FINAL REPORT

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COMPARATIVE EVALUATION OF SOLAR, FISSION, FUSION, AND FOSSIL ENERGY RESOURCES

PART II

POWER FROM NUCLEAR FISSION

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ACKNOWLEDGMENT

The permission of the Nuclear Assurance Corporation to reproduce graphs from its Fuel-trac service (references 5 and 6) for use in this report is gratefully acknowledged.

POWER PRODUCED BY NUCLEAR FISSION REACTORS

INTRODUCTION

Nuclear power is now (1974) producing approximately 5% of the electrical power in the United States. It has been estimated that by the year 2000 power from nuclear energy will equal or exceed that produced by fossil sources. It appears that the recent crisis in energy and oil has led to a series of events which will speed up dramatically the role of nuclear power in the United States.

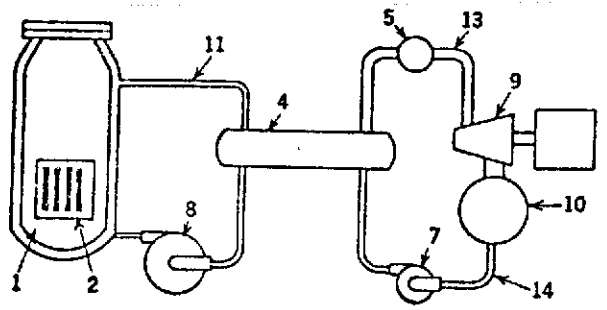
It is well known that the fissile nulcear fuels are: uranium-235, which composes 0.7% of natural uranium, the odd isotopes (Pu-239 and Pu-241) of plutonium which are produced by the neutron irradiation of the fertile U-238, and U-233 which is produced by the neutron irradiation of thorium as found in nature. The present generation of power reactors in the United States are mostly light water reactor (LWR) moderated and cooled, using slightly enriched uranium as uranium dioxide for the fuel. Of the light water reactors, the pressurized water reactor is manufactured by the Westinghouse Electric Corporation, the Babcock and Wilcox Corporation, and the Combustion Engineering Corporation. The boiling water reactor is manufactured by the General Electric Corporation. The Gulf General Atomic Corporation is producing a gas-cooled graphite moderated thermal reactor. The LWR's and HTGR's are converter reactors, that is, the fissile isotopes which are produced in the course of energy production are less than those used up.

Another class of fission reactors which is predicted to become important in approximately the year 2000 is the breeder reactor. In a breeder reactor the fissionable material which is produced is in excess of that which is

utilized for the energy production. For example, in a neutron irradiation of U-238 more plutonium could be produced than uranium consumed. There are two important candidates for breeders which shall be considered later, namely the liquid/metal cooled fast breeder reactor (LMFBR) and the gas cooled fast breeder reactor (GCFBR). Figure 1 (taken from reference 1) indicates the four most important reactor systems under consideration.

Some of the important differences between nuclear fuels and fossile fuels are as follows:

1. Nuclear fuels, as compared to fossile fuels, are fabricated in a chain of development processes which encompass a large high technology nuclear fuels industry, and involves a complex fuel cycle.
2. The procurement of nuclear fuels requires very long lead times. In order to procure a core loading, orders for nuclear fuels must be made several years before the fuels are in the reactor.
3. Nuclear fuels are costly and require a large initial investment many months before use. As a result, one must consider carrying charges as an important factor in computing the nuclear fuel costs.
4. Another difference between the nuclear fuels and fossil fuels is that the irradiated reactor fuel when taken out of the reactor has a high residual value. This is a consequence, of course, of the fact that all of the Uranium-235 or fissile fuel is not burned up in the reactor and also the fact that plutonium may be produced by the irradiation of the U-238. Hence, the high residual value of the fuel requires a reprocessing operation and a storage operation which must be considered in fuel cycle cost calculations.
5. The irradiated fuel discharged from the reactor is radioactive and

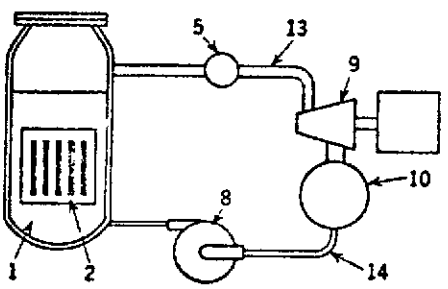


Legend*:

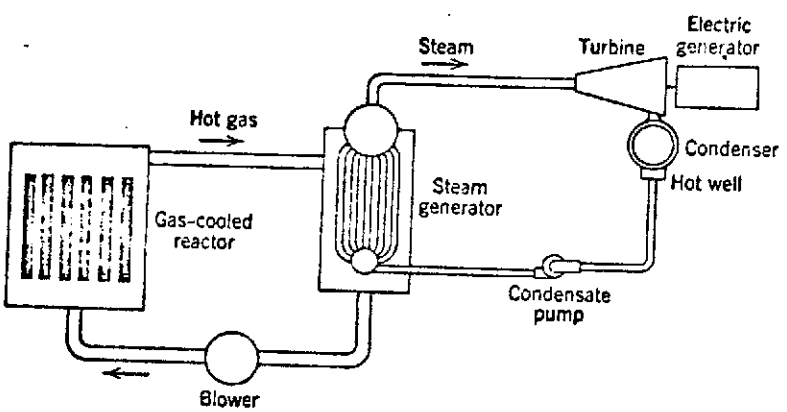
1 Reactor
2 Core
3 Blanket
4 Boiler
5 Steam drier
6 Intermediate heat exchanger
7 Feed water pump
8 Circulating pump
9 Turbogenerator
10 Condenser
11 Primary coolant
12 Intermediate coolant
13 Steam
14 Condensate

Figure 1. Diagrams of Reactor System[1]

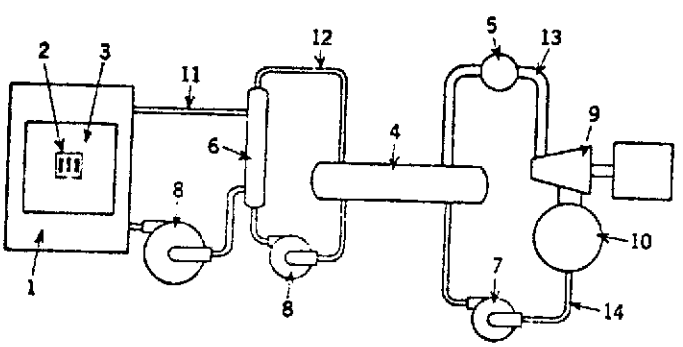
Pressurized-water reactor system.



Direct-cycle boiling-water reactor.



Gas-cooled reactor system for steam generation.



Fast-breeder reactor system.

poses a problem of storing for a time interval before the fuel can be shipped. It also entails a very difficult problem for the ultimate disposal of the nuclear wastes.

THE REACTOR AS A COMPONENT IN THE FUEL AND THE POWER SYSTEMS

In understanding the role of a nuclear fission reactor in the production of nuclear power it is convenient to consider the reactor as a component in the fuel cycle system and also as a component in the power system. The cost of power produced by the nuclear reactors is strongly influenced by its utilization as a component in the power system.[2,3]

Figure 2 illustrates the viewpoint of considering a reactor as a component in two complex systems. Notice that, in looking at the left hand side of Figure 2, one notes that the reactor is a component in the fuel cycle. In this system, the fuel is obtained from the mine, the raw ore is used in a processing operation to produce yellowcake U_3O_8. The U_3O_8 is converted to uranium hexaflouride in conversion operation, followed by the enrichment operation in which the U-235 isotopic concentration is enhanced. After the enriched uranium dioxide powder is produced, fuel elements are produced which after a series of operations are put into a form of fabricated fuel assemblies for insertion into the reactor. The spent fuel from the reactor is stored for cooling and reprocessed to obtain the remaining U-235 and any plutonium which has been produced. The extracted U-235 can in turn be reenriched and continued through the cycle and the plutonium can be extracted and used in plutonium recycle.

Figure 3 is a more detailed diagram of the fuel cycle, also depicting the thorium cycle and plutonium recycle. The right-hand side of the figure indicates the reactor as a component in the power system. The production of power by the reactor is used to satisfy the demand as set by the consumer. The utility has the option of meeting the consumer demands by committing and dispatching other electrical power generating equipment in the power system. These include fossil plants, possibly other reactors, hydraulic and

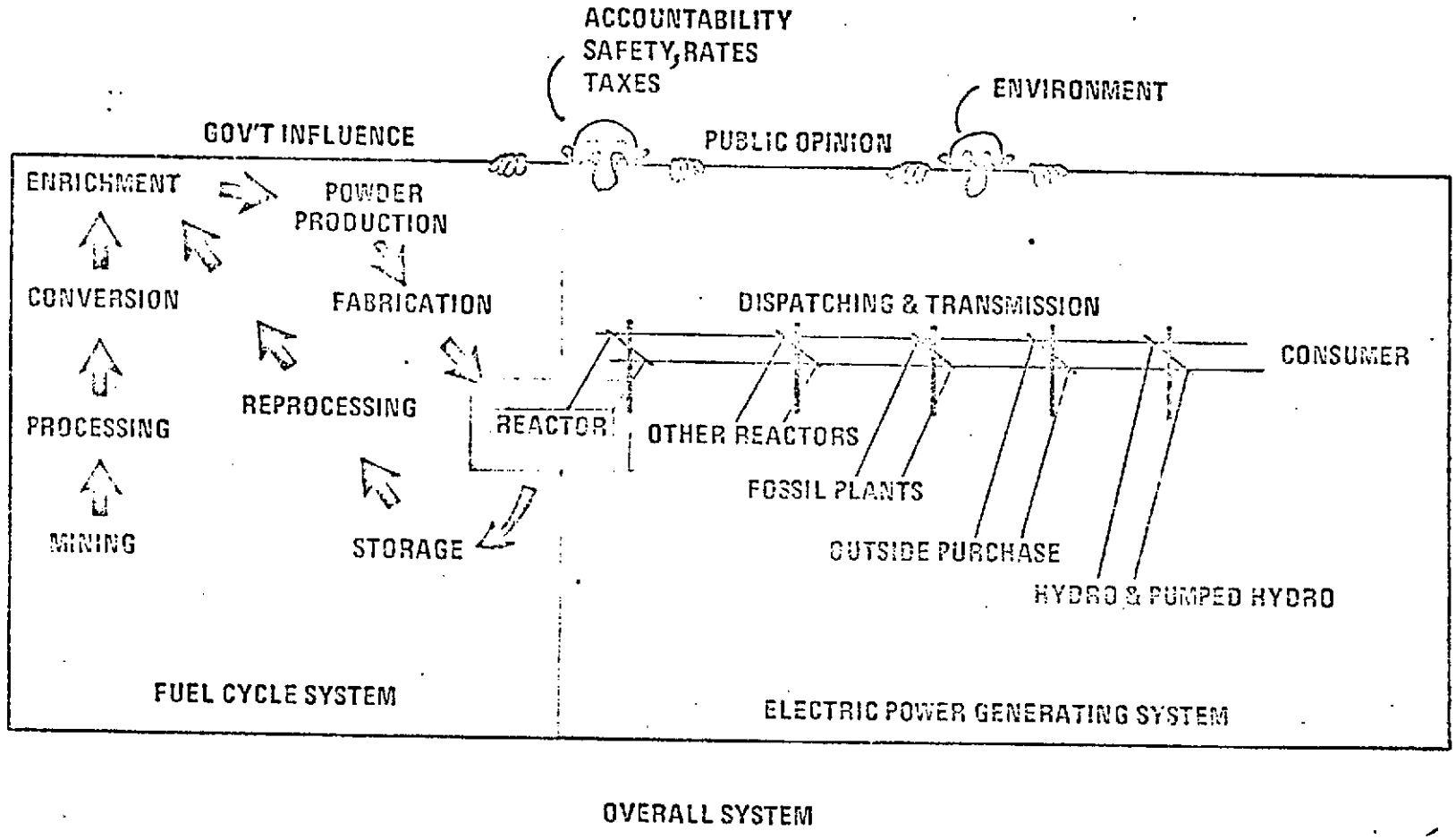


Figure 2. The Reactor as a Component in the Fuel System and the Power System

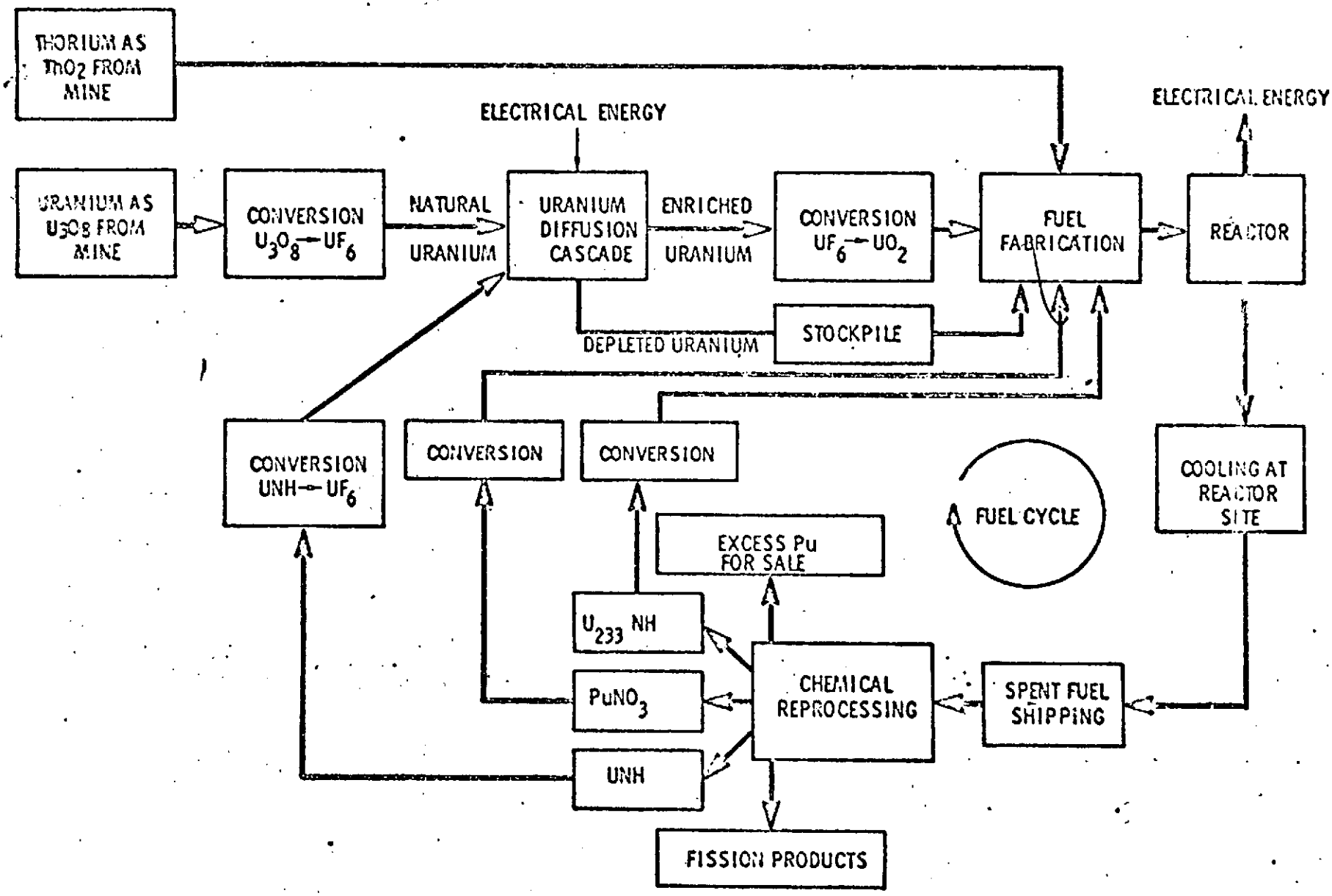


Figure 3. The Nuclear Fuel Cycle

pumped hydraulic plants which may be available. Energy can also be purchased from an outside neighboring utility. The decision of unit commitment and dispatching requires an economic optimization which may be influenced by obviously the demand but also such other government influences as safety, rates, taxes, from the environment, from public opinion. According to Hoskins[3]

> "Early in the development of nuclear power it was recognized that the operating constraints and economic considerations in the operation of nuclear units on a power system are quite different than for conventional fossil-fueled generating units. In the past, economic optimization of power system operation has, for the most part, been based primarily on incremental generating cost from fossil units, which is essentially a function of instantaneous fuel cost and variation of heat rate with plant operating level. With the large scale introduction of commercial nuclear power plants it became increasingly apparent that traditional methods are inadequate for planning the operation of power system operation. This is due to the complex nature of the fuel cycle, fuel cycle economics and constraint imposed incore fuel management. If utilities are to effectively utilize nuclear units, new power system operational methods must be developed which encompass the ability to manage nuclear fuel from an overall power system viewpoint. Such power systems include various combinations of nuclear plants, fossil fuel fired plants, gas turbine peaking plants, conventional, and pumped-storage plants."

UNITED STATES ENERGY CONSUMPTION BY SOURCE
1971 - 2000
(QUADRILLION BTU's)

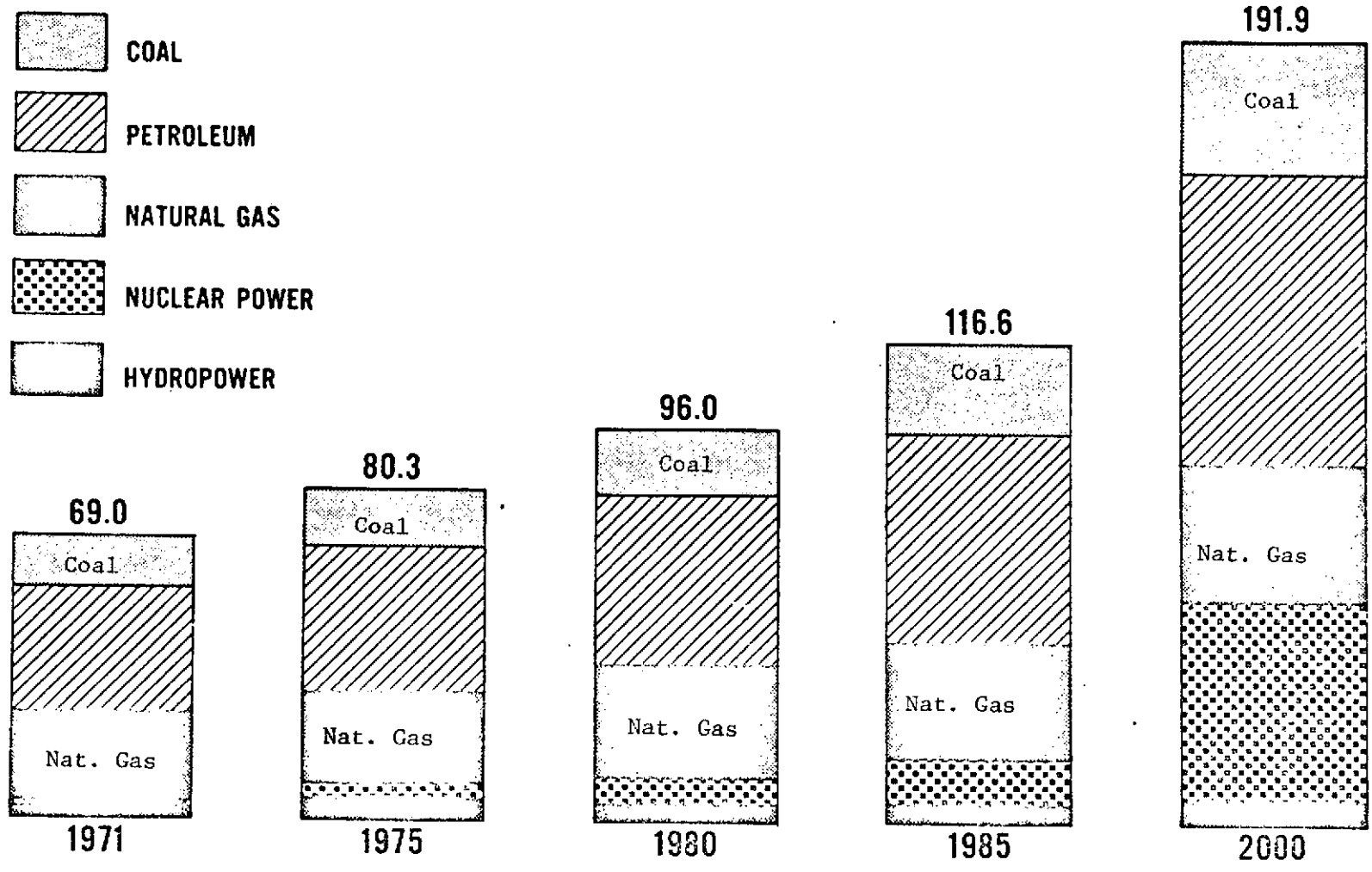


Figure 4

NUCLEAR POWER STATUS AND PROJECTIONS

There are a large number of projections for the production of electrical power by various kinds of power sources. Although some of the projections are not in exact agreement, all of the projections indicate that nuclear power will provide a significant supply of the electrical power in the United States by the year 2000. For example, Figure 4 (Reference 4) indicates that by the year 2000 electrical power production from nuclear sources will be equal to about 160% of the power provided by coal. Table 1 taken from Reference 4 is an estimate provided by the Department of Interior which predicts that by the year 2000 nuclear power will provide $49,230 \times 10^{12}$ BTU's as compared with coal which provides $31,360 \times 10^{12}$ BTU's.

TABLE 1

Energy Source	1971[1]	1975	1980	1985	2000
Coal	12,560	13,825	16,140	21,470	31,360
Petroleum	30,492	35,090	42,190	50,700	71,380
Natural Gas	22,734	25,220	26,980	28,390	33,980
Nuclear Power	405	2,560	6,720	11,750	49,230
Hydropower	2,798	3,570	3,990	4,320	5,950
Total	68,989	80,265	96,020	116,630	191,900

(All figures in trillions of BTU)

1 Actual

Most estimates, including those in Table 1, were made before the oil crisis of the past few months. Undoubtedly the role of nuclear power will be enhanced as a result of actions taken by the United State Government. One whould look also toward other possible sources such as solar and geothermal.

To provide the most authoritive data on the exact present status and firmly committed development of nuclear power in the United States, the Nuclear Assurance Corporation of Atlanta has a data bank encompassing firm committments and also predictions as indicated by the Futura program through the year 1981. Authors of this report wish to thank the Nuclear Assurance Corporation for their permission to include in this section the results of their Fuel-Tract and Futura services.

Following are quotations and figures from References 5 and 6.

"This Nuclear Fuel Status and Forecast section of the fuel-trac Quarterly Report is concerned with the current and projected requirements for materials and services throughout the fuel cycle.

The information contained herein is unique to fuel-trac since it it is based upon the operating and fuel management plans of the individual utilities and not upon average quantity assumptions and projections. Fuel-trac assumptions are incorporated into the system only for those powerplants that are anticipated but not yet ordered; hence, only for those contemplated plants where no information is available from outside sources. Information on these projected reactors is generally separately shown throughout the report and is indicated as FUTURA.

The fuel cycle requirements (quantity and timing) data are generated within the fuel-trac computer system by modeling of the entire industry and the operations of individual suppliers. Timing of feed materials requirements for a particular step in the fuel cycle is therefore a function of the individual supplier's plant capacity and his commitments at that time. Also, his feed materials requirements for a specific product output include processing losses which are incorporated into the fuel-trac computer system through analysis of historical information.

Commitments information is obtained from both the electric utilities and their contractors. The quantities ordered at any particular time, for example U_3O_8, may not be exactly identical to the quantities required to fuel a specific reactor and/or may include requirements for more than one reactor. The fuel-trac computer system prorates the ordered quantities according to requirements and thereby ascertains a true picture of requirements not committed or excesses purchased.

In general this Quarterly Report provides an industry summary picture that is built up from the detailed requirements and commitments status of each utility and supplier."[5,6]

GLOSSARY OF U.S.A. PRODUCTION FACILITIES

URANIUM ORE PROCESSING FACILITIES

Map I.D. #		
	1	Anaconda Co., Grants, N.M.
	2	Atlas Corp., Moab, Utah
	3	Cotter Corp., Canon City, Colo.
	4	Dawn Mining Co., Ford, Wash.
	5	Federal-American Partners, Gas Hills, Wyo.
	6	Kerr-McGee Corp., Grants, N.M.
	8	Petrotomics Co., Shirley Basin, Wyo.
	12	Union Carbide Corp., Gas Hills, Wyo.
	13	Union Carbide Corp., Uravan, Colo.
	14	United Nuclear-Homestake Partners, Grants, N.M.
	15	Utah International, Inc., Gas Hills, Wyo.
	16	Utah International, Inc., Shirley Basin, Wyo.
	17	Western Nuclear Corp., Jeffrey City, Wyo.
	18	Continental Oil – Pioneer Nuclear, Falls City, Texas
	19	Exxon Co., U.S.A., Douglas, Wyo.
	20	Rio Algom Corp., La Sal, Utah

U_3O_8 – UF_6 CONVERSION FACILITIES

Map I.D. #		
	1	Allied Chemical Corp., Metropolis, Ill.
	2	Kerr-McGee Corp., Sallisaw, Okla.

SPENT FUEL REPROCESSING FACILITIES

Map I.D. #		
	1	Nuclear Fuel Services, Inc., West Valley, N.Y.
	2	General Electric Co., Morris, Ill.
	3	Allied-Gulf Nuclear Services, Barnwell, S.C.

ZIRCONIUM METAL PROCESSING FACILITIES

Map I.D. #		
	1	General Electric Co., Wilmington, N.C.
	2	Sandvik Special Metals Co., Kennewick, Wash.
	3	Westinghouse Electric Corp., Blairsville, Penn.
	4	Wolverine Tube, Allen Park, Mich.
	5	Zirconium Technology Corp., Albany, Ore.
	6	AMAX Specialty Metals, Inc., Akron, N.Y.
	7	Teledyne Wah Chang Albany Corp., Albany, Ore.

URANIUM ENRICHMENT FACILITIES

Map I.D. #		
	1	USAEC, Oak Ridge, Tenn.
	2	USAEC, Paducah, Ky.
	3	USAEC, Portsmouth, Ohio

UO_2 FUEL FABRICATION FACILITIES

Map I.D. #		
	1	Babcock & Wilcox Co., Lynchburg, Va.
	2	Combustion Engineering, Inc., Windsor, Conn.
	3	General Electric Co., Wilmington, N.C.
	4	Exxon Nuclear Co., Inc., Richland, Wash.
	5	NUMEC Division - Babcock & Wilcox Co., Apollo.
	6	Gulf United Nuclear Fuels Corp., New Haven, Conn.
	7	Westinghouse Electric Corp., Columbia, S.C.
	9	Kerr-McGee Corp., Cimarron, Okla.
	10	Nuclear Fuel Services, Inc., Erwin, Tenn.
	11	Gulf United Nuclear Fuels Corp., Hematite, Mo.
	12	Nuclear Fuel Services, Inc., West Valley, N.Y.
	13	Gulf General Atomic Co., San Diego, Calif.

GLOSSARY OF EUROPEAN PRODUCTION FACILITIES

URANIUM ORE PROCESSING FACILITIES

1. CEA, Gueugnon, France
2. SIMO (Society Industrielle des Minerais de l'Quest), Ecarpiere, France
3. SIMO (Society Industrielle des Minerais de l'Quest), Bessines, France
4. SIMO (Society Industrielle des Minerais de l'Quest), Forez, France
5. Junta de Energia Nuclear, Andujar, Spain
6. Junta de Energia Nuclear, Salamanca, Spain
7. A B Atomenergi, Rantstad, Sweden
8. Versuchsanlage fuer Uranerz der Gewerkschaft Brunhilde-Ellweiler, Federal Republic of Germany
9. Junta de Energia Nuclear, Urgeirica, Portugal

U_3O_8 – UF_6 CONVERSION FACILITIES

1. COMURHEX (Societe pour la Conversion de l'Uranium en Metal et en Hexafluoure), Pierrelatte, France
2. BNFL (British Nuclear Fuels Limited), Springfields, Lancashire, UK

POWER GENERATION

OPERATING POWER PLANTS

Name	Utility	NSSS Vendor	Net MWe	Commercial Operation Date
U.S.A.				
Dresden – Unit 1	Commonwealth Edison Company	GE	200	August 1960
Yankee – Unit 1	Yankee Atomic Electric Company	West.	175	February 1961
Indian Point – Unit 1	Consolidated Edison Co. of N.Y.	B&W	275	December 1962
Big Rock Point	Consumers Power Co.	GE	72	January 1963
Connecticut Yankee	Connecticut Yankee Atomic Power Company	West.	575	January 1968
San Onofre – Unit 1	Southern California Edison Co.	West.	425	January 1968
R. E. Ginna	Rochester Gas & Electric Company	West.	500	December 1969
Oyster Creek – Unit 1	Jersey Central Power & Light	GE	650	January 1970
Nine Mile Point – Unit 1	Niagara Mohawk Power Company	GE	625	January 1970
Dresden – Unit 2	Commonwealth Edison Company	GE	800	July 1970
Point Beach – Unit 1	Wisconsin Electric Power Company	West.	500	December 1970
Millstone – Unit 1	Millstone Point Co.	GE	650	December 1970
Oconee Nuclear Station – Unit 1	Duke Power Company	B&W	875	August 1973
Robinson – Unit 2	Carolina Power & Light	West.	700	March 1971
Monticello	Northern States Power Company	GE	550	July 1971
Dresden – Unit 3	Commonwealth Edison Company	GE	800	September 1971
Palisades	Consumers Power Co.	C-E	700	July 1972
Quad Cities – Unit 1	Commonwealth Edison Company	GE	1050	July 1972
Quad Cities – Unit 2	Commonwealth Edison Company	GE	1050	August 1972
Point Beach – Unit 2	Wisconsin Electric Power Company	West.	500	October 1972
Surry – Unit 1	Virginia Electric & Power Company	West.	900	December 1972
Turkey Point – Unit 3	Florida Power & Light	West.	700	December 1972
Turkey Point – Unit 4	Florida Power & Light	West.	700	July 1973
Maine Yankee	Maine Yankee Atomic Power Co.	C-E	800	December 1972
Vermont Yankee	Vermont Yankee Nuclear Power Corp.	GE	825	December 1972
Pilgrim – Unit 1	Boston Edison Company	GE	650	December 1972
Surry – Unit 2	Virginia Electric & Power Company	West.	900	March 1973
Oconee – Unit 1	Duke Power Company	B&W	875	June 1973

ABBREVIATIONS

(U.S.A.)
SUPPLIERS, ENGINEERS, CONSTRUCTORS

B&W	Babcock & Wilcox Company
C-E	Combustion Engineering, Inc.
GE	General Electric Company
GGA	Gulf General Atomic Company
West.	Westinghouse Electric Corporation
AEPSC	AEP Service Corporation
Bechtel	Bechtel Corporation
Brown	Brown & Root, Inc.
B&R	Burns & Roe, Inc.
Daniel	Daniel Construction
Ebasco	Ebasco Services, Inc.
G&H/D&R	Gibbs & Hill/Durham & Richardson
Gilbert	Gilbert Associates, Inc.
Jones	J. A. Jones Construction Company
Kaiser	Kaiser Engineers
Kiewit	Peter Kiewit Sons' Company
Parsons	Ralph M. Parsons Company
Pioneer	Pioneer Services & Engineering
S&L	Sargent & Lundy
S-S	Southern Services
SS/BC	Southern Services/Bechtel Corporation
S-R	Stearns-Roger Corporation
S&W	Stone & Webster Engineering Corporation
UE&C	United Engineers & Constructors, Inc.
Indep.	Independent Constructor
OPS	Offshore Power Systems

NUCLEAR POWER CAPACITY

FIRMLY COMMITTED REACTORS

		Prior to 1972	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Total of Country
U.S.A. &	MWe	7498	6379	8220	14512	13094	6976	11572	7892	18585	26608	24923	146259
Puerto Rico	# of Reactors	15	9	10	18	14	7	12	8	18	24	22	157
Austria	MWe						692						692
	# of Reactors						1						1
Belgium	MWe			390		1260							1650
	# of Reactors			1		2							3
Brazil	MWe							600					600
	# of Reactors							1					1
Federal Republic	MWe	810	1270		1146	1635	4653	864	2485	1260			14123
of Germany	# of Reactors	3	2		1	2	5	1	2	1			17
Finland	MWe						420		1080				1500
	# of Reactors						1		2				3
France	MWe	266				898	898	903	1920	995			5880
	# of Reactors	1				1	1	1	2	1			7
India	MWe	380											380
	# of Reactors	2											2
Italy	MWe	396				800							1196
	# of Reactors	2				1							3
Japan	MWe	1060	470	1160	2041	2570	5333	4912	2779	4740	5521		30586
	# of Reactors	3	1	2	3	4	6	6	3	5	6		39
Mexico	MWe							640					640
	# of Reactors							1					1
Netherlands	MWe			450									450
	# of Reactors			1									1
Republic of China	MWe					604	604		900	900			3008
	# of Reactors					1	1		1	1			4
Republic of Korea	MWe					564							564
	# of Reactors					1							1
Spain	MWe	593					1804	2761	1804				6962
	# of Reactors	2					2	3	2				9
Sweden	MWe	440			2142	580	900	1480	900	900			7342
	# of Reactors	1			3	1	1	2	1	1			10
Switzerland	MWe	350	656						1860	918			3784
	# of Reactors	1	2						2	1			6
Total by Year	MWe	11793	8775	10220	19841	22005	22280	23732	21620	28298	32129	24923	225616
	# of Reactors	30	14	14	25	27	25	27	23	28	30	22	265

NUCLEAR POWER CAPACITY

Firmly Committed and Futura

GRAPH 1

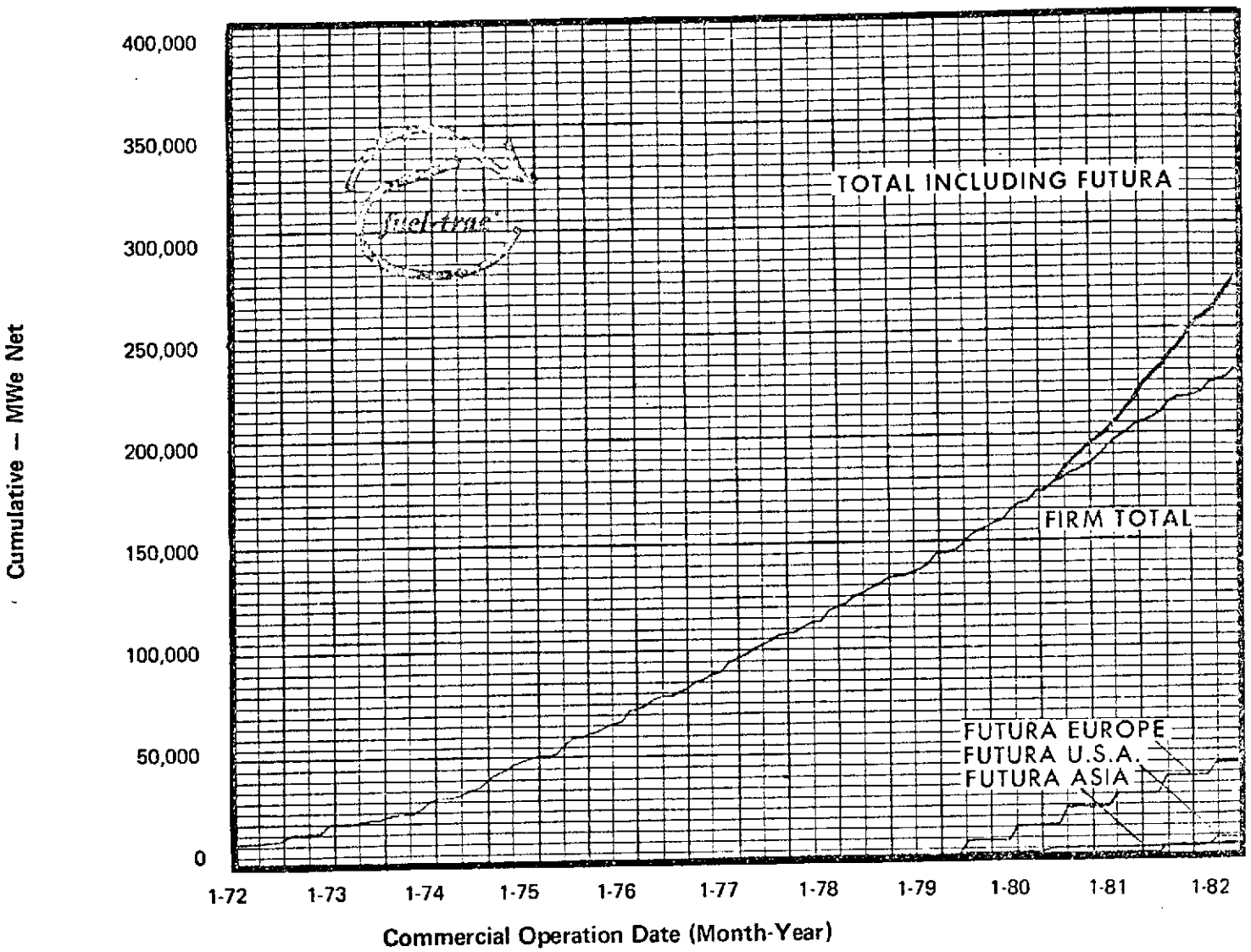


TABLE 3

ANNUAL CAPACITY BREAKDOWN — MWe NET

	1972**	1973	1974	1975	1976	1977	1978	1979	1980	1981
FIRM TOTAL*	20568	10220	19841	22005	22280	23732	21620	28298	32129	24923
FUTURA										
U.S.A.	0	0	0	0	0	0	0	0	0	9943
EUROPE	0	0	0	0	0	0	0	14110	14050	13910
ASIA	0	0	0	0	0	0	0	0	0	6600
FUTURA TOTAL	0	0	0	0	0	0	0	14110	14050	30453
FIRM + FUTURA TOTALS	20568	10220	19841	22005	22280	23732	21620	42408	46179	55376

* Includes Reactors not in U.S.A., Europe and Asia

** Cumulative through 1972

COMMITTED NUCLEAR POWER CAPACITY

Firmly Committed

GRAPH 2

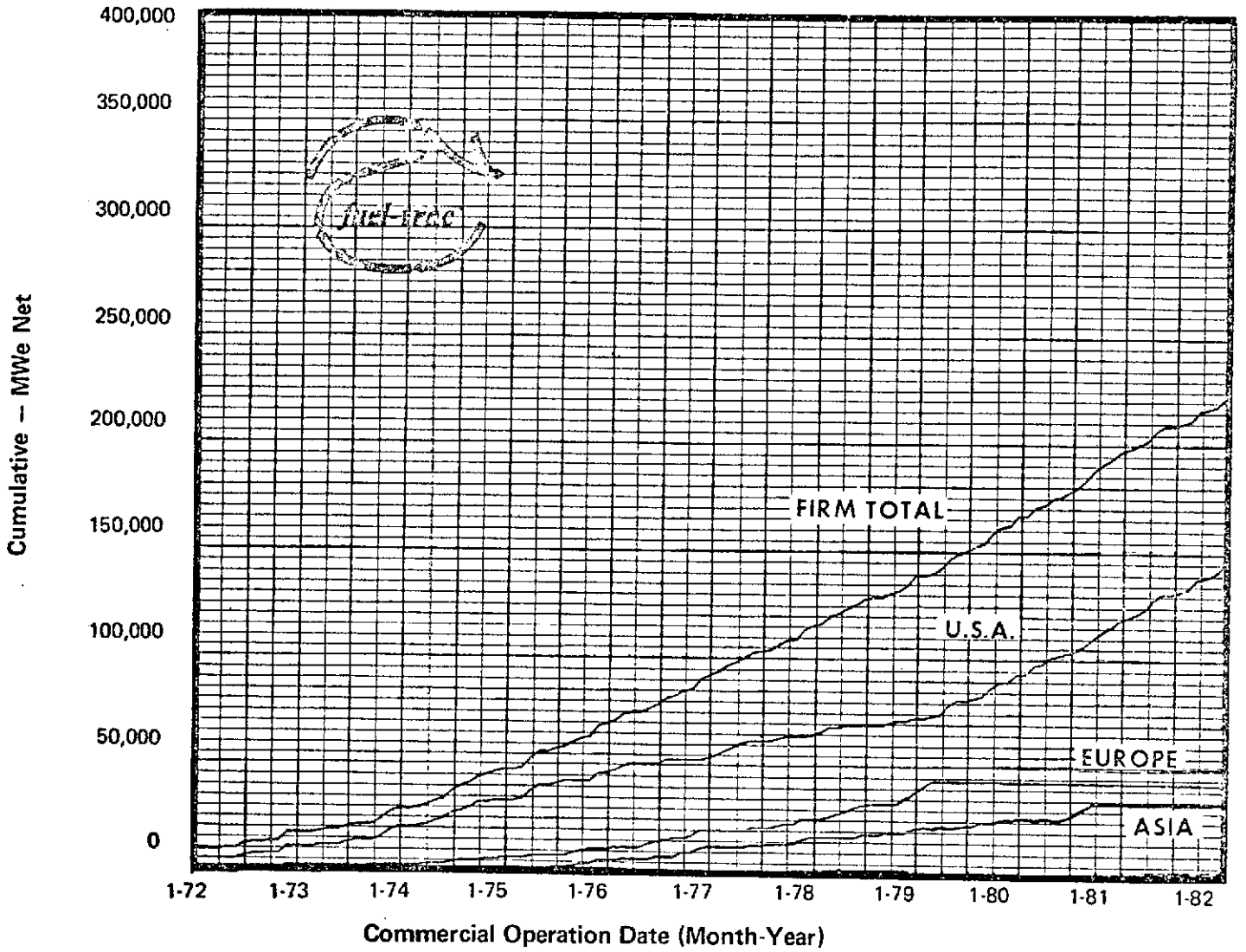


TABLE 4

ANNUAL CAPACITY BREAKDOWN – MWe NET

	1972**	1973	1974	1975	1976	1977	1978	1979	1980	1981
U.S.A.	13877	8220	14512	13094	6976	11572	7892	18585	26608	24923
EUROPE	4781	840	3288	5173	9367	6008	10049	4073	0	0
ASIA	1910	1160	2041	3738	5937	4912	3679	5640	5521	0
TOTAL	20568	10220	19841	22005	22280	22492	21620	28298	32129	24923

** Cumulative through 1972

U. S. A. NUCLEAR POWER CAPACITY

Firmly Committed By NSSS

GRAPH 3

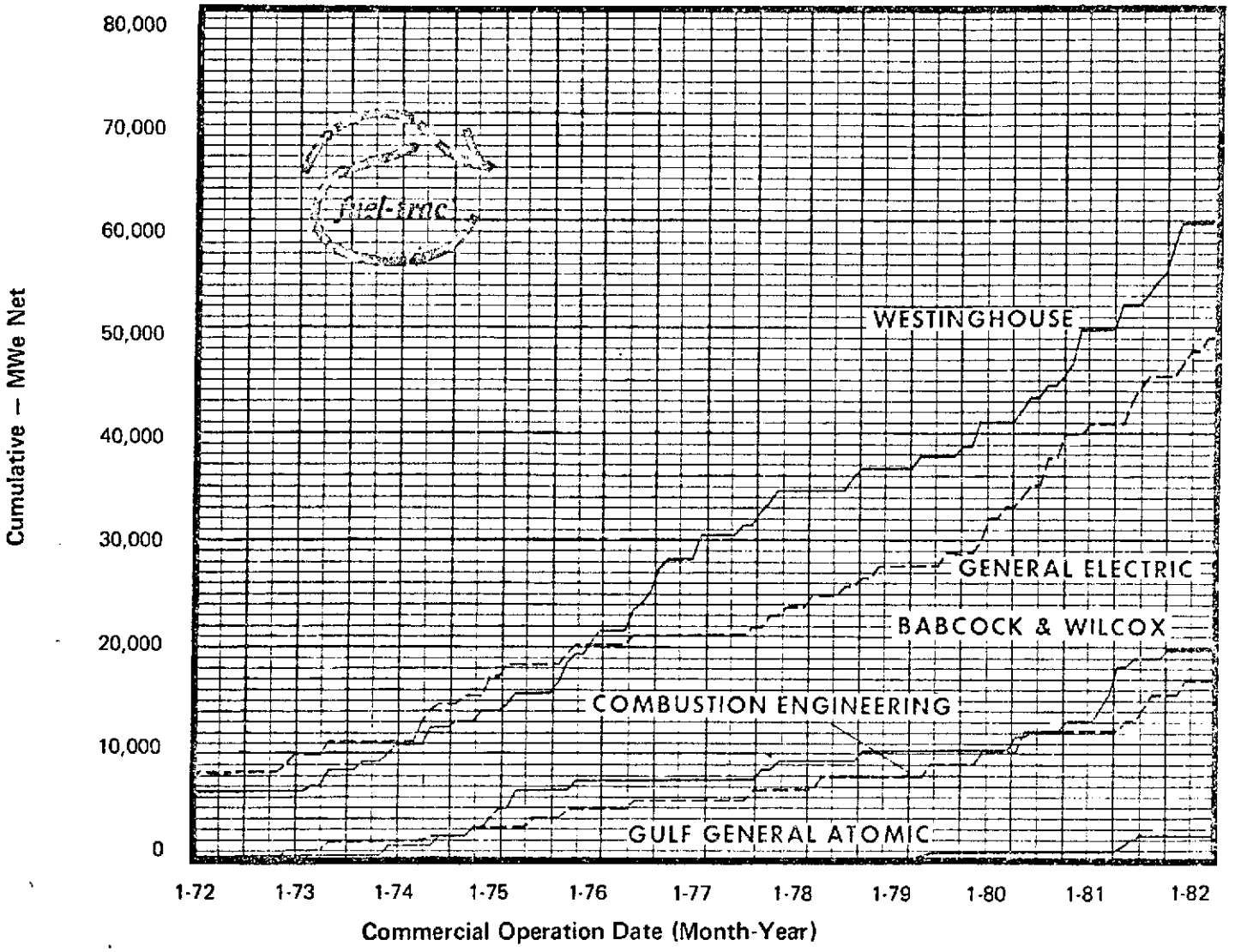


TABLE 5

ANNUAL CAPACITY BREAKDOWN MWe NET

	1972**	1973	1974	1975	1976	1977	1978	1979	1980	1981
NSS SUPPLIERS										
BABCOCK & WILCOX	265	886	5153	906	0	1724	893	1763	6785	893
% ANNUAL TOTAL	2	11	36	7	0	15	11	9	25	4
COMBUSTION	1614	457	1628	1601	0	2085	1150	3030	2050	3900
% ANNUAL TOTAL	12	6	11	12	0	18	15	16	8	16
GENERAL ELECTRIC	7153	2943	4237	2706	0	3679	2671	7580	9068	8012
% ANNUAL TOTAL	52	36	29	21	0	32	34	41	34	32
GULF GENERAL ATOMIC	0	0	330	0	0	0	0	770	0	1160
% ANNUAL TOTAL	0	0	2	0	0	0	0	4	0	5
WESTINGHOUSE	4845	3934	3164	7881	6976	4084	3178	5442	8705	10958
% ANNUAL TOTAL	35	48	22	60	100	35	40	29	33	44
TOTAL ANNUAL CAPACITY	13877	8220	14512	13094	6976	11572	7892	18585	26608	24923

** Cumulative through 1972

URANIUM ORE PROCESSING

U.S.A., Europe, Asia

Cumulative Requirements

GRAPH 4

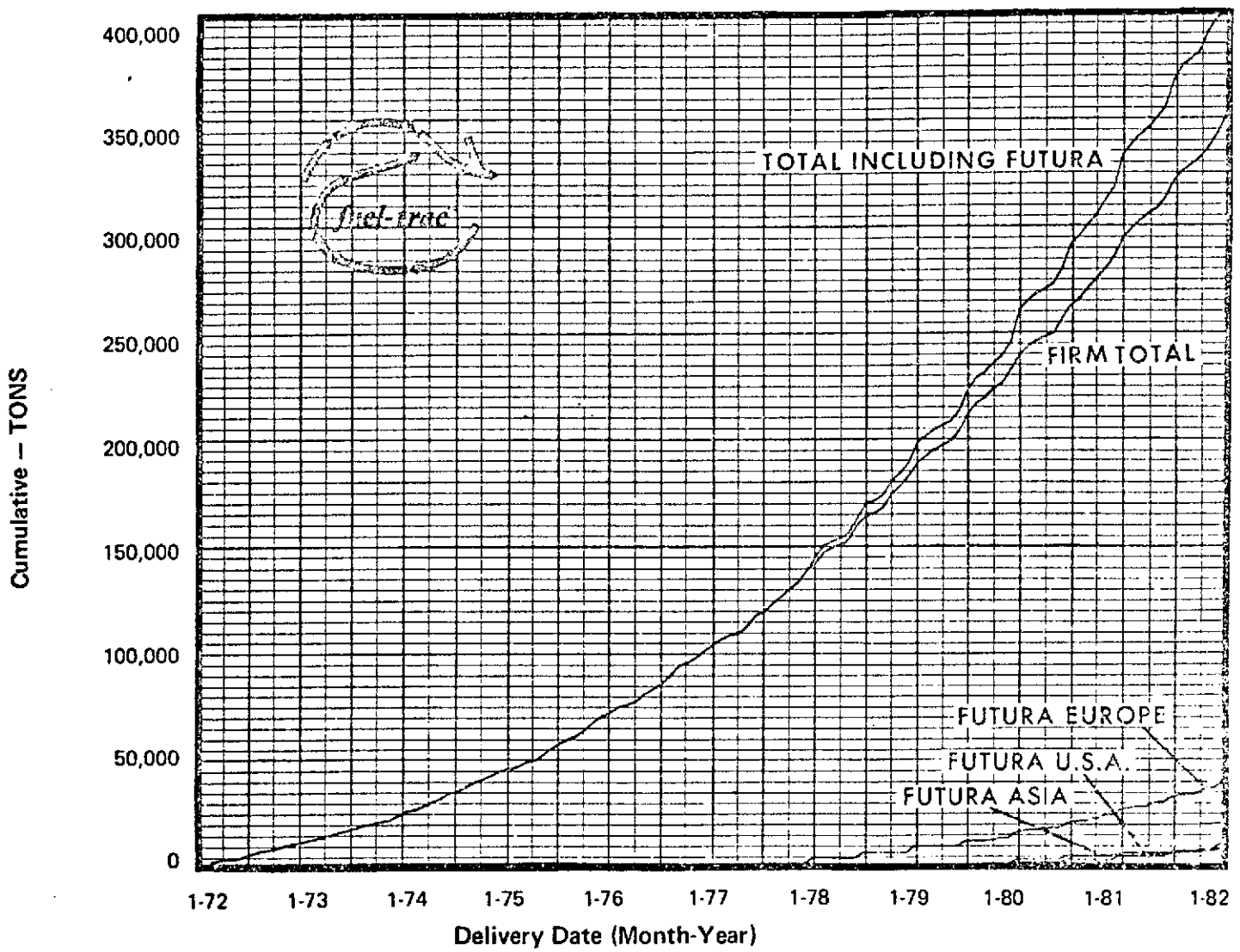


TABLE 1.1

ANNUAL REQUIREMENTS BREAKDOWN—TONS

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
FIRM TOTAL *	10849	14195	19835	27124	32271	38097	46500	50439	55140	56562
FUTURA										
U.S.A.	0	0	0	0	0	0	0	3494	3204	5721
EUROPE	0	0	0	0	0	3094	6292	6938	10841	13894
ASIA	0	0	0	0	0	0	0	1811	3217	4163
FUTURA TOTAL	0	0	0	0	0	3094	6292	12243	17262	23778
FIRM + FUTURA TOTALS	10849	14195	19835	27124	32271	41191	52793	62683	72402	80340

* Includes Reactors not in U.S.A., Europe and Asia

U. S. A. URANIUM ORE PROCESSING

Firm Cumulative Requirements – First Cores & Reloads

GRAPH 5

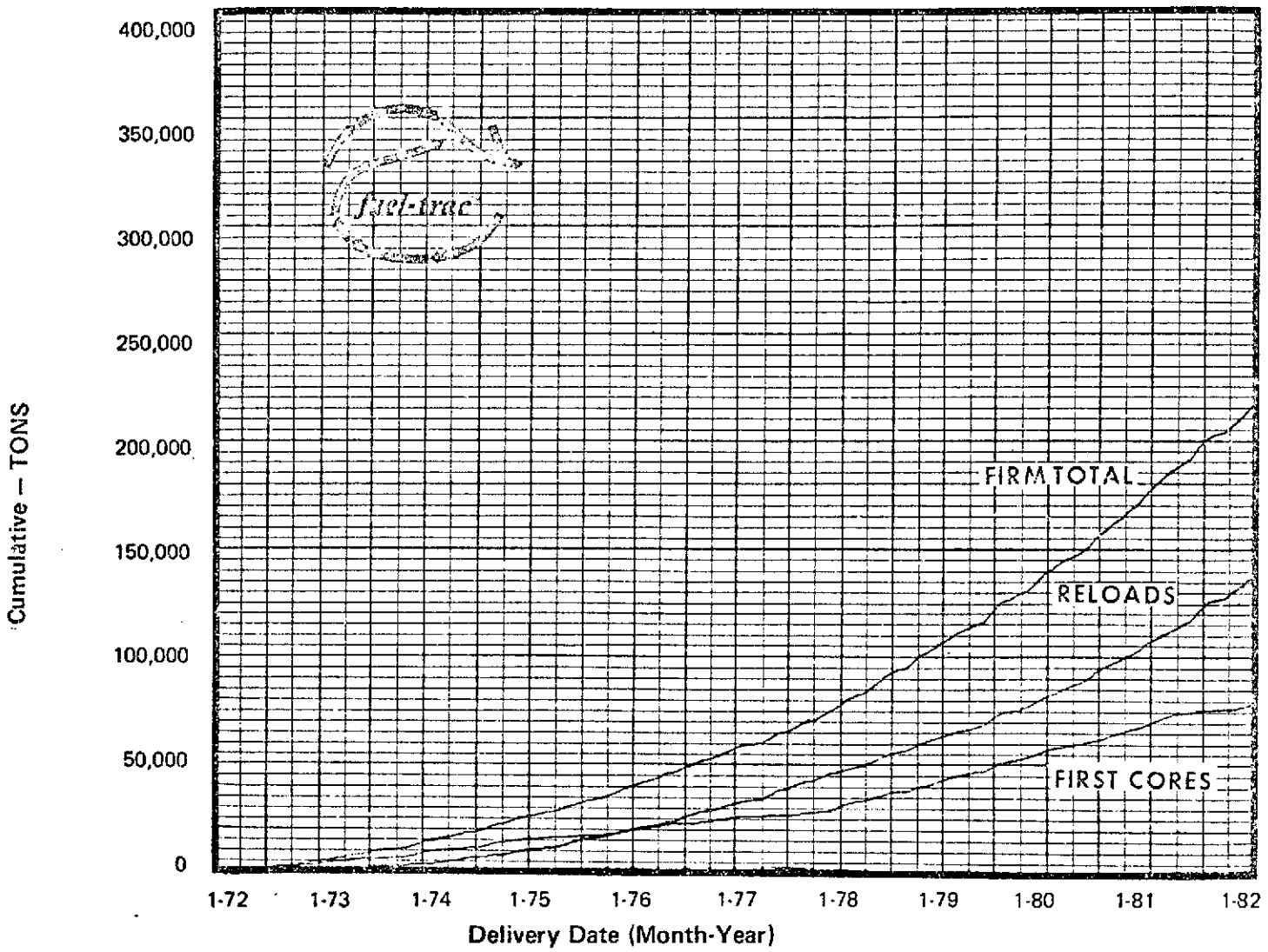


TABLE 1.2

FIRM ANNUAL REQUIREMENTS BREAKDOWN–TONS
First Cores & Reloads

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
FIRST CORES	4967	5376	5822	4396	5761	4284	12549	14083	12023	8603
% ANNUAL TOTAL	77	61	49	31	32	22	43	43	32	22
RELOADS	1455	3436	6003	9564	12231	14769	16418	18203	25412	30610
% ANNUAL TOTAL	23	39	51	69	68	76	56	56	67	78
FIRM TOTAL	6422	8812	11825	13960	17992	19053	28967	32286	37435	39213
CUMULATIVE TOTAL	6422	15234	27059	41019	59011	78064	107031	139317	176752	215965

U_3O_8 – UF_6 CONVERSION

U. S. A., Europe, Asia

Cumulative Requirements

GRAPH 6

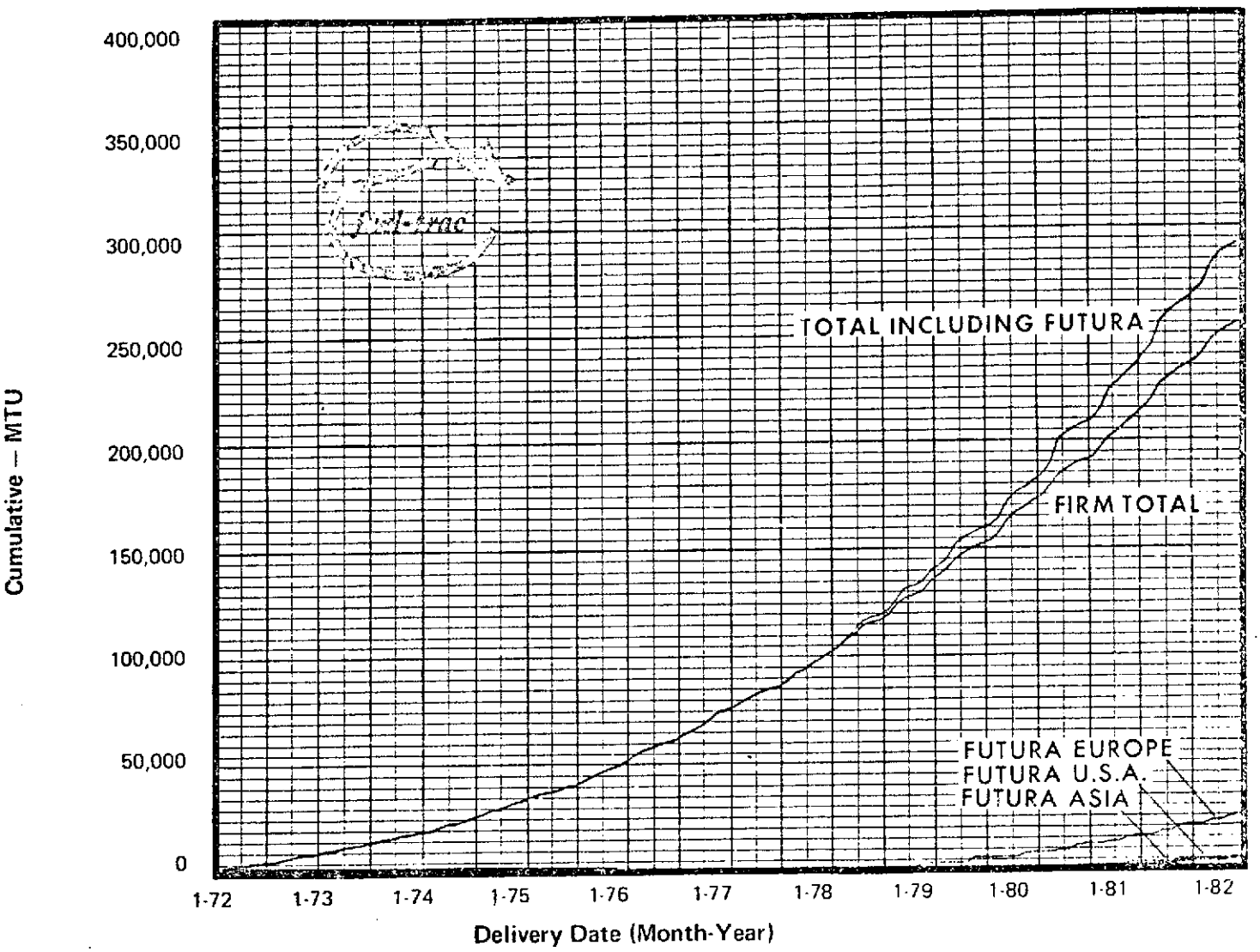


TABLE 2.1

ANNUAL REQUIREMENTS BREAKDOWN – MTU

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
FIRM TOTAL*	7032	10070	15138	18189	24225	26650	34146	37462	39452	43318
FUTURA										
U.S.A.	0	0	0	0	0	0	0	0	4027	2003
EUROPE	0	0	0	0	0	0	4566	4618	7497	9671
ASIA	0	0	0	0	0	0	0	0	2439	2930
FUTURA TOTAL	0	0	0	0	0	0	4566	4618	13963	14604
FIRM + FUTURA TOTALS	7032	10070	15138	18189	24225	26650	38712	42080	53415	57922

* Indicates Reactors Not in U.S.A., Europe and Asia

U. S. A. U_3O_8 – UF_6 CONVERSION

Cumulative Requirements – First Cores & Reloads

GRAPH 7

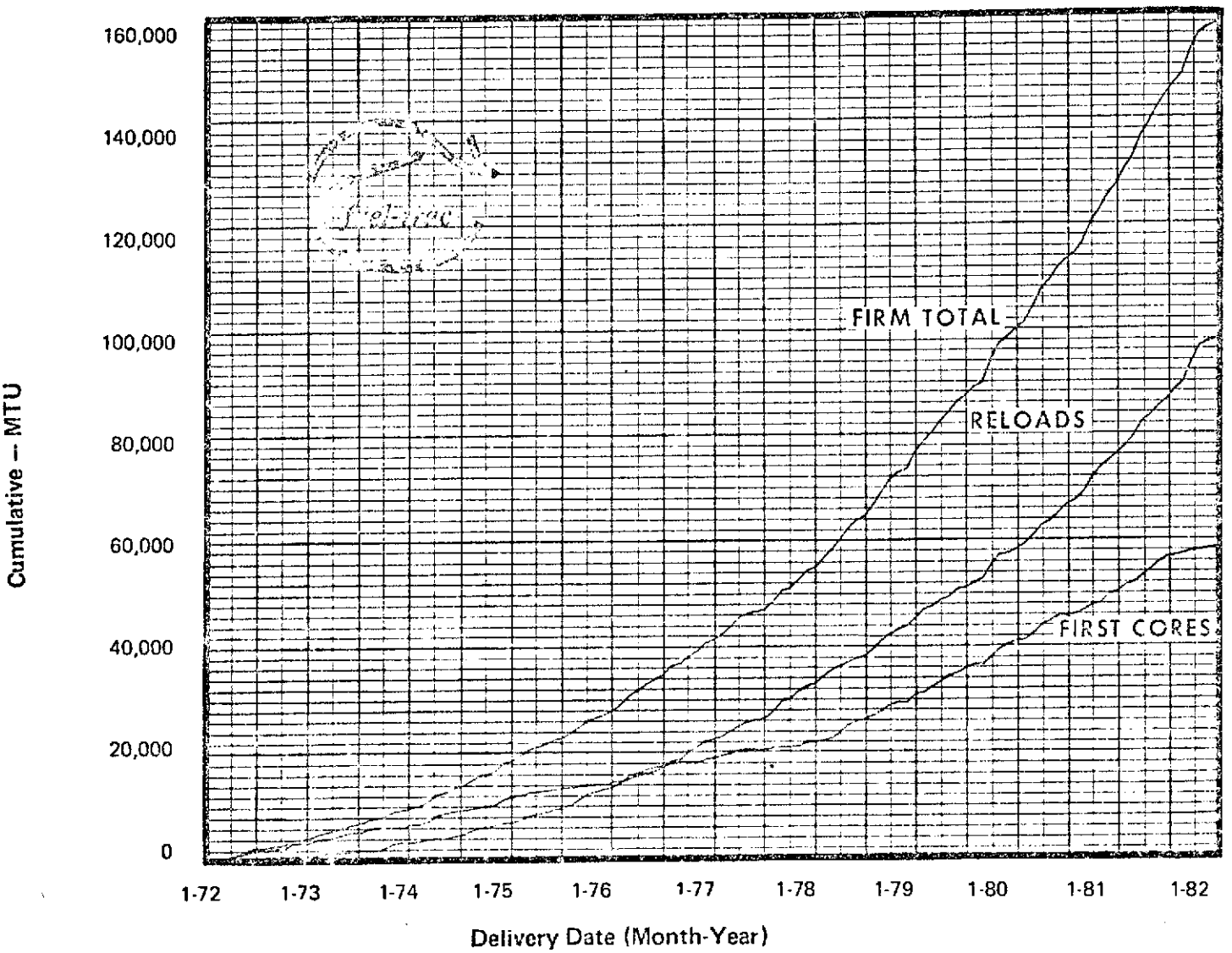


TABLE 2.2

ANNUAL REQUIREMENTS BREAKDOWN – MTU
First Cores & Reloads

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
FIRST CORES	3234	3277	5218	2822	4409	3075	9053	10162	9126	8345
% ANNUAL TOTAL	79	57	59	29	33	23	41	43	34	28
RELOADS	744	2472	3603	6776	9088	10507	12416	13020	18027	21691
% ANNUAL TOTAL	18	43	41	71	67	77	57	56	66	72
FIRM TOTAL	3978	5749	8821	9598	13497	13582	21469	23182	27153	30036
CUMULATIVE TOTAL	3978	9727	18548	28146	41643	55225	76694	99876	127029	157065

U. S. A. Reactors

Firm Cumulative Requirements & Commitments

GRAPH 8

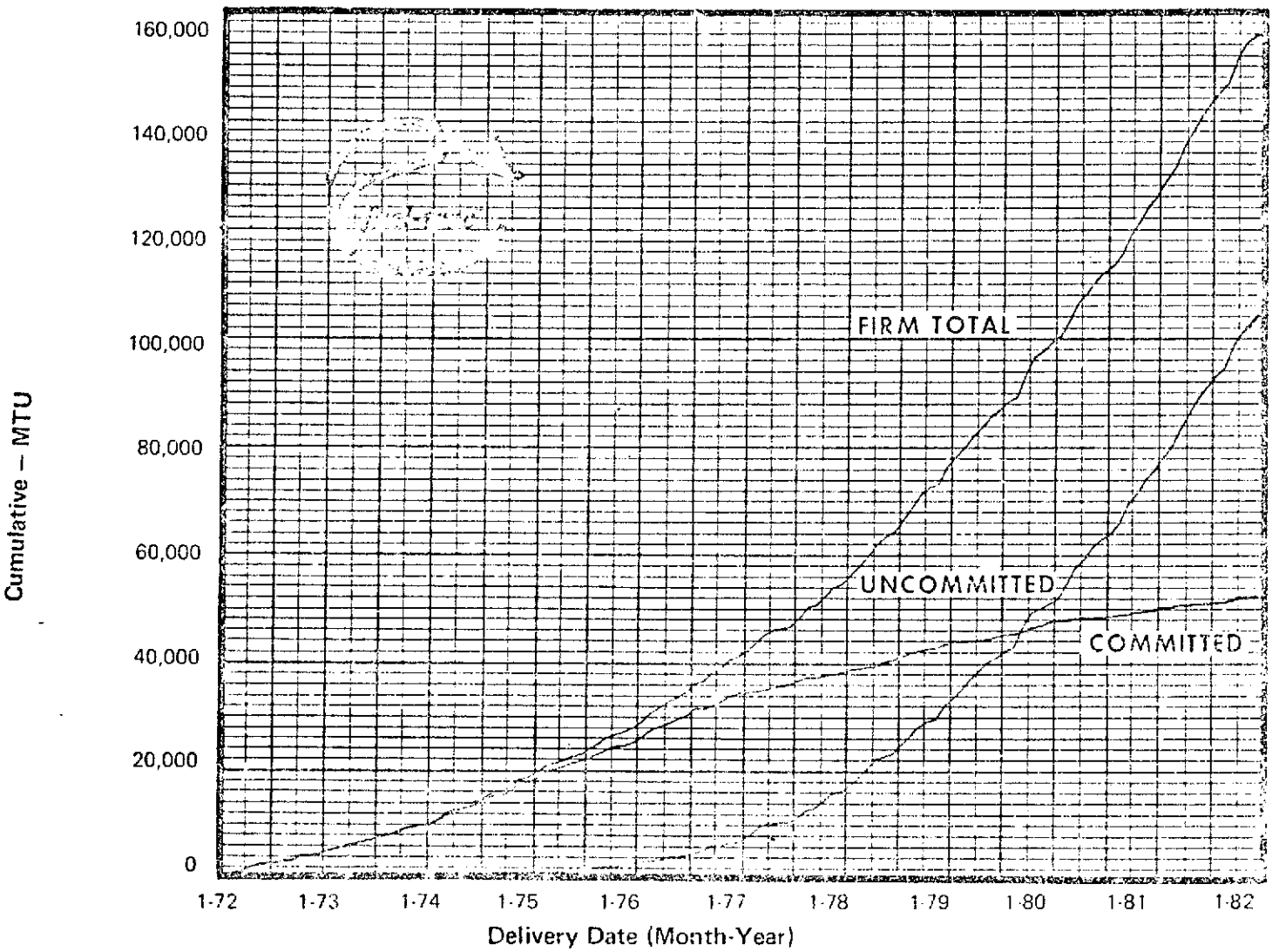


TABLE 2.5

FIRM ANNUAL COMMITMENTS BREAKDOWN – MTU
First Cores & Reloads

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
COMMITTED TO PROCESSORS	3948	5646	8049	7717	9437	3984	5522	4260	2283	2824
% ANNUAL TOTAL	99	98	91	80	70	29	26	18	8	9
UNCOMMITTED TO PROCESSORS	30	104	772	1881	4060	9598	15947	18922	24871	27407
% ANNUAL TOTAL	1	2	9	20	30	71	74	82	92	91
FIRM TOTAL	3978	5750	8821	9598	13497	13582	21469	23182	27154	30231
CUMULATIVE TOTAL	3978	9728	18549	28147	41644	55226	76695	99877	127031	157262

URANIUM ENRICHMENT

U. S. A., Europe, Asia

Cumulative Requirements

GRAPH 9

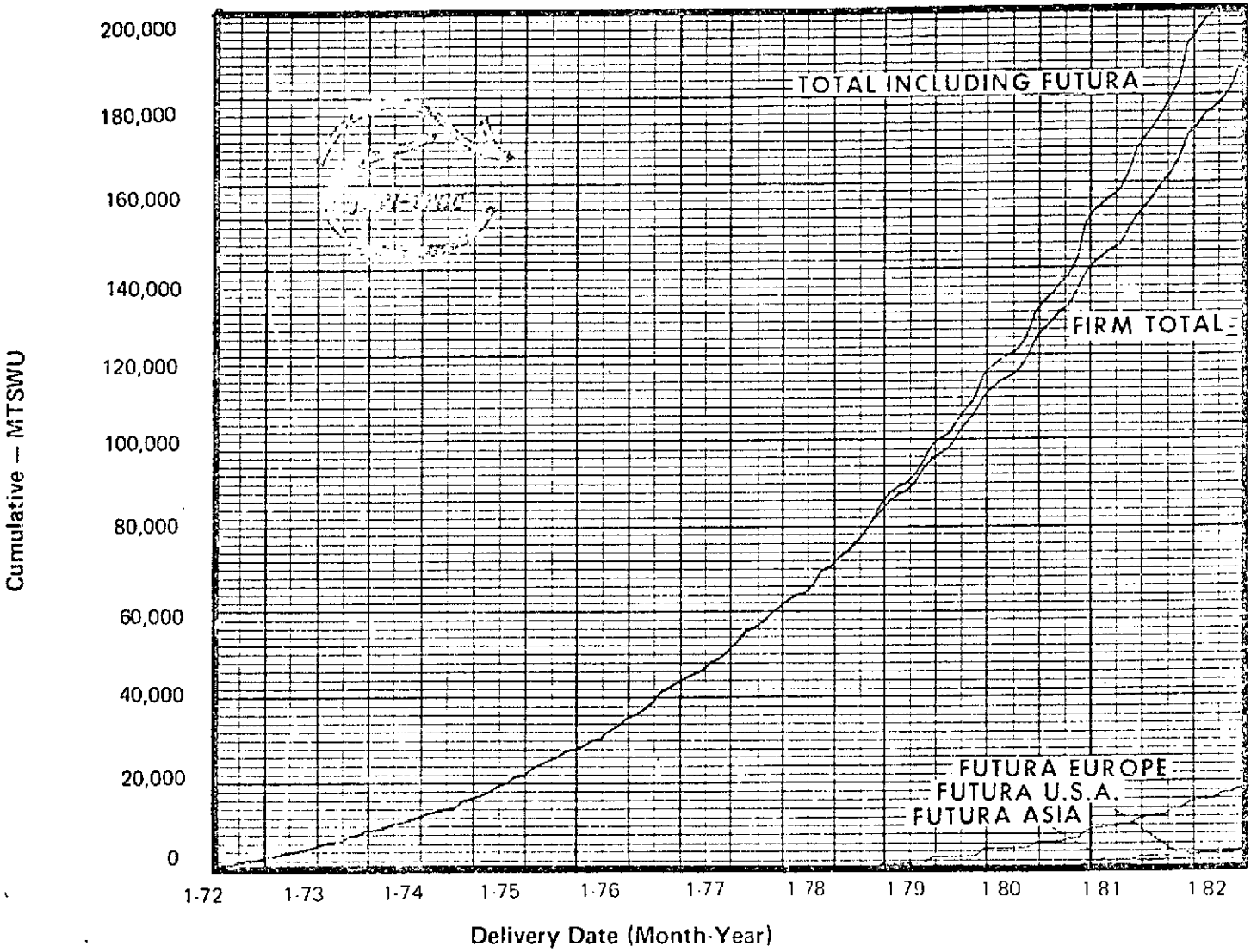


TABLE 5.1

ANNUAL REQUIREMENTS BREAKDOWN–MTSWU

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
FIRM TOTAL *	4605	6968	9777	13286	16183	19607	25274	27618	29339	34758
FUTURA										
U.S.A.	0	0	0	0	0	0	0	0	2668	1008
EUROPE	0	0	0	0	0	0	3353	2936	5177	6985
ASIA	0	0	0	0	0	0	0	0	1618	2002
FUTURA TOTAL	0	0	0	0	0	0	3353	2936	9462	9995
FIRM + FUTURA TOTAL	4605	6968	9777	13286	16183	19607	28627	30554	38802	44753

* Includes Reactors not in U.S.A., Europe and Asia.

WORLDWIDE SEPARATIVE WORK REQUIREMENTS AND USAEC CAPABILITY

(With CIP and CUP at 0.3% Tails Assay)

GRAPH 10

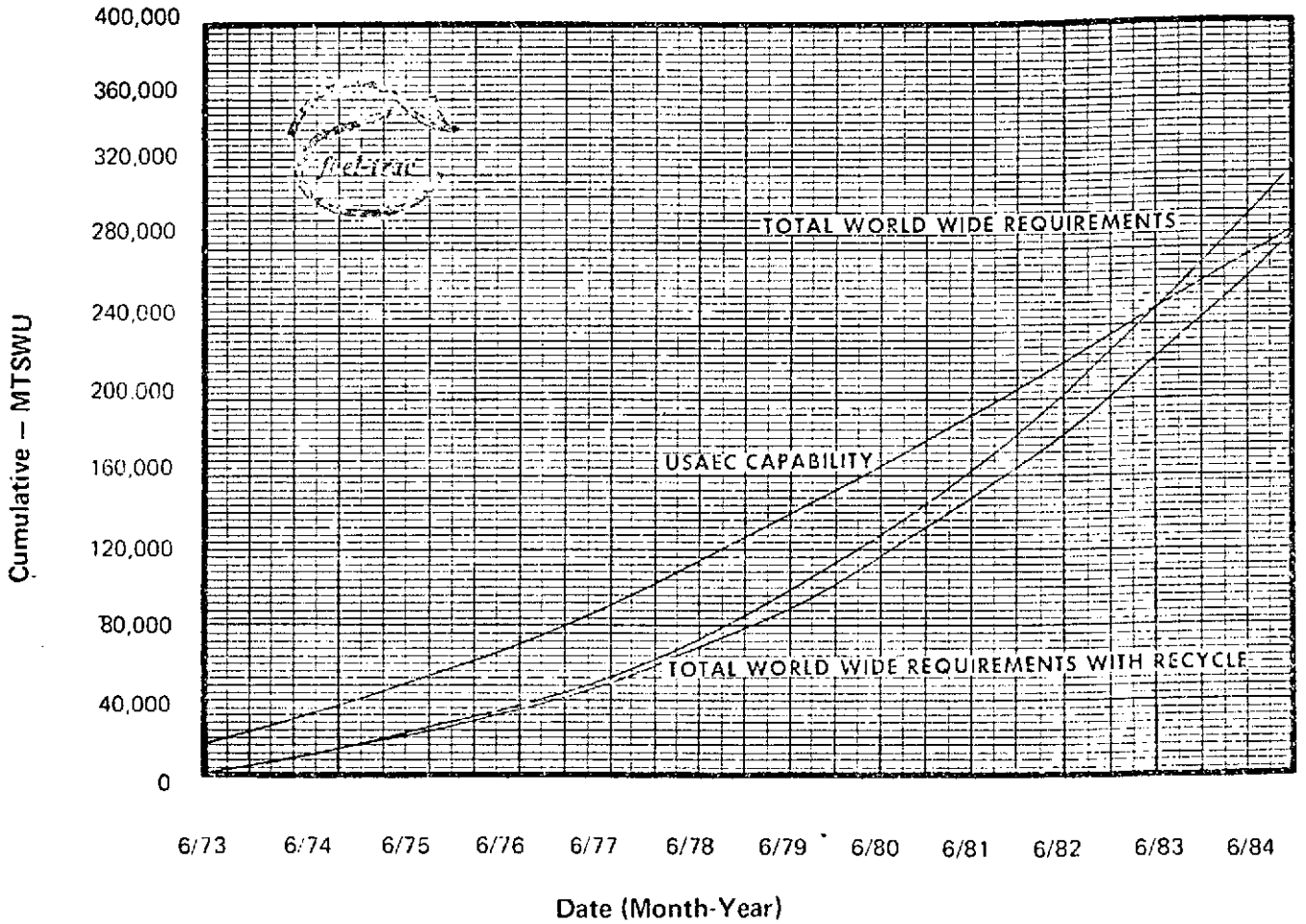


TABLE 5.5

ANNUAL REQUIREMENTS BREAKDOWN--MTSWU

	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
[illegible]*	5500	7729	10514	12799	15527	20007	21875	23269	27581	27425	27346	26857
FUTURA**	0	0	0	0	0	2738	2297	7542	7915	13037	16988	23572
TOTAL	5500	7729	10514	12799	15527	22745	24172	30811	35496	40462	44334	50429
VALUE OF RECYCLED URANIUM	421	177	192	237	298	412	452	579	756	851	1056	1128
VALUE OF RECYCLED PLUTONIUM	246	289	442	783	1282	1783	2152	2634	3364	4325	5418	6212
DEMAND W/RECYCLE	4833	7263	9880	11779	13947	20550	21568	27598	31376	35286	37860	43089

* [illegible] allowance for uranium recycle and includes allowance for plutonium recycle where definite plans exist.
** [illegible] allowance for uranium recycle and includes allowance for plutonium recycle where vendor expresses definite plans.

U. S. A., Europe, Asia

Firm Reactors

GRAPH - 11

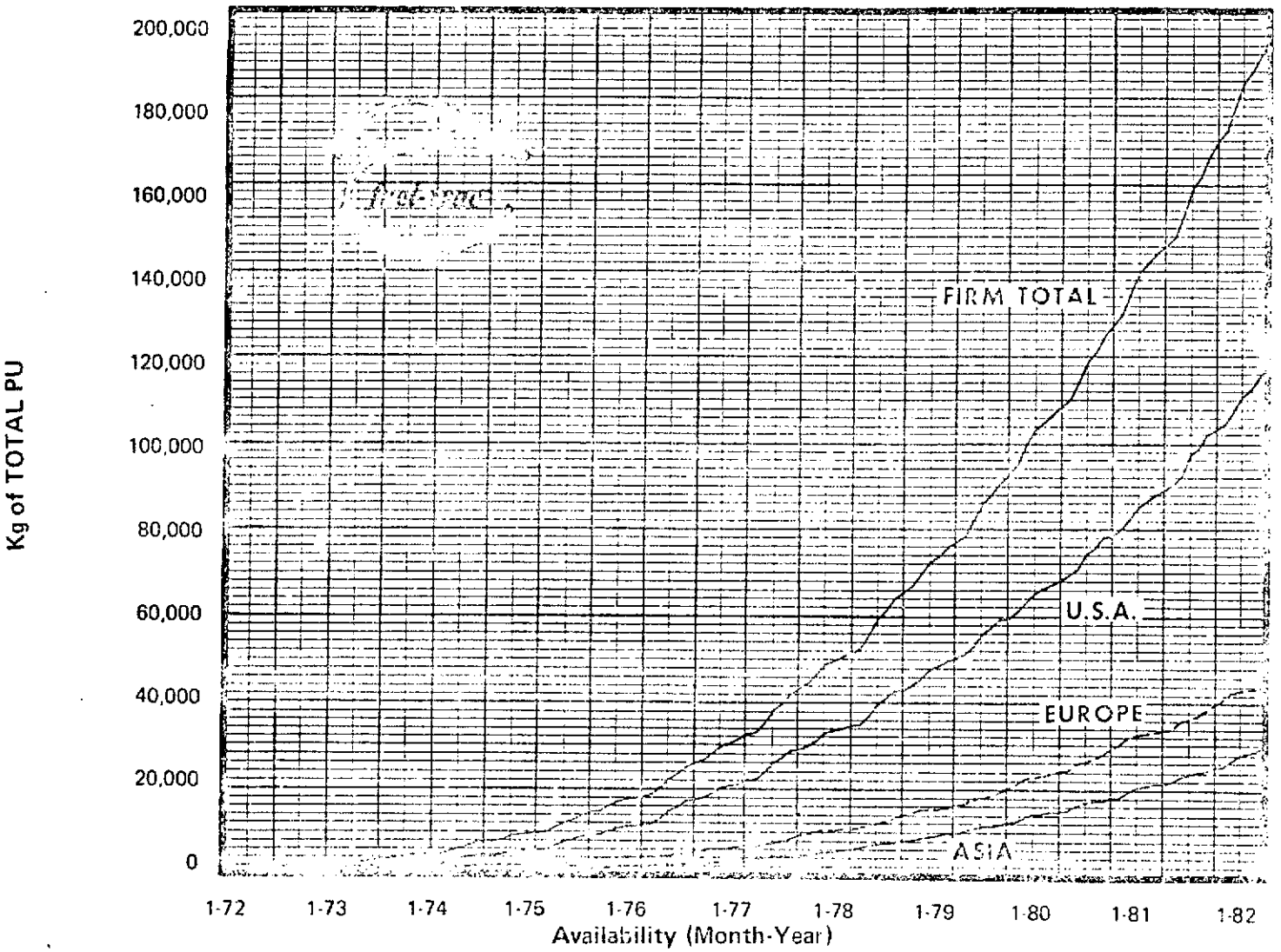


TABLE 9.2

ANNUAL DISCHARGE BREAKDOWN – KgPU

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
U.S.A.	1268	1178	3184	5764	9935	12993	15853	17728	20944	27079
EUROPE	583	1052	1100	1703	2403	4212	5095	7820	9798	10464
ASIA	183	495	661	749	1572	2339	3999	5195	6567	7792
FIRM TOTAL	1947	2725	4945	8216	13910	19544	24947	30743	37309	45335
CUMULATIVE TOTAL	1947	4672	9617	17833	31743	51287	76234	106977	144286	189621

PLUTONIUM

U. S. A., Europe, Asia

GRAPH 12

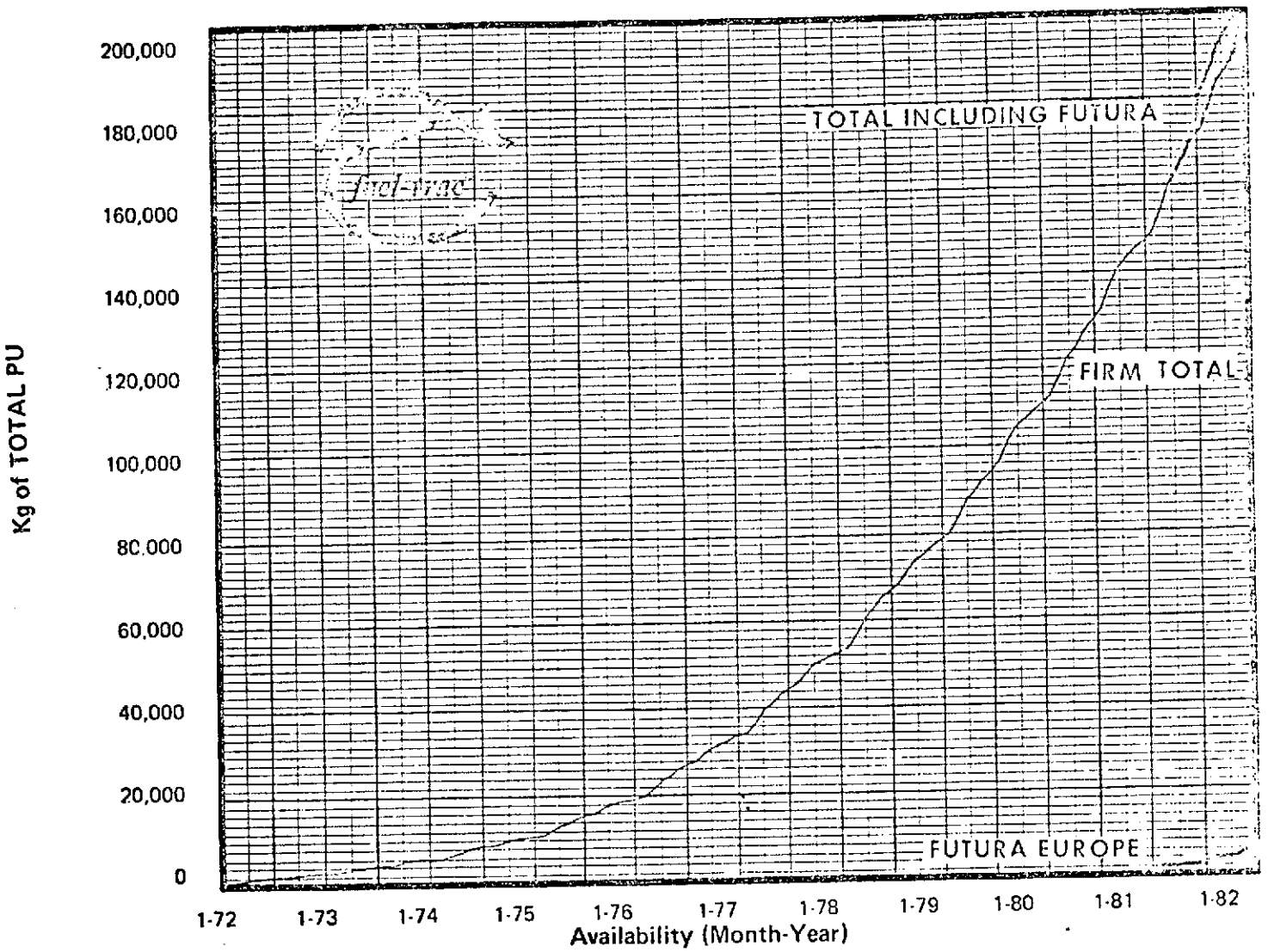


TABLE 9.1

ANNUAL DISCHARGE BREAKDOWN – KgPU

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
FIRM TOTAL*	2034	2725	4945	8215	13910	19543	25140	31039	37615	45638
FUTURA										
U.S.A.	0	0	0	0	0	0	0	0	0	0
EUROPE	0	0	0	0	0	0	0	0	1318	4732
ASIA	0	0	0	0	0	0	0	0	0	0
FUTURA TOTAL	0	0	0	0	0	0	0	0	1318	4732
FIRM + FUTURA TOTALS	2034	2725	4945	8215	13910	19543	25140	31039	38933	50371

* Includes Reactors not in U.S.A., Europe and Asia.

UO_2 POWDER PRODUCTION

U. S. A., Europe, Asia

Cumulative Requirements

GRAPH 13

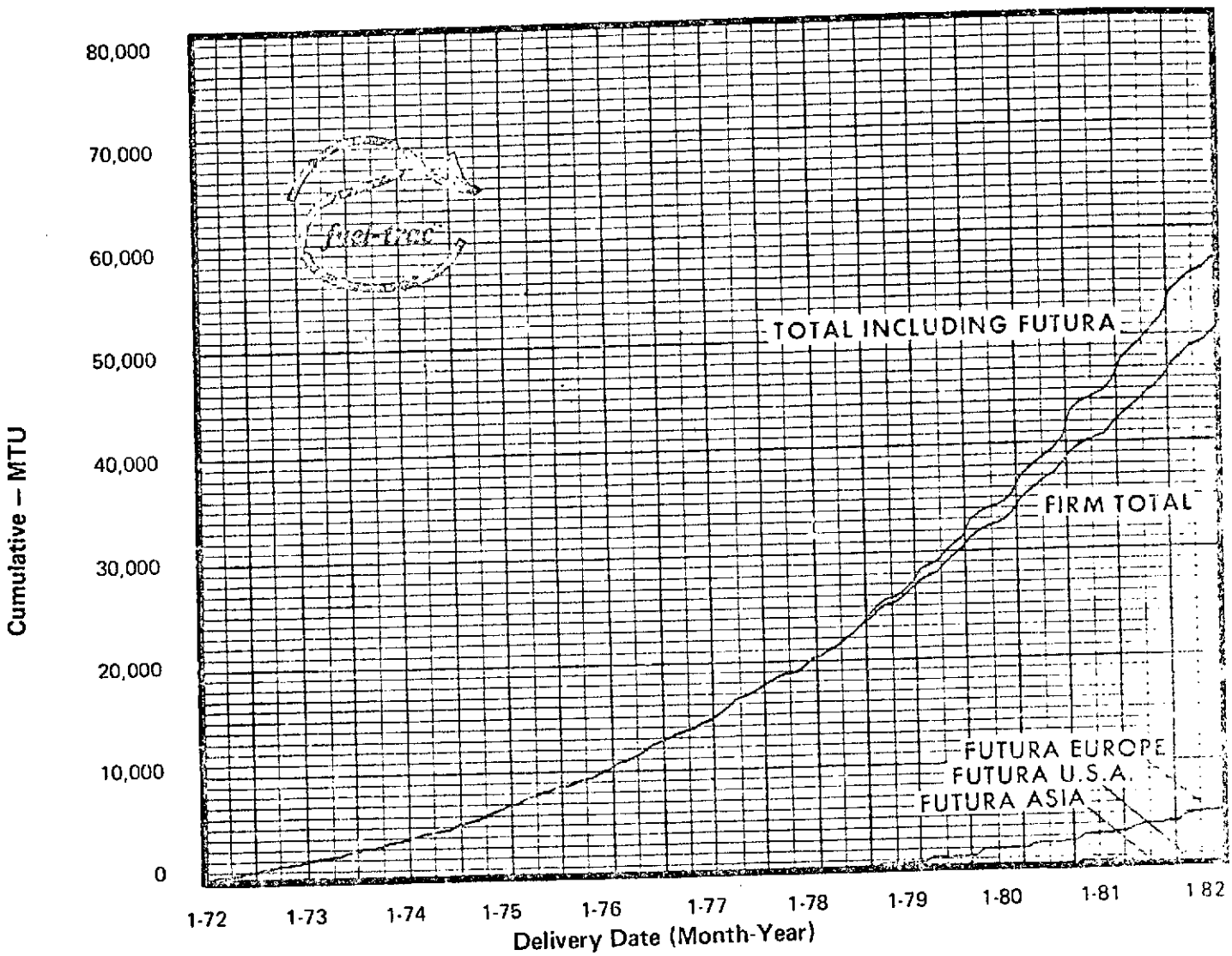


TABLE 6.1

ANNUAL REQUIREMENTS BREAKDOWN – MTU

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
FIRM TOTAL *	1552	2025	3080	3226	4490	5711	6582	6928	8109	8690
FUTURA										
U.S.A.	0	0	0	0	0	0	0	0	674	655
EUROPE	0	0	0	0	0	0	495	1198	1428	1921
ASIA	0	0	0	0	0	0	0	0	352	589
FUTURA TOTAL	0	0	0	0	0	0	495	1198	2454	3224
FIRM + FUTURA TOTALS	1552	2025	3080	3226	4490	5711	7077	8126	10564	11914

* Includes Reactors not in U.S.A., Europe and Asia

UO_2 POWDER PRODUCTION

U. S. A., Europe, Asia

Firm Cumulative Requirements

GRAPH 14

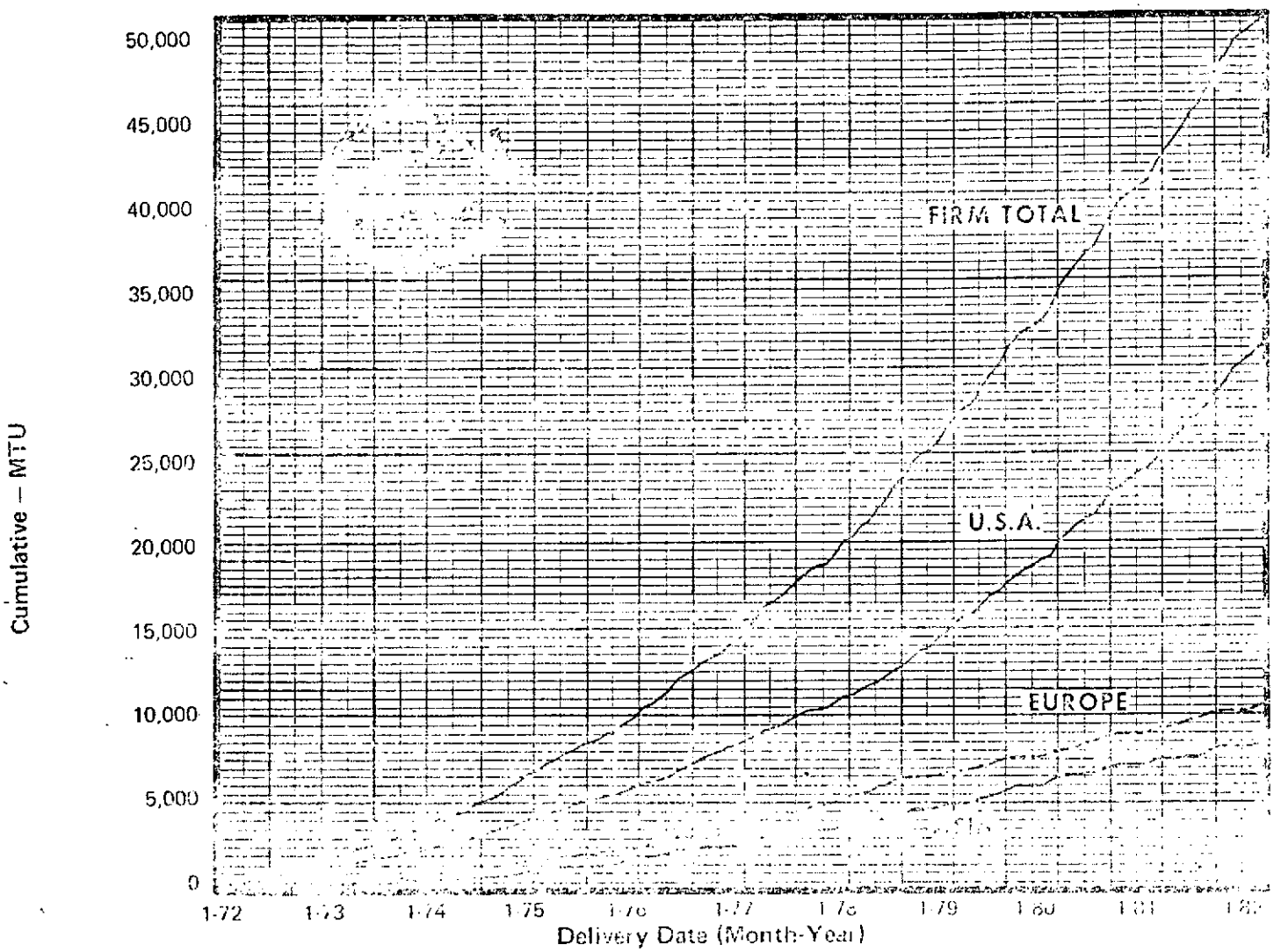


TABLE 6.2

ANNUAL REQUIREMENTS BREAKDOWN – MTU

	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981
U.S.A.	1209	999	1809	1849	2409	2803	3782	4354	5478	6485
EUROPE	136	663	831	712	1093	1768	1540	1135	1412	1232
ASIA	197	362	440	665	851	1140	1224	1398	1183	936
FIRM TOTAL	1542	2024	3080	3226	4353	5711	6546	6887	8073	8653
CUMULATIVE TOTAL	1542	3566	6646	9872	14225	19936	26482	33369	41442	50095

PROJECTION BY REACTOR TYPES

There is a vast literature on the engineering and design considerations safety and environmental - for the various reactors which are under construction. Figure 1 illustrates the various reactor types. These are the pressurized water reactor system, the direct cycle boiling water reactor, the gas-cooled reactor system, and the fast breeder reactor system. Of course, both the pressurized water reactor and the boiling water reactor systems utilize light water as moderator and coolant. The advantage of course being that water is a well documented heat transfer medium and a relatively simple cooling system. It is interesting to note that the present development of the light water reactors holds its present status largely due in part to federal sponsorship. To quote from Rose[7]

> "The light water devices were developed either with federal money (as part of the nuclear submarine program of Westinghouse Electric Corporation) or with conscious acceptance of initial losses such as those incurred by the General Electric Company in promoting the boiling water reactor. The high temperature gas-cooled reactor may actually be safer than the water reactors, more economical of uranium resources, more efficient, meaning that less heat is rejected to the environment and perhaps even cheaper to build, although not all of these advantages are confirmed. Its development lagged because the sponsor, the General Atomic Division of General Dynamics could not afford to accept losses on initial units. Now that General Atomic is part of the Gulf Oil corporation, that limitation has been removed; a first reactor is nearing operation and there are six more on order."

Graph 3 taken from Reference 5 (page 18 of this report is a projection of the share of the reactor market which the gas-cooled reactor is expected to power. The same figure also projects reactors to be built by, Westinghouse, General Electric, Babcock and Wilcox, and Combustion Engineering.

THE FAST BREEDER REACTOR

The economically recoverable energy from ordinary nuclear reactors is ~300 x 10^{12} watt-years in the United States according to Starr[8], as shown in Table 2.

TABLE 2[8]

Depletable Supply (10^{12} Watt-Years)	World	U.S.
Coal	670 - 1,000	160 - 230
Petroleum	100 - 200	20 - 35
Gas	70 - 170	20 - 35
Subtotal	840 - 1,370	200 - 300
Nuclear (Ordinary Reactor)	~3,000	~300
Nuclear (Breeder Reactor)	~300,000	~30,000
Cumulative Demand 1960 to Year 2000 (10^{12} Watt-Years)	350 - 700	100 - 140

> "Economically recoverable fuel supply is an estimate of the quantities available at no more than twice present costs. U.S. reserves of all fossil fuels are slightly less than a fourth of the world total. Fossil-fuel reserves are barely equivalent to twice the cumulative demand for energy between 1960 and 2000. Even nuclear fuel is none too plentiful if one were to use only the ordinary light-water reactors. By employing breeder reactors, however, the nuclear supply can be amplified roughly a hundred fold. (10^{12} x 10^{15} BTU)"[8]

The fast breeder reactor permits the recovery of much of the available energy in uranium and thorium. This occurs because during fission in the fast breeder more than two neutrons are released per neutron absorbed. On the average, slightly more than one neutron is necessary for sustaining the

fission process, and the extra neutron can be absorbed in non-fissionable uranium-238. As the uranium-238 absorbs the extra neutron it is transformed into fissionable plutonium-239. Thus, while the fast reactor is sustaining the fission process and thereby creating energy, it is also generating fresh fuel which can later be used to create more energy. Reactors which have a breeding ratio greater than 1 create more fuel than they need for their own purposes, and the extra plutonium transformed from uranium-238 can be used to fuel new breeder reactors. By this means, up to 80% of the available energy in uranium can be recovered and used in reactors.

The fast breeder reactor gets its name from its ability to breed, that is to create more fissionable material than it consumes; and from the fact that its neutrons travel faster than they do in a thermal reactor. The breeding process depends, in part, upon the neutrons maintaining a high speed, or high energy. If their speed or energy is allowed to degrade as occurs in thermal reactors, the number of neutrons produced per absorption in uranium or plutonium decreases. Furthermore, at lower velocities, neutrons tend to be captured in various structural materials of the reactor, and this further reduces the breeding potential. It is important, therefore, in fast reactors to keep the velocity of the neutrons high. Water, which is used as a coolant in some thermal reactors, tends to slow the neutrons down and thus prevent efficient breeding. Therefore it is necessary to use a coolant which does not slow the neutrons or capture them as they travel through the coolant.

Considerable research and development has been carried out on the liquid metal cooled fast breeder reactor LMFBR. Another reactor concept, chiefly developed in the United States by Gulf General Atomic is the gas cooled fast

breeder reactor GCFRB.

Creagan[9] summarized the LMFBR work to date as of February 1973. Table 3[9] represents national commitments of several countries toward development of the LMFBR.

TABLE 3 National Investments in LMFBR

	Country					
	U. S.	U.S.S.R.	France	U. K.	Japan	Germany
LMFBR/year ($ millions)	200	200	100	70	50	30
1972 GNP ($ billions)	1113	538	162	128	232	195
Percentage of GNP	0.018%	0.04%	0.06%	0.055%	0.02%	0.015%

World status and plans for LMFBR power plants are given in Table 4[9], which lists LMFBR projects that are operable, under construction, planned and decommissioned with country location, megawatts thermal and electric, and initial operation date. Table 4 also shows whether a loop or pool configuration is used.

Present plans for the U.S. LMFBR program in the 1970's consist of completion of the 400 MWt Fast Flux Test Facility (FFTF) on the AEC's Hanford Reservation in the state of Washington. It will not produce electric power but will reject heat to an air heat exchanger. The Hanford Engineering Development Laboratory is operated for the AEC by Westinghouse Hanford Company. The FFTF, when completed in the mid-1970's will be used for testing fuels and materials. It will provide an environment typical of that to be found in future LMFBR's. The reactor will contain closed loops for advanced fuel tests, which will be isolated from process sodium in the main

TABLE 4 Liquid-Metal-Cooled Fast-Reactor Projects

Name	Country	Power		Pool or Loop	Initial Operation
		MWt	MWe		
Operable					
BR-5	U.S.S.R	5[a]	-	Loop	1959
DFR	U.K	72	14	Loop	1959
EBR-II	U.S.	62.5	16	Pool	1963
BN-350	U.S.S.R.	1000[b]	150	Loop	1973
PHENIX	France	600	250	Pool	1973
RAPSODIE	France	40	-	Loop	1967
BR-60 (BOR)	U.S.S.R.	60	12	Loop	1970
Under Constr.					
PFR	U.K.	600	250	Pool	1972
FFTF	U.S.	400	-	Loop	1977
JOYO	Japan	100[c]	-	Loop	1974
BN-600	U.S.S.R	1500	600	Pool	1976
KNK-11	W. Germany	58	20	Loop	1973
PEC	Italy	140	-	Modified Pool	1976
SNR	W. Germany[d]	730	300	Loop	1977
DEMO No. 1	U.S.	750-1250	300-500	Loop	?
MONJU	Japan	750	300	Loop	1978
DEMO No. 2	U.S.	750-1250	300-500	Not Decided	?
CFR	U.K.	3125	1320	Not Decided	1979
PHENIX 1000	France[e]	2500	1000	Pool	1979
SNR 2000	Germany	5000	2000	Loop	1983
Decommissioned					
FERMI	U.S.	200	60.9	Loop	1963
SEFOR	U.S.	20	-	Loop	1969
CLEMENTINE	U.S.	0.025	-	Loop	1946
EBR-1	U.S.	1	0.2	Loop	1951
BR-2	U.S.S.R.	0.1	-	Loop	1956
LAMPRE	U.S.	1	-	Loop	1961

a- To be increased to 10 MWt in 1972; b- Dual purpose; 150 MWe for electric power and 200 MWe equivalent for desalination.
c- To be operated at 50 MWt initially; d- In cooperation with Belgium and The Netherlands; e- Tripartite effort France, German and Italian electric utilities

reactor coolant loop so that test failures will not harm the reactor.

In addition to the FFTF, the highest priority U.S. LMFBR program is construction of a demonstration plant.

In the latter part of 1973 contracts were signed for the breeder demonstration plant. According to Nuclear News[10] the Project Management Corporation will provide over-all management and coordination design contractor and operation of the facility.

> "The AEC and PMC each signed a contract with the Breeder Reactor Corporation which represents the public utilities contributing to the project.
>
> Under the terms of the main contract, the AEC will seek statutory authority to have two representatives on the PMC board, which now has two members from the TVA, two from Commonwealth Edison, and one designated by the BRC. The parties to the contract had previously established a project steering committee composed of three members -- one each designated by the AEC, the TVA, and Edison. The steering committee will implement management of the project and will administer the contract. The steering committee would become an executive committee of the PMC board, when the AEC is represented on the board, subsequent to the passing of the legislation.
>
> By contract, the general project management authority and responsibility are vested in the PMC board and the steering committee."

Over $240 million has already been pledged by the electric utility industry for the first demonstration plant, which will be built on the Tennessee Valley Authority system. The total cost of this plant is estimated at about $500 million.

Two organizations have been established to implement this project. The Breeder Reactor Corporation's (BRC) 17 man board represents both investor-owned and consumer-owned utilities, plus the Edison Electric Institute, the American Public Power Association, and the National Rural Electric Cooperative Association. The BRC will provide senior counsel, manage financial contributions to the project, serve as a liaison with the Nation's utilities, and

handle the dissemination of information.

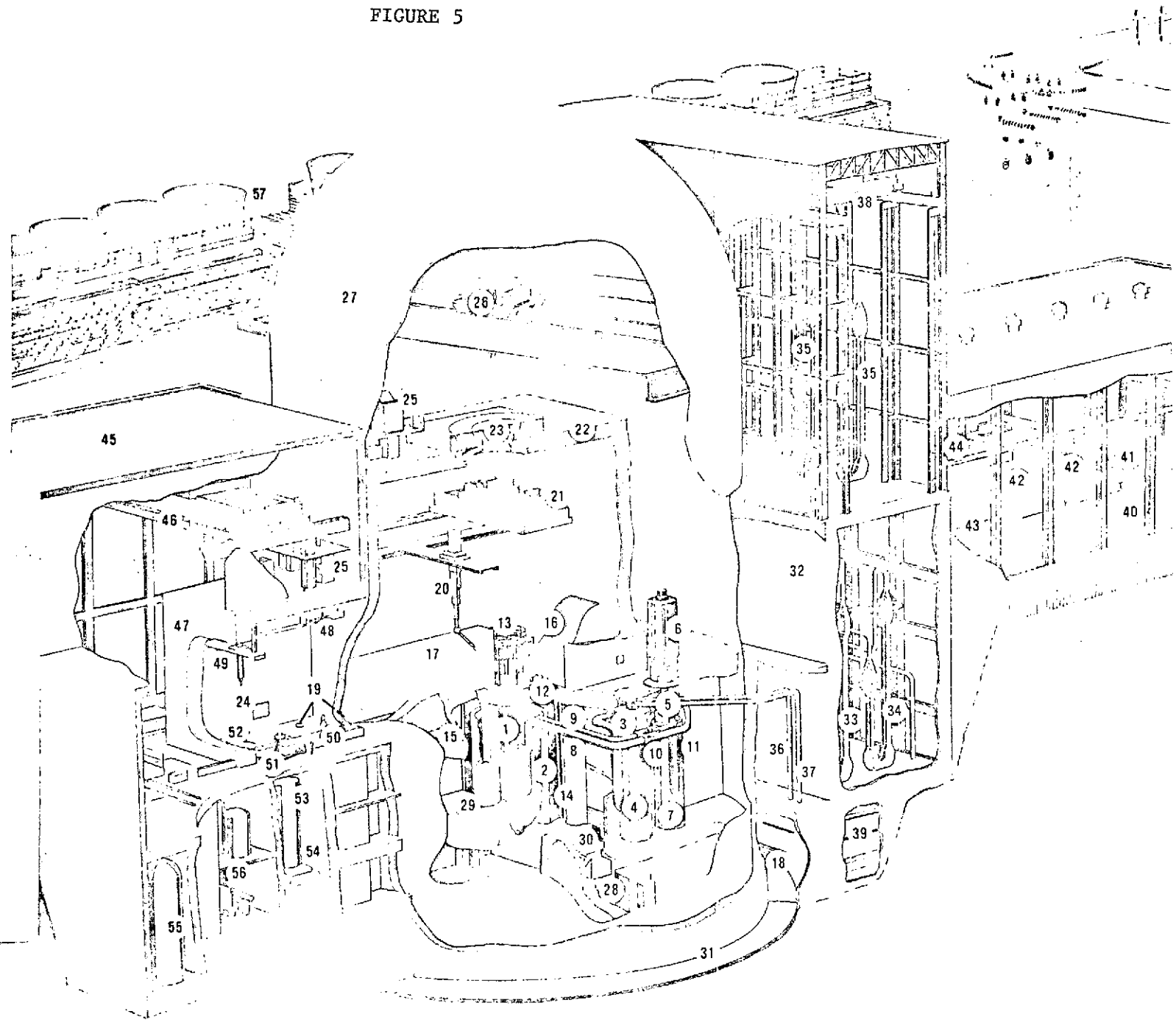
The board of the Project Management Corporation (PMC) includes one representative from the BRC, two from TVA, and two from Commonwealth Edison Company, Chicago. The PMC will be responsible for overall design, engineering, and construction of the plant. Commonwealth Edison has provided the Project Manager and Engineering Manager, and TVA will start up, operate, and maintain the demonstration plant. Represented on the PMC Steering Committee are TVA, Commonwealth Edison, and the AEC. The Westinghouse Electric Corporation has been awarded the contract for the reactor system. Figure 5 is a schematic design of the LMFBR demonstration plant.

Westinghouse LMFBR Demonstration Plant

FIGURE 5

1. REACTOR VESSEL
2. REACTOR VESSEL GUARD VESSEL
3. INTERMEDIATE HEAT EXCHANGER (IHX)
4. IHX GUARD VESSEL
5. PRIMARY SODIUM PUMP
6. PRIMARY SODIUM PUMP DRIVE MOTOR
7. PRIMARY SODIUM PUMP GUARD VESSEL
8. PRIMARY SODIUM HOT LEG PIPING
9. PRIMARY SODIUM COLD LEG PIPING
10. ISOLATION VALVE
11. CHECK VALVE
12. REACTOR CLOSURE HEAD
13. CONTROL ROD DRIVE MECHANISMS
14. SHIELDING
15. OPERATING FLOOR
16. POWER AND INSTRUMENT CABLES
17. EX-VESSEL TRANSFER MACHINE HOUSING
18. PRIMARY SODIUM STORAGE TANKS
19. FUEL STORAGE TANK PORT
20. OVERHEAD MANIPULATOR
21. REFUELING HOT CELL CRANE
22. REFUELING HOT CELL (UNDER-THE-PLUG CONCEPT)
23. REFUELING HOT CELL ROOF PORT
24. VIEWING WINDOW
25. HOT CELL ENVIRONMENTAL CONTROL EQUIPMENT
26. CONTAINMENT BUILDING POLAR CRANE
27. CONTAINMENT BARRIER
28. COMPARTMENT COOLING AND INERTING EQUIPMENT
29. PRIMARY SODIUM PURIFICATION EQUIPMENT
30. STAIRWELL

31. REACTOR BUILDING FOUNDATION
32. STEAM GENERATOR BUILDING
33. STEAM GENERATOR-EVAPORATOR
34. STEAM GENERATOR-SUPERHEATER
35. STEAM GENERATOR-SPARE
36. INTERMEDIATE SODIUM COLD LEG PIPING
37. INTERMEDIATE SODIUM HOT LEG PIPING
38. STEAM GENERATOR BUILDING CRANE
39. SODIUM STORAGE TANK
40. TURBINE-GENERATOR BUILDING
41. ELECTRIC GENERATOR
42. L.P. TURBINE
43. H.P. TURBINE
44. MOISTURE SEPARATOR AND REHEATER
45. FUEL SERVICE BUILDING
46. FUEL SERVICE BUILDING CRANE
47. FUEL SERVICE HOT CELL
48. FUEL SERVICE HOT CELL CRANE
49. FUEL SERVICE MANIPULATOR
50. FUEL SERVICE ROTOR DRIVE
51. NEW FUEL TRANSFER VALVE
52. SHIPPING CASK PORT
53. IRRADIATED FUEL SHIPPING CASK AND CART
54. FUEL CANEL
55. EMERGENCY GASEOUS RADWASTE STORAGE TANK
56. GASEOUS RADWASTE SYSTEM EQUIPMENT
57. FORCED DRAFT COOLING TOWERS
58. MAIN STEPUP TRANSFORMER SUBSTATION
59. SWITCHYARD



FBR DESIGN CONSIDERATIONS

There is extensive leterature on fast reactor engineering considerations and fast reactor physics.[10-22] Rather than repeat the detailed papers we will summarize here only the general characteristics which have a beneficial or adverse effect.

Sodium is a metal melting at about 210°F. It has a low cross-section for absorbing and thermalizing neutrons, but its ability to transfer heat is excellent. It has a high boiling point (1640°F) and a low vapor pressure at most temperatures. These properties make it almost ideal for use as a coolant in a reactor. It can be heated to high temperatures without generating pressure and its excellent ability to transfer heat makes it less sensitive to short term disturbances in the surfaces from which the heat is being transferred. Because the coolant system is operating at a low pressure, in the event of a pipe failure, the liquid will not escape as rapidly as occurs with high pressure systems.

Chemical reactivity of sodium is a safety aspect in some respects. During irradiation of fuel many radioactive isotopes are formed known as fission products. Some of the fission products are radioactive in unstable species of elements which decay gradually to stable forms. In some of the fast reactors these fission products are vented or discahrged from the fuel to the reactor into the sodium coolant. In other fast reactors failure in the fuel outer cladding can release these fission products to the sodium. Because of its unique chemical properties, sodium tends to retain some of these fission products, so they are not so readily released to the inert gases such as helium and argon which are used to blanket the sodium. Radioactive

iodine, for example, combines with the sodium to form sodium iodide and cesium is retained in the solution. Niobium and certain other solid fission products also tend to be retained in the sodium. However, the sodium does not retain all fission products. Nearly all of the radioactive zenon and krypton gases bubble up to the sodium and are released into the inert cover gas. Thus, the property of sodium to retain some materials acts as a safety advantage since accident or spillage of sodium does not free quantities of fission products. If the sodium were to violate or break through its containers and to burn in the air, the burning is at a constant rate of the order of 2-14 lbs/hr-square foot of exposed surface, and the fission products would not be released rapidly. This would give time to cope with other problems such as containing the fire. Reactiveness of the sodium causes certain undesirable aspects. For example, when exposed to air sodium oxidizes rapidly if it is in the solid state, and, if in the liquid state, it will burn. This burning is at a constant rate and can be extinguished by eliminating oxygen. When exposed to water, sodium will react violently to form hydrogen. The hydrogen in turn can combine with oxygen and increase the reaction energy. Other features, of sodium also make it undesirable for reactor coolant. Irradiation sodium forms the radioisotopes Na-22 and Na-24 which emit gamma radiation. However most of this radiation will decay within a few days. The characteristic of sodium to become radioactive and to contain radioactive products from other sources makes it potentially hazardous.

In practice, the accessability of sodium to human access is limited. One way for accomplishing this in a fast-breeder reactor is to include two separate cooling circuits containing sodium and one containing water. The first circuit circulates the sodium to the reactor core and becomes highly radioactive.

This radioactive circuit is shielded from human acess, and any maintanence can be accomplished by remote mechanisms. The second circuit picks up the heat from the first and in turn transfers the heat to the water circuit without becoming radioactive. Because of the excellent heat transfer characteristics of sodium, these circuits can be used and still have an economically attractive system. Nevertheless, the extra sodium loop is a safety feature which is included at the expense of extra cost.

The undisrable aspects of sodium can be treated in two ways:

(1) All equipment containing radioactive sodium is placed in gas-type cells which exclude oxygen.

(2) Water is used only to transfer the heat from the nonradioactive sodium circuits, and these circuits are designed to withstand the effects of a sodium water reaction.

There are two important neutronic characteristics of fast reactors which are significantly different from those of thermal reactors. These are:

(1) The shortness on lifetime.

(2) The possibility of secondary criticality.

Neutron lifetime is a measure of the time interval between the birth of the neutron when fission occurs, and its capture in uranium or other materials. Thermal reactor neutrons are slowed down by bouncing off hydrogen atoms, if water is a coolant. Neutrons "live" longer than in a fast reactor in which there is no hydrogen or moderator material to slow them before they are captured. This short neutron lifetime was originally thought to be an undesirable feature. Subsequent research has shown that a short lifetime need not be a significant disadvantage provided the instantaneous power coefficient is negative. With a negative instataneous power coefficient, the lengths of neutron lifetime has

little effect on amplitude or duration of the energy ramps from the reactor.

Secondary criticality is a somewhat more complex situation. In any reactor system it is necessary for a certain minimum amount of fissionable material to be present before a self-sustaining chain reaction can take place. The self-sustaining chain reaction takes place when the number of neutrons lost from the system or captured is exactly balanced by the number of neutrons which are being generated in the process of the fission. A thermal reactor is so designed that this balance occurs only after the neutrons have been slowed to a thermal energies. In a fast reactor since the neutrons are not slowed down, criticality is achieved without a moderating material.

The characteristic of fast reactors to be able to be critical without the coolant present can result in "secondary criticality". If for example, some of the fuel which would melt and fall to the bottom of the reactor while at the same time rearranging itself into a more dense assembly or arrangement by filling up the passages normally occupied by the coolant then a critical mass could be possible and this new configuration could become an uncontrolled reactor. This potential problem has resulted in considerable study with the consequence that fast reactors are deisgned with great care to avoid possibilities which can lead to a rearrangement of the core and to a more reactive configuration. This can be accomplished by designing the coolant so that the possibility of the loss of a large amount of the coolant capacity is very low, and also by selecting a geometric arrangement which makes the assembly into a more reactive configuration difficult. Additionally, instrumentation to detect the onset of abnormal circumstances which might lead to meltdown can be included. In the past, two fast reactors have acutally experienced partial core meltdown and of both of these reactors the coolant

systems and core geometry where such that the secondary criticality did not occur. Although the probability of secondary criticality is very low, most fast reactor systems designed today have included provisions for accommodating energy released during uncontrolled transient from secondary criticality.

One way to avoid secondary criticality is to insure that the coolant integrity is always maintained. To achieve such assurance reactor systems engineers often take great care in the design of the primary coolant circuit. For example, in one type of fast reactor design the core and all the pumps, valves, pipes and heat exchangers which must circulate the primary sodium from the core are positioned within a large tank vessel which is filled with sodium.

It was previously mentioned that an instantaneous negative power coefficient was desirable. A power coefficient is simply a term which describes the response of the reactor to certain stimuli. For instance, if the power is increased by withdrawing control rods which control the nuclear chain reaction, this would normally cause the fuel to increase in temperature and to expand physically. As the core expands from the higher temperature its height grows slightly and its outside surface area becomes larger. This wll permit a greater number of neutrons to leak out of the core and to be lost from the reactor system, thus tending to reduce the amount of neutrons which are fissioning. This in turn will cause the reactor power increase to be reduced, compared to what would have been the case if the thermal expansion had not occurred. The entire effect is described as thermal expansion power coefficient. It is negative. If the coefficient were positive instead of negative the opposite effect would occur, namely that as power increases in the reactor by withdrawing control rods, this increase would be amplified beyond

the movement implied by the control rods.

During the early developments of the fast reactor it became obvious that two particular reactor characteristics were desirable. One of the characteristics was a long fuel lifetime, and the other is a negative power coefficient. A long fuel lifetime which permits leaving the fuel in the reactor for an extended time can yield a low fuel cost. Most of the early reactor designs included uranium fuel in the form of metal. However, under irradiation this metal gradually damaged and had to be removed from the core. By changing the form of the uranium or plutonium metal to uranium or plutonium oxide it is found that the lifetime of the fuel can be extended substantially. Fortunately it is found that using ceramic fuel not only improved the fuel lifetime characteristics but also introduced a prompt negative power coefficient which was as predictable as expansion coefficient in metal fuel. This particular coefficient is known as a Doppler coefficient. Since the ceramic fuel is high temperature material, in order for the fuel to undergo damage it must reach very high temperatures. It is the change in temperature from the operating point to some higher temperature which produces the Doppler effect. This effect which is caused by the heating up of the atoms of the uranium fuel, causing them to move faster. Neutrons which are passing through the fuel tend to be captured by some of the U-238 atoms at what is known as a resonance energy. The increased velocity of uranium atoms increases the number of these atoms which are at the resonance capture and would be relative to the passing neutrons. Thus these U-238 atoms therefore stop some of the neutrons which otherwise would have continued their travel until capture in the fission process, and this effect tends to lower reactivity and power. Again a

reactivity or negative power coefficient results. Discovery of the Doppler effect in a fast reactor was an extremely important development.

One of the power coefficients in a sodium cooled fast breeder reactor which is not negative is a sodium void coefficient. If the sodium were to boil down it could be expelled from the coolant channels. Depending upon the geometry of a fast reactor core in the manner in which the sodium is removed, this can result in a positive reactivity effect. This happens because sodium tends to slow neutrons down and reduce the number of fast neutrons available for fissioning. Therefore, when sodium is removed from the core by boiling, not as many atoms are slowed and more fast neutrons are present for the fission process. A competing effect is that the removal of sodium also tends to allow more neutrons to leak from the core and this results in a decrease in the total number of neutrons. The net result of these two competing effects is dependent upon the geometric pattern of the sodium being removed from the core. Under proper conditions the net effect can be to increase the number of neutrons available for fission with a consequent reactivity increase and increase in power level of the core. As previously mentioned, the sodium operates very much like a below the boiling point of the reactor and this reduces the likelihood of boiling. Furthermore, instruments are present to detect conditions which might cause boiling, the reactor can be shut down if anomalies develop.

From the previous discussion of the characteristics of fast breeder reactors, it is clear that some of the characteristics have a beneficial effect on the safety of the reactor and others have an adverse effect. Considerable amounts of experience and design work permit the selection of parameters and design features so as to amplify the desirable characteristics and to deemphasize or properly cope with the undesirable characteristics.

Safety and environmental effects of fast reactors have been questioned even more than thermal reactors. This subject will be discussed further in the next section.

ENVIRONMENTAL ASPECTS OF NUCLEAR POWER STATIONS

To the nuclear industry the term environment means those parts of nature which interact with nuclear operations, namely the atmosphere, the land, surface water, ground water, coastal waters, and the sea. In normal operation nuclear power plants have an interaction on the environment as well as a potentially adverse effect on the environment in the event of an accident. In normal operation, nuclear power plants release a small amount of radioactivity in the effluents - air and water. Additionally, there is a problem of the effect of quantities of waste heat on the water bodies or the atmosphere to which the heat is discharged. The impact of nuclear power stations on the environment have been a continuing study ever since electrical power generation using nuclear reactors became feasible. These problems have been debated, for example at the first United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955. Extensive research has continued.[26] Over the years the International Commission on Radiological Protection has prepared a number of recommendations on dose limits for external and internal exposures and the concentration limits of various radionuclides in air and water. These have been used as a base for determining safe working levels for various forms of radioactive material both occupationally exposed workers and the general public.

The general public has become slowly aware of the side effects resulting from the many spectacular advances in various fields - medicine, agriculture, motor and air transport, and power generation. By the late 1960's the problem of "pollution" had become a topic that aroused strong feelings in the general public in most industrial countries. Many people have expressed concern with atomic energy, nuclear power stations, existing

and proposed. The safety of the reactor and relative importance of the effect on the environment has become an extremely controversial issue. In some cases, as a result of intervention, radioactivity limits have been decreased and nuclear power plant startup schedules have been delayed as a result of long public hearings and arguments.

According to reference 24

> "Prior to the issuance of a construction permit or an operating license for a nuclear power plant, the USAEC is required to assess the potential environmental effects of that plant in order to assure that the issuance of that permit or license will be consistant with the national environmental goals as set forth by the Public Environmental Act of 1969 (Public Law 91-190). In order to obtain information essential to this assesment the commission requires each applicant for a permit or license to submit a report on the potential environmental effects of the proposed plant and associated facilities.
>
> The national environmental goals as expressed by the environmental policy act (NEPA) are as follows: 'It is the continuing responsibility of the federal government to use all practical means, consistant with other essential considerations of national policy, to improve and coordinate federal plans functions, program and resources."

Additionally, a cost benefit analysis is required in which alternate site-plant combinations and plant systems are to be examined in order to determine whether the proposed facility is a cost effective choice considering economic, social and other environmental factors and any institutional constraints.

One of the most often discussed topics concerning the social aspects of nuclear power is the biological effect of radiation. It is widely known that sufficient exposure to radiation can be harmful to man. Both the nuclear industry and the population as a whole will be exposed to increased levels of radiation as nuclear power plants increase in number. The genetic effects of radiation have been studied for some time now and a relatively large amount of data is available. However, extrapolation of the data to the low

exposures levels from an operating nuclear power plant has been the subject of much controversy[25]. Consequently the effects have been generally estimated by linear extrapolation from data at higher exposure levels. These results must be then compared to exposures from natural and other man-made sources to determine the social impact of increased radiation levels. Exposures from natural sources are shown in Table 5, and exposures from man-made sources are shown in Table 6.

Estimates of the exposure to radiation due to operation of nuclear reactors are given in the Argonne study as the impact on the whole U.S. and world populations. Tables 7 and 8 show the total radiation dose to these populations for nuclear power industries in 1980 and 1990. These data already include factors to account for population size and mean lifetime of radioactive nuclide; and to compare with previous tables 130 man-rad exposure to Kr-85 in the U.S. for the 1980 industry corresponds to a dose rate of about 3×10^{-4} mrad/yr. Thus one can see that the influence of nuclear plant operation to the general population exposure is very slight.

Global effects are not the only exposure effects that must be included however, since the fraction of the population that works in the nuclear industry will receive a proportionally higher exposure. The majority of the exposure comes from reactor operation, mining, and fuel reprocessing. The exposure breakdown for several reactor types is given in Table 9.

It is necessary now to translate the exposure levels that have been estimated to arise from nuclear plants into health effects. There is a large amount of data on somatic effects on laboratory animals, but we would like the somatic and genetic effects on humans. The induction of malignant neoplasms has had the largest attention of radiobiolotists, and estimates

TABLE 5 Average Dose Rates Due to Natural Background

Source of Irradiation	Dose Rate (mrad/yr) Gonad (=total body)	Bone Marrow
External irradiation		
Extraterrestrial sources		
Ionizing component	28	28
Neutrons	0.7	0.7
Terrestrial sources	50	50
Internal irradiation		
K-40	20	15
C-14	0.8	1.6
Ra-226 and decay products, 35% equilibrium	-	0.03
Ra-228 and decay products equilibrium	-	0.03
Po-210 and decay products 50% equilibrium	0.3	0.3
Rn-222 dissolved in tissues	0.3	0.3
External irradiation (excl. neutrons)	56-150	56-150
Terrestrial γ rays from building materials (measured inside of buildings)	17-180	17-180

TABLE 6 Average Exposure Due to Man-Made Sources of Radiation

	Genetically-Significant dose rate (mrad/yr)	Dose/exposure or treatment (mrad)	
Medical sources		Gonad	Bone Marrow
Diagnostic x-ray	7-58	0.1-5100	0.2-2000
External radiotherapy	2-13	0.1-160,000	0.5-100
Internal radioisotopes	0.2-0.4		
Weapons test fallout (1954-1962 testing)		Dose commitment/individual (mrad)	
		Gonad	Bone Marrow
To year 2000	2	80	140
After year 2000 (due to C-14		180	180
Occupational	0.2		
Miscellaneous (e.g., consumer products)	2		

TABLE 7

Predictions of World and U.S. Population Exposure
Resulting From the Expected Release of Kr-85 and H-3 Formed
During Operation of a 1000 MWe Reactor in 1980

Energy System	PWR	BWR	HTGR
World (whole body man-rad)			
Kr-85	130	130	256
H-3	22	21	21
Total	152	151	277
U.S. only (whole body man-rad)			
Kr-85	6.1	6.1	12.0
H-3	2.4	2.3	2.3
Total	8.5	8.4	14.3

Note: The predictions of the global model are order of magnitude estimates. The number of significant digits shown is not indicative of precision.
(Hub, 1973)

TABLE 8

Global Model Predictions of World and U.S. Population Expsoure Resulting From the Expected Release of Kr-85 and H-3 Formed During Operation of a 1000 MWe Reactor in 1990

	PWR	LMFBR
World (whole body man-rad)		
Kr-85	0.55	0.39
H-3	29	31
Total	30	31
U.S. (shole body man-rad)		
Kr-85	0.02	0.02
H-3	2.7	2.9
Total	2.7	2.9

Note: The predictions of the global model are order of magnitude estimates. The number of significant digits shown is not indicative of precision. (Hub, 1973)

TABLE 9

Radiation Dose From Nuclear Energy Systems

Energy System	1980 PWR	1980 BWR	1980 HTGR	1990 PWR	1990 LMFBR
General Population					
Global model (man-rad)*	150	150	280	30	31
Local model (man-rad)	4	25	5	4	4
Occupational Personnel					
Miners' exposure (man-WLM)	110	100	58	110	0
All other fuel cycle steps (man-rad)	345	350	327	345	339

Man-rad population doses are whole body doses.

WLM (Working Level Months) is the unit of miner exposure used in uranium mining.

of risk from radiation exposure are often found in this category. Table 10 gives such a risk summary. Other somatic effects include reduced fertility and reduced lifespan.

Several factors may be similar between fossil and fission power plants such as thermal pollution. Thermal discharges from nuclear plants are relatively higher than for fossil plants due to lower thermal efficiency. Particulate emission is essentially zero for a nuclear plant, although quite large for a fossil plant.

The dollar cost of social impact from operation of a nuclear power plant has been estimated in the Argonne study[25], and values can be compared to the fossil plant costs notes earlier. For a 1000 MWe plant operating in 1990 the annual external costs were estimated to be $1,100,000 for a light water reactor plant and $900,000 for a LMFBR. As with the fossil plant the largest fraction of the cost was due to thermal discharge. These costs were higher than the fossil plant and were $1,000,000 and $800,000 for the LWR and LMFBR respectively. The total health related effects in terms of man-days lost is 1790 for the LWR and 1310 for the LMFBR. A more detailed breakdown of the Argonne results with comparison to the fossil plant is given in Table 11.

TABLE 10

Summary of Risk of Radiation Induced Somatic and Genetic Effect

Somatic

Estimates of the probability that a person will develop a malignant neoplasm following irradiation in various periods of life.

Type of Neoplasm	Period of Life at Time of Irradiation	Probability of Developing Malignancy During Time Period Specified	Exposure Condition to Which Probability Applies
Leukemia	Adulthood	Lifetime: $0(2 \times 10^{-5}/\text{rad})$[a]	Approximately uniform irradiation of red bone marrow.
Thyroid cancer	Infancy (< 6 months)	Lifetime: $0(10 \times 10^{-5}/\text{rad})$	Approximately uniform irradiation of thyroid.
	Adulthood	Lifetime: $0(3 \times 10^{-5}/\text{rad})$	Not applicable to I-131 uptake by thyroid.
Total Malignancies (including leukemia)	Adulthood	Lifetime: $0(20 \times 10^{-5}/\text{rad})$	Approximately uniform irradiation of whole body.
	In utero	Before age 10: $0(60 \times 10^{-5}/\text{rad})$	

Genetic

Estimates of the probability that a mutation will be transmitted to a conceived offspring as a result of irradiation of a parent.

Period of Life at Time of Irradiation	Sex of Parent	Probability That Mutation is Transmitted	Exposure Condition to Which Probability Applies
Before end of reproductive lifetime	male	$0(2 \times 10^{-3}/\text{rad})$[b]	Approximately uniform irradiation of gonads.
	female	$0(0/\text{rad})$[b]	

[a] The notation of 0(r) indicates that r is an order-of-magnitude value

[b] Male and female are assumed to be equally irradiated, the situation which is encountered in exposure of the general population. If female alone is irradiated, then the estimate of zero transmitted mutations should not be applied.

(Hub, 1973)

TABLE 11

Annual Costs for 1000 MWe Energy Systems for Nominal 1990

$ = Millions of dollars
MDL = Man-days lost

Item	Energy System	PWR $	PWR MDL	Coal $	Coal MDL
INTERNAL COSTS					
Conventional Costs					
Capital		80		50	
Operation & Maintenance		9		14	
Fuel		21		74	
Health & Accident					
Occupational Accident		C	490	C	2400
Occupational Health		C	240		~0
Public Injuries in Transportation		C	30		S
Total Internal		110	760	138	2400
EXTERNAL COSTS					
Public Health & Accidents					
Routine Pollutant Release		.003	40		U
Accidental Radiation Release		S	S		-
Large Accident at Power Plant		-	U		-
Transportation Accidents		.002	30		S
Genetic Effects		.006	70		U
Occupational Health & Accidents					
Accidents		.04	490	0.2	2400
Health		.02	240		~0
Genetic Effects		.07	920		U
Damage					
Water Base	Thermal Discharge	1.0		0.7	
	Other				
Air Base	SO_2 & Particulates	0.		S	
	Other				
Land Base	Mining			~0	
Total External Man-Days Lost			1790		2400
Total External Cost		1.1		0.9	
Total Internal & External MDL (Rounded)			2600		4800
Total Internal & External Cost (Rounded)		111		139	

S = Small
U = Unevaluated
C = Included in conventional cost

NUCLEAR POWER ECONOMICS

Energy cost can be broken into three components:

1) investment costs

2) fuel costs

3) operating and maintenance costs

An additional cost, the social cost, is discussed in the previous section. A major justification for the nuclear fuel industry is the low nuclear fuel cycle cost. The calculation of the nuclear fuel costs involves the cost of each unit operation in the nuclear fuel cycle and also the schedule, since carrying charges must be accounted for. Fuel cycle costs can be calculated by simplified hand techniques[27-31] or more complex computer programs.[32-35]

Table 12 lists the results of a typical calculation.[2]

TABLE 12[2]

PWR Fuel Cycle Cost Projection
1,150 MWe
1975 Startup
80% Capacity Factor

	Fuel Cost (mills/kwh)			
	Consumption Costs	Financing Costs	Total	% of Costs
Fabrication (@ $70/kg U)	0.34	0.08	0.42	20
Uranium Ore (@ $8/lb U_3O_8)	0.56	0.18	0.74	36
Conversion (@ $2.52/kg U) and reprocessing (@ $45/kg U)	0.62	0.16	0.78	37
Spent fuel shipping and reprocessing (@ $45/kg U)	0.19	−0.04	0.15	7
Plutonium (@ $7.50/gm Pu) and uranium credits	−0.35	0.08	−0.27	—
Totals	1.36	0.46	1.82	100

Notes: Consumption costs include interest during construction in the first core. Cost of money and interest during construction at 7%/yr and total fixed charge rate on non-depreciable capital at 14%/yr. The first three items include 4% sales tax.

The cost of power plants, nuclear and fossil, has been increasing.[37,38] This trend can be expected to continue. In general the nuclear plant costs more than the fossil plant, but the lower fuel costs for the nuclear make the nuclear plant competitive, depending on the price of coal.. Table 13 also taken from reference 2 gives an estimate of energy cost for a typical nuclear plant.

TABLE 13[2]

Estimates of Energy Costs
(mills/kwh)

	1975 LWR
Investment costs	4.0-4.8
Fuel costs	1.7-1.9
Operating and maintenance costs	0.3
Total energy costs	6.0-7.0

The use of a plant in a power network depends, of course, on the other power plants available, the characteristics of each plant, incremental power costs of each unit, and the power demand on the system. In case of the outage for nuclear refueling, other units must be committed and dispatched (or power purchased from the outside) to meet the demand. The cost of nuclear power is system dependent. According to Hoskin[3]

> "The strong interdependence between management of nuclear fuel and overall power system management leads to a very large and complex multi-stage optimization problem which can best be treated, in principal, by the systematic application of simulation, systems analysis, and operations research techniques. Over the past four or five years a great deal of work has been done on various approaches to and various aspects of this or closely related optimization problems. Some products of these efforts are now in routine use, others are approaching the power of moving from the development stage to practical application, while some are still in the formative and experimental stage."

Table 14[36] is a comparison of important characteristics of types of electric generating units, which must be considered for optimizing the mix of types of power plants in a system.

TABLE 14

CHARACTERISTICS OF TYPES OF ELECTRIC GENERATING UNITS

TABLE 1
CHARACTERISTICS OF TYPES OF ELECTRIC GENERATING UNITS

	Dimensions	Nuclear Steam (LWR)	Fossil Steam	Fast-Start Peaking	Hydro	Pumped-Hydro
System use		Base-load	Base-load and cyclical	Peaking	Inventory dependent	Peaking
Capacity factor	Percent	60-90	30-90	Up to 20	Up to 100	Up to 50
Capital cost	$/kwe	300-450	250-400	100-150	300-500	100-200
Unit capacity	MW	500-1200	200-1200	10-50	10-600	50-400
Minimum power	% unit capacity	10-40	10-50	75-90	0-10	25-40
Avg heat rate	MBTU/MWH	10.5-11	8.5-14	12-17	N/A	N/A
Avg net energy conversion efficiency	Percent	31-34	25-40	20-28	85-93	65-80
Fuel cost	¢/MBTU	16-20	35-80 (coal) 50-100 (oil)	50-100	0	Cost of pumping power
Energy cost	$/MWH	1.7-2.2	3.0-8.4	6.5-20	0	~1.5 x pumping power cost
Comments on fuel inventory		Depends on fuel cycle	Approximately constant at 100 days supply	4-8 hours (oil)	Depends on season	Depends on operating cycle
Transmission losses	Percent	Up to 10	Up to 10	Up to 5	Up to 10	Up to 15
Startup shutdown heat requirement	MBTU/MW Capacity	3-6	3-8	0-2	~0	~0
Min shutdown time	Hours	<2	2-10	<0.3	<0.5	<0.5
Maintenance requirement	Weeks/year	4-8 wk/ refueling	3-5	1-4	1-2	1-2
Forced-outage rate	Percent	Up to 15	Up to 20	Up to 40	Up to 5	Up to 10
Performance probability	Percent	85-100	80-100	90-100	95-100	95-100

PLUTONIUM RECYCLE IN LIGHT WATER REACTORS

According to Graph 12 page 27 of this report, by 1976 plutonium will be discharged from operating power plants at a rate of about 14,000 kilograms of fissile material per year with a total worth of over 100 million dollars. In 1978 the production rate will be 25,140 kilograms of fissile material per year. The cumulative value of plutonium produced in the next ten years is approximately 200,000 kg. It is apparent that there will be a strong economic incentive for recycling plutonium in thermal reactors in the United States in the mid-1970's to mid-1980's.

The concept of plutonium recycle has been with the nuclear industry for a long time because the nuclear fuel cycle economics depends upon how well the plutonium generated by thermal reactors can be utilized. The credit for plutonium has a potential value of more than 10% of the fuel costs of the lightwater reactors now committed. But that plutonium must be recycled economically for this credit to be achieved. Nuclear fuel costs analyses have taken into account the credit for plutonium since the 1950's and today's light water reactors receive a plutonium credit of about .2 mil/kil hr. This value had been supported in the United States by the Atomic Energy Commission's guaranteed buy-back, which had been used to supply various research and development requirements in providing for demonstration programs. In December 1970 the guaranteed government buy-back of plutonium ended. So that as more reactors come on the line in the 70's, substantial quantities of plutonium over and above any requirements for breeder development will become available.

It is presently estimated that the first large scale commercial breeder

reactor will not be able to go on the line until during the mid-1980's. If so, the requirements for fast breeder inventories would not become a substantial factor in the plutonium market before the 1990's. Without plutonium recycle by 1984, many tons of fissile plutonium would accumulate, which would amount to well over one billion dollars. It would be uneconomical of course to stock pile large amounts of plutonium for an extended period of time.

It should be pointed out that we are producing and burning plutonium in place in current day reactors since as much as 40% of the energy is produced by the plutonium in the core after 30,000 megawatt days per metric ton of uranium. Although the economic importance of plutonium recycle starting in the mid-70's has been generally recognized, the preparations and the development programs required for the necessary recycle ability are not as fully appreciated. There are several important differences between plutonium and uranium fuel that require careful design consideration. The plutonium, which is produced in a reactor consists of several isotopes. Important characteristics of these isotopes are listed in Table 15. Unlike uranium fuel, for example as shown in the table, the designer must work with plutonium that is 71% fissile, the remainder being nuclear poison. As shown by Puechl[39] the details of nuclear analysis to calculate the depletion of the higher isotopes must be accounted for and since they affect the reactivity lifetime in an important fashion. The designer must account then for the product buildup that has taken place after the material is being recycled.

There are significant nuclear differences in the characteristics between plutonium and uranium. These characteristics are summarized in Table 16. Some characteristics for the mixed oxide PuO_2UO_2 reactor are worth mentioning.

TABLE 15. Characteristics of PWR-Grade Plutonium

Isotope	Fraction*	Fissile	Major Radiation Sources for		
			Alpha	X, Gamma	Neutron
Pu^{238}	< 0.01				X
Pu^{239}	0.58	X	X		
Pu^{240}	0.23		X		X
Pu^{241}	0.13	X		X	
Pu^{242}	0.06		X		X
Am^{241}	**			X	
U^{237}	**			X	

* Based on recycling plutonium generated after 3 cycles of operation in a large PWR.

** Daughter products of ^{241}Pu which has a 13 year half life.

TABLE 16. Capsule Comparison of Uranium and Plutonium Nuclear Design Characteristics

Parameter	Plutonium Core	Reason for Difference	Consequence
Moderator Temperature Coefficient	More Negative	Increased resonance absorption and spectrum shift	Improved stability and transient characteristics except for steam break
Doppler Coefficient	More Negative	Pu-240 resonances	Improved transient characteristics
Cold-to-Hot Reactivity Swing	Increased	Larger moderator temperature coefficient	None-boron used for compensation
Installed Reactivity	Reduced	Reduced depletion rate-Reactivity saturates	None
Control Rod Requirement	Increased	Larger moderator and doppler coefficients	Possible increase in number of rods
Control Rod Worth	Reduced	Thermal flux reduced	Possible increase in number of rods
Boron Worth	Reduced	Thermal flux reduced	None
Xenon Worth	Reduced	Thermal flux reduced	Improved stability
Fission Product Poisons	Increased	Increased yields-Increased resonance absorptions	Reactivity penalty
Local Power Peaking	Increased	Increased water worth	Fuel management action required
Delayed Neutron Fraction	Reduced	$\beta_{pu} < \beta_u$	Rod ejection accident

Qualifications: 1. Effects can be modified by changes in design H/F; 2. Successive recycles influence the parameters

The temperature and Doppler coefficient are both more negative in the partial plutonium core. However, the former results in improved stability and the latter in imporved transient response. Xenon worth is also decreased resulting in improved stability of the thermal reactor. On the negative side, however, the fission products increase, resulting in reactivity penalty. Local power peaking also becomes a problem, but this can sometimes be solved by certain fuel management requirements. Also control rod worth decreases result in a necessity for more control rods.

Another deviation from a uranium experience involves an enriching step accomplished in the plutonium fuel fabrication plant. For the uranium fuel fabricated this function is provided by the AEC. Fuel fabrication is another important problem area in developing plutonium recycle capability. A number of problems unique to plutonium are not encountered during uranium fabrication. These differences from uranium fabrication include toxicity, radiation and criticality considerations, all of which affect the development of the required fuel facilities.

Because of its toxicity, plutonium must at all times be isolated from the personnel until the product is encapsulated. It is therefore necessary that it be confined by effective barriers such a glove boxes which completely contain the processing equipment. Directionally controlled air flow is needed to limit the spread of airborne contaminants.

Shielding is another problem for plutonium recycle fabrication. Neutron and gamma radiation from the plutonium isotopes, as summarized in Table 15, constitute sources of external exposure when handling plutonium. The magnitude of the gamma radiation from the american depends on the time between reprocessing and fabrication. Neutron radiation levels depend on the fuel burnup

and the recycle history of the plutonium. Criticality safety is another important aspect in which plutonium processing is different from uranium processing. Much of the fabrication process plutonium enrichment is equivalent to 93% enriched uranium. This high enrichment means that small batches are required for those parts of the process which involve undiluted plutonium. By contrast the maximum uranium enrichment employed in fabricating uranium fuel is of the order of 3%. Still another basic problem in developing plutonium recycle capabilities will concern licensing. Different licensing criteria for the AEC, Department of Transportation, and IAEA have to be satisfied regarding toxicity, radiation, nuclear considerations, and safeguards. For example the plutonium plant must meet different licensing criteria than a similar uranium facility. Also new licenses will be required for the containers which are needed to ship the fabricated fuel to the reactor site. It would not be suprising if they were intervenor groups which would delay or prevent licensing for the use of plutonium recycle.

The introduction and success of the fast reactor may actually result in short lived LWR plutonium recycle programs as power requirements will be filled by fast reactors which breed their own fuel which may be plutonium if the fertile complement is uranium. Hence, plutonium fuel requirements may be restricted to operating lightwater reactors. Plutonium requirements for recycle will possibly peak around 1990-1995, assuming of course, that breeder reactors are being ordered in the early or mid-1980's. There have been a number of programs sponsored by the AEC and by the Edison Electric Institute to study the characteristics of plutonium needed for recycle. The overall plutonium recycle program which started in 1964 included 4 years of operation and post irradiation examination of Saxton plutonium fuel, two

joint projects with the Edison Institute, and the criticality studies for the Empire State Atomic Development Association. It also included operation of a Westinghouse fuel development laboratory which was completed in 1969. Further data are listed in Reference 40.

CURRENT DESIGN PARAMETERS OF THE VARIOUS CONCEPTS OF NUCLEAR POWER PLANTS

(This section was prepared by Dr. R. A. Karam, Associate Professor of Nuclear Engineering, Georgia Institute of Technology.)

Table 17 summarizes the pertinent design parameters of the PWR, BWR, HTGR, LMFBR, and GCFR. In terms of plant efficiency, the LMFBR and the gas cooled reactors, i.e., the HTGR and GCFR, are superior to the water reactors. The main reason for this is the higher steam-cycle temperatures. Thermal pollution from the LMFBR and gas-cooled reactors is lower than the water reactors, due to better thermal efficiency.

The power density in the the LMFBR is about an order of magnitude larger than the thermal reactors and almost a factor of 2 larger than the GCFR. The equilibrium condition for fission product accumulation is not well established in fast reactors. However, it is safe to say that the conversion of fission products through neutron absorption is significantly lower in fast reactors than it is in thermal reactors.

17. CURRENT DESIGN PARAMETERS OF THE VARIOUS CONCEPTS OF NUCLEAR POWER PLANTS

	PWR	BWR	HTGR	LMFBR	GCFR
A. GENERAL					
PLANT	OCONEE	BROWNS FERRY	PHILADELPHIA ELEC.		GULF GENERAL
MANUFACTURER	BABCOCK & WILCOX	GENERAL ELECTRIC	GULF GENERAL ATOMIC	GENERAL ELECTRIC	ATOMIC
OUTPUT	2584 MW(t) 922 MW(e)	3293 MW(t) 1098 MW(e)	3000 MW(t) 1174 MW(e)	2500 MW(t) 1000 MW(e)	1093 MW(t) 420 MW(e)
EFFICIENCY	34.5%	33.3%	39.1%	40%	38.4%

17 CURRENT DESIGN PARAMETERS OF THE VARIOUS CONCEPTS OF NUCLEAR POWER PLANTS

	PWR	BWR	HTGR	LMFBR	GCFR
B. OPERATING CHARACTERISTICS					
FUEL T	$FUEL_{max}$ 4250°F $CLAD_{max}$ 653°F	$FUEL_{max}$ 4380°F $FUEL_{avg}$ 1100°F	$FUEL_{max}$ 4400°F $FUEL_{avg}$ 1634°F	$FUEL_{max}$ 4340°F $FUEL_{avg}$ 2670°F	
COOLANT T	INLET 554°F OUTLET 604°F	376.1°F 562 °F	INLET 606°F } He OUTLET 1366°F }	INLET 800°F OUTLET 1100°F	INLET 470°F OUTLET 1112°F
PRESSURE	COOLANT 2200 psig	1000 psia operating	710 psig; Δp=10 psig	~ 100 psi Δp=66.5 psig	1000 psia (He) Δp=32.6 psig
STEAM	572°F @ 910 psig	562°F @ 1146	1000°F @ 1450 psi	1000°F @ 3500 psia	~ 1000°F @ ~ 1400 psi

17 CURRENT DESIGN PARAMETERS OF THE VARIOUS CONCEPTS OF NUCLEAR POWER PLANTS

	PWR	BWR	HTGR	LMFBR	GCFR
C. CORE PARAMETERS					
POWER DEN.	84.1 kw/ℓ	50.8 kw/ℓ	8.4 kw/ℓ	~ 500 kw/ℓ	218.6 kw/ℓ
PEAKING FAC	1.011	2.6	1.6	1.94	---
DOPPLER COEFFICIENT	-2.0×10^{-5} to -3.1×10^{-5} $\Delta k/k/°C$	-2.3×10^{-5} $\Delta k/k/°C$	$ -2 \times 10^{-5}/°C	$-1.0 \times 10^{-5}/°C$	$-1.0 \times 10^{-5}/°C$
VOID COEFFICIENT	$+1.8 \times 10^{-4}$ to -5.4×10^{-3} $\Delta k/k/°C$	-1.8×10^{-3} to -2.9×10^{-3} $\Delta k/k/°C$	(- SMALL ?)	+ $ 2.5 CORE ONLY + $ 4.25 MAXIMUM	$+1.8 \times 10^{-6}/°K$
TEMPERATURE COEFFICIENT	FUEL--2.0 to -3.0×10^{-3} % $\Delta k/k/°C$ CLAD--0 to -5.4×10^{-2} % $\Delta k/k/°C$	-9.0×10^{-5} $\Delta k/k/°C$	$-9.30 \times 10^{-5}/°C$ at 300°K $-3.3 \times 10^{-5}/°C$ at 1100°K		
NEUTRON LIFETIME	1.6×10^{-5} sec	~ 1×10^{-4} sec	3.4×10^{-4} sec	~ 5×10^{-6} sec	4.37×10^{-7} sec
DELAYED N FRACTION	.0072	~ .007	~ .007	~ .0035	~ 0.0035

17 CURRENT DESIGN PARAMETERS OF THE VARIOUS CONCEPTS OF NUCLEAR POWER PLANTS

	PWR	BWR	HTGR	LMFBR	GCFR
D. FUEL					
COMP.	UO_2 SINTERED PELLETS	UO_2	U:Th COATED PARTICLES	$PuO_2 + UO_2$ MIXED OXIDES	$PuO_2 + UO_2$ MIXED OXIDES
ENRICHMENT (%)	3 ZONES: 2.05, 2.10, 2.15	2.19%	93%	17.9% (INITIAL)	
PINS	CLAD--ZIRCALOY-4 O.D. 0.430"	CLAD--ZIRCALOY-2 O.D. 0.562"	O.D. 0.619"	SS-316 CLAD O.D. 0.245"	STAINLESS STEEL OR HASTELLOY CLAD O.D. 0.439"
ASSEMBLY	208 RODS [15 x 15 array](less 17 positions for control)	7 x 7 ROD ARRAY	132 RODS	282 SUBASSEMBLY	100 RODS PER BOX (5.2" square)
TOTAL	177 ASSEMBLIES 207, 486 lb UO_2	764 ASSEMBLIES 327, 571 lb UO_2	3486 lb U 82,500 lb Th	121,000 FUEL PINS 4910 lb CORE 192 lb BLANKET	21,300 RODS (100 boxes) FISSILE LOADING 3894 lb

17 CURRENT DESIGN PARAMETERS OF THE VARIOUS CONCEPTS OF NUCLEAR POWER PLANTS

	PWR	BWR	HTGR	LMFBR	GCFR
E. CONTROL					
	5% Cd, 15% In, 80% Ag RODS--SS 340 CLAD	SS CLAD B_4C	INCOLOY 800 CLAD	SCRAM	SS CLAD B_4C
		185 CRUCIFORM RODS 144" LONG	B_4C/GRAPHITE	32 RODS B_4C	29 RODS
	69 ASSEMBLIES		73 ROD PAIRS		
	16 RODS/ASSEMBLY	TEMPORARY CURTAINS	EMERGENCY SHUTDOWN	SHIM	
	POISON LENGTH 134"	STAINLESS STEEL w/5700 ppm B;	B_4C/GRAPHITE	32 RODS B_4C	
	B-10 in H_2O TEMPORARY POISON	356 SHEETS BETWEEN FUEL CHANNELS	73 CANNISTERS		

17 CURRENT DESIGN PARAMETERS OF THE VARIOUS CONCEPTS OF NUCLEAR POWER PLANTS

	PWR	BWR	HTGR	LMFBR	GCFR
F. STRUCTURE	PRESSURE VESSEL SS CLAD CARBON STEEL CYLINDER ID 14.3' h 37.4' DESIGN PRESSURE 2500 psig	PRESSURE VESSEL STAINLESS STEEL CARBON STEEL CYL ID 20.9' h 72.6' DESIGN PRESSURE 1000 psia	PRESTRESSED CONCRETE REACTOR VESSEL (PCRV) ID 37' IH 47.3' OD 100' OH 91.5' DESIGN PRESSURE 765 psig	PRESSURE VESSEL CORE d = 11.5' h = 16" STEEL MAIN Na TANK d = 52' x 1" THICK h = 47' DESIGN PRESSURE 10 psig (NO TANK)	d_{core} 7.65' $d_{blanket}$ 10.7' L/D RATIO 0.5 PRESTRESSED CONCRETE CYLINDER w/FLAT ENDS LINED WITH STEEL

17 CURRENT DESIGN PARAMETERS OF THE VARIOUS CONCEPTS OF NUCLEAR POWER PLANTS

	PWR	BWR	HTGR	LMFBR	GCFR
G. CONTAINMENT					
	PRESTRESSED CONCRETE CYLINDER ID 116' h 208.5'	REINFORCED CONCRETE SS LINED			
	DESIGN PRESSURE	DESIGN PRESSURE	DESIGN PRESSURE	DESIGN PRESSURE	DESIGN PRESSURE
	58 psig	62 psig		----	----

17 CURRENT DESIGN PARAMETERS OF THE VARIOUS CONCEPTS OF NUCLEAR POWER PLANTS

	PWR	BWR	HTGR	LMFBR	GCFR
REFERENCES FOR DATA IN THIS SECTION	OCONEE	BROWNS FERRY	GGA DESIGN	G.E. DESIGN	G.E. DESIGN
A-CE	"Nuc. Engr. Int'l" Apr 70 15:337-344 World's Reactors #50	Nuclear Energy Conversion, M. M. Wahil Intext Educational Publs. 1971, p. 114 (table)	HTGR Fact Sheet Gulf Oil Corp. 1973	Argonne National Laboratory ANL 7120 Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors October 11-14, 1965	
	USAEC DOCKET 50269-1 "Preliminary Safety Analysis Report" Duke Power Co. 1 Dec '66	USAEC DOCKET 50259-1 "Design & Analysis Report" (TVA) 7 July '66		p. 185 1000 MW(e) Fast Sodium Cooled Reactor Design Cohen & O'Neill General Electric	p. 230 Safety Characteristics of Large Gas Cooled Fast Power Reactors Fortesque et al. General Atomic
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