR program consists of a cooperative effort (15% funding) with West Germany, Belgium, and Luxembourg on the SNR-300 project. Emphasis is on establishing the technology of large sodium components such as steam generators, heat exchangers, valves, pumps, etc., and on their prooftesting. Work centers around prooftesting SNR-300 steam generators at the Hengelo 55 MWt test facility, which is capable of testing heat exchangers and other sodium components. The Dutch have other sodium and nuclear test facilities at Petten and Appeldoorn.

Belgium. A sodium loop in the BR-2 (not to be confused with the Russian BR-2) test reactor at Mol, which has been in operation since the

mid-1950s, is Belgium's main LMFBR test facility. This loop is being used for in-reactor fuel tests. Belgium, like the Netherlands, contributes 15% of the funding of the SNR-300 project.

Italy. The LMFBR program in Italy, initiated in the early 1960s by the Comitato Nazionale per l'Energia Nucleare (CNEN), consists mainly of research and development in support of the fuels and materials irradiation tests. Major test facilities are located at Ispra (a Euratom R&D facility) in northern Italy, and at Casaccia, near Rome. PEC, a reactor for testing LMFBR fuel elements under normal, transient, and meltdown conditions, is being built at Brasimone; operation is expected in 1982. Italy also has a tripartite interest in the large commercial LMFBRs (Super-Phenix and SNR-2) to be built in France and Germany.

United States. The unique characteristics and potential of LMFBRs were recognized early in the United States reactor development program. The basic feasibility of the concept has been demonstrated and system characteristics have been explored. Significant factors that have been established include: stable operation of a fast system (the 25 kW Clementine in 1946-48), the breeding principle (the 1 MW EBR-I in the early 1950s), the steady operation of a fast reactor as a heat source for electricity generation (the 20 MWe EBR-II since 1962), fuel recycle (demonstrated by EBR-II), and the capability of a fast reactor to be subjected to maloperation without danger to the public (EBR-I in 1956, the 61 MWe Fermi in 1966, and the 20 MW SEFOR in 1969-72). The 400 MW Fast Flux Test Facility (FFTF) is under construction. Plans to build the 350 MWe CRBR demonstration plant have been indefinitely postponed.

Sodium-cooled reactors not classed as fast reactors also have been built and operated in the United States. These plants, the Submarine Intermediate and the Hallam Nuclear Power Facility (HNPF), provided experience in the use of molten sodium as a reactor coolant and in the technology of high-temperature systems.

The experience with breeders and other sodium-cooled reactors has demonstrated the technical feasibility of designing, building, and operating LMFBR power plants. What remains to be demonstrated is that such plants in sizes of commercial interest can be designed and built to operate economically with assured safety, adequate reliability, and acceptable environmental effects. Proliferation concerns have led to the current reconsideration of the future of breeder reactors.

ADVANCED ISOTOPE SEPARATION (AIS)

As in the United States, the gaseous diffusion process for separating isotopes was used initially in other countries (the U.K., France, USSR, and China) to produce weapons material. In September 1971, the U.K., West Germany, and the Netherlands formed Urenco to design, develop, and test gas-centrifuge technology for enriching uranium. Urenco made its first commercial delivery of enriched uranium in 1975; full-scale operation of a pilot plant is expected in 1978.

APPENDIX C—FISSION ENERGY DEVELOPMENT IN OTHER COUNTRIES

The only detailed public knowledge of AIS development programs in other countries involves a chemical exchange process in France and an aerodynamic method in South Africa.

Research programs involving laser-induced chemistry are known to exist in many countries having nuclear power programs. These efforts vary from scattered university research to structured programs. Structured programs are classified so it is difficult to define their status. No public information exists on projected plans of other countries in the area of laser separation of isotopes.

France. A uniquely French area of activity has been the development of a new technique for uranium enrichment based on a chemical exchange system. Specific technical information about the process has been limited, but it is claimed that this method can be used to go from normal uranium (0.7% U-235) to the nominal 3% U-235 required for LWRs, but that the process cannot be used to prepare the highly enriched (over 90%) material desired for nuclear weapons. This characteristic has stimulated great interest because it would not contribute to the proliferation of nuclear explosives in the world. The French government has reported that it has spent about \$200 million on developing the process, that about 500 people are working on it, and that by the mid-1980s they will have a demonstration facility operational.

West Germany. The Becker nozzle is an aerodynamic method of isotope separation. Presently this process requires about twice the power as the gaseous-diffusion enrichment process or 40 times the power of the gas-centrifuge process. Considerable effort is being devoted to improving the nozzle process.

Union of South Africa. South Africa is developing an aerodynamic method of isotope separation. Although the exact details of this process have not been made public, South Africa expects to demonstrate the commercial feasibility of this technology in the early 1980s.

United States. Selective laser-induced chemistry emerged in the United States in 1971 at the Los Alamos Scientific Laboratory. Since then, the United States program to develop advanced isotope separation technology has grown to an annual program of about \$50 million, in which atomic vapor, molecular, and plasma processes are being developed.

FUEL PROCESSING

Technology for recycling irradiated fuels has been applied successfully to low-exposure fuels in multi-ton/day plants in the United States and abroad for three decades.

France. The French have modified an existing fuel-recycling facility at La Hague to provide capability for recycling high-exposure oxide fuels. The French also participated in United Reprocessors, a group formed for cooperation in the technology and marketing of fuel reprocessing.

U.K. At its British Nuclear Fuels Limited (BNFL) B-205 plant at Windscale, the U.K. has been reprocessing the Magnox-clad natural-uranium fuel from its domestic gas-cooled reactors on a regular basis since 1964 at a rate of 1000 metric tons per year; that plant also is processing the fuel from the two similar reactors exported to Italy and Japan.

In 1969, an older adjacent plant (B-204) was refurbished and head-end equipment was installed for the initial stages of reprocessing high-exposure oxide power reactor fuel. The product from the B-204 head-end process was transferred to the B-205 plant, which had excess capacity. After about 40 MTU of high-exposure oxide fuel had been reprocessed, operation of the B-204 head-end plant was discontinued following a minor accident in 1973. Substantial modifications were decided upon for a capacity of 400 MTU/year. BNFL is planning construction of a new oxide-fuel reprocessing plant, which could be available in the late 1980s and would have a capacity of 100 MTU/year of high-exposure fuel.

The U.K. also participates in United Reprocessors, an international group that cooperates in the technology and marketing of fuel reprocessing.

West Germany. The Federal Republic of Germany (FRG) has formed a loosely organized group, United Reprocessors, with the United Kingdom and France. This group was formed for cooperation in the Technology and marketing of fuel reprocessing. As part of this group, FRG plans to build a plant for reprocessing some 1,400 MTU of spent fuel per year. This plant is scheduled to begin operation in the late 1980s.

Belgium. The Eurochemic plant at Mol had processed a total of about 120 MTU of LWR-type fuel by the time operations were suspended in July 1974. This plant had a rated capacity of 70-100 MTU/year of high-exposure power-reactor fuel. A Belgian group is planning to restart this plant in the 1980s.

United States. The first plant built for recycling high-exposure power-reactor fuels, the Nuclear Fuel Services (NFS, a subsidiary of Getty Oil) plant at West Valley, N.Y., operated for about 5 1/2 years, and was shut down in late 1971 for modification and expansion. This plant, designed to recycle about 300 metric tons of uranium (MTU) of fuel per year, actually recycled a total of about 244 MTU of high-exposure power reactor fuel, plus about 375 MTU of relatively low-exposure fuel from an AEC production reactor. NFS had planned to increase the capacity of this plant to 600-700 MTU per year, but the high cost estimated for the expansion resulted in a decision by NFS to withdraw from recycling.

The General Electric Co. (GE) constructed a fuel recycling plant in Morris, Ill., and had scheduled startup in 1972. However, after identification of problems during cold testing, GE concluded in mid-1974 that the plant was not operable in its present configuration, and that modifications requiring several years and costing in excess \$100 million would be required for the plant to be made operable. At this time, GE has no known plans for operation of this facility.

Allied-General Nuclear Services (AGNS), a partnership of Allied Chemical Corp. and General Atomic Co. (a partnership of Gulf and Shell

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geologic-disposal programs of other countries. There is interest in seabed disposal and the island-disposal concept. International reprocessing, which includes terminal disposition of the waste, would be an extremely attractive service for Japan.

Italy. CNEN, with involvement of Casaccia and Trisaia personnel, is studying the properties of plastic argillaceous sediments in the Trisaia area as possible sites for geologic disposal of radioactive wastes. Plans for future work include additional coring of the clay formation, in situ tests (heat and radiation effects), geological and geophysical surveys, engineering development studies, and risk assessment. If results are favorable, a pilot plant repository could become operational by the mid-1980s.

Switzerland. As a partner in Eurochemic, Switzerland helps support the Eurochemic waste-solidification program. The Federal Institute for Reactor Research at Wurenlingen is evaluating the disposal of nuclear wastes in the Swiss Alps and anhydrite formations in various locations. The program calls for the eventual construction of a disposal facility for low- and intermediate-level (and possibly high-level) wastes.

India. Trombay, India, has developed a pot calcination-vitrification process for high-level waste. The waste is pre-concentrated, fed with additives into the melter, and drained into the waste canister.

Work is underway on geologic waste disposal technology and on the selection of a site for India's radioactive wastes. In India, because of the lack of suitable evaporite deposits, the choice is restricted to igneous rock formations and selected sedimentary deposits. Evaluations are underway of various geological formations in selected nonseismic zones of the country. Data on geological and hydrogeological characteristics, the physical and chemical properties of the rocks, and climatological and seismic data are being collected in coordination with various field agencies. Compilation, correlation, and analysis of the data to screen and identify possible areas to locate repository sites are underway prior to detailed site investigations.

Iran. Iran has a contract with the UKAEA to evaluate the problems associated with the storage and management of spent fuel and radioactive waste from Iran's nuclear power plants. Options considered for spent-fuel handling will include extended pool storage after re-canning, reprocessing either in Iran or abroad, vitrification, and geologic isolation of high-level wastes. Second-stage spent-fuel facilities are required by 1984, when it will be necessary to remove fuel elements from storage ponds at Bushehr. In response to an Iranian invitation, Austria and Iran are negotiating on a proposal to store Austria's nuclear wastes in the Iranian desert.

Argentina. In 1985, work is to be completed on the first stage of an experimental plant for solidifying radioactive liquid waste. No plans have been announced by Argentina for fuel reprocessing, so the waste presumably will come from reactor operations.

Belgium. Once the present Eurochemic high-level waste has been solidified, the Eurochemic facility is to be transferred to the Belgowaste

Co. for treatment of future high-level waste. Belgium is studying the use of caverns in the clay beds that underlie the Mol region for geologic disposal of waste.

Canada. Current Canadian waste-management interest is primarily on interim spent-fuel storage concepts and packaging designs because their nuclear power system is based on the CANDU reactor fueled with natural uranium, from which the spent fuel has limited recoverable economic value. However, Canada is becoming interested in developing a reprocessing capability for the CANDU thorium U-233 fuel cycle, including waste processing. Canada is also interested in geologic terminal storage of high-level waste from reprocessed fuel and is seeking suitable geologic formations for both secure retrievable storage and terminal storage. Canada has followed the United States geologic program closely. Canada has the same types of geologic formations as the United States; if a need for disposal capability arises, Canada could use technology developed and demonstrated by the United States. The Canadian Geologic Survey is continuing to stay abreast of the salt technology and they are investigating the use of crystalline rocks and also are interested in the possible use of a suitable offshore island.

United States. The United States is developing and demonstrating safe processes and facilities for packaging, transporting, storing, and disposing of radioactive wastes. For wastes from commercial power stations, the key goals are to have in operation by the late 1980s a deep geological repository, together with the technical support base for the retrievable storage of spent fuels, and the disposal of high-level and transuranium-contaminated wastes. For defense wastes from the operation of weapons-material production reactors, naval propulsion reactors, and R&D reactors, the goal is to complete the long-term disposition of the backlog of high-level wastes by about the end of this century.

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APPENDIX D

U.S. PARTICIPATION IN INTERNATIONAL COOPERATIVE FISSION ENERGY PROGRAMS

All the technologically advanced nations, as well as some developing countries, have major development programs for the generation of power by fission energy systems. They are investing significant resources to achieve their nuclear power goals. For the United States to participate in, influence, and take advantage of the nuclear development being accomplished worldwide, the United States government has entered into agreements with many countries for the exchange of information, personnel, and cooperative endeavors.

This appendix indicates and briefly describes the cooperation agreements the United States has with other countries. Among the countries with which the United States has bilateral or multilateral agreements for cooperation in fission energy technology are: Canada, France, the Federal Republic of Germany, Japan, Sweden, Switzerland, United Kingdom, and the USSR.

CANADA

There is a DOE/AECL memorandum of understanding effective for 4 years from September 8, 1976 that includes:

- Systems analysis of heavy-water power reactors
 - Economic data, including costs
 - Reactor system, power plant, and associated fuel-cycle characteristics
 - Plant design and regulatory criteria
 - Fuel cycle and support facilities
 - Systems analysis assessment and methodology
- Waste Management
 - Terminal storage in geologic formations
 - Retrievable storage
 - Waste processing
 - Environmental effects

FRANCE

There is a DOE/CEA agreement effective for 4 years from January 18, 1977 covering LMFBF technology that includes:

- Reactor safety
- Reactor physics

APPENDIX D—U.S. PARTICIPATION IN INTERNATIONAL COOPERATIVE FISSION ENERGY PROGRAMS

There is a DOE/BMFT/CEA/AWF agreement in the field of gas-cooled-reactor concepts and technology that extends for 10 years from February 11, 1977 (see Germany, Federal Republic of).

GERMANY, FEDERAL REPUBLIC OF

There is a DOE/BMFT agreement effective for 5 years from December 20, 1974 on radioactive-waste management that includes:

- Incorporation of high-level-waste in glass
- Disposal of radioactive waste in salt deposits
- Retrievable surface-storage facilities
- Waste management R&D
- Waste from decommissioning of nuclear installations
- Operating aspects of storage or disposal of low- and intermediate-level wastes
- Transportation of radioactive wastes

There is a DOE/BMFT agreement effective for 10 years from June 8, 1976 on LMFBFR technology, whose details of cooperation are being negotiated, that includes:

- Reactor neutronics
- Fuels and materials
- Fuel recycle, including reprocessing, refabrication, waste processing, and storage
- Reactor and reactor coolant systems and components
- Coolant technology
- Nonnuclear test facilities in support of LMFBFR programs
- Quality assurance and nondestructive test practices
- Design and operation of LMFBFRs

There is a DOE/BMFT/CEA/AWF agreement effective for 10 years from February 11, 1977 in the field of gas-cooled-reactor concepts and technology that includes:

- Fuels, materials, components, fission products, coolants
- High temperature gas reactor fuel recycle, including reprocessing (inactive), refabrication, waste treatment, and disposal
- High temperature gas reactor steam cycle
- High temperature gas reactor direct cycle
- Very high temperature reactor and process heat, including fuels, materials, systems, and components
- Gas-cooled fast breeder reactors, including components and fuel fabrication
- Safety
- Economic and environmental considerations

There is a DOE/BMFT letter of cooperation effective for 5 years from September 29, 1977 on nuclear-materials safeguards and physical-security R&D, whose details of cooperation are being negotiated, that includes transportation systems for plutonium and highly enriched uranium.

JAPAN

There is a DOE/PNC arrangement effective for 10 years from March 4, 1969 on LMFBR technology that includes:

- Reactor physics
- Reactor safety
- Fuels and materials
- Sodium technology
- Component technology
- Plant experience

Negotiations are underway to expand the scope to include:

- Fuel recycle
- Qualify assurance
- Economic and environmental considerations
- Nonnuclear test facilities

SWEDEN

There is a DOE/SKBF agreement effective for 3 years from July 1, 1977 on radioactive-waste storage that includes:

- Temperature effects in granite rocks
- Long-term effects of waste heat in fractured granitic rock
- Fracture hydrology
- Locations of fracture systems in granitic rock
- Properties of granitic rocks
- Measurement of gross seepage in low-permeability granitic rock
- Virgin stress in fractured granitic rock

SWITZERLAND

There is a DOE/BMFT/CEA/AWF agreement in the field of gas-cooled reactor concepts and technology that extends for 10 years from February 11, 1977 (see Germany, Federal Republic of).

UNITED KINGDOM

There is a DOE/UKAEA agreement effective for 5 years from September 20, 1976 covering LMFBR technology that includes:

- Reactor neutronics
- Reactor safety
- Fuels and materials
- Fuel cycle including fabrication, reprocessing, waste processing, and storage
- Reactor and sodium systems and their associated components
- Sodium technology
- Nonnuclear test facilities that support LMFBR programs
- Quality assurance and nondestructive practices

APPENDIX D—U.S. PARTICIPATION IN INTERNATIONAL COOPERATIVE FISSION ENERGY PROGRAMS

- Overall LMFBF programs of the United States and U.K.
- Operation of LMFBFs
- Economic and environmental considerations
- Topics of interest to industry (design, construction experience, quality assurance)

Details of the cooperation are being negotiated.

USSR

There is a DOE/State Committee for Utilization of Atomic Energy protocol of October 8, 1974 effective indefinitely covering LMFBF technology that includes:

- Fuels and materials
- Reactor systems and components
- Safety
- Reactor design, construction, and operation
- Nonnuclear test facilities
- Economic and environmental considerations

OTHER INTERNATIONAL COOPERATION

The United States is a member and is cooperating with a number of international activities and organizations. These include International Fuel Cycle Evaluation (Table 11), the International Atomic Energy Agency (Tables 12 and 13), the International Energy Agency (Table 12), the Organization for Economic Cooperation and Development (Table 15), and the Nuclear Energy Agency (Table 16).

**APPENDIX D—U.S. PARTICIPATION IN INTERNATIONAL
COOPERATIVE FISSION ENERGY PROGRAMS**

Table 11. International Nuclear Fuel Cycle Evaluation (INFCE)

(Chapter IV, Part B, Section 2.0 describes this activity in greater detail.)

An organizing conference was held October 19-21, 1977.

Goal: To conduct technical and analytical studies of measures which can and should be taken at the national level and through international agreements to minimize the danger of the proliferation of nuclear weapons without jeopardizing energy supplies or the development of nuclear energy for peaceful purposes.

Participants. The 39 countries participating in the evaluation are:

Algeria	India	Philippines
Argentina	Indonesia	Poland
Australia	Iran	Portugal
Belgium	Ireland	Romania
Brazil	Israel	Spain
Canada	Italy	Sweden
Czechoslovakia	Japan	Switzerland
Denmark	Korea	Turkey
Egypt	Mexico	USSR
Finland	Netherlands	United Kingdom
France	Nigeria	United States
German Democratic Republic	Norway	Venezuela
Federal Republic of Germany	Pakistan	Yugoslavia

Scope and Working Groups:

Fuel and Water Availability
 Enrichment Availability
 Assurances of Long-Term Supply
 of Technology, Fuel and
 Heavy Water and Services in
 the Interest of National
 Needs Consistent with
 Nonproliferation
 Reprocessing, Plutonium Han-
 dling, Recycle
 Fast Breeders
 Spent Fuel Management
 Waste Management and Disposal
 Advanced Fuel Cycle and
 Reactor Concepts

Co-Chairmen of Working Groups from:

Canada, Egypt, India
 France, Fed. Rep. of Germany, Iran
 Australia, Philippines, Switzerland

 Japan, UK

 Belgium, Italy, USSR
 Argentina, Spain
 Finland, Netherlands, Sweden
 Republic of Korea, Romania, United
 States

**APPENDIX D—U.S. PARTICIPATION IN INTERNATIONAL
COOPERATIVE FISSION ENERGY PROGRAMS**

Table 12. International Atomic Energy Agency (IAEA)

Established July 29, 1957.

Objectives:

- Promotion of civil applications of nuclear energy.
- Ensuring that assistance provided under IAEA auspices will not be used to further any military purpose.

Major areas of United States DOE involvement:

- Safeguards accountability of nuclear material
- Nuclear safety and standards
- Scientific-information exchange
- Technical-assistance program

110 member states as of December 31, 1977:

Afghanistan	France	Luxembourg	South Africa
Albania	Gabon	Madagascar	Spain
Algeria	German Democratic Republic	Malaysia	Sri Lanka
Argentina		Malitius	Syrian Arab Republic
Australia	Germany, Fed. Republic of	Mauritius	Thailand
Austria		Mexico	Tunisia
Bangladesh	Ghana	Monaco	Turkey
Belgium	Greece	Mongolia	Uganda
Bolivia	Guatemala	Morocco	Ukranian Soviet Socialist Republic
Brazil	Haiti	New Zealand	Union of Soviet Socialist Republics
Bulgaria	Holy See	Nicaragua	United Arab Emirates
Burma	Hungary	Niger	United Kingdom of Great Britain and Northern Ireland
Byelorussian Soviet Socialist Rep.	Iceland	Nigeria	United Rep. of Cameroon
Cambodia	India	Norway	United Rep. of Tanzania
Canada	Indonesia	Pakistan	United States of America
Chile	Iran	Panama	Uruguay
Colombia	Ireland	Paraguay	Venezuela
Costa Rica	Israel	Peru	Yugoslavia
Cuba	Italy	Philippines	Zaire
Cyprus	Ivory Coast	Poland	Zambia
Czechoslovakia	Jamaica	Portugal	
Democratic People's Republic	Japan	Qatar	
Denmark	Jordan	Republic of South Viet-Nam	
Dominican Republic	Kenya	Romania	
Ecuador	Korea, Republic of	Saudi Arabia	
Egypt	Kuwait	Senegalrabia	
El Salvador	Lebanon	Sierra Leone	
Ethiopia	Liberia	Singapore	
Finland	Libyan Arab Republic	Socialist Republics	
	Liechtenstein		

Table 13. Selected IAEA Activities

Safeguards (began in 1968)

Scope: (a) Studies, analyses, and training, (b) technology, and (c) information processing.

Status: (a) Training courses at Vienna and LASL in physical-protection techniques and materials accountability, (b) United States member of senior safeguards advisory committee, and United States provides advanced nondestructive radioassay instruments to inspectors, and (c) software and data base being established.

Nuclear Safety and Environmental Protection (began in 1959)

Scope: (a) Regulations for safe transport of radioactive material (DOT, DOE, and NRC are lead United States agencies), (b) definition and recommendations on high-level radioactive waste materials unsuitable for sea dumping (DOE and EPA are lead United States agencies), and (c) safety guides agencies and codes of practice for nuclear power reactor design, siting, quality assurance, operation, and government organization and regulations (NRC is lead agency).

Status: (a) Updated in 1973, (b) preliminary data in 1974, (c) to be updated in 1978, and (d) initiated in 1975 with a target date of 1980.

Participants: Most IAEA member nations.

Scientific Information Exchange & Technology Assistance Program (began in 1959)

Scope: (a) Scientific-information exchange, including International Nuclear Information System (INIS), (b) fusion technology, (c) world power and research reactors, and (d) technical symposia and publications.

Status: (a) Atomindex now contains nearly 100% of world's nuclear literature (computerized), (b) includes catalog of all fusion installations and devices and key personnel (not computerized as yet), (c) an updated listing of essential information on all power and research reactors in the world, and (d) nuclear-energy information is disseminated annually through numerous technical publications and about 16 international seminars and symposia.

Participants: All IAEA members.

Technical Assistance Program (began in 1959)

Scope: The United States has been providing fellowship training in the United States, equipment and experts for IAEA's less-developed members, expert advisors, training courses, safeguards assistance (experts, equipment, training), and sponsored the regional fuel-cycle-centers study.

Participants: The majority of IAEA members (i.e., less-developed countries).

**APPENDIX D—U.S. PARTICIPATION IN INTERNATIONAL
COOPERATIVE FISSION ENERGY PROGRAMS**

Table 14. International Energy Agency (IEA)

Established November 18, 1974, by the treaty, Agreement on an International Energy Program.

Purposes:

- Promotion of secure oil supplies
- Emergency oil self-sufficiency
- Cooperation with producer and consumer nations
- Long-term cooperative efforts in
 - Conservation
 - Alternative sources
 - Energy R&D
 - Nuclear energy

19 member nations:

Austria	Greece	Netherlands	Switzerland
Belgium	Ireland	New Zealand	Turkey
Canada	Italy	Norway	United Kingdom
Denmark	Japan	Spain	United States
Germany (FRG)	Luxembourg	Sweden	

**IEA R&D Working Parties (established by
IEA Committee on Energy R&D)**

Lead Country

Energy R&D strategy	United States
Nuclear safety	United States
Radioactive-waste management	Nuclear Energy Agency
High-temperature reactors for process heat	FRG
Conservation	United States
Coal	U.K.
Biomass	Ireland
Geothermal	Italy
Solar energy	Japan
Small solar systems	Austria
Wind power	Netherlands
Ocean energy systems	U.K.
Hydrogen	European Community
Fusion	European Community

Table 15. Organization for Economic Cooperation and Development (OECD)

Formed in 1948 as the Organization for European Economic Cooperation; expanded in 1960 and renamed as the OECD.

Objectives:

- Promote economic growth of member countries
- Assist less-developed countries
- Promote trade expansion worldwide

Areas of interest:

- Economic policy cooperation	- Trade, financial, and fiscal policy
- Energy (see IEA)	- Social affairs, manpower, education
- Nuclear energy for peaceful purposes (see NEA)	- The environment
- Development cooperation	- Science, technology, industry
	- Agriculture and fisheries
	- Waste management

24 member nations:

Australia	Greece	Norway
Austria	Iceland	Portugal
Belgium	Ireland	Spain
Canada	Italy	Sweden
Denmark	Japan	Switzerland
Finland	Luxembourg	Turkey
France	Netherlands	United Kingdom
Germany (FRG)	New Zealand	United States
	Special status: Yugoslavia	

**APPENDIX D—U.S. PARTICIPATION IN INTERNATIONAL
COOPERATIVE FISSION ENERGY PROGRAMS**

Table 16. Nuclear Energy Agency (NEA) of the OECD

Established in 1957, as the European Nuclear Energy Agency; name changed to NEA in 1972 when Japan and Canada joined.

Objective:

- Orderly development of the peaceful uses of atomic energy.

Priority activities:

- Nuclear safety
- Radioactive waste management
- Radiation protection and public health
- Nuclear energy development including:
 - uranium resources
 - production demand forecasts
 - nuclear science

Members:

There are 23 OECD member countries, plus New Zealand as associate member. The United States became a full member in October 1976, after 19 years as an associate member.

APPENDIX E

DEFINITIONS

Advanced Converter Reactor (ACR). A reactor system that provides better uranium fuel utilization than that obtained in present commercial Light Water Reactors (LWRs), but not as good as a breeder.

Applied Research. Systematic study directed toward fuller scientific knowledge or understanding for direct use in fulfilling specific energy requirements.

Basic Research. Systematic, fundamental study directed toward fuller scientific knowledge or understanding of subjects bearing on national energy needs.

Blanket. Consists of fertile material. See also "Fertile Material" and "Seed and Blanket."

Breeder Reactor. A reactor that creates more fissile fuel than it consumes; the process by which this occurs is known as breeding. The new fissile material is created by capture in fertile materials of neutrons from fission.

Commercial Demonstration Plants. The purposes of commercial demonstration plants are: (1) to resolve commercial investment uncertainties by establishing the actual economic factors, environmental feasibility, socioeconomic impact, capital and resource requirements, and constraints, and (2) to encourage creation of a viable industry using these technologies. They have an output of about 3 to 5 times that of demonstration plants, and they demonstrate the viability of a facility and/or the process to compete commercially.

Conversion Ratio. The ratio of fissile fuel that can be reclaimed after reactor operation to that installed in a reactor prior to operation. For an Advanced Converter Reactor, this ratio is less than 1; for a Breeder Reactor the ratio is greater than 1.

Demonstration. Verification of economic and environmental viability for commercial application, through design, construction, test, and evaluation of large-scale energy systems in operational circumstances.

Demonstration Plants. The purpose of demonstration plants is to demonstrate and validate economic, environmental, and productive capacity of a near-commercial-size plant by integrating and operating a single modular unit using commercial-size components. They are still developmental in the sense that technological scaleup problems may occur and require engineering modification; however, the risk is much lower because the plant production process was developed and tested at the pilot stage.

APPENDIX E—DEFINITIONS

Depleted Uranium. Uranium having a lower concentration of the uranium-235 isotope than that found in natural uranium, i.e., uranium containing less than 0.71% U-235.

Engineering Development. Systematic use of the knowledge and understanding gained from research and technology development to achieve the detailed design, construction, and test for performance, producibility, and reliability of energy-system prototypes and pilot plants.

Enriched Uranium. Uranium having a higher concentration of the uranium-235 isotope than found in natural uranium, i.e., uranium containing more than 0.71% U-235.

Fertile Material. Normally nonfissile material that can, through the absorption of a neutron, be transmuted to a fissile fuel.

Fissile Fuel. Nuclear fuel that is capable of sustaining a nuclear chain reaction. Examples of fissile fuel are uranium-233 and uranium-235.

Light Water. Ordinary water (H_2O), as distinguished from heavy water (D_2O).

Naval Reactors. Explanation of plant designations:

First symbol is ship type:

- A = aircraft carrier
- C = cruiser
- D = formerly destroyer/frigate, now cruiser
- S = submarine

Second symbol is sequential number of plant designed by prime contractor for that type ship.

Third symbol is prime contractor that designed plant:

- C = Combustion Engineering
- G = General Electric
- W = Westinghouse
- e.g., SIW = First submarine plant designed by Westinghouse

- Explanation of reactor core designations:

First three symbols designate plant for which core is designed

Following symbols show sequential number of core designed for plant application; e. g., SIW-3 = third core designed for SIW type plant

- Explanation of ship designations:

Alphabetic prefix indicates ship class:

- CV = aircraft carrier
- CG = guided-missile cruiser
- SS = attack submarine
- SSB = ballistic-missile submarine
- NR = research ship

Symbol "N" denotes ship is propelled by a nuclear reactor plant; symbol is omitted for conventionally propelled ships.

Numerical suffix identifies a particular ship of the class; e.g., CVN 68 identifies USS NIMITZ, a nuclear powered aircraft carrier.

Pilot Plant. The purpose of a pilot plant is to establish the integrated process feasibility by combining commercial-type (not commercial-size) components into a small model plant to test and evaluate the critical parameters of scaleup and to acquire engineering data needed to assess economic feasibility and design a larger near-commercial-size plant. Pilot plants are usually short lived and are the first scaleup facility to produce enough end product to permit product testing and refinement.

Prebreeder Core. A reactor core fueled with available fissile fuel such as uranium-235 that produces uranium-233 fissile material from thorium while generating commercial electric power. The uranium-233 is then available for manufacture of fuel for a reactor core.

Reflector. A layer of material that immediately surrounds a reactor core and scatters back (reflects) into the core, or captures in a fertile material, many neutrons that would otherwise escape. These neutrons can then cause more fissions or create fissile fuel and thus improve the neutron economy of the reactor. Common reflector materials are graphite, water, and beryllium, and fertile materials include natural uranium, depleted uranium, and thorium.

Seed and Blanket. A reactor core that includes a relatively small volume of material with a high content of fissile material (the seed) surrounded by a much larger volume of material with a high content of fertile material (the blanket). As a result of fissions in the seed, neutrons are supplied to the blanket where new fissile fuel is produced and more fission takes place. In this way, the blanket is made to furnish a substantial fraction of the total power of the reactor.

Technology Development. Systematic use of the knowledge and understanding gained from research to achieve technical feasibility and to gauge economic and environmental potential of energy concepts, processes, materials, devices, methods, and subsystems.

Thoria. Thorium oxide (ThO_2).

Urania. Uranium oxide (UO_2).

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APPENDIX F

ACRONYMS AND LETTER DESIGNATIONS

ACR	Advanced Converter Reactor
ACRP	Advisory Committee on Reactor Physics
ACRS	Advisory Committee on Reactor Safeguards
A-E	Architect-Engineer
AEA	Atomic Energy Authority (UK)
AEC	United States Atomic Energy Commission (subsequently ERDA and NRC, now DOE and NRC)
AECL	Atomic Energy of Canada Ltd.
AEM	Acoustic Emission Monitoring
AETL	Approved Engineering Test Laboratories
AETR	Advanced Epithermal Thorium Reactor
AFCT	Alternate Fuel Cycle Technologies
AFR	Away From Reactor (Water Storage Basins)
AFSR	Argonne Fast Source Reactor (ANL)
AGC	Aerojet General Corp.
AGHCF	Alpha Gamma Hot Cell Facility (ANL)
AGNS	Allied General Nuclear Services
AGR	Advanced Gas-cooled Reactor
AI	Atomics International Div., Rockwell International Corp.
AIF	Atomic Industrial Forum, Inc.
AIPA	Accident Initiation and Progression Analysis
AIS	Advanced Isotope Separation (Program)
AMCF	Alkali Metal Cleaning Facility (HEDL)
ANC	Aerojet Nuclear Co.
ANL	Argonne National Laboratory
ANL-E	Argonne National Laboratory, East (Illinois)
ANL-W	Argonne National Laboratory, West (Idaho)
ANS	American Nuclear Society
AlG	Aircraft carrier reactor--core design by GE (operator of KAPL); number indicates model
AlW	Aircraft carrier reactor--core design by Westinghouse Electric Corp. (operator of BAPL); number indicates model
APDA	Atomic Power Development Associates, Inc.
ARD	Advanced Reactors Div., Westinghouse Electric Corp.
ARCO	Atlantic Richfield Co.
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
ASNPP	Advanced Submarine Nuclear Propulsion Plant
ATR	Advanced Test Reactor (ANC/INEL)
ATSR	Argonne Thermal Source Reactor (ANL)
AVR	Arbeitsgemeinschaft Versuchsreaktor
AWBA	Advanced Water Breeder Applications
AWF	Office of Science and Research (Switzerland)
BAPL	Bettis Atomic Power Laboratory
BIPS	Brayton Isotope Power System
BLTC	Bottom Loading Transfer Cask
BMFT	Federal Ministry for Research & Technology (FRG)

APPENDIX F—ACRONYMS AND LETTER DESIGNATIONS

BMI	Battelle Memorial Institute
BN-350,-600	Russian LMFBRs
BNFS	Barnwell Nuclear Fuel Services
BNL	Brookhaven National Laboratory
BNWL	Battelle Northwest Laboratories
BOM	Bureau of Mines (U.S. Dept. of Interior)
BOR-60	Russian LMFBR
BR-1,-2,-5,-10	Russian LMFBRs
BR-2	Belgian test reactor
BRC	Breeder Reactor Corp.
B&W	Babcock and Wilcox Co.
B&R	Burns & Roe, Inc.
BWR	Boiling-Water-cooled reactor
CANDU	Canadian-Deuterium-Uranium Reactor
CDA	Core Disruptive Accident
CDF	Component Development Facility
CE	Combustion Engineering, Inc.
CEA	Commissariat a l'Energie Atomique (France)
CEL	Carbon Equilibrium Loop (Ward)
CERN	European Organization for Nuclear Research
CESNEF	Centro Studi Nucleari 'Enrico Fermi' (Italy)
CFR	Commercial Fast Reactor (United Kingdom)
CGN	Cruiser, Guided-missile, Nuclear propulsion
CHCF	Component Handling and Cleaning Facility (LMEC)
CLEM	Closed Loop Ex-Vessel Handling Machine (FFTF)
CLIRA	Closed Loop In-Reactor Assembly (FFTF)
CMM	Core Mechanical Mockup (HEDL)
CNEN	Comitato Nazionale per l'Energia Nucleare (Italy)
CNWW	Commercial Nuclear Waste Vitrification (Demonstration)
CP-5	Chicago Pile No. 5 (ANL-E)
CRBR	Clinch River Breeder Reactor
CRBRP	Clinch River Breeder Reactor Plant
CRCTA	Composite Reactor Component Test Activity (HEDL)
CRDM	Control-Rod Drive Mechanism
CRM	Core Restraint Mechanism
CRT	Cathode-Ray Tube
CSDD	Conceptual System Design Description
CVN	Carrier (aircraft) Vessel, Nuclear propulsion
DAS	Data Acquisition System (EBR-II)
DBA	Design Basis Accident
DCHT	Direct-cycle Helium Turbine
DFR	Dounreay Fast Reactor (United Kingdom)
DHX	Dump Heat Exchanger
DOD	United States Department of Defense
DOE	United States Department of Energy
DIG	Destroyer reactor-core design by GE (operator of KAPL); number indicates model
D2W	Destroyer reactor-core design by Westinghouse Electric Corp. (operator of BAPL); number indicates model
D2O	Deuterium Oxide, Heavy Water
EACRP	European-American Committee on Reactor Physics
EANDC	European-American Nuclear Data Committee
EBR-I	Experimental Breeder Reactor No. 1 (ANL-W/INEL)

APPENDIX F—ACRONYMS AND LETTER DESIGNATIONS

EBR-II	Experimental Breeder Reactor No. 2 (ANL-W/INEL)
ECCS	Emergency Core Cooling System
ECF	Expended Core Facility (Naval Reactors Program)
ECF	Electricite de France
EDF	Engineering Demonstration Facility
EEI	Edison Electric Institute
EIS	Environmental Impact Statement
ENDF	Evaluated Nuclear Data File
ENEA	European Nuclear Energy Agency
Enr.	Enriched
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
ERDA	United States Energy Research and Development Administration (now DOE)
ET	Energy Technology (DOE)
ETN	Directorate of Nuclear Energy Programs in DOE
ETR	Engineering Test Reactor (INEL)
EURATOM	European Atomic Energy Community
EURFNR	United States -EURATOM Fast Reactor Exchange Program
EVHM	Ex-Vessel Handling Machine
FBR	Fast Breeder Reactor
FEDAL	Failed-Element Detection and Location System
FEFPL	Fuel Element Failure Propagation Loop (now SLSF)
FERMI	Enrico Fermi Atomic Power Plant (LMFBR)
FFM	Fuel Failure Mockup (ORNL)
FFTF	Fast Flux Test Facility (HEDL)
FIV	Flow-Induced Vibration
FMEF	Fuels and Materials Evaluation Facility
FNG	Fast Neutron Generator (ANL)
FRG	Federal Republic of Germany (West Germany)
FSA	Fueled Sphere Assembly
FSAR	Final Safety Analysis Report
FSV	Fort St. Vrain, United States 330-MWe HTGR plant
FTR	Fast Test Reactor (now FFTF)
FWC	Foster-Wheeler Corp.
FY	Fiscal Year
GA	General Atomic (Gulf/Shell)
GCFR	Gas-Cooled Fast Reactor
GCR	Gas-Cooled Reactor
GCRA	Gas-Cooled Reactor Associates (HTGR utility group)
GCTR	Gas-Cooled Thermal Reactor
GCUA	Gas-Cooled Reactor Umbrella Agreement (international agreement on Gas-cooled Reactors)
GDS	Ground Demonstration System (isotopic generator, Space Program)
GE	General Electric Co.
GEIS	Generic Environmental Impact Statement
GE-NED	General Electric Co. Nuclear Engineering Div.
GETR	General Electric Test Reactor (GE)
GPHS	General Purpose Heat Source (Solar Polar Mission, Space Program)
GPL	General Purpose Loop (WARD)
GRT	Gulf Radiation Technology, Inc.

APPENDIX F—ACRONYMS AND LETTER DESIGNATIONS

HCDA	Hypothetical Core Disruptive Accident
HCM	Hydraulic Core Mockup (HEDL)
HCTF	Helium Component Test Facility
HEDL	Hanford Engineering Development Laboratory
HEF	Hot Experimental Facility
HEPA	High-Efficiency Particulate Absolute (Filter)
HETF	Hot Engineering Test Facility
HEU	High-Enrichment Uranium
HFEF	Hot Fuel Examination Facility (ANL-W/INEL)
HFEF/N	HFEF-North
HFEF/S	HFEF-South (formerly FCF and FEF)
HFIR	High-Flux Isotope Reactor (ORNL)
HHT	German direct cycle HTGR program
HNPF	Hallam Nuclear Power Facility
HPFL	High Performance Fuel Laboratory
HPP	Hot Pilot Plant (fuel reprocessing)
HRI	Houston Research Institute
HRLEL	High-Radiation-Level Examination Laboratory (ORNL)
HTGR	High-Temperature Gas-cooled Reactor
HTGR-GT	HTGR with gas-turbine power conversion
HTGR-SC	HTGR with steam-cycle power conversion
HTL-1	Heat Transfer Loop No. 1 (HEDL)
HTR	High Temperature Reactor
HTSF	High Temperature Sodium facility (HEDL)
HTTF	High Temperature Critical Test Facility (Naval Reactors Program)
HVEM	High Voltage Electron Microscope (HEDL)
HWOCR	Heavy-Water-moderated Organic-Cooled Reactor
HWR	Heavy-Water Reactor
H ₂ O	Ordinary (light) Water
I&C	Instrumentation and control
IAEA	International Atomic Energy Agency
ID	Idaho
IEA	International Energy Agency
IEMTF	Interim Examination and Maintenance Training Facility (HEDL)
IET	Integrated Equipment Test (Facility)
IHX	Intermediate heat exchanger
ILW	Intermediate-Level Waste
IMGA	Irradiated Microsphere Gamma Analyzer System
INC	Idaho Nuclear Corp.
INCOT	In-Core Instrument Test Facility (EBR-II)
INEL	Idaho National Engineering Laboratory (formerly NRTS)
INFCE	International Nuclear Fuel Cycle Evaluation
INSAT	Instrumented Subassembly Test Facility (EBR-II)
INSRP	Interagency Nuclear Safety Review Panel
INTERATOM	Internationale AtomreaktorbaugmbH
IR	Infrared
ITF	Interstitial Transfer Facility (WARD)
IVHM	In-vessel handling machine
IWGFR	International Working Group for Fast Reactors (IAEA)
JAERI	Japan Atomic Energy Research Institute
JAEC	Japanese Atomic Energy Commission
JOP	Jupiter Orbiter/Probe Mission (NASA)

APPENDIX F—ACRONYMS AND LETTER DESIGNATIONS

JOYO	Japanese LMFB
JAPFRX	
JCAE	Joint Committee on Atomic Energy (United States Congress)
KAPL	Knolls Atomic Power Laboratory
KFA	German Energy Research Laboratory
KIPS	Kilowatt Isotope Power System
KNK	Kompakte Natriumgekuhlte Kernreaktorenlage, compact sodium-cooled nuclear-reactor plant (Germany)
KW	Kilowatt
LAMPRE	Los Alamos Molten Plutonium Reactor Experiment
LASL	Los Alamos Scientific Laboratory
LES	Lincoln Experimental Satellites
LINAC	Linear Accelerator
LLL	Lawrence Livermore Laboratory
LLTR	Large Leak Test Rig (LMEC)
LLTV	Large Leak Test Vessel
LLW	Low-Level Waste
LMEC	Liquid Metals Engineering Center
LMFBR	Liquid Metal Fast Breeder Reactor
LOFT	Loss-of-Flow Test (INEL)
LRL	Lawrence Radiation Laboratory
LSBR	Large Seed Blanket Reactor
LWBR	Light Water Breeder Reactor
LWR	Light Water Reactor
M \$	Millions of dollars
MARF	Modifications and Additions to Reactor Facilities (Naval Reactors Program)
M bbl/yr	Millions of barrels per year
MEU	Medium-Enrichment Uranium
MHW	Multi-Hundred Watt (Heat Source, Space Program)
MJS	Mariner Jupiter/Saturn (Voyager Mission)
MLM	Mound Laboratory Monsanto
M lb/yr	Millions of pounds per year
MPS	Mechanical Properties Systems -1,-2,-3 (WARD)
MSAR	Mine Safety Appliances Research Corp.
MSBR	Molten Salt Breeder Reactor
MSG	Modular Steam Generator
MSRE	Molten-Salt Reactor Experiment (ORNL)
MTI	Mechanical Technology, Inc.
MTL-1,-2	Materials Test Loops-1,-2 (WARDS)
MTU	Metric tons of uranium
MWe	Megawatts, electrical
NACF	Sodium Cleaning Facility (WARD)
NAL	National Accelerator Laboratory
NASA	National Aeronautics and Space Administration
NASAP	Nonproliferation Alternate Systems Assessment Program
NBS	United States National Bureau of Standards
NCBR	Near-Commercial Breeder Reactor
NEP	Nuclear Fuel Services, Inc.
NFS	National Oceanographic and Atmospheric Administration
NOAA	Nuclear-propelled research submarine ("Deep Submergence Research Vehicle")
NR-1	

APPENDIX F—ACRONYMS AND LETTER DESIGNATIONS

NRC	United States Nuclear Regulatory Commission
NRF	Neutron Radiography Facility (HEDL)
NRL	Naval Research Laboratory
NRTS	National Reactor Testing Station (now INEL)
NWCF	New Waste Calcining Facility
NWTS	National Waste Terminal Storage (Program)
OECD	Organization for Economic Cooperation and Development
OER	Office of Energy Research (DOE)
ORELA	Oak Ridge Electron Linear Accelerator
ORNL	Oak Ridge National Laboratory
ORNL-ILW	ORNL Intermediate-Level Waste
ORR	Oak Ridge Research Reactor (ORNL)
ORRDACS	Oak Ridge Reactor Data Acquisition System
PAHR	Post-Accident Heat Removal
PAL	Prototype Applications Loop (HEDL)
PBF	Power Burst Facility (ANC/INEL)
PCI	Pellet-Clad Interaction
PCML	Plutonium Critical Mass Laboratory (PNL)
PCRV	Prestressed Concrete Reactor Vessel
PCTF	Plant Component Test Facility
PDU	Process Development Unit
PEC	Prova Elementi di Combustibile, fuel-element test reactor (Italy)
PEF	Plutonium Experimental Facility
PFR	Prototype Fast Reactor (United Kingdom)
PHENIX	French LMFBR
PICS	Post-Impact Containment Shell
PLFFF	Pilot-Line Fuel Fabrication Facility
PISA	Post-Impact Shell Assembly
PLBR	Prototype Large Breeder Reactor
PMC	Project Management Corp.
PMIS	Precision Mechanisms in Sodium Loop (LMEC)
PNC	Power, Reactor and Nuclear Fuel Development Corp. (Japan)
PNL	Pacific Northwest Laboratory (Battelle)
PNP	German process heat program
PNR	Pittsburgh Naval Reactors Office
PRDC	Power Reactor Development Co.
PSAR	Preliminary Safety Analysis Report
PSTF	Pump Seal Test Facility (LMEC)
Pu	Plutonium
PuFF	Plutonium Fuel Facility (SRP)
PWR	Pressurized-Water-cooled Reactor
QA	Quality Assurance
R&D	Research and development
RAPSODIE	French LMFBR
RAPTUS	Rapido Toria Uranio Sodiu (Italy)
RF	Rocky Flats
RFP	Request for proposal
RFREPP	Remote Fuels Refabrication Engineering Pilot Plant
RHO	Rockwell Hanford Operations
RL	Richland Operations Office (ERDA, now DOE)
RM	Reactor manufacturer
RML	Radiometallurgy Laboratory (HEDL)
RSCL	Radioactive Sodium Chemistry Loop (EBR-II)

APPENDIX F—ACRONYMS AND LETTER DESIGNATIONS

RTG	Radioisotope Thermoelectric Generator
RV	Reactor vessel
S/A	Subassembly
SAEA	Southwest Atomic Energy Associates
SAR	Safety Analysis Report
SAREF	Safety Research Experiment Facility (formerly STF)
SCEL	Small Components Evaluation Loop (HEDL)
SCTI	Sodium Components Test Installation (LMEC)
SCTL	Small Components Test Loop (LMEC)
SDD	System Design Description
SEFOR	Southwest Experimental Fast Oxide Reactor
S8G	Submarine reactor-core design by GE (operator of KAPL); number indicates model
SELNI	Societe Elettronucleare Italiana
SERF	Special Environment Radiometallurgy Facility (HEDL)
SSBN	Submarine Ship, Ballistic missile, Nuclear propulsion
SG	Steam generator
SGR	Sodium-Graphite Reactor
SGTF	Steam Generator Test Facility (ANL-E)
SGTR	Steam Generator Test Rig (GE)
SIG	Selenide Isotope Generator
SIPS	Stirling Isotope Power System
SIR	Submarine Intermediate Reactor
SLSF	Sodium Loop Safety Facility (formerly FEFPL) (ANL-W/INEL)
SNR	Schnell Natriumgekuhlt Reaktor, fast sodium-cooled reactor (Germany)
SNR	Schenectady Naval Reactors Office
SLC	Submarine reactor-core design by Combustion Engineering, Inc.; number indicates model
SIW	Submarine reactor-core design by Westinghouse Electric Corp. (operator of BAPL); number indicates model
SOWAT	Small-Leak Sodium-Water Reaction Test Rig (GE)
SPL	Sodium Purification Loop (HEDL)
SPTF	Sodium Pump Test Facility (LMEC)
Sr	Strontium
SR	Savannah River, South Carolina
SRDA	Sodium Removal Development Apparatus (HEDL)
SRE	Sodium Reactor Experiment
SRI	Stanford Research Institute
SRL	Savannah River Laboratory
SRP	Savannah River Plant
SSN	Submarine Ship, Nuclear propulsion
STCL	Source Term Control Loop (HEDL)
STF	Safety Test Facility (now SAREF)
STP	Space Test Program (DOD)
SURF	Surface Unreprocessed Fuel (Facility)
SWRI	Southwest Research Institute
SWEATER	Sodium Friction and Wear Test Rig (<u>WARD</u>)
S&W	Stone & Webster, Inc.
TFCT	Thorium Fuel Cycle Technology (Program)
Th	Thorium
ThO ₂	Thoria, Thorium Oxide, Thorium Dioxide
THTR	Thorium High-Temperature Reactor

APPENDIX F—ACRONYMS AND LETTER DESIGNATIONS

TITLE I	Preliminary Design Phase of Project
TITLE II	Final Design and Construction Phase of Project
TREAT	Transient Reactor Test Facility (ANL-W/INEL)
TRU	Transuranium
TSF	Tower Shielding Facility (ORNL)
TTF	Thermal Transient Facility (LMEC)
TTL	Transient Test Loop (HEDL)
TTTF	Thermal Transient Test Facility (ORNL)
TURF	Thorium-Uranium Recycle Facility (ORNL)
TVA	Tennessee Valley Authority
U-233	Uranium isotope with atomic weight 233
U-235	Uranium isotope with atomic weight 235
U ₃ O ₈	Uranium oxide
UE&C	United Engineers & Constructors, Inc.
UKAEA	United Kingdom Atomic Energy Authority
UNC	United Nuclear Corp.
UO ₂	Urania, Uranium oxide, Uranium Dioxide (the form suitable for reactor fuel)
UV	Ultraviolet
VHTR	Very High Temperature (gas-cooled) Reactor for high-temperature process applications
VTRZ	Variable Temperature Rodded Zone
W	Westinghouse Electric Corp.
WARD	Westinghouse Advanced Reactors Div.
WASH (#)	The serial numbering system for reports issued by the AEC
WCBR	Water-Cooled Breeder Reactor
WHC	Westinghouse Plant Apparatus Division
WIPP	Waste Isolation Pilot Program
WPAD	Westinghouse Plant Apparatus Division
YAG	Yttrium Aluminum Garnet
ZPPR	Zero Power Plutonium Reactor (ANL-W/INEL)
ZPR	Zero Power Reactor

SYMBOL DEFINITIONS

Begin Milestone

End Milestone

Decision Milestone

Begin Milestone (Task Completed)

End Milestone (Task Completed)

Decision Milestone (Decision Made)

● Release Request for Proposals
1 or Program Opportunity Notice

●
2 Proposals Received

●
3 Contractor Selected

●
4 Contract Award

●
5 Order Long-Lead Items

●
6 Final Environmental Impact Statement

Information Transfer

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