APTD-1077

Survey #2

PB-209 283

Survey of Nuclear Power Supply Prospects

Hittman Associates, Inc.

February 1972

Distributed By:

NTIS

National Technical Information Service
U. S. DEPARTMENT OF COMMERCE
5285 Port Royal Road, Springfield Va. 22151

BIBLIOGRAPHIC DATA SHEET	1. Report No. APTD-1077	2.	3. Recipient's Accession No. PB 209-283
4. Title and Subtitle Survey of Nuclear Power Supply Prospects			5. Report Date February 1972 6.
7. Author(s)			8. Performing Organization Rep. No. HIT-501
9. Performing Organization Name and Address HITTMAN ASSOCIATES, INC. COLUMBIA, MARYLAND 21045			10. Project/Task/Work Unit No. 11. Contract/Grant No. EHSD-71-43
12. Sponsoring Organization Name and Address ENVIRONMENTAL PROTECTION AGENCY Office of Air Programs			13. Type of Report & Period Covered 14.

15. Supplementary Notes

16. Abstracts A detailed review of the nuclear segment of the power industry and an evaluation of the ways in which nuclear power will impact on national air quality and emissions from fossil-fueled steam electric plants are presented. Electrical power from nuclear-fueled reactors will continue to grow at a substantial rate and will ultimately supply a major portion of electrical power in the United States. The development and use of nuclear plants will directly affect the use of fossil fuels to supply electrical power and will also directly affect the amounts of combustion products released to the environment. Forecasts of the proportion of nuclear power, and thus of fossil-fueled power, vary widely although the majority of forecasts are in reasonable agreement through about 2020. The forecast of future power supplied by nuclear reactors has been shown by optimization analysis to be strongly dependent upon the reactor types and their technological development. Depending on the placement in service of various reactor systems, the emissions of sulfur dioxide and other pollutants to the atmosphere may vary by a factor of 10 within the next 80 years, independent of the allowable limits of emissions or fuel contents. Fossil-fueled generating plants will continue to produce power at levels equal to or greater than 1970 levels for the next 80 years. Therefore, emission reductions can only be achieved through control devices or fuel modifications to remove undesirable emissions from effluent gases. Light water reactors will provide the bulk of nuclear-fueled power through the next 15 years, although some use of advanced converter nuclear systems is projected, notably in the High Temperature Gas-Cooled Reactor System. Breeder reactor technology will develop sufficiently in the next decade to enable utilities to place breeder reactors in commercial service by the middle of the next decade. Fusion reactor systems are currently under development but are not expected to become commercially feasible for the production of electrical power before the first decade of the next century.

17. Key Words and Document Analysis. 17a. Descriptors

Air pollution
Nuclear power plants
Nuclear reactors
Fuels
Combustion products
Sulfur dioxide

17b. Identifiers/Open-Ended Terms

17c. COSATI Field/Group 13B

18. Availability Statement	19. Security Class (This Report) UNCLASSIFIED	21. [No. of Pa]ges
Unlimited	20. Security Class (This Page) UNCLASSI[FIED]	22. Price

FORM NTIS-35 (10-70)

SURVEY OF
NUCLEAR POWER SUPPLY
PROSPECTS

HIT-501

February 1972

Prepared Under
Contract No. EHSD 71-43
Environmental Protection Agency
Office of Air Programs

HITTMAN ASSOCIATES, INC.
COLUMBIA, MARYLAND

LEGAL NOTICE

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES (Continued)

I. INTRODUCTION

Following the successful controlled fission reaction by Enrico Fermi and his co-workers at the University of Chicago on December 2, 1942, the immediate applications of nuclear power were devoted to the existing war effort. Following World War II, scientific attention was directed toward the application of nuclear energy to electrical power generation. This development program led to the installation of the first commercially-operated nuclear plant at Shippingport, Pennsylvania. This 90 megawatt plant, placed on line in 1957, is owned by the Atomic Energy Commission and operated by the Duquesne Light Company. Since that time, many more and larger nuclear plants have been placed in service and are successfully producing ever-increasing percentages of national power demands. At present, over 110 nuclear fueled electric generating units are in operation, under construction or ordered.

Most of the nuclear plants presently planned for operation are of the light water cooled and moderated variety. Long-term operating experience with such plants as Shippingport (1957; 90 megawatts), Dresden 1 (1959; 200 megawatts), and Yankee Rowe (1962; 175 megawatts) has adequately demonstrated the reliability and practicality of light water reactor systems. In 1967, a high temperature gas-cooled reactor at Peach Bottom, Pennsylvania was placed in commercial operation, and has demonstrated the practicality of this concept.

Overall, nuclear power reactors are presently delivering almost three percent of national electrical supply, nearly 10,000 megawatts from the modest beginnings of 90 megawatts in 1957. Nuclear power has made impressive inroads and contributions to the problems of supplying reliable and pollution-free electrical power in the United States.

Nuclear energy systems are experiencing steady growth in sophistication and in technological advancements and improvements. Current on-going efforts are being directed toward developments in all aspects of nuclear power systems, including future power requirements, economics, resource conservation, reactor technology, safety, and long-range planning. As a result of the Atomic Energy Acts of 1946 and 1954, the Atomic Energy Commission has the major responsibility for regulating and promoting the use and development of nuclear energy to meet the expanding needs for safe and reliable electric power. In cooperation with the utilities and other involved organizations in the public and private sectors, the Atomic Energy Commission is presently committed to the development of new reactor technology, with primary priority focussed on the Liquid Metal Cooled Fast Breeder Reactor (LMFBR). Simultaneously ongoing projects include the advanced convertor reactors such as the High Temperature Gas-Cooled Reactor (HTGR) and the Fusion Reactor.

Where foreign research efforts, notably these of Canada in the areas of heavy-water-moderated organic cooled reactors and BLW reactors, have coincided with those of the United States or have been performed as part of a joint technical effort with the United States, these have been described briefly in this report.

The advanced convertor reactor technology has advanced to a point where a demonstration plant (Peach Bottom, 40 Mwe) has been operating for several years, and a commercial-sized plant (Fort St. Vrain, 330 Mwe) will be placed on line in 1972. From all indications, breeder reactors will be built and ready for commercial operation about 1985. Potentially, the breeder reactors, which will produce more fissionable material than is consumed and which will operate at thermal efficiencies equal to or greater than comparable fossil plants, will provide the utility industry with an economically and technologically attractive means of producing electric power. Breeders will in all likelihood become a primary source of steam for power generation, outstripping present fission reactors and fossil plants in new plant construction once the breeder is sufficiently developed for commercial application. Theoretically, a fusion reactor, which will utilize the principle of fusing light elements into heavier elements with a release of energy, is a possibility for future power production requirements. However, the necessary technology has not yet been developed. It is unlikely that fusion power reactors will be commercially available until after the year 2000.

The report following was prepared for the Environmental Protection Agency, Office of Air Programs for the purpose of forecasting the nation's future nuclear power supply prospects. Nuclear power is expected to gain an increasingly larger share of the future power supply market because of its more attractive economic conservation and pollution picture as compared to fossil fuel power. Sections I-IV of the following report detail the reactor types now in operation, as well as those under development, and indicate the existing and planned nuclear generating status of each state through 1980. The final sections of this report evaluate several previous nuclear power supply forecasts and formulate a forecast based on this previous work as well as data collected for this report. The reactor types described in Sections I-IV are evaluated in terms of energy projections and cost, fuel availability, safety considerations and advantages and disadvantages. The beneficial effect of nuclear power on air quality is also detailed.

II. SUMMARY AND CONCLUSIONS

For a variety of reasons, among them rising fossil fuel costs, increased costs of air and water pollution control measures, and economies of scale, the utility industry has been turning to nuclear powered steam plants to provide base loading power supplies, particularly when larger sized plants are involved. Of all new plants ordered for construction over the next several years, approximately 40 percent are nuclear fueled (Ref. II-1). It is expected that nuclear plants will supply 19 percent of electrical power in 1980, 35 percent in 1990, and over 45 percent in 2000.

At present, the primary reactor sales for nuclear power generation are the light water types, including the boiling water reactor and the pressurized water reactor. In addition, high temperature gas-cooled reactors have also been purchased for commercial operations. These reactor systems will provide basic nuclear steam supplies to the utility industry through about 1985, when it is anticipated that breeder reactors will become commercially available for power generation. Thereafter, continuing developmental work will ultimately provide the necessary technology for commercial application of a fusion reactor system, although it is unlikely that a working fusion reactor steam supply can be in service until sometime after the turn of the century.

From various forecasts of the quantity and distribution of steam supply systems for electrical power generation, a single forecast of the utility industry has been formulated (Ref. II-2) extending through the next 80 years. Based on this information, a prediction of electrical generating capacity through 2050 by fuel types has been derived (Ref. II-1) and is shown in Table II-1.

TABLE II-1. PROJECTED CUMULATIVE POWER CAPACITY BY FUEL (Mwe x 10^3)

Fuel	1970	1980	1990	2000	2010	2020	2030	2040	2050
Coal	176	280	412	566	640	615	595	510	416
Oil	22	34	49	60	76	58	52	45	40
Gas	61	84	108	119	115	103	93	85	80
Hydro	51	62	69	89	90	95	100	102	104
Nuclear	10	106	343	671	1080	1825	2660	3458	4560
Total	320	570	980	1500	2000	2700	3500	4200	5200

It may be noted from Table II-1 that nuclear power plants are expected to assume the major share of power production through the forecast period. Nevertheless, fossil fuels will continue to grow in usage through about 2010, when they are expected to commence to decline. Fossil fuels will, over the next 30 to 40 years, provide a major portion of power supply but will decrease in percentage from over 80 percent in 1970 to 70 percent in 1980, 58 percent in 1990, and 50 percent in 2000. After 2000, total fossil fuel usage for electrical power generation will decline on both percentage and actual bases. Figures II-1 and II-2 illustrate these forecasted changes. Figure II-2 particularly illustrates the continued expected growth of the use of fossil fuels in generating electrical power and indicates that a four-fold increase in fossil fuel combustion will cause a continuing rise in combustion products to the environment through about 2010, when a gradual decline is forecast to begin. Because of the forecasted growth in the use of fossil fuels, and despite the overwhelming forecasted growth of nuclear plants, there is no time within the foreseeable future when fossil fuel use for power generation will be any less than it is today and, in fact, after a 40-year decline beginning in 2010, will only reach 1985 levels. Thus, it may be expected that, even with the very large trend to nuclear fuels, fossil plants either in existence or yet to be built will contribute increasing amounts of combustion waste products to the environment through the middle of the next century at levels equal to or greater than at present, barring any major technological advances to prevent this occurrence.

This report includes a detailed review of the nuclear segment of the power industry and an evaluation of the ways in which nuclear power will impact on national air quality and emissions from fossil-fueled steam electric plants. are [illegible]. The following is a list of primary conclusions drawn from this report:

- Electrical power from nuclear-fueled reactors will continue to grow at a substantial rate and will ultimately supply a major portion of electrical power in the United States.
- The development and use of nuclear plants will directly affect the use of fossil fuels to supply electrical power and will also directly affect the amounts of combustion products released to the environment.

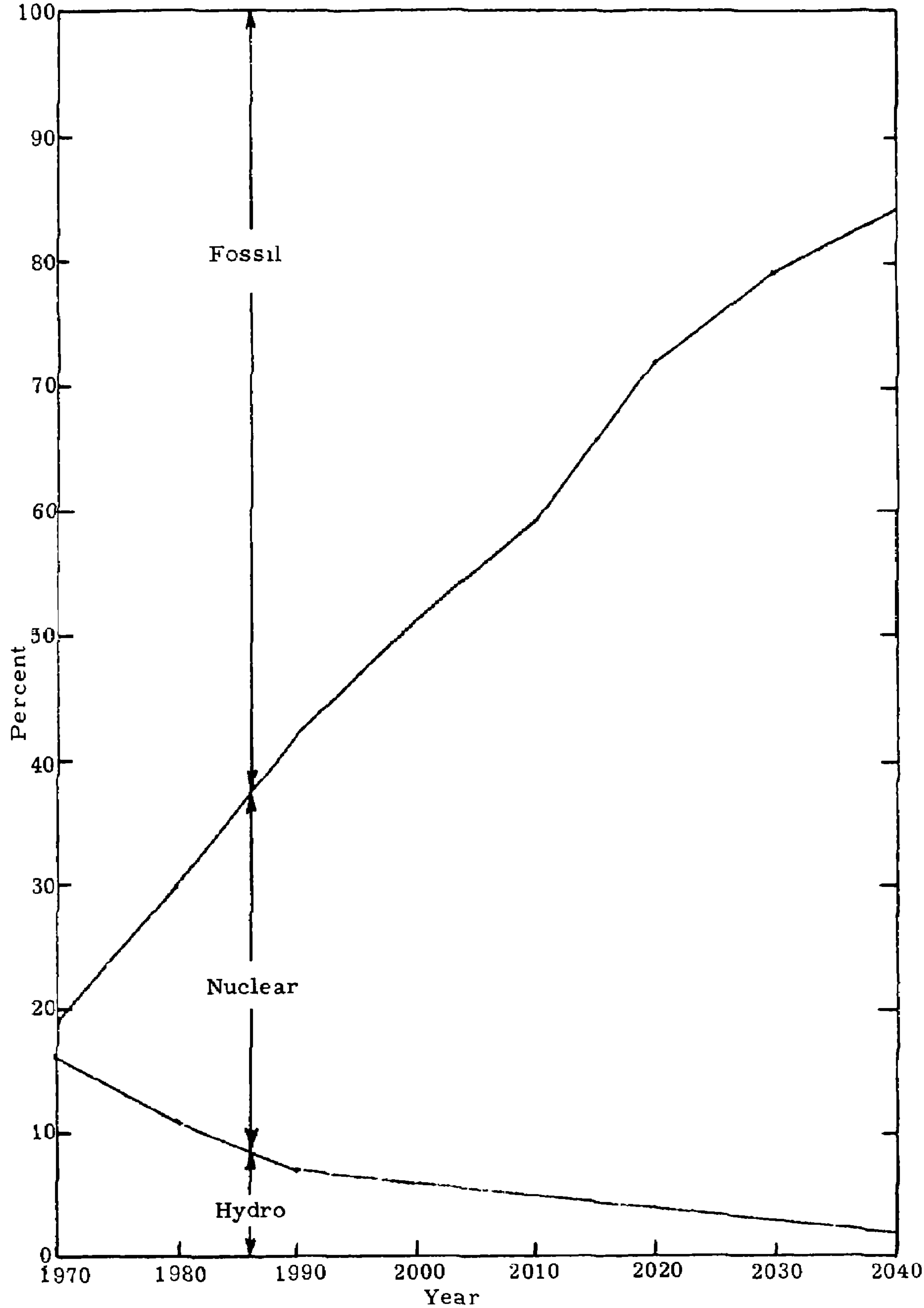


Figure II-1. Percent Fuel Distribution

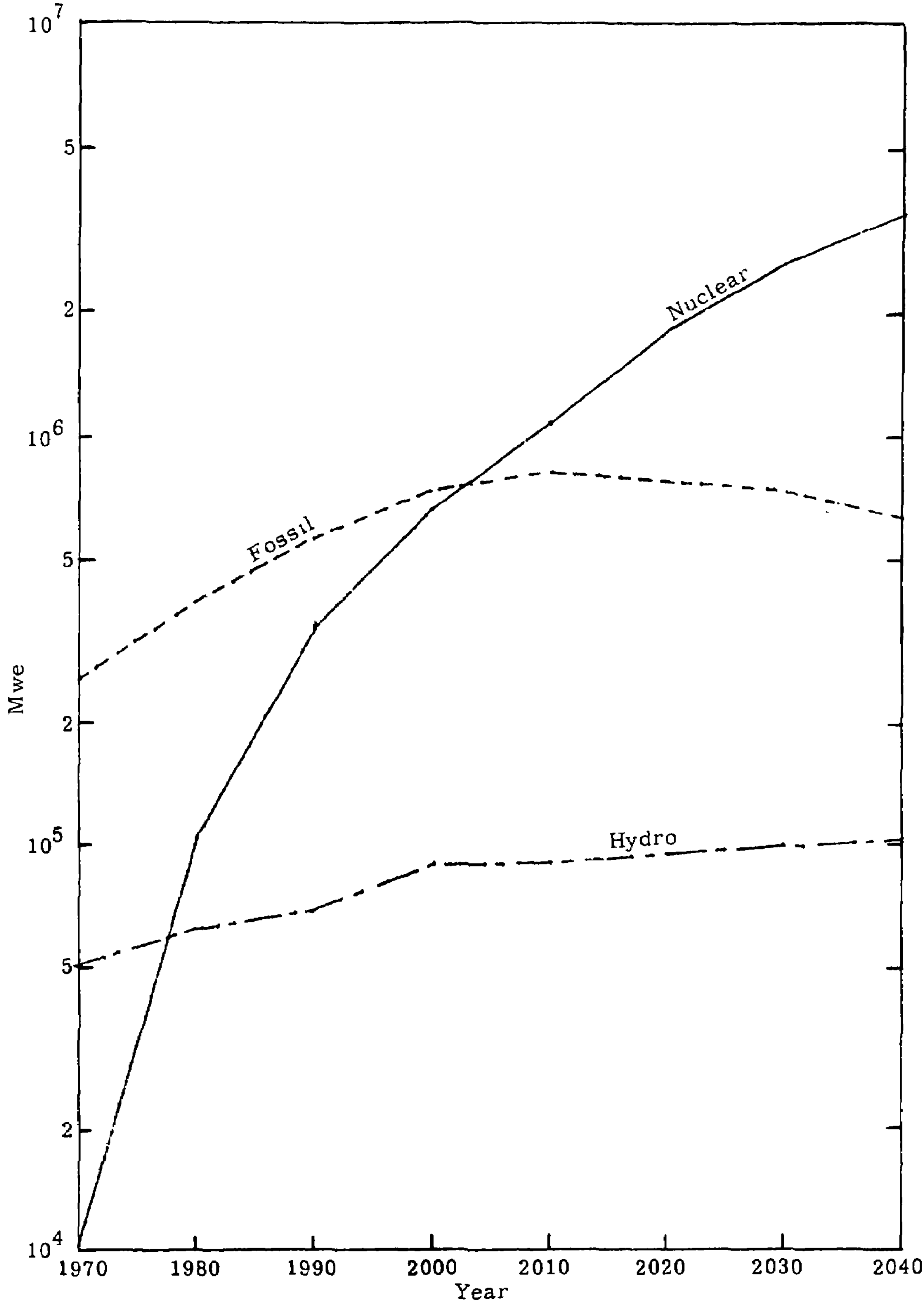


Figure II-2. Actual Fuel Distribution

- Forecasts of the proportion of nuclear power, and thus of fossil-fueled power, vary widely although the majority of forecasts are in reasonable agreement through about 2020.

- The forecast of future power supplied by nuclear reactors has been shown by optimization analysis to be strongly dependent upon the reactor types and their technological development. Thus, several differing nuclear power projections result from differences in the expectations for timely deployment of light water reactors, advanced converters, thorium systems, breeder reactors, and combinations of these with fossil-fueled plants.

- Depending on the placement in service of various reactor systems, the emissions of sulfur dioxide and other pollutants to the atmosphere may vary by a factor of 10 within the next 80 years, independent of the allowable limits of emissions or fuel contents. This is directly proportional to the percentage of total electrical power produced by nuclear and fossil steam plants.

- Fossil-fueled generating plants will continue to produce power at levels equal to or greater than 1970 levels for the next 80 years. Therefore, emission reductions can only be achieved through control devices or fuel modifications to remove undesirable emissions from effluent gases.

- Light water reactors will provide the bulk of nuclear-fueled power through the next 15 years, although some use of advanced converter nuclear systems is projected, notably in the High-Temperature Gas-Cooled Reactor System.

- Breeder reactor technology will develop sufficiently in the next decade to enable utilities to place breeder reactors in commercial service by the middle of the next decade. Breeder reactors will comprise the largest majority of nuclear systems constructed from 1985 through the beginning of the next century.

- Fusion reactor systems are currently under development but are not expected to become commercially feasible for the production of electrical power before the first decade of the next century.

REFERENCES

II-1. A Review and Comparison of Selected United States Energy Forecasts, Pacific Northwest Laboratories, Battelle Memorial Institute, Columbus, Ohio, December 1969.

II-2. Bibliography and Digest of U.S. Electric and Total Energy Forecasts, 1970-2050, Publication 69-23, Edison Electric Institute, New York, 1969.

III. BACKGROUND AND DEFINITION

A. Fission Power

The power reactors discussed in this section are classified into four categories:

(1) Light water reactors

(2) Advanced converters

(3) Thorium systems, and

(4) Breeder reactors

In total, there are approximately 10 of these reactor types as shown in Table III-1. This classification follows that of the Systems Analyses Task Force organized by the AEC Division of Reactor Development and Technology to evaluate the various reactor types. These basic types may be further subdivided as seen in the text to follow.

1. Light Water Reactors

This category of reactors includes the boiling water reactor (BWR) and the pressurized water reactor (PWR). As can be seen from Table III-2 (Ref. III-1), the performance characteristics for the BWR and PWR are similar. The basic source of these performance data is Reference III-2.

Boiling water reactors are currently manufactured by the General Electric Company. They are sold as steam supply systems in a range of sizes from 500 to 1100 Mwe.

The first large commercial application of the BWR concept was the Dresden I reactor located near Morris, Illinois. This reactor plant was initially rated at 180 Mwe and is currently operating at 200 Mwe. It employed the dual cycle concept with an elevated steam drum and a recirculation system contained in a spherical steel shell containment structure. Through March 1968 the plant was operated at an average availability factor of approximately 90 percent. A recent BWR reactor design (1075 Mwe net capacity) has the performance parameters shown in Table III-2.

TABLE III-1. TYPES OF NUCLEAR POWER REACTORS

Classification	Reactor Type	Assigned Acronym
1. Light water reactor (LWR)	Boiling water reactor	BWR
	Pressurized water reactor	PWR
2. Advanced converters	Heavy water-moderated organic-cooled reactor	HWOCR
	Heavy water-moderated boiling-light water-cooled reactor	HWBLW
	High temperature gas-cooled reactor	HTGR
3. Thorium systems	Molten salt reactors:	MSR
	Molten salt converter	MSCR
	Molten salt breeder	MSBR
4. Breeder reactors	Liquid-metal-cooled fast breeder reactor	LMFBR
	Steam-cooled fast breeder reactor	SCFR
	Gas-cooled fast reactor	GCFR

Table III-2 SUMMARY OF DESIGN AND PERFORMANCE PARAMETERS FOR LIGHT WATER REACTORS

	Reference BWR	Reference PWR	1980-1990 LWR	1990-on LWR
Core Region:				
Active height, ft.	12.0	12.0		
Diameter, ft.	15.6	11.1		
Number of control rods	185	53		
Number of fuel elements	764	193		
Fuel pin diameter, inches	.562	.422		
Number of fuel pins per element	49	204		
Power, MWt	3293	3083		
Average specific power, MWt/MT	22.0	34.8	39.9	44.8
Peak linear rod power, kW/ft	18.3	18.0		
Fuel average discharge exposure, MWD/MT	27,500	30,000	30,000*	30,000*
Hot-spot cladding temperature, °F	565	657		
Maximum fuel temperature, °F	4380	4100		
Maximum heat flux, B/hr-ft^2	425,000	553,700		
Minimum critical heat flux ratio	1.90	1.86		
Core subcooling, Btu/lb	25.5	147.8		
Reactor pressure, Psi	1050	2235		
Core inlet/outlet temperature, °F	376.1/546.4	545/610		
Average core power density, Kw/l	50.8	93.1		
Average core exit quality, w/o	13.6			
H/U atom ratio, average, full power	3.74	4.23		
Specific inventory, Kg fissile/MWe	3.0	2.2	1.72	1.41
Pu Yield, Kg fissile/MWe-yr, at avg. exposure	.22	.23	.23	.23
U_3O_8 consumption, Kg/MWe-yr	188	200	200	200
Thermal to electrical conversion efficiency, %	32.8	32.5	32.5	32.5

Reference: WASH 1082, "Current Status and Future Tecnnical and Economic Potential of Light Water Reactors," March 1968.

*30,000 MWD/MT data shown here for comparison of performance with reference designs but actual exposures used in calculations varied.

Improvements were made in the design parameters of boiling water reactors increasing the operating efficiency and reducing the cost of generated power.

(a) More power delivered per unit weight of loaded uranium, or greater specific power

(b) More power delivered per unit core volume, or greater power density

(c) Improved exit steam quality

(d) Increased exposure of fuel in terms of integrated thermal energy delivered per unit weight of fuel (MWD/MT)

These improvements result principally from improved optimization of core power distributions, design innovations, improvements in anticipated heat transfer characteristics and application of internal steam separation.

The BWR steam supply system includes the reactor vessel and its internal components and all of the primary and auxiliary circulating equipment such that the system can provide input steam to the turbine-generator for producing power. Components within the reactor vessel include the core, control rod drive systems, and jet pumps. The core consists of the fuel assemblies, channels, control blades, and instrumentation to monitor its status at various stages of operation. Each fuel assembly consists of a 7 x 7 array of fuel rods enclosed in a Zircaloy channel. The fuel rods consist of slightly enriched uranium dioxide pellets contained in a Zircaloy tube cladding. The cruciform-shaped control rods occupy alternate locations between fuel assemblies and are withdrawn into the guide tubes of the core during operation in accordance with a predetermined pattern.

Pressurized water reactors are currently manufactured by the Westinghouse Electric Company, Babcock & Wilcox Company and Combustion Engineering, Inc. The design approach employed by these companies is quite similar. The Babcock & Wilcox Company concept employs a once-through steam generator capable of producing steam having a limited degree of superheat with an associated heat rate improvement. Otherwise, the technical approach employed by all pressurized water reactor manufacturers is similar with respect to design pressure, use of chemical shim control, and fuel design.

The first pressurized water cooled reactor to operate at a sufficiently high reactor temperature for power generation was the submarine thermal reactor STR Mark I (now known as SIW), the prototype PWR used in the submarine "Nautilus". The favorable experience with STR led to the full scale central-station nuclear power project of the Shippingport reactor, which began operation in 1957. The Shippingport reactor (90 Mwe) pioneered the use of Zircaloy clad and uranium dioxide (UO_2) as a reactor fuel material. These steps represented two of the more important advances made to date in the water cooled reactor field.

The Yankee reactor plant, rated at 175 Mwe and located at Rowe, Massachusetts, represents the first commercial application of a pressurized water plant in the Dresden I size range. The Yankee plant has operated successfully for eleven years and as of March 1967 was operating with an availability factor of approximately 90 percent.

While identical in principle, the current pressurized water reactor concepts have substantially improved performance characteristics when compared with the Yankee initial design. These include important advances in core design performance characteristics, incorporation of chemical shim and rod cluster control, and optimization of fuel management programs. The significant design parameters affecting both capital and fuel cycle costs are similar to those previously listed for boiling water reactor plants, i.e.:

(a) More power delivered per unit weight of loaded uranium, or greater specific power

(b) More power delivered per unit core volume, or greater power density

(c) Increased exposure of fuel in terms of integrated thermal energy delivered per unit weight of fuel (MWD/MT)

These improvements in design characteristics result principally from the employment of chemical shim and rod cluster control, improved fuel management techniques and heat transfer correlations. A recent PWR reactor design (1035 Mwe net capacity) has the performance parameters shown in Table III-2 for comparison purposes.

PWR nuclear power plants each incorporate a closed-cycle pressurized water nuclear steam supply system and a turbine generator system utilizing dry and saturated steam. The nuclear steam supply system consists of the reactor vessel and its internal components, two or more closed reactor coolent loops (depending on station size and design) connected in parallel to the reactor vessel, an electrically heated pressurizer and all necessary auxiliary systems. The vessel is cylindrical with a hemispherical bottomed head and a flanged and removable upper head. The reactor core is divided into three concentric regions. All fuel assemblies are mechanically identical, but the fuel enrichment for the initial core is different in each region. The inner region has the lowest enrichment in fissionable uranium-235 and the outer region has the highest enrichment. This variation results in more uniform power distribution throughout the core. A scatter pattern is employed in the arrangement of the inner region. The fuel rods consist of pellets of slightly enriched UO_2 assembled in cold-worked Zircaloy tubes which are welded closed at the ends.

Each control rod cluster assembly consists of cylindrical absorber rods fastened together by a spider-type bracket at the top of the cluster. The control rod clusters provide reactivity control with an adequate safety margin. The relatively homogeneous distribution of absorber material tends to provide a more uniform power distribution than the cruciform control used in earlier PWR's. This system also provides more reactivity control per unit weight than the cruciform control rods because of the larger surface to volume ratio.

Performance characteristics for future LWR's are projected and presented in Table III-2. These characteristics are for a future PWR which was considered to adequately represent LWR status in the analyses to determine future nuclear energy projections (Section V. B).

2. Advanced Converters

The first attempt to evaluate the future role of advanced converter reactors in the U.S. civilian power program was initiated at the Oak Ridge National Laboratory near the end of 1963. Designs generated by sponsors of various 1000-Mwe concepts were evaluated. These included the spectral shift control reactor (SSCR), pressurized heavy water-cooled reactors (HWR-U and HWR-Th), the sodium graphite reactor (SGR), the high temperature gas-cooled reactor (HTGR), and the molten salt converter reactor (MSCR).

A pressurized light water reactor (PWR) was also considered for comparative purposes. The results of these studies indicated that only two of the advanced converters considered, the HTGR and the MSCR, had the potential of both lower power costs and better fuel utilization than the PWR. The remaining systems were encumbered either with high power costs (SSCR and HWR-Th) or poor fuel utilization (SGR and HWR-U). On the basis of these early evaluations, it appeared that there was insufficient incentive for continuing the development of heavy-water-moderated reactors in the U.S. Canadian studies show that reductions in power costs of HWR's can be obtained by using organic or boiling-light-water coolants instead of pressurized heavy water. To verify this, an evaluation of heavy water-moderated organic-cooled reactors was undertaken at ORNL in the spring of 1965.

Three concepts were compared: slightly enriched UC-fueled, thorium oxide-fueled, and thorium metal-fueled systems. The results of the evaluation verified that under the assumed ground rules the HWOCR concepts had lower power costs and better fuel utilization characteristics than the pressurized heavy water-cooled reactors and the PWR considered previously. Thus, by January 1967, three advanced converter concepts--the HTGR, the HWOCR, and the MSCR--were identified as having better costs and fuel utilization than light water reactors. The role of these particular advanced converters in the future U.S. power system still remained unclear, however, because the evaluations conducted up to that time were limited to a comparison of converter reactors with each other, rather than with all reactors that might comprise the future U.S. power system. The evaluations also did not take into consideration the effect of the introduction of these converters on the economics of the system as a whole, the competitiveness of converters with breeders, and the influence of rising ore prices on the competitive position of converters relative to light water reactors and breeders.

Based on the degree of continued success of current light water systems, as well as the availability of low-cost uranium ore and the progress of breeder reactor development which is being assessed as part of the Commission's ongoing civilian power analysis program, it was determined early in 1967 that sufficient data had been developed on heavy water reactor alternatives to permit readjustment of the program. On this basis, the developmental work on the heavy water organic-cooled concept has been deferred until the role of this class of reactors is further clarified. In the meantime, a heavy water reactor search and development program involving a modest expenditure of U.S. resources is underway. This program, coupled with active international technical liaison, provides an economical means for the U.S. to maintain the option to exploit the heavy-water reactors in the future.

HWOCR's are of interest when considering the engineering characteristics of advanced converters as shown in Table III-3 based on Reference III-3. Five types of heavy water-moderated organic-cooled reactors (HWOCR) exist due to fueling type variations:

(a) Slightly enriched uranium carbide

(b) Natural uranium carbide

(c) Natural uranium metal

(d) Thorium-uranium oxide, and

(e) Thorium-uranium metal

Also shown are a heavy water-moderated boiling-light water-cooled reactor (HWBLW) and two high temperature gas-cooled concepts (HTGR backup and reference designs). The molten salt reactors are not included as they are classified under Thorium Systems.

All five HWOCR concepts have vertical pressure tubes (calandria) and bidirectional on-line refueling, bidirectional coolant flow, carbon steel primary loops, recovery facilities for decomposed organic coolant, and primary heat transfer components located outside the containment building. Superheated steam is generated in all cases by the hot Santowax OM coolant (primarily a mixture of ortho- and metaterphenyls). However, steam temperatures and pressures vary somewhat, depending on the type of fuel element used. Principal differences of the natural uranium-fueled concepts compared with the slightly enriched UC-fueled reactor are:

(a) Twice as many fuel channels for the enriched UC fueled reactor because the reactivity required a larger core, higher D_2O inventory, and larger fuel element diameter, and

Table III-3 Engineering Characteristics of Advanced Converters

	HWOCR					HWBLW Reactor	HTGR	
	EUC	NUC	NUM	TUO	TUM		Backup	Reference
Fuel type	Uranium carbide	Uranium carbide	Uranium metal	$(Th-U)O_2$	Thorium-uranium	UO_2	$(Th-U)O_2$ or $(Th-U)C$	$(Th-U)O_2$ or $(Th-U)C$
Fuel configuration	37-pin assemblies	19-pin assemblies	3 nested cylinders	37-pin cluster	4 nested cylinders	19-pin assemblies	Coated particles	Coated particles
Cladding material	SAP	SAP	Ozhennite 0.5	SAP	Ozhennite 0.5	Zircaloy-4	None	None
Moderator	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	Graphite	Graphite
Coolant	Santowax OM plus high boilers					Boiling H_2O	Helium	Helium
Thermal capacity, Mw	3093	3268	3166	3100	3187	3242	2460	2320
Net electrical capacity, Mw	1067	1118	1070	1076	1048	965	1001	1000
Net station efficiency, %	34.5	34.2	33.8	34.7	32.9	29.8	40.7	43.1
Steam conditions, psig/°F	900/725	900/725	900/685	900/740	600/660	765/514	2400/1000	3500/1050
Coolant temperature, inlet/outlet, °F	590/745	580/750	569/710	605/766	535/685	513/518	757/1449	802/1525
Coolant pressure, psi	285	360	320	280	353	1080	690	695
Coolant flow rate, lb/hr	110×10^6	108×10^6	128×10^6	106×10^6	119×10^6	49.8×10^6	10.3×10^6	9.3×10^6
Average core power density, kw/liter	16.1	8.1	5.6	12.3	19.6	9.1	7.5	7.0
Average core specific power, kw/kg of fuel	24.8	13.6	7.6	26.4	32.2	13.3	60	60
Maximum linear heat rating, kw/ft	23.4	18.4	123	16.5	261	18.7		

(b) The use of zirconium alloys for process tubes for the natural uranium concepts sintered aluminum product (SAP)

Both UC concepts have SAP-clad fuel pins. However, the uranium metal fuel element consists of three nested cylinders clad with a zirconium alloy (Ozhennite 0.5). The reference HWOCR is considered to be fueled with slightly enriched uranium carbide. However, for the purpose of system analysis studies, a throwaway fuel cycle core was also proposed for this concept.

One thorium-fueled HWOCR concept has 37-pin cluster fuel elements containing ThO_2-UO_2, and the other has fuel elements consisting of four nested cylinders of zirconium-alloy-clad thorium-uranium metal. The conceptual design of a 37-pin cluster ThC-UC-fueled reactor was also prepared by Babcock & Wilcox, but was not included in the ORNL evaluation because its economic performance was no better than that of the other systems, and the use of this fuel involved more technical problems. The overall plant design for the thorium-fueled concepts is basically the same as that for the Atomic International-Combustion Engineering uranium-fueled reactors, except that the core design was changed to accommodate the thorium fuels.

The design of the 1000 Mwe HWBLW reactor represents an extrapolation from the Canadian CANDU-BLW-250 Mwe reactor, which is smaller in overall size but in many other important respects is essentially the same as the 1000 Mwe reactor. The BLW-250 reactor is being designed and built by Atomic Energy of Canada, Ltd. (AECL), and much of the research and development has been a part of the USAEC-AECL Cooperative Program for heavy-water reactors. The design features a vertical pressure-tube calandria-type reactor cooled with boiling light water. Coolant enters at the bottom of the reactor at about 1000 psi and exits at the top as 30 percent quality steam. The steam, after separation, goes directly to the turbine. Fuel assemblies consist of 19-rod Zircaloy-4-clad natural UO_2 pellets. Five assemblies, each 1.5 m long, are stacked in each of 688 pressure tubes. Although the reference HWBLW reactor was considered to be fueled with natural UO_2, enrichment with plutonium was also included in the data provided for analytical studies.

The HTGR backup design is essentially a scaleup of the 330-MWe Fort St. Vrain plant. This concept consists of a graphite-moderated helium-cooled reactor operating on the thorium-U-233 fuel cycle. The graphite serves as the moderator, reflector, and fuel-bearing structure, and the fuel is in the form of carbon-coated particles of UO_2 + ThO_2 or UC_2 + ThC_2. The core and complete helium circulating system are housed in a prestressed-concrete reactor vessel.

The HTGR reference design includes the following improvements over the backup design·

(a) On-line refueling

(b) A prestressed concrete pressure vessel

(c) A radial-flow steam generator, and

(d) Supercritical steam conditions

In both the backup and reference HTGR's, the initial operation of the reactor is normally with uranium highly enriched in U-235 as fissile material. Bred uranium is recovered from the spent fuel and is recycled along with enough fully enriched uranium to maintain criticality. In the typical equilibrium cycle, the fuel elements have a mixture of two types of coated particles. makeup particles containing only highly enriched U-235 and fertile particles containing the thorium and all the recycled uranium. This permits separation of the bred material from makeup material.

3. Thorium Systems

There is an overlap between the advanced converter and thorium systems classifications. The HWOCR and HTGR advanced converters that utilize thorium fuel types can also be classed as thorium systems. However, this section will consider only the remaining types, molten salt reactors, in the following discussion. Power reactor types employing thorium are discussed to considerable detail in Reference III-4.

Molten salt technology has been studied extensively at ORNL since 1950. There have been two molten salt reactors--the Aircraft Reactor Experiment in 1954 and the Molten Salt Reactor Experiment (MSRE)--as well as a broad base of related applied research in this concept and other fluid-fuel reactors. These experimental reactors provided a varied background of experience in complete circuits of circulating fuel, including reactor kinetics response, pumping of fluid fuels, heat removal, and remote maintenance. Since it achieved critically in June 1965, the MSRE operated successfully until placed on standby in December 1969, mostly at a power level of 8.0 Mwt. This operation has served to demonstrate the following important design features of the experiment-sized single-region molten salt concept:

(a) The practicality of high temperature (1200 F) operation of a molten salt fuel

(b) The sustained performance of basic system components, such as pumps, heat exchangers, and instrumentation, with molten salt fuel

(c) Satisfactory performance of remote maintenance

(d) Removal of xenon and other volatile fission products from the molten salt

(e) On-line refueling and fuel adjustment, and

(f) Self-regulation and good response to changes in power demand

Preliminary reactor designs, including the 1000 Mwe MSBR as well as an advanced converter, are currently under investigation. Program plans include:

(a) Demonstration of dimensional and structural stability of graphite during long exposure to fast-neutrons

(b) Establishment of long term compatibility of Hastelloy N in the molten salt and neutron environment

(c) Development of remote maintenance equipment

(d) Removal of fission products and Pa-233 from molten salts during reactor operation

(e) Scale-up of system components, especially the pumps and heat exchangers

As in all reactor development programs, there is a difficult transition from an experimental facility such as the MSRE to a large scale commercial plant such as the MSBR. This concept has not yet received significant industrial or utility support, and major R&D efforts will be required to develop the concept commercially.

Previously, the reference design for the development of the MSBR has been the ORNL two-region, two-fluid system with fuel salt separated from the blanket salt by graphite tubes. The fluids consisted of lithium and beryllium fluorides containing UF_4 and ThF_4 for the fuel and blankets materials, respectively. The on-site fuel reprocessing employs fluorides-volatility and vacuum distillation operations for the fuel steam and direct protactinium removal for the blanket steam. This reference design was assessed by the Thorium Task Force and was the basis for the Systems Analyses Task Force overall assessment effort.

Graphite irradiation experience has shown that dimensional changes can occur which result in an initial volumetric contraction followed by expansion. The rate of expansion, after the initial contracted volume is attained, increases with increasing exposure so that eventually the expansion limits the useful life of the graphite. In addition, the factors which control the lifetime dosage are graphite strength and changes in pore structure under irradiation.

A consequence of the irradiation experience was the further reassessment of the MSBR development effort due to the considerable uncertainty as to the practicality of using graphite as a structural material to separate fluids in the reference two-fluid MSBR concept. Simultaneously chemical research results indicated that molten salt reactors potentially could be operated economically as single fluid systems. These developments were associated primarily with the evidence that protactinium as well as rare earth fission products could be separated from single fluid salts. Thus in mid 1968, a single fluid, two-region MSBR concept was proposed, and a preliminary conceptual design prepared in which the graphite no longer serves as a structural material to separate two distinct fluids, but primarily serves as a moderator and a separation medium for two fuel regions of a single fluid. An important consideration in the new design was theoretical and preliminary experimental evidence that the U-233, and possibly rare earth fission products, could be separated from a mixed thorium uranium fuel salt by reduction extraction employing liquid bismuth. This, combined with nuclear consideration of the single fluid design, indicated that fuel breeding gains and economics comparable to the reference two fluid system could be achieved by the proposed single fluid concept.

An alternative for the MSBR is provided by the Molten Salt Converter Reactor (MSCR), a single-region, single-fluid reactor moderated by graphite, which is essentially the same as the single-fluid MSBR except that the fuel is processed on a much longer processing cycle. Thus, an MSCR can be converted to an MSBR by appropriate installation of processing equipment. The MSCR is a reactor which utilizes fluoride volatility and vacuum distillation processing.

Performance characteristics for the MSCR and the MSBR are summarized in Table III-4. Uranium-thorium and plutonium-thorium fuel cycles can be utilized in MSR's.

4. Breeder Reactors

Nuclear power reactors of the breeder type (Ref. III-5) produce more nuclear fuel than they consume. Thus they would make it feasible to utilize enormous quantities of low-grade uranium and thorium ores dispersed in the rocks of the earth as a source of low-cost energy for thousands of years. In addition, these reactors would operate without adding noxious combustion products to the air. It is in the light of these considerations that the U.S. Atomic Energy Commission, the nuclear industry, and the electric utilities have mounted a large-scale effort to develop the technology whereby it will be possible to have a breeder reactor generating electric power on a commercial scale by 1984.

Nuclear breeding is achieved with the neutrons released by nuclear fission. The fissioning of each atom of a nuclear fuel, such as uranium 235, liberates an average of more than two fast (high-energy) neutrons. One of the neutrons must trigger another fission to maintain the nuclear chain reaction; some neutrons are nonproductively lost, and the remainder are available to breed new fissionable atoms, that is, to transform "fertile" isotopes of the heavy elements into fissionable isotopes. The fertile raw

TABLE III-4. SUMMARY OF DESIGN AND PERFORMANCE
PARAMETERS FOR MOLTEN SALT REACTORS

	Molten Salt Converter	Molten Salt Breeder
Core height, feet	20.8	13.7
Core diameter, feet	16.6	9.7
Blanket thickness, feet		2.0
Core power, Mwt	2222	1812
Blanket power, Mwt		410
Average core power density, Kw/l	17	64
Graphite replacement life, years		2.1
Specific fuel inventory, Kg/Mwe	1.63	1.06
Breeder ratio	0.96	1.07
Annual fuel yield, %/year		4.8
Fuel doubling time, years		14.4
Thermal to electrical conversion efficiency, %	45.0	45.0

materials for breeder reactions are thorium-232, which is transmuted into uranium-233, and uranium-238, which is transmuted into plutonium-239.

Breeding occurs when more fissionable material is produced than is consumed. A quantitative measure of this condition is the doubling time: the time required to produce as much net additional fissionable material as was originally present in the reactor. At the end of the doubling time the reactor has produced enough fissionable material to refuel itself and to fuel another identical reactor. An efficient breeder reactor will have a doubling time in the range of from seven to 10 years.

Two different breeder systems are involved, depending on which raw material is being transmuted. The thermal breeder, employing slow neutrons, operates best on the thorium-232-uranim-233 cycle (usually called the thorium cycle). The fast breeder, employing more energetic neutrons, operates best on the uranium-238-plutonium-239 cycle (the uranium cycle). Nonproductive absorption of neutrons is less in fast reactors than it is in thermal reactors, resulting in a decrease in the doubling time. For this reason, only fast breeder reactors are discussed below.

The breeder reactor types are defined based on the type of coolant employed to carry off the heat of fission and deliver it to a power generating system. The coolants proposed were water and molten salts for thermal breeding and inert gas, liquid metal, and steam for fast breeding. The molten salt breeder reactor (MSBR) was discussed in the previous section. As shown in Table III-1, the remaining breeders (LMFBR, SCFR, and GCFR) make up the breeder reactor class. The liquid-metal-cooled fast breeder reactor is discussed further in the following section. The SCFBR and the GCFR are discussed under the heading "Alternate Coolant Fast Breeder Reactors."

a. Liquid-Metal-Cooled Fast Breeder Reactor (LMFBR). In the U.S. and several other countries decisions were made that a fast breeder reactor cooled with liquid metal was an attractive concept to develop. Several design features of the LMFBR are of interest. The core of a fast reactor can be quite small. For economic reasons the reactor must be operated at a much higher power density than ordinary fission reactors. The active core volume is therefore only a few cubic meters and is roughly proportional to the power output. The power density is approximately 400 Kw/liter.

In order to carry off the heat while maintaining the fuel at a reasonable temperature, sodium must flow through the core at a rate of tens of thousands of cubic meters per hour. To provide channels for the flow of sodium, the fuel is divided into thousands of slender vertical rods.

The fuel is preferably in a ceramic form such as oxide or carbide. These ceramics are stable during long exposures to heat and radiation, have very high melting points, and are relatively inert in liquid metal. The fissionable component of the fuel can be enriched uranium-235, plutonium-239, or a mixture of the two. Performance characteristics of LMFBR's utilizing various fuel forms are shown in Table III-5.

Much of the breeding takes place in the blanket that surrounds the fast reactor core. The blanket consists of uranium-238 in stainless steel tubes. Since there is a certain amount of fission in the blanket, it too must be cooled by the sodium.

Interspersed through the core region are numerous rods with safety and control functions. They maintain the power output at the desired level and provide the means for starting and stopping the reactor. The rods are filled with neutron absorbing material such as boron carbide or tantalum metal. Fast breeder reactors require fewer control rods than thermal reactors.

Three major reactors will carry the burden of the A.E.C.'s program to develop an LMFBR. Two of them are already in operation:

(1) The Experimental Breeder Reactor II (EBR-II), since 1964

(2) The Zero Power Plutonium Reactor (ZPPR), since 1969

The third reactor, Fast Flux Test Facility (FFTF), is being designed on the basis of data obtained from EBR-II, ZPPR, and smaller facilities.

EBR-II is a fast-neutron test reactor operated by the Argonne National Laboratory at the A.E.C.'s National Reactor Testing Station in Idaho. This reactor is the focal point of the program for testing fuels and irradiating materials for the LMFBR. At mid-1971, EBR-II has produced more than 1,000,000 Mw-hr of power (thermal) and over 250 x 10^6 Kw-hr electricity. It has achieved its design power of 62.5 MWt.

TABLE III-5. PERFORMANCE CHARACTERISTICS OF LIQUID-METAL-COOLED REACTORS

	Reference Oxide	Advanced Oxide-Negative Sodium	Advanced Oxide-Positive Sodium	Oxide Converter	Reference Carbide	Advanced Carbide	Carbide Converter
Thermal power, Mw	2197	1975	2103	2162	2300	2295	2032
Net electrical capacity, Mw	880	860	915	910	850	900	860
Average core power density, [a]Kw/liter							
Average core specific power, [a]Mwt/MT	175	230	230	106.4	113	167.9	126.7
Core fissile plutonium inventory, [a]Kg							
Total fissile plutonium inventory, [a]Kg	3430	2410	2100	3800	2410	1300	3250
Net gain of fissile plutonium, Kg/yr							
Core average burnup, Mwd/MT	80,000	101,000	97,000	80,000	79,360	110,300	80,000
Breeding ratio[b]	1.27	1.34	1.31	852	1.45	1.50	.89
Exponential doubling time, [c]years	15.00	8.00	7.00	---	8.6	4.8	---

[a]Calculated as linear averages over equilibrium cycle.

[b]Defined as fertile captures divided by absorptions in fissile plutonium

[c]Based on one-year out-of-core holdup of fissile plutonium.

Reference: WASH 1098, Tables E-5 and E-6.

The Zero Power Plutonium Reactor is also operated by Argonne National Laboratory. It has the size and a large enough inventory of plutonium ($\geq$ 3000 Kg) to allow full-scale mockups of the plutonium fuel arrangements that will be used in the large commercial breeders envisioned for the 1980's and beyond.

The Fast Flux Test Facility will operate at a very high neutron flux to produce the radiation effects on fuel and structural materials that will take place in a commercial breeder reactor. It will operate at 400 Mwe power and will be built at the A.E.C.'s site in Richland, Washington. Initial operation is anticipated in 1974.

b. Alternate Coolant Fast Breeder Reactors. Most of the development work of fast breeder reactors has been based on the use of liquid metal cooling, but in the early 1960's some interest was expressed in the use of other coolants as a means of alleviating some of the problems associated with liquid metals. Numerous coolants were considered. However, most were eliminated either by chemical or metallurgical evaluations (air, hydrogen, carbon monoxide) or heat transfer considerations (neon, argon). In addition to relatively poor heat transfer, argon also has problems of neutron activation, and neon is very expensive. Nitrogen has a high neutron-absorption cross section and might cause nitriding at high temperatures. Supercritical SO_2 cooling has been studied at the Oak Ridge National Laboratory and appears to be feasible. However, experience with this system at the required operating conditions is very limited. Thus, the list of alternate coolants of interest for fast breeder reactors was reduced to carbon dioxide, helium, and steam. Although from purely thermal and cost standpoints there appears to be little choice between CO_2 and helium, CO_2 cooling leads to higher core pressure drops, higher fuel ratings, and greater potential for flow-induced vibrations. Thus, present designs of alternate-coolant fast breeder reactors are based on the use of helium or steam cooling. These reactor types are discussed in depth in Reference III-6.

Steam-cooled fast breeder reactors have been studied in varying degrees of depth since 1961. At least four groups in the United States have studied the concept, and studies have also been made in Germany by the

Karlsruhe Nuclear Research Center, Institute of Reactor Development; in Sweden by A.B. Atomenergi; and by the Euratom-Belgium Fast Reactor Association.

The study reported in 1965 by the Babcock & Wilcox Company jointly with American Electric Power Service Corporation was based on use of supercritical steam as a coolant and, although the concept appeared to be economically competitive, the breeding ratio was only about 1.03. It was concluded that the concept was not consistent with the objectives of the AEC breeder reactor program but that it did fulfill the requirements for an advanced converter. The low breeding ratios found in this study, and the conviction that the core could be cooled with low-pressure steam, led the Babcock & Wilcox Company to propose at the Argonne National Laboratory in October 1965 that a lower steam pressure would result in a higher breeding ratio. Their paper indicated that, with a steam pressure of 1200 psi, it would be possible to attain an overall breeding ratio of about 1.40. They had not developed a complete design based on the parameters of the lower pressure steam-cooled reactor. Thus in 1966 and 1967, under contract to the Oak Ridge National Laboratory, Babcock & Wilcox Company and Sargent & Lundy Engineers developed a more complete design study of the low pressure concept for use by the Alternate Coolant Task Force in the evaluation of steam-cooled breeder reactors. A rather complete study by the Karlsruhe group was published in August 1966 based on the use of steam at a pressure of approximately 2650 psia. The German study showed a breeding ratio of about 1.15 and a doubling time of about 37 years. Thus, considerations of the range of steam pressures from 1250 to 3700 psia was available to the task force in three design studies. The Oak Ridge National Laboratory attempted to normalize design parameters of these three studies sufficiently to evaluate the potential of steam cooling for fast breeder reactors, and to obtain some understanding of the economics, physics, safety, unsolved problems, and the effect of steam pressure on the concept. As a supplement to this evaluation, the Pacific Northwest Laboratory was asked to perform a parametric study of reactor design over the range of steam conditions of current interest.

Three steam-cooled fast breeder reactors (SCBR) have been considered in the evaluation (Ref. III-6). Some of their performance characteristics are shown in Table III-6. The first is a high pressure concept cooled by steam which enters the reactor at 3700 psia and 750° F. Steam enters the turbine at 330 psia and 994° F, with reheat to 950° F. This high pressure SCBR concept was developed jointly by the Babcock & Wilcox Company and American Electric Power Service Corporation.

An intermediate pressure SCBR concept developed by the Karlsruhe group is the second concept considered. In this system, the steam enters the reactor vessel at 2680 psia and 710° F and enters the turbine at 2350 psia and a temperature of 1000° F. The steam is reheated in a surface-type heat exchanger to 950° F.

The third SCBR considered was developed by Babcock & Wilcox Company and Sargent & Lundy Engineers and is referred to as the low-pressure SCBR design. In this concept steam enters the core at 1250 psia and 575° F, and the turbine throttle conditions are 1050 psia and 925° F. There is no reheat of the steam in this power cycle.

In all three SCBR concepts, the steam used for cooling the reactor core is generated in a Loeffler-type cycle that uses steam from the reactor exit as a heat source. Steam generated by mixing the high-temperature exit steam with feedwater is circulated through the reactor by a steam circulator. In the low pressure design, the boiler and circulator are integral with the reactor vessel so that piping between the Loeffler boiler, the steam circulator, and the reactor is eliminated. In the other two concepts, the steam generators and circulators are located separately from the reactor vessel and are connected by piping.

Conceptual design studies of the GCFR were started in 1961 by the General Atomic Division of General Dynamics Corporation. These studies were continued and expanded in additional work by Gulf General Atomics under an AEC contract starting in 1963. A conceptual design study to establish the practicality of a 1000 Mwe GCFR power plant was undertaken in August 1965 as a joint effort by GGA and the East Central Nuclear Group (ECNG). The results to date of the ECNG study were made available to the AEC and the Alternate Coolant Task Force. This study was updated and reoptimized for the current assessment of civilian power reactors under contract to ORNL, and the results were published in a report by GGA.

Table III-6 Performance Characteristics of Alternate Coolant Fast Breeder Reactors

	Steam-Cooled Reactors			Gas-Cooled Reactors		
	1250-psi Plant	2680-psi Plant	3700-psi Plant	Derated Plant	Reference Plant	Carbide-Fueled Plant
Thermal power, Mw	2,900	2,519	2,326	2,681	2,530	2,678
Net electrical capacity, Mw	1,012	1,000	970	1,000	1,000	1,000
Average core power density,[a] kw/liter	355	285	445	245	265	545
Average core specific power,[a] Mw/kg fissile	0.74	0.66	0.94	0.80	0.82	1.30
Core fissile plutonium inventory,[a] kg	3,555	3,530	1,980	2,987	2,727	1,688
Total fissile plutonium inventory,[a] kg	4,085	3,835	2,325	3,797	3,523	2,520
Net gain of fissile plutonium, kg/yr	318	95	65	349	356	450
Core average burnup, Mwd/MT	70,000	57,500	58,000	74,800	75,000	112,200
Breeding ratio[b]	1.38	1.14	1.11	1.48	1.48	1.60
Exponential doubling time,[c] years	12	38	41	11	10	5.5
Fuel life, years at a power factor of 0.8	3.0	2.65	1.6	2.5	2.5	2.5

[a] Calculated as linear averages over equilibrium cycle.

[b] Defined as fertile captures divided by absorptions in fissile plutonium.

[c] Based on one-year out-of-core holdup of fissile plutonium.

Incentives for developing the helium-cooled fast reactor (GCFR) include the attainment of a high breeding ratio in a system utilizing gas cooling technology. The breeding ratios obtained in the GCFR can be equal to or better than those of the LMFBR. Principal components, including the helium circulators, steam generators, prestressed concrete pressure vessel, and helium piping, would draw heavily on technology for existing thermal gas-cooled reactors and on current HTGR development programs. Fuel element design would utilize the current and planned AEC development programs for sodium-cooled fast reactors. The high internal breeding ratio of the GCFR, along with the provision for on-load variation of flow orificing, would permit a fuel cycle in which the entire core was reloaded at one time. This would minimize the frequency of outages for refueling.

Three helium-cooled designs were also considered:

(1) A derated oxide fueled design (GCFR-4D) with a limit on the cladding temperature set 50°C below that specified in the reference design

(2) A reference oxide fueled design (GCFR-4) with relatively high values for maximum pin heat rating and cladding temperature

(3) A carbide fueled design operating at higher pin rating, gas pressure, power density, and specific power

Some performance characteristics for these breeder reactors are shown in Table III-6.

In a typical GCFR, the reactor and steam generators are contained in the prestressed concrete pressure vessel which also serves as the biological shielding. Electricity is generated from a high pressure steam cycle with a net thermodynamic efficiency of almost 40 percent. The steam generators are supplied with 1250- to 1750-psi high-temperature helium. The coolant flow is downward through the core, with the fuel elements cantilevered from a deep top-mounted grid plate. Each fuel element is comprised of metal clad mixed uranium-plutonium oxide or carbide fuel rods. Overall conversion ratios of 1.45 to 1.60 are obtained, and the internal conversion ratios are about 1.0.

B. Fusion Reactor Power

The generation of nuclear power by controlled nuclear fusion can probably be achieved. An optimistic projection is that fusion power will be available in appreciable quantity by the year 2000. Controlled fusion research has passed through several epochs, the first of which was initiated by four items. First came measurements of reaction energies and rates between hydrogen isotopes and other light elements, which showed that under proper conditions large energy releases would be possible. Second, the well-known laws of single particle physics seemed to show how an assembly of high energy ions and electrons could be confined in magnetic fields long enough to establish the proper conditions. Third, the radioactive ingredients and by-products of fusion appear to be much less hazardous than those associated with nuclear fission; therefore, fusion reactors would be simpler and safer than fission reactors. Fourth, deuterium is a fusion fuel in plentiful supply--one part in 7000 of ordinary hydrogen, and extraction from ordinary water is not difficult. So matters stood in the early days, say up to 1955.

Under the present scientific development program various schemes are being studied and postulated to confine the fusion plasma. The main schemes being developed so far involve use of large volumes of high magnetic fields. Plasma ions and electrons are hindered by magnetic forces from moving across the direction of magnetic fields, but can spiral along the field lines. Thus (naively), confinement in the two directions perpendicular to the field direction is achieved, and one might have to worry only about confinement along the field direction. The various devices which may lead to a feasible fusion power reactor are:

(1) Stellarator

(2) Tokamak

(3) Internal conductor devices

(4) Fast pulsed (θ - pinch)

(5) Hot electron mirror

(6) Ion injection mirrors

(7) Astron (a mirror device)
(8) Continuous flow pinch, and
(9) Laser ignition

The description and status of these various devices are presented by Rose in Reference III-7. At this time hopes are high that the Tokamak device will establish the feasibility of a fusion reactor. It utilizes a toroidal-shaped plasma which is also the secondary loop of a transformer.

Problems must be solved before a controlled fusion reactor can be attained. There are some problems that are substantially independent of the particular geometric model or device:

(1) Plasma conditions in imagined practical devices, such as ion and electron temperatures, the fraction of fuel burned up per pass through the reactor, and radiation from the plasma surface must be calculated. However, a major difficulty with all calculation of electron and ion temperatures with increasing fractional fuel burnup is that the rather restrictive assumptions used, i.e., no turbulence or space charge effects in the plasma, render the calculations nebulous. A subfield of fusion plasma engineering needs developing before a fusion reactor can be sensibly designed.

(2) Regenerating tritium (for a Deuterium-Tritium reactor) in a surrounding moderator-blanket by means of the 14.1-Mev neutrons in a Li reaction seems adequately assured by using a liquid Li or Li salt coolant with a neutron moderator between the blanket and vacuum walls of the reactor.

(3) Heat deposition, temperature of the moderator and vacuum wall, and heat removal must be determined since heat deposition (and removal) per unit volume on the vacuum wall facing the plasma determines the power capability of the whole system.

(4) Providing large quantities of high magnetic field and structure to withstand high stress is a problem. For example, a simple solenoid generating 150,000 gauss has a magnetic bursting force of 900 atm. on its windings.

(5) The effects of material radiation damage to the vacuum wall by the 14.1-Mev neutrons must be minimized since the consequences of such damage may be frequent and expensive replacement of much of the structure.

(6) Size and cost, which are implicit in many of the above are, of course, the major factors in determining the feasibility of a fusion reactor. Size is large for lowest power cost and will increase in the future so that 10,000 Mwt is liable to be quite acceptable in 2000. Cost estimates indicate that cost might be $6-20 per Kwt, an attractive cost range. Other problems are model-dependent; some device concepts seem to require additional developments.

These problems can be described and a physical feel can be obtained with the aid of Figure III-1, a fusion reactor schematic consisting of a series of concentric cylinders.

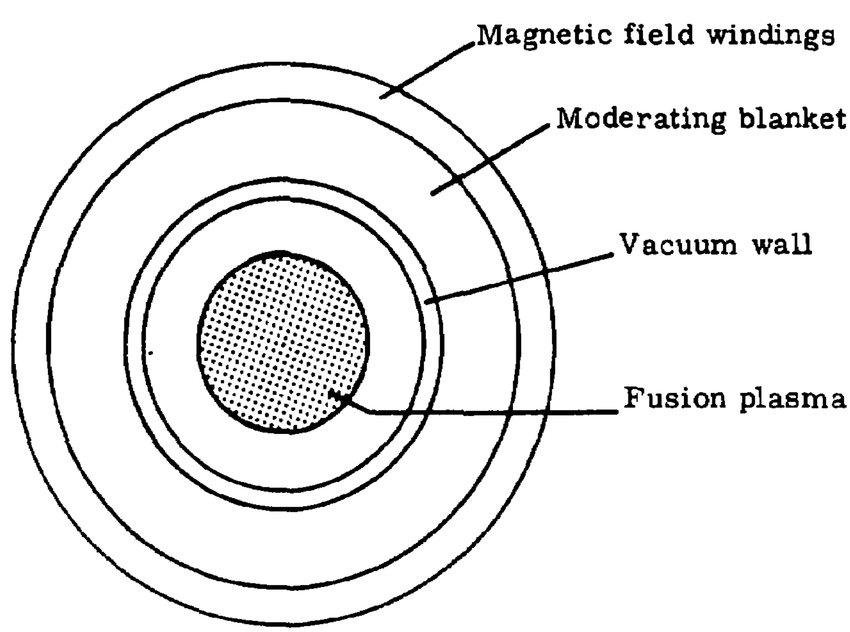


Figure III-1. Schematic of Controlled Fusion Reactor

The main confining magnetic field is into (or out of) the paper; whether the cylinder is the center section of a stabilized mirror device or is wrapped into a torus need not concern us here. The fusion plasma occupies the evacuated center, is surrounded by a neutron-moderating blanket and, at large radius, by a set of magnetic field coils.

Since its containment is imperfect, some of the plasma fuel is continually being lost from the ends or sides. Thus, it is being replaced by some injection process into the center. A certain throughput of plasma is needed to keep up its density. The plasma is at least partly heated by its own reaction.

For a deuterium-tritium reactor, tritium must be regenerated. The general approach, then, in Figure III-1 is to make the vacuum wall and blanket supporting structure of thin section refractory metal. Within it, there would be liquid lithium or a lithium salt coolant, plus an artfully disposed neutron moderator (probably partly of graphite). Leading choice for metals is niobium in that it can be formed and welded, retains its strength at 1000°C, and is transparent to tritium. This transparency helps in two ways: tritium generated in the lithium-bearing coolant is not trapped in the metal; and tritium can be recovered by diffusion through thin section walls into evacuated recovery regions. Some additional neutrons also come from the niobium via (n, 2n) reactions, but in this particular respect molybdenum would be a better material. The tritium fuel doubling time in a fusion reactor might be less than one year. This short doubling time for fusion is contrasted to the longer one ($\approx$ 20 years in some designs) for fission breeder reactors.

The approximate size of the fusion reactor can be estimated. Fairly simple nuclear calculations establish that the blanket plus a radiation shield (not shown) to protect the outer windings must be 1.2 to 2.0 m thick. This substantial thickness implies not only substantial blanket cost, but also very high magnetic field cost, to energize such a large volume. The only way to make the system pay is to have it generate a great deal of power, but nearly all this power must pass from the plasma into or through the vacuum wall. Engineering limits of power density and heat transfer then dictate large plasma and vacuum wall radii as well--between 1 and 4 m, say. Then overall size will be large, and total power will be high--almost certainly more than 1000 megawatts (electric) and perhaps 5000 megawatts.

Energy is deposited in the vacuum wall facing the plasma, mainly from three sources:

(1) Some of the fusion neutrons suffer inelastic collisions as they pass through

(2) Gamma rays from deeper inside the blanket shine onto the back side

(3) All electromagnetic radiation from the plasma is absorbed there

The plasma itself makes no additional load, being imagined to be pumped out elsewhere. The three sources may constitute 10 to 20 percent of the total reactor power. This is a modest fraction; but the vacuum wall region is thin, and heat depostion (and removal) per unit volume determines the power capability of the whole system.

The size of the magnetic field windings generating 150,000 gauss causes a problem with the stresses that arise resulting from the magnetic bursting force. Almost all conceptions involve superconducting coils at 4^{o}K, or at least cryogenically cooled ones at 10^{o} to 20^{o}K. This is the reason for placing them outside the blanket, outside a radiation shield; otherwise the refrigeration problem would be intolerable. To make a reinforcing structure for operation at such a temperature, with the size and stress loads described, is a task yet to be fully contemplated. Titanium is very strong at such low temperatures; but it is also very brittle--as are most other materials under those conditions.

Neutron damage is a very serious problem, for either a fission or fusion reactor. In one way, fusion appears at a substantial disadvantage, as follows. One fission reaction produces 200 Mev and about 2.5 neutrons, each with no more than about 2 Mev. One fusion reaction produces 17.6 Mev, of which 14.1 Mev appears in one high energy neutron. Thus, the "energetic neutrons/watt" is an order of magnitude higher in fusion than in fission, and the structural damage caused by these neutrons is correspondingly high.

REFERENCES

III-1. "Potential Nuclear Power Growth Patterns," WASH 1098, U.S.A.E.C. Division of Reactor Development and Technology, December 1970.

III-2. "Current Status and Future Technical and Economic Potential of Light Water Reactors," WASH 1082, U.S.A.E.C. Division of Reactor Development and Technology, March 1968.

III 3. "An Evaluation of Advanced Converter Reactors," WASH 1087, U.S.A.E.C. Division of Reactor Development and Technology, April 1969.

III-4. "The Use of Thorium in Nuclear Power Reactors," WASH 1097, U.S.A.E.C. Division of Reactor Development and Technology, June 1969.

III-5. Seaborg, G.T. and J.L. Bloom, "Fast Breeder Reactors." Scientific American, Vol. 223, No. 5, November 1970, pp. 13-21.

III-6. "An Evaluation of Alternate Coolant Fast Breeder Reactors," WASH 1090, U.S.A.E.C. Division of Reactor Development and Technology, April 1969.

III-7. Rose, D.J., "Controlled Nuclear Fusion: Status and Outlook," Science, Vol. 172, No. 3985, May 21, 1971, pp. 797-808.

IV. EXISTING AND PLANNED GENERATING STATUS BY TYPE AND REGION

The existing and near-term planning status of central station nuclear plants is presented for each census region in Tables IV-1 through IV-9. This status is defined as encompassing those units operable, under construction, or planned in the near future. For each state, the nuclear plant name is given. The net plant power capacity in Mwe and type for each unit are listed in the year projected as being the startup time. In the latter years, such as those spanning the period from 1976 through 1980, additional new plant purchases can be expected. The total electrical capacities for a given year are shown after summing the individual capacities over the particular census region. The data in these tables were compiled from Refs. IV-1 through IV-3. More recent information for update purposes was obtained from U.S.A.E.C. immediate releases and recent issues of Nuclear News and Nuclear Safety. Therefore, the tables are considered to be current through September 1971.

The geographical distribution of nuclear power plants in the United States is presented in Figure IV-1 (Ref. IV-2). This figure includes one operable and two planned dual purpose plants which generate electrical power.

TABLE IV-1. CENTRAL STATION NUCLEAR PLANTS
CENSUS REGION: NEW ENGLAND
(Maine, New Hampshire, Vermont, Massachusetts, Rhode Island, Connecticut)

State	Nuclear Plant Name	Pre-1970	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Me.	Me. Yankee				790(PWR)								
N.H.	Seabrook										885		
Vt.	Vt. Yankee			514(PWR)									
Mass.	Pilgrim				655(BWR)								
	Yankee-Rowe	(1961) 175 (PWR)											
R.I.	(None)												
Conn.	Conn. Yankee	(1967) 575 (PWR)											
	Millstone 1		652(BWR)										
	Millstone 2						828(PWR)						
Region Total		750	652	514	1445	0	828	0*	0*	0*	885*	0	0

*New purchases may be expected.

TABLE IV-2
CENSUS REGION: MIDDLE ATLANTIC
(New York, New Jersey, Pennsylvania)

State	Nuclear Plant Name	Pre-1970	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
N.Y.	Bell (Inactive)										838(BWR)		
	Fitzpatrick					821(BWR)							
	Indian Point 1	(1963)265 (BWR)											
	Indian Point 2				873(PWR)								
	Indian Point 3					965(PWR)							
	Indian Point 4							1115(BWR)					
	Indian Point 5									1115(BWR)			
	Shoreham							819(BWR)					
	Ginna		420(PWR)										
	Nine Mile Pt. 1		500(BWR)										
	Nine Mile Pt. 2									1100(BWR)			
N.J.	Forked River									1140(PWR)			
	Newbold Island 1							1088(BWR)					
	Newbold Island 2									1088(BWR)			
	Salem 1					1090(PWR)							
	Salem 2						1115(PWR)						
	Oyster Creek	(1969)560 (BWR)											
Pa.	Limerick 1							1065(BWR)					
	Limerick 2									1065(BWR)			
	Peach Bottom 1	(1967) 40 (GCGM)											
	Peach Bottom 2					1065(BWR)							
	Peach Bottom 3						1065(BWR)						
	Three Mile Island 1				831(PWR)								
	Three Mile Island 2						907(PWR)						
	Pa. Pwr. & Lt. 1										1052(BWR)		
	Pa. Pwr. & Lt. 2											1052(BWR)	
	Beaver Valley 1					847(PWR)							
	Beaver Valley 2										847(PWR)		
	Shippingport	(1957) 90 (PWR)											
Region Total		955	920	0	1704	4788	3087	4087	0*	5508*	2737*	1052*	0*

*New purchases may be expected.

TABLE IV-3
CENSUS REGION: SOUTH ATLANTIC
(Delaware, D.C., Florida, Georgia, Maryland, N. Carolina, S. Carolina, Virginia, W. Virginia)

State	Nuclear Plant Name	Pre-1970	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Del.	(None)												
D.C	(None)												
Fla.	Crystal River 3				858(PWR)								
	Crystal River 4										897(PWR)		
	Hutchinson Isl. 1						800(PWR)						
	Hutchinson Isl. 2								800(PWR)				
	Turkey Point 3			693(PWR)									
	Turkey Point 4				693(PWR)								
Ga.	Hatch 1					786(BWR)							
	Hatch 2								786(BWR)				
	Hancock 1										1100(BWR)		
	Hancock 2											1100(BWR)	
Md.	Calvert Cliffs 1					845(PWR)							
	Calvert Cliffs 2						845(PWR)						
N.C.	Brunswick 1							821(BWR)					
	Brunswick 2						821(BWR)						
	McGuire 1							1088(PWR)					
	McGuire 2									1088(PWR)			
	Wake County 1									915			
	Wake County 2										915		
	Wake County 3											915	
	Wake County 4												915
	Wilmington 1								821(BWR)				
S.C.	Robinson 2		700(PWR)										
	Oconee 1			841(PWR)									
	Oconee 2				886(PWR)								
	Oconee 3					886(PWR)							
	Parr 1									900			
Va.	North Anna 1						845(PWR)						
	North Anna 2							845(PWR)					
	Surry 1				780(PWR)								
	Surry 2				780(PWR)								
	North Anna 3									940			
	North Anna 4										940		
W. Va.	(None)												
Region Total		0	700	1534	3997	2517	3311	1933	2407*	3843*	3852*	2015*	915*

*New purchases to be expected.

TABLE IV-4
CENSUS REGION· EAST NORTH CENTRAL
(Illinois, Indiana, Michigan, Ohio, Wisconsin)

State	Nuclear Plant Name	Pre-1970	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Ill.	Dresden 1	(1960)200 (BWR)											
	Dresden 2		809(BWR)										
	Dresden 3				809(BWR)								
	LaSalle 1							1078(BWR)					
	LaSalle 2								1078(BWR)				
	Quad Cities 1			809(BWR)									
	Quad Cities 2				809(BWR)								
	Zion 1				1050(PWR)								
	Zion 2					1050(PWR)							
Ind.	Bailly 1								660(BWR)				
Mich.	Cook 1					1054(PWR)							
	Cook 2						1060(PWR)						
	Fermi 2						1123(BWR)						
	Midland 1								492(PWR)				
	Midland 2									818(PWR)			
	Palisades			700(PWR)									
	Big Rock Pt.	(1962)70 (BWR)											
	Fermi	(1966)61 (SC, F)											
Ohio	Davis-Besse						872(PWR)						
	Zimmer 1						840(BWR)						
	Zimmer 2								810(BWR)				
Wisc.	Kewaunee				540(PWR)								
	Pt. Beach 1		497(PWR)										
	Pt. Beach 2			497(PWR)									
	LaCrosse	(1968) 50 (BWR)											
Region Total		381	1306	2106	3208	2104	3895	1078	3040*	818*	0*	0*	0*

*New purchases may be expected.

TABLE IV-5
CENSUS REGION EAST SOUTH CENTRAL
(Alabama, Kentucky, Mississippi, Tennessee)

State	Nuclear Plant Name	Pre-1970	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Ala.	Browns Ferry 1				1065(BWR)								
	Browns Ferry 2				1065(BWR)								
	Browns Ferry 3					1065(BWR)							
	Farley 1							829(PWR)					
	Farley 2									829(PWR)			
Ky.	(None)												
Miss.	(None)												
Tenn.	Sequoyah 1						1124(PWR)						
	Sequoyah 2						1124(PWR)						
	Unit 6								1150(PWR)				
	Unit 7									1150(PWR)			
	Unit 8									1230(PWR)			
	Unit 9										1230(PWR)		
	Watts Bar 1								1170(PWR)				
	Watts Bar 2									1170(PWR)			
Region Total		0	0	0	2130	1065	2248	829	2320*	4379*	1230*	0*	0*

*New purchases may be expected.

TABLE IV-6
CENSUS REGION: WEST NORTH CENTRAL
(Iowa, Kansas, Minnesota, Missouri, Nebraska, N. Dakota, S. Dakota)

State	Nuclear Plant Name	Pre-1970	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Iowa	Arnold					530(BWR)							
Kan.	(None)												
Minn.	Prairie Island 1				530(PWR)								
	Prairie Island 2						530(PWR)						
	Monticello			545(BWR)									
Mo.	(None)												
Neb.	Cooper					778(BWR)							
	Fort Calhoun				457(PWR)								
N. D.	(None)												
S. D.	(None)												
Region Total		0	0	545	987	1308	530	0	0*	0*	0*	0*	0*

*New Purchases may be expected.

TABLE IV-7
CENSUS REGION: WEST SOUTH CENTRAL
(Arkansas, Louisiana, Oklahoma, Texas)

State	Nuclear Plant Name	Pre-1970	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Ark.	Ark. Nuclear 1					820(PWR)							
	Ark. Nuclear 2							920(PWR)					
La.	Waterford 3								1165(PWR)				
Okla.	(None)												
Tex.	(None)												
Region Total		0	0	0	0	820	0	920	1165*	0*	0*	0*	0*

*New purchases may be expected.

TABLE IV-8
CENSUS REGION: MOUNTAIN
(Arizona, Colorado, Idaho, Montana, Nevada, N. Mexico, Utah, Wyoming)

tate	Nuclear Plant Name	Pre-1970	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
riz.	(None)												
olo.	Fort Saint Vrain				330(HTGR)								
da.	(None)												
Mont.	(None)												
Nev.	(None)												
N. M.	(None)												
Jtah	(None)												
Wyo.	(None)												
Region Total		0	0	0	330	0	0	0	0*	0*	0*	0*	0*

*New purchases may be expected.

TABLE IV-9
CENSUS REGION PACIFIC
(California, Oregon, Washington)

State	Nuclear Plant Name	Pre-1970	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Cal.	Diablo Canyon 1						1060(PWR)						
	Diablo Canyon 2							1060(PWR)					
	Rancho Seco					804(PWR)							
	San Onofre 2							1140(PWR)					
	San Onofre 3									1140(PWR)			
	San Onofre 1	(1967)430 (PWR)											
	Bay	(1963) 68 (BWR)											
	Point Arena 1 & 2									1128		1128	
Ore.	Trojan						1130(PWR)						
Wash.	Hanford 2									1135			
	N Reactor	(1966)790 (G)											
Region Total		1288	0	0	0	804	2190	2200	0*	3403*	0*	1128*	0*

*New purchases may be expected.

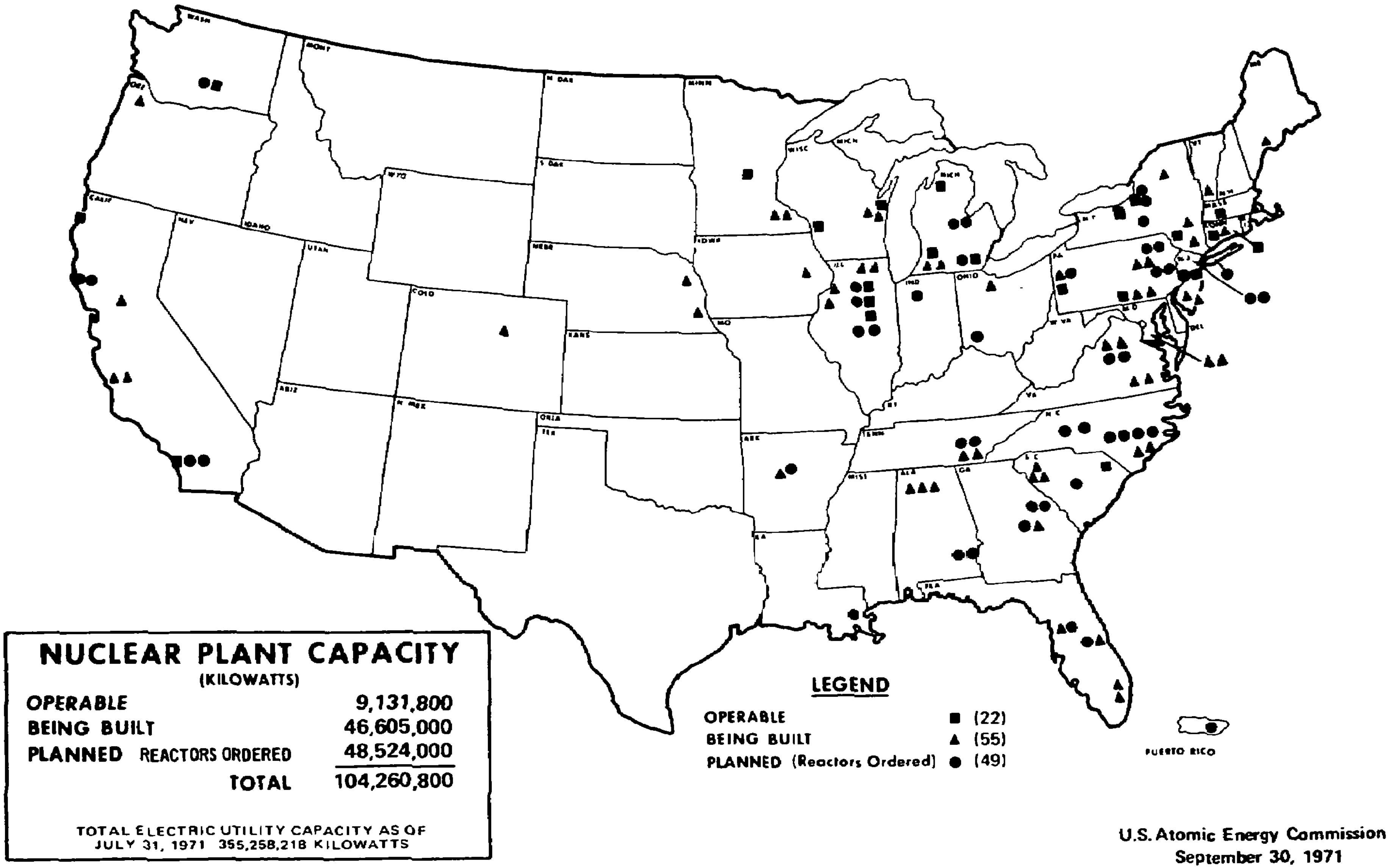


Figure IV-1. Nuclear Power Reactors in the United States

REFERENCES

IV-1. "Central Station Nuclear Plants - Units Operable, Under Construction or On Order," U.S.A.E.C. Division of Industrial Participation, March 31, 1971.

IV 2. "Nuclear Reactors Built, Being Built, Or Planned in the United States as of December 31, 1970," TID 8200 (23rd Rev.), Office of the Assistant General Manager for Reactors of the U.S.A.E.C., 1971.

IV-3. Lyerly, R.L., "A Listing of Commercial Nuclear Power Plants, Edition No. 3," Southern Nuclear Engineering, Inc., August 1970.

V. LONG-TERM PROSPECTS

A. *Comparison of Nuclear Power Projections*

There have been many attempts to project the future requirements for nuclear power. In this section these various projections are presented for comparative purposes along with their underlying sources. Several current estimates of nuclear power requirements were not included since there was no source or basis given for the quantities. Basic references in which energy forecasts have been presented and on which others have based their forecasts are listed as References V-1 through V-15.

Along with this wealth of forecast sources,there has been made a corresponding range of assumptions, and a variety of terminology has been employed. Thus, confusion arises during the attempt to compare and evaluate the magnitudes of nuclear power required in the future years. Fundamentally, there is an electrical energy demand and an electrical energy supply. Electrical energy demand or requirement is a function of population growth and per capita consumption. Electrical energy supply or availability is a function of natural resources available, market place considerations, and the development of technology. In this report an attempt will be made to adhere to this terminology.

Pacific Northwest Laboratories (PNL), in Ref. V-16,collected and compared the projections of a number of forecasters. They then assessed the methodologies employed. Most of the forecasts studied (including some from the basic listing of references above) provided only limited information about their methodology and practically none provided quantitative statements of the actual forecasting relationships. Very few forecasts provided standard errors of estimate of other measures of uncertainty. Some forecasts gave ranges but no information on the probability that future values would be within the range. Many projections do not set forth their underlying assumptions nor do they all consider the same factors in the construction of their projection. Indeed it was apparent that variations in definitions for such terms as energy consumption, energy requirement, and energy demand existed.

The variations in terminology become evident when functional relationships for nuclear power projections were expressed for some of the forecasts whose abstracts appeared in Ref. V-16. These are listed in Table V-1. In parentheses is the abbreviation of the reference assigned by the authors of Ref. V-16. The corresponding page of the PNL report is also given. As an example, in the EUS reference (Ref. V-13), installed capacity is a function of the 1960-1985 time period, per capita energy consumption, the population growth, and future fuel costs.

The first step in the preparation of the curve showing the forecasts of future nuclear power requirements was to determine the present and near term installed capacity. Figure V-1 shows the growth of this on-line nuclear capacity. The solid bars show the net nuclear capacity installed in a given year. These magnitudes were determined from Tables IV-1 through IV-9 by summing the total capacities shown for all nine census regions for each year. The cumulative nuclear generating capacity curve was then determined by the summation of all the installed capacities up through a given year. As anticipated, the cumulative capacity curve tails-off in the latter years shown since more plant purchases can be expected but are unknown at this time.

The cumulative capacities from Figure V-1 were plotted on Figure V-2 as the near-term installed capacity that will be available. Then the forecasts of future nuclear power requirements were plotted from the various sources. The first 10 sources are identified by the abbreviations assigned by Ref. V-16. The additional references, V-17 to V-19, discussed in the PNL report were not previously assigned the designation "basic references" as per this report. More recent forecasts of installed nuclear power requirements attributed to Gambs and Rauth, Earhart, and Yarcosa are found in Refs. V-20 through V-23. In order to satisfy the future demand as shown by the trends of these data, the nuclear generating capacity available must be an extension of the solid line. Ideally, this extended curve (now shown) of future nuclear power available would be an upper bound curve to the required nuclear power data points. Later in Section V.B, it is shown, after employing a linear programming computer model, that future nuclear generating capacity at minimized cost can be a family of curves. The family of curves results from the choices of various reactor types being developed and employed for future power generation.

TABLE V-1. NUCLEAR FORECAST FUNCTIONAL RELATIONSHIPS

Installed capacity (EUS, p. 57)	=	F (1960-85 period, per capita consumption, population growth, future fuel costs)
Energy demand (10^{12} B) (RAF, p. 61)	=	F (fuel source)
Energy demand (EMUS, p. 66)	=	F (1947-65 period, fuel uses, utility electricity uses, raw material non-fuel and non-power uses, sector, source)
Nuclear power capacity (FGNP, p. 38)	=	F (present and future installed capacity)
Gross energy (PEC, p. 39)	=	Σ all types of commercial energy
Electrical energy requirements (NPS, p. 53)	=	F (population, GNP, geographical areas)
Energy consumption (PCCP, p. 76)	=	F (population, labor force, productivity, GNP, Index of Industrial Production & Gross Product Originating)

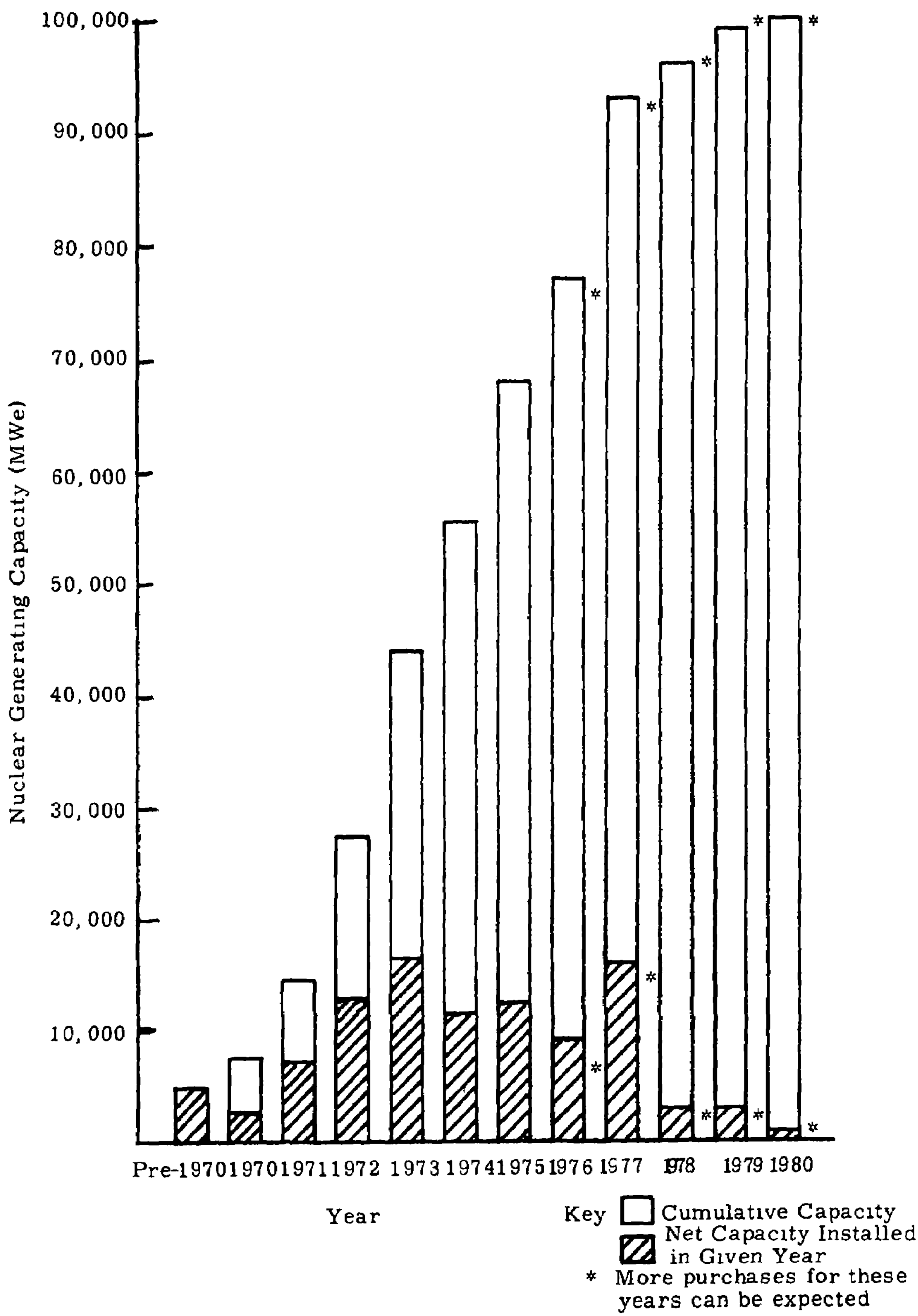


Figure V-1. Growth of On-Line Nuclear Capacity

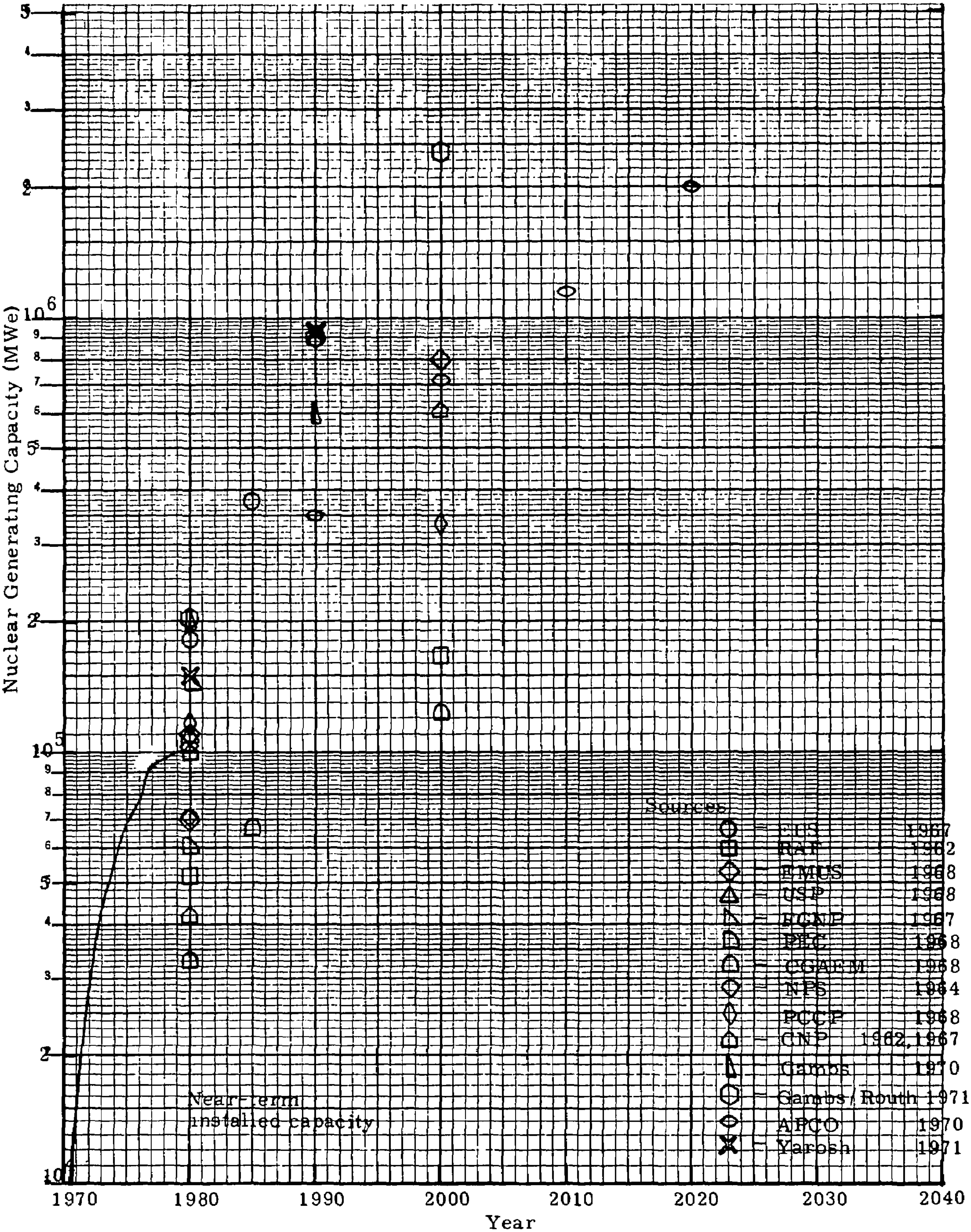


Figure V-2. Forecasts of U.S. Nuclear Power Requirements

The nuclear energy requirements from the RAF, PEC, CGAEM, and PCCP references were given in Ref. V-16 in the units of trillions of Btu's. These are converted to capacities and plotted on Figure V-2 after assuming a heat rate = 10200 Btu/kwh and an 80 percent load factor (7000 hr/yr plant operation). For the first 10 references listed on Figure V-2, two assumptions generally applied are:

(1) Rate of growth of the gross national product (GNP) equal to $\approx$ 4 percent/year

(2) Projected population growth from the Bureau of Census equal to ~1.6 percent/year

The forecasts for total electricity requirements in the United States are presented on Figure V-3 for comparison purposes with Figure V-2. The future total electricity requirement consists of contributions from fossil, nuclear and hydroelectric power utilities, and industrial power generation.

Many future perturbations affect the accuracy of the nuclear power forecasts. Some of these are:

(1) Environmental constraints

(2) Siting

(3) Nuclear waste disposal

(4) Politics

(5) Labor availability

(6) Inflation

(7) Fuel availability and cost

They are such as to cause delays in meeting future power demands. The results are strongly dependent on the commercial marketing assumptions including those concerning the establishment of large fuel fabrication and reprocessing industries, acceptance of safety and siting features, and the availability of trained personnel and resources. Timely introduction of the breeder reactor is necessary to reduce power costs by substantial margins while providing good nuclear fuel utilization. Environmental factors now play a significant part in power plant selection as well as costs. In a case such as California,

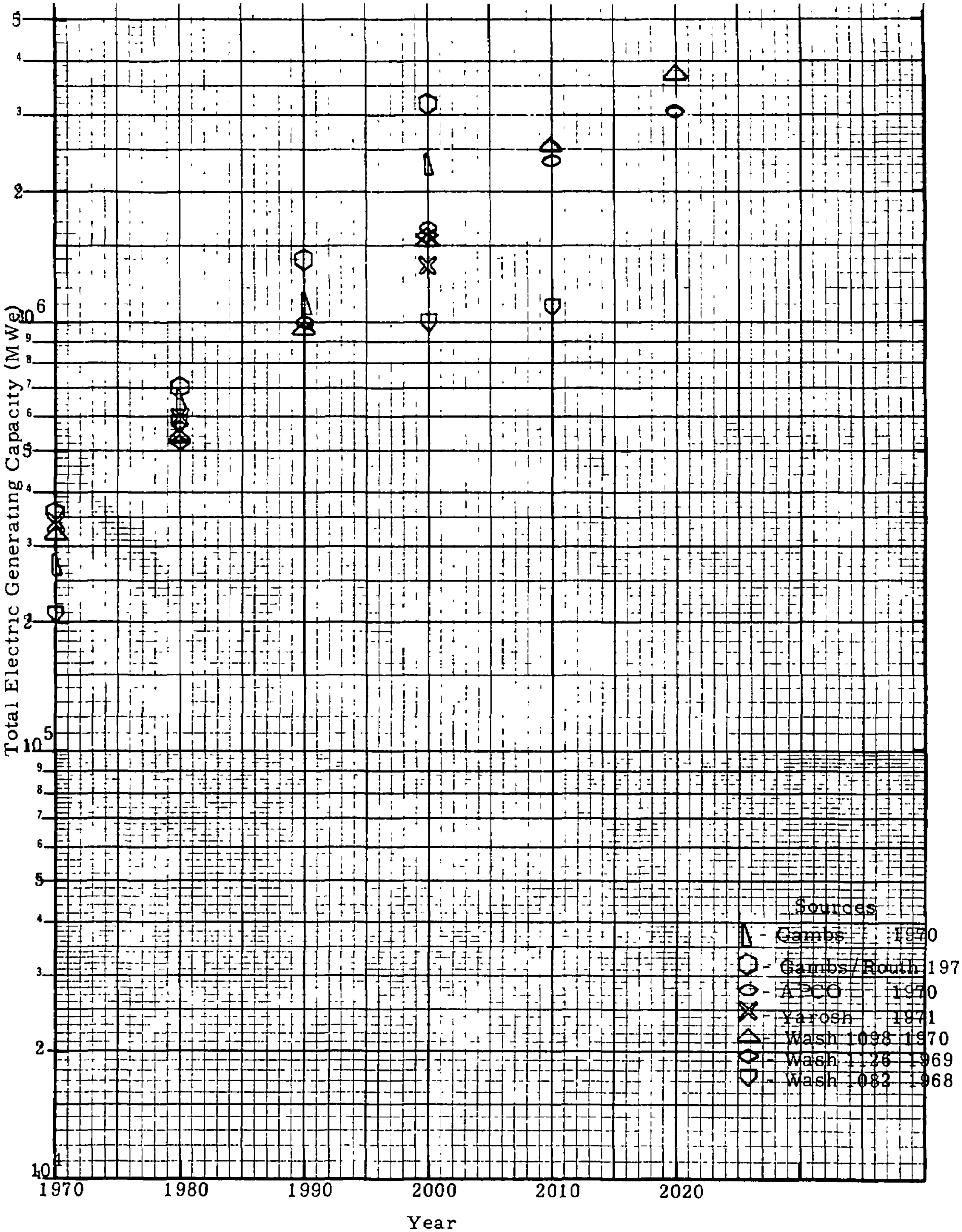


Figure V-3. Forecasts of U.S. Total Electricity Requirements such stringent smoke control ordinances have been passed that no more fossil fuel plants are possible without major control equipment. In other states, such as Maryland and southern Florida, some groups are deeply concerned about the thermal effects of nuclear power plants on surrounding bodies of water. Also, in the future it is possible that a constant or even an increasing amount of energy per unit GNP may be required if present practices of encouraging the use of energy are continued.

B. Effects of Reactor Types and Their Development on Energy Projections and Costs

A linear programming (LP) model of the U.S. electrical power economy has been developed under the Civilian Nuclear Power (CNP) program. The model was applied (Ref. V-24) to evaluate the benefits of alternate courses of electrical power system developments. The reactor types treated in the study were:

(1) Light water reactors (LWR)

(2) Heavy water reactors (HWR)

(3) High temperature gas-cooled reactors (HTGR)

(4) Liquid metal fast breeder reactors (LMFBR)

(5) Gas-cooled fast breeder reactors (GCFR)

(6) Steam cooled fast breeder reactors (SCFR)

(7) Molten salt breeder reactors

A 50-year period, from 1970 to 2020, was projected. Optimum cost cases are presented and the nuclear percentage of future total power capacity (installed after 1970) was estimated for each case.

Some of the ground rules and assumptions utilized in the analysis were:

(1) Nuclear power plant designs are based on technology feasible in 1967.

(2) The linear programming model obtains an optimal solution over the 1970-2020 period on the basis of minimal costs discounted at seven percent.

(3) Price inflation or escalation is not accounted for.

(4) LWR's are represented by PWR's.

(5) The rising uranium price schedule assumed is based on information from the A.E.C.'s Division of Raw Materials.

(6) Power plants built before 1970 will not be included in the study, but the effect of their omission is reflected in the capacity factor.

(7) LP model determines best load factor history for each power plant so that the present worth of cash flows will be minimized.

(8) LP model selects plants to minimize total discounted power cost over study period.

(9) U_3O_8 cost will be represented as a function of cumulative consumption and predicted reserves.

(10) Costs are based on June 1967 prices and privately owned utility practices.

(11) Nominal plant size will be 1000 Mwe.

(12) Power plant life will be 30 years.

The ground rules, selected in 1967, assumed a capital cost of $133/Kw for a 1000 Mwe pressurized water power plant and $100/Kw for a 1000 Mwe coal-fired plant. In the latter part of 1970 for the 1000 Mwe size range, the capital cost of a nuclear plant is about $250/Kw and a coal-fired plant, $195/Kw. While nuclear fuel costs have remained essentially constant, fossil fuel costs have increased dramatically and are now much higher than those used in the study. Since inflation applies to the entire electric power industry, it has been hypothesized in Ref. V-24 that inflation would not have a marked effect on the power plant mix developed in the study. Also in reviewing the cost estimates in the study, one should consider the relative cost differences from case to case and not their absolute magnitudes.

The major part of the LP model input data includes prediction of plant performance, capital costs, fuel cycle costs, and introductory dates for new reactor types. It was provided by the assigned task forces under the CNP program that were charged with the evaluation of light water reactors, advanced converter reactors, the liquid metal fast breeder reactor, alternate coolants to sodium for fast reactors, the role of thorium, and fuel recycle. The task forces generated this information on the basis of specific 1000 MWe reactor designs which were evaluated for each of the reactor concepts considered in the studies. These designs were based on information provided by proponents of the systems and therefore generally reflected their viewpoint and enthusiasm. The evaluations of the reactor types are presented in Refs. III-2 through III-4 and III-6.

The computer analysis of Ref. V-24 treated three categories of cases:

(1) Fossil only

(2) Combined nuclear and fossil

(3) Nuclear only

The following discussion is only concerned with results from the first two categories, and the reader is referred to WASH 1098 for the complete details of these detailed analyses. The case 1-A with only fossil fueled power plants was necessary to establish a reference power cost in a system without nuclear power. Performance in the model is based upon average capital costs of $100/Kw for coal-fired plants and $90/Kw for gas-fired plants and upon fuel costs ranging from 1.6 to 3.4 mills/kwh depending upon the geographical region. Two variations in the cost of power supplied by fossil fuel were also examined:

(1) Perturbation due to a 1 percent per year increase in fuel cost

(2) Perturbation due to an increase in capital costs by $10/Kw

Thus, the sensitivity of the reference fossil-fuel case to fuel and capital costs was obtained.

Seven cases, 2-A through 2-G, with both fossil and nuclear power plants available were computed. Thus, an assessment of the major possible developments in reactor technology in competition with fossil was obtained. The cases combining nuclear and fossil power plants are defined at the top of Table V-2.

TABLE V-2. RESULTS OF SYSTEMS ANALYSIS TASK FORCE CALCULATIONS FOR FOSSIL AND NUCLEAR PLANTS WITH RISING URANIUM COSTS

Case Number	1-A	2-A	2-B	2-C	2-D	2-E	2-F	2-G
Case Definition								
Plants Included	Fossil	Fossil LWR	Fossil LWR LMFBR (Reference Oxide only)	Fossil LWR LMFBR	Fossil LWR LMFBR HTGR HWOCR	Fossil LWR LMFBR HTGR HWOCR GCFR SCFR MSR	Fossil LWR HTGR LMFBR	Fossil LWR LMFBR HTGR
Characteristics of Optimum Solution								
Total Cost, 1970 Present Worth, billions 1970 to 2020 at 7% Discount Rate	216.3	205.3	201.4	189.0	185.6	178.9	179.8	187.7
Levelized Power Cost 1970 to 2020 mills/Kwh	4.939	4.688	4.599	4.316	4.238	4.085	4.106	4.286
Total Nuclear Capacity, Mwe								
1980		124×10^3	155×10^3	158×10^3	158×10^3	162×10^3	171×10^3	143×10^3
2000		484×10^3	760×10^3	$1,056 \times 10^3$	$1,059 \times 10^3$	$1,122 \times 10^3$	$1,155 \times 10^3$	$1,062 \times 10^3$
% of Total Capacity in 2000								
Fossil	100	64.2	44.4	21.9	21.6	17.0	14.6	22.7
LWR		35.8	22.2	20.9	7.8	8.2	10.7	8.4
HWR					7.4	8.0		
HTGR					20.7	5.9	25.1	44.4
LMFBR			33.2	57.2	42.5		49.6	24.5
GCFR						32.8		
SCFR						2.4		
MSR						25.7		
Total Uranium Used, Thousands of Tons U_3O_8 Through 2020		1906	1568	979	1255	1004	1199	1616
Max. Price through 2020, \$/lb U_3O_8		37.50	27.50	13.75	22.50	17.50	17.50	32.50
Ave. Price through 2020		16.73	12.00	11.34	14.04	11.70	12.99	16.32

Case 2-B considers only the reference oxide design of the LMFBR which would be available in 1980, thus limiting LMFBR development to an early design. Case 2-C allows for the full development of the LMFBR technology. Thus for this case the following sequence of plant introductions is considered:

(1) The reference oxide LMFBR in 1980

(2) The reference carbide design in 1984

(3) The advanced oxide design in 1990, and

(4) The advanced carbide design in 1994

Case 2-D considers the introduction of the advanced converters in 1976 which then displace the LWR to the extent allowed by new reactor growth. Case 2-E treats fossil plants and all of the reactor types. All cases up to this point consider that for each specific reactor type, the fuel costs are subject to both rising uranium costs and improving fuel technology, but the capital costs remain essentially fixed. Cases 2-F and 2-G, in addition to the uranium and fuel technology assumptions, include the effects of improving technology, experience, and unit size on capital costs. Both cases, therefore, use declining capital costs for the LWR's and HTGR's. Also, LMFBR capital costs are assumed to be constant at $135/Kwe and $165/Kwe, respectively, for Cases 2-F and 2-G.

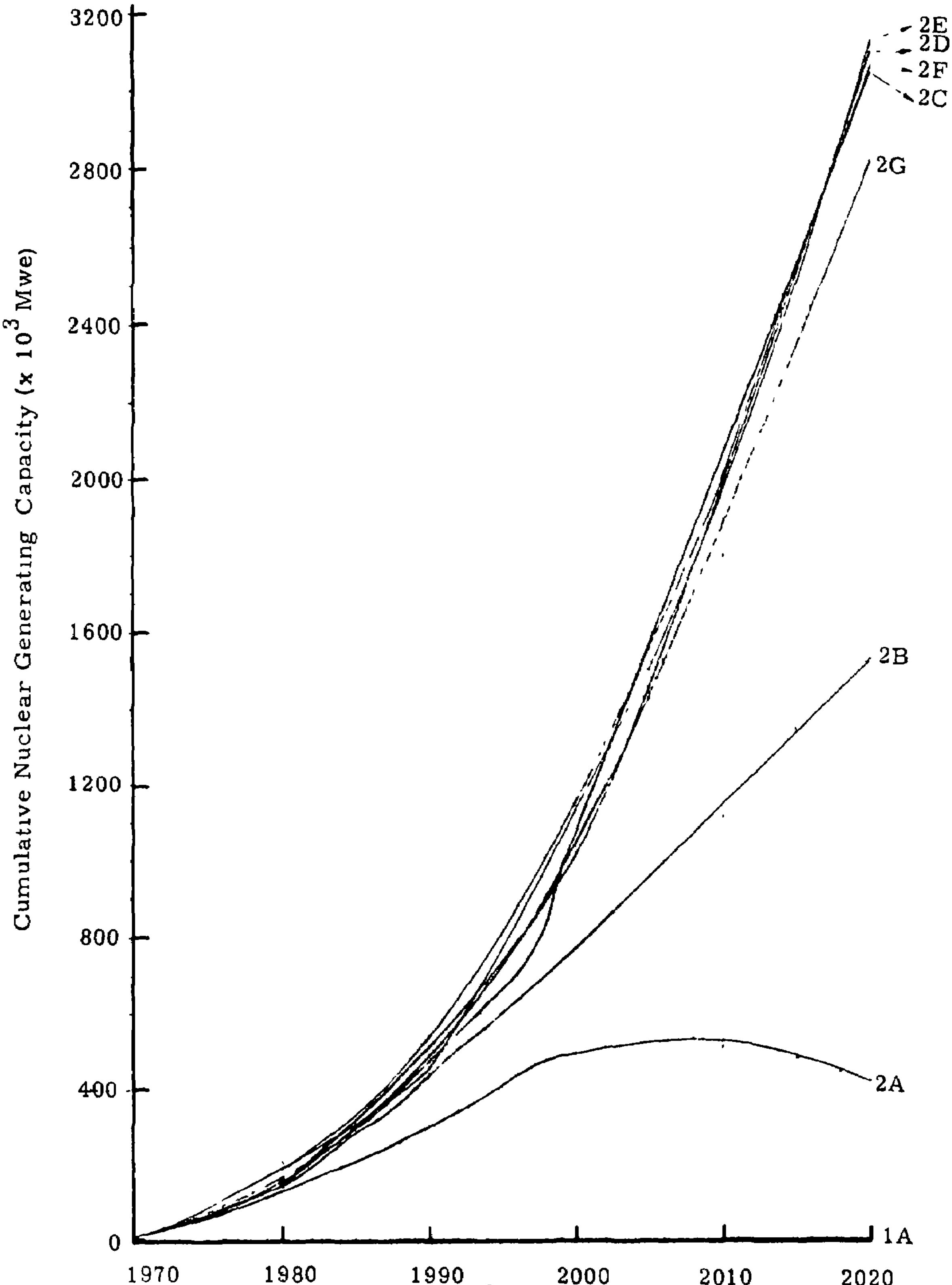
The results of the systems analysis employing the LP model are summarized in Table V-2. The most meaningful cost evaluation parameter shown in the table is the discounted cost of power. This is defined as the present worth of all power costs between 1970 and 2020. The levelized power cost in mills/Kw-hr is defined as the total present worth of all power costs divided by the present worthed energy production. Since the present worth of all energy is a constant for all cases studied, any solution which yields the minimum discounted cost of power must also give the minimum levelized power cost. For each case considered, the selection of reactor types was such that the total power costs would be minimized over the period 1970-2020. It is seen from Table V-2 that the minimum cost case is 2-E, an optimum combination of fossil plants and all types of reactors.

The total or cumulative nuclear capacity (Mwe) is given in Table V-2 for the years 1980 and 2000. Cumulative nuclear capacity for each case is also plotted in Figure V-4. The cumulative LWR sales through 1972 are incorporated into the model and consequently into all cases except 1-A: 12,000 Mwe by 1970 and 34,000 Mwe by 1972. These magnitudes are somewhat higher than the values read from the near-term installed capacity line of Figure V-1. The curve for Case 2-A results as shown because the LWR captures the market for heavy load power in every power supply area during the 1970's and 1980's. However, with a rise in uranium prices, the fossil plants recapture the market. Thus, LWR's supply the demand only in the high cost fossil areas and only through the mid 1980's. Table V-2 also presents for each case the uranium quantity used and U_3O_8 unit costs.

The nuclear percentage of the future total power capacity installed after 1970 is shown in Figure V-5. This percentage is itemized per reactor type and is shown for each case for the year 2000 in Table V-2. In the trivial case 1-A, this percentage is zero. Figure V-5 indicates that for Case 2-A the LWR's supply 45 percent of the total generating capacity (installed after 1970) by the year 1980, approximately 37 percent by the year 2000, and 15 percent by the year 2020.

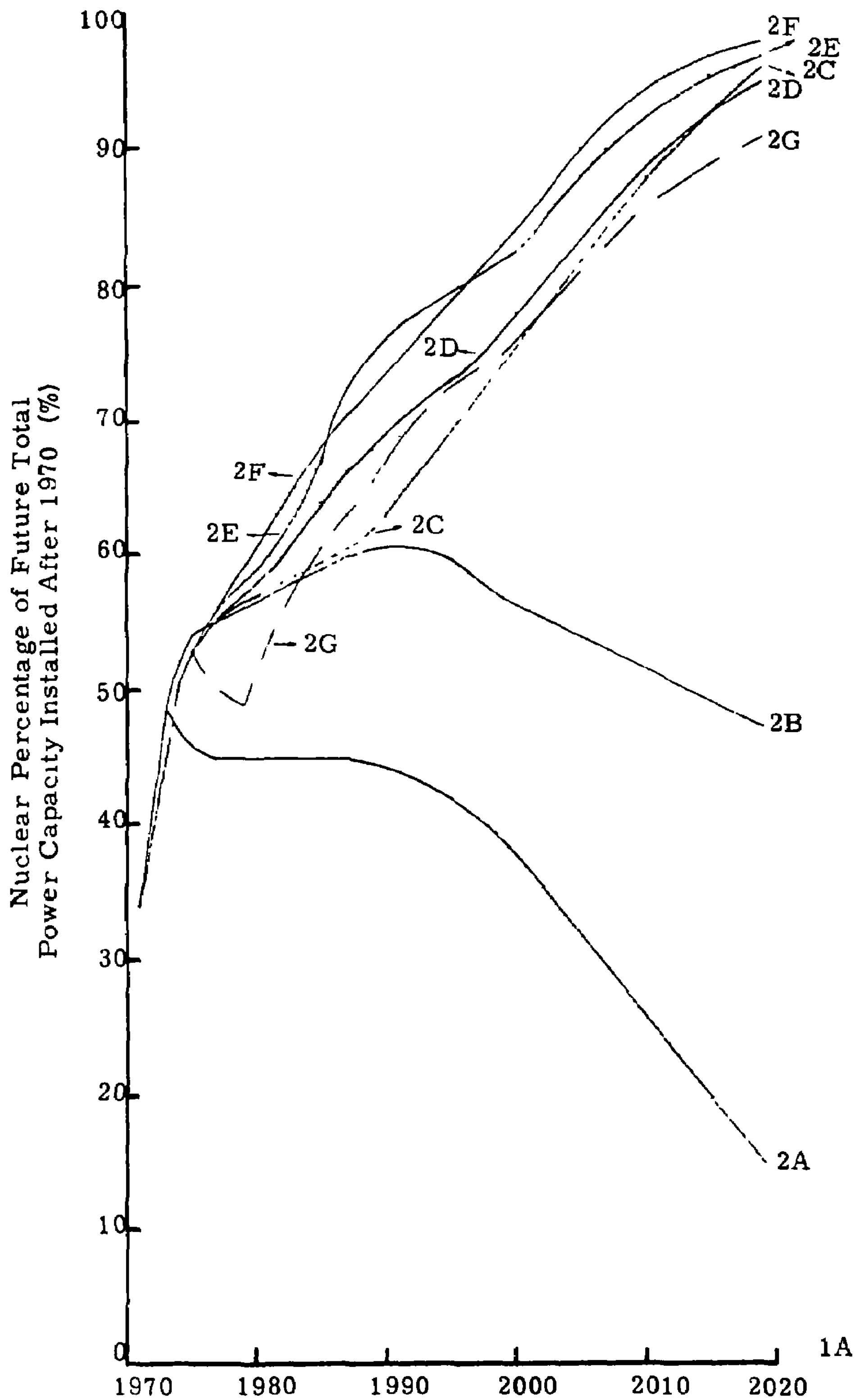
In Case 2-A, where only the LWR is available, increasing uranium prices force the LWR out of the optimum generating system. However, in Case 2-B the reference oxide LMFBR, by holding down the uranium consumption, makes possible an increased penetration of nuclear plants into the generating system (Figure V-4). Also after the year 2002 the reference oxide LMFBR produces plutonium for new reactors at a rate greater than that required by the economy. Thus, it will be economically attractive to burn the excess Pu in plutonium-fueled light water reactors. Consequently, it is calculated that about 100 reactors fueled by excess plutonium from the reference oxide LMFBR will be constructed by the year 2020.

In Case 2-C, the timely introduction of each successively, advanced LMFBR design causes not only that design to replace all previous LMFBR designs, but also breeder reactors to capture more of the total capacity market. Thus, by the year 2004, no more fossil plants are built for the nation's power economy. As in Case 2-B, overproduction of plutonium breeders causes plutonium-fueled light water reactors to be introduced after the year 2002.



Case 1A - Fossil
Case 2A - Fossil + lwr
Case 2B - Fossil + lwr + lmfbr (RO)
Case 2C - Fossil + lwr + lmfbr (ALL)
Case 2D - Fossil + lwr + lmfbr + advanced conv.
Case 2E - Fossil + lwr + hwr + htgr + gcfr + msr + scfr
Case 2F - Fossil + lwr + htgr + lmfbr ($135/KW)
Case 2G - Fossil + lwr + htgr + lmfbr ($165/KW)

Figure V-4. Cumulative Nuclear Generating Capacity



Case 1A - Fossil
Case 2A - Fossil + lwr
Case 2B - Fossil + lwr + lmfbr (RO)
Case 2C - Fossil + lwr + lmfbr (ALL)
Case 2D - Fossil + lwr + lmfbr + advanced conv.
Case 2E - Fossil + lwr + hwr + htgr + gcfr + msr + scfr
Case 2F - Fossil + lwr + htgr + lmfbr ($135/KW)
Case 2G - Fossil + lwr + htgr + lmfbr ($165/KW)

Figure V-5. Nuclear Percentage of Future Total Power Capacity Installed After 1970

In Case 2-D advanced converters are represented by the HTGR, the heavy water-moderated organic-cooled reactors (HWOCR) and the heavy water-moderated boiling-light water-cooled reactor (HWBLW) designs. They are introduced in 1976 and displace the LWR to the extent allowed by new reactor growth. The advanced converters displace most of the reference-oxide LMFBR's and some advanced-oxide LMFBR's. This requires more uranium than that saved by displacing the LWR's, so the uranium consumption by the year 2020 increases. When the advanced-carbide LMFBR becomes available, the advanced converter construction stops. As in Case 2-C, fossil-fueled plants are no longer built after the year 2004 because they are competing against the same advanced carbide LMFBR. The system begins to build Pu-fueled LWR's by the year 2002. After this time, nuclear plants in Case 2-D capture about the same percentage of the total installed market as in Case 2-C (Figure V-5).

Three additional designs (GCFR, MSR, and SCFR) were made available to the power economy in Case 2-E. The Pu-fueled molten salt converter (MSR) and the advanced gas cooled fast breeder (GCFR) were built in large numbers, while only a few steam-cooled breeders were built (SCFR). The advanced GCFR has a doubling time equal to that of the advanced LMFBR and an estimated \$20/Kw lower capital cost. Because of this lower capital cost, the GCFR is essentially the only breeder built in Case 2-E. The excess plutonium produced by the GCFR is consumed by the Pu-fueled MSR which thus displaces the LWR. In Case 2-E both advanced converters (HTGR and HWR) are made available but to a lesser extent than in Case 2-D. They are eventually displaced by the MSR.

Cases 2-F and 2-G were designed to assess how the LMFBR would offset power costs when the LWR and HTGR plant costs are decreasing with time. In Case 2-F the capital cost of the LMFBR is assumed to be \$135/Kw, that of the HTGR to drop from \$122 to \$102 per Kw, and that of the LWR to drop from \$130 to \$104 per Kw. In this case, the HTGR's are built at the maximum rate, and the LWR's are initially built in great numbers so that in 1980 the total nuclear capacity reaches 171,000 Mw (Figure V-4). No LWR's are built after the 1982-83 period, and thereafter the HTGR and the LMFBR share the nuclear market until the year 2000. The HTGR has lower power costs than either the reference oxide or reference carbide LMFBR's, but these are built at the maximum allowable rate in order to function as plutonium producers. However, since the building rate of LMFBR's is limited by plutonium availability, the number of HTGR's being built in the 1980's is large. Even in the early 1990's, when the advanced oxide LMFBR becomes available, the HTGR's are still being built. It is worthwhile to note that no LWR's are built to provide plutonium in this period, even though their capital costs are approximately the same as those of the HTGR's. The reason for this is that the rising uranium costs ($11/lb by 1988 and $17.50/lb by 2000) make the LWR's too costly. Although the advanced carbide LMFBR essentially supplies all new capacity after 1994, some HTGR's are still being built in the 1996-97 period becuase of plutonium limitations. However, these temporary plutonium shortages do not prevent a plutonium surplus in the nuclear system around the year 2000, and LWR's are built again as plutonium burners. The discounted power cost to year 2020 is $179.8 billion. This is $5.8 billion less than in Case 2-D. The nuclear share of the power economy market is about 3 percent more than in Case 2-D by 1980, about 8 percent more by 2000, and about 3 percent more by 2020.

Increasing the capital cost of the LMFBR to 165/Kw in Case 2-G makes it advantageous to build primarily HTGR's from 1982 to 1995 because the HTGR's have an assumed capital cost of about $50/Kw less than that of the LMFBR's. Although some reference carbide and advanced oxide LMFBR's are also built as plutonium producers, HTGR's are built in large numbers as late as year 2004 because of plutonium limitations. The advanced carbide LMFBR design, which would capture all the nuclear market after its introduction, is held back by a shortage of plutonium. The system in Case 2-G does not obtain a surplus of plutonium until 2008, at which time LWR's are built again as plutonium burners. What makes nuclear capacity in year 1980 drop to 143×10^3 Mwe is increased uranium prices. This uranium price increase is due to increased uranium requirements brought about by large numbers of LMFBR's being displaced by HTGR's, and so making it uneconomical to build more LWR's. Nuclear power plants in Case 2-G capture about the same market by year 2000 as in Case 2-D and then lose about 4 percent of it by year 2020. The discounted power cost in Case 2-G increases to $187.7 billion, which is $7.9 billion more than in Case 2-F.

The advanced carbide LMFBR, even at a capital cost of $165/Kw, supplies the required generating capacity upon its introduction. However, the incentive for its rapid buildup is much greater at $135/Kw than at $165/Kw. Case 2-F reflects this in the increased number both of oxide and carbide LMFBR's and of plutonium-supplying LWR's and in the decreased number of HTGR's.

Cases 2-F and 2-G also indicate that a high-performance breeder such as the advanced carbide LMFBR with its low inventory, short doubling time, and insensitivity toward changing fissile material costs can:

(1) Probably capture a major portion of the market even with capital costs that are about $65/Kw greater than those of fossil-fueled plants and about $40/Kw greater than those of the HTGR or LWR

(2) Rapidly approach and meet the total electrical demand, even in the absence of a large number of plutonium producing reactors that supply plutonium for its growth

In summary then, Cases 2-A through 2-G demonstrated that fossil-fueled power plants and LWR nuclear plants only can produce moderate total savings. In order for nuclear plants to capture and hold a major portion of the electrical power market and provide discounted savings of $30-40 billion by the year 2020, high-performance breeders will be required. It appears (Figure V-5) that these high-performance breeders will permit nuclear plants to capture over 70 percent of the market (installed capacity after 1970) by year 2000 and over 90 percent by year 2020, assuming that fossil-fuel prices remain constant and that capital cost differentials between nuclear and fossil-fueled plants range between $20 and $50/Kw. Advanced converters offer savings when capital costs are constant and the uranium prices are rising.

The nuclear power forecasts shown in Figure V-5 show the strong dependency upon reactor type technology and development. These can be compared to the forecasts in Figure V-2. In the cases treating nuclear power only, it was shown that a six-year delay in the anticipated schedule of fast breeder development results in a substantial decrease in the savings associated with the breeder.

C. Fuel Availability

1. Uranium

Estimates of uranium resources available as of January 1, 1970 are shown in Table V-3 (Ref. V-24). Reasonably assured resources are in known ore deposits and occur in such grade, quantity, and configuration that they can be profitably produced with current technology at the given prices. At $8/lb and $10/lb prices, reasonably assured resources can be considered equivalent to ore reserves in the usual sense. Data on resources in the higher price ranges are not as well developed, and the estimates are less accurate.

Estimated additional resources are in the extensions of known deposits or in undiscovered deposits in known or postulated uranium districts. Estimated resources in the price category of $15/lb and lower are almost entirely tabular ore bodies in sedimentary rocks of the Western States.

By-product resources include uranium recoverable in conjunction with production of phosphoric acid, principally in Florida, and uranium recoverable from copper leach solutions. Projected production from such by-product sources, estimated to be available through the year 2000, is included in the estimates. Some of this by-product is expected to be available at prices in the $8-$10/lb range.

Research work indicates that uranium recovery from the oceans (about 4000 million tons of U_3O_8) may be possible at the high price range. Although there are no foreseen needs to exploit these resources, they do represent a virtually inexhaustible source of high-cost fuel which eventually could be used in breeder reactors if lower cost resources become exhausted.

The quantities of U_3O_8 fuel required each year were projected by Gambs and Routh (Ref. V-21). These are presented in Table V-4:

TABLE V-3. ESTIMATED U. S. URANIUM RESOURCES

January 1, 1970

Cumulative Thousands of Tons of U_3O_8

	Reasonably Assured			Estimated Additional			
U_3O_8 Price Per Pound	Conventional Deposits	By-Product	Total	Conventional Deposits	By-Product	Total	Total
$ 8.00	204	--	204	390	--	390	594
10.00	250	90	340	600	--	600	940
15.00	390	110	500	950	--	950	1,450
30.00	530	110	640	1,600	--	1,600	2,240
50.00	5,400	110	6,000	4,000	--	4,000	10,000
100.00	11,400	110	12,000	13,000	--	13,000	25,000

TABLE V-4. QUANTITIES OF U_3O_8 FUEL REQUIRED EACH YEAR

Year	U_3O_8 Quantity, tons/yr
1965	200
1970	600
1975	9,000
1980	30,000
1985	65,000
1990	100,000
1995	140,000
2000	150,000

The cumulative quantity of U_3O_8 fuel required up to the year 2000 was determined from these data to be 2100 x 10^3 tons. Thus from Table V-3 one sees that the uranium resources are abundant enough in the U.S. such that it is not necessary to consider any material above a cost of \$50/lb.

2. Thorium

Although there has been little demand for thorium to date, and hence little prospecting for it, the estimated U.S. thorium resources are large, as shown in Table V-5. They are located predominantly in Lemhi County, Idaho, and in neighboring Montana, where preliminary investigation indicates the presence of deposits containing some 100,000 tons of ThO_2 from which thorium can be produced at prices comparable to present-day prices for uranium.

TABLE V-5. ESTIMATED U.S. THORIUM RESOURCES

	Millions of Short Tons		
\$/lb ThO_2	Reasonably Assured	Estimated Additional	Total
10	0.1	0.5	0.6
30	0.2	0.6	0.8
50	3.2	7.6	10.8
100	11.2	24.6	35.8

3. <u>Bred Fuels</u>

The bred fuels, plutonium and U-233, are produced by certain reactor types and are consumed by others. The LMFBR and GCFR reactors use uranium-238 fuel and breed plutonium-239 which can then be consumed in Pu-fueled LWR's. Thermal breeder reactors use thorium-232 fuel and breed uranium-233.

REFERENCES

Basic References on Forecasts of Nuclear Power Requirements

V-1. "Statistical Year Book of the Electric Utility Industry for 1969," Edison Electric Institute, New York, September 1970.

V-2. "National Power Survey," The Federal Power Commission, Washington, D.C., 1964, 1969, and 1970. (NPS)

V-3. "The Economy, Energy and the Environment," Joint Economic Committee, Congress of the U.S., September 1, 1970.

V-4. Landsberg, H.H., et.al., "Resources in America's Future, Patterns of Requirements and Availabilities 1960-2000," The Johns Hopkins Press, Baltimore, 1963. (RAF)

V-5. "Nuclear Power and the Environment," U.S. Atomic Energy Commission, Division of Technical Information, 1969, p. 6.

V-6. Roddis, L.H., "The Future of Nuclear Power," Science and Technology Advisory Council to the Mayor, New York, New York.

V-7. "Considerations Affecting Steam Power Plant Site Selection," Office of Science and Technology, December 1968.

V-8. "Research and Development Problems in Power Plant Siting," Office of Science and Technology, Washington, D.C., 1970 (to be published).

V-9. Civilian Nuclear Power, The 1967 Supplement to the 1962 Report to the President, February 1967, U.S.A.E.C. (CNP)

V-10. Sporn, P., "Nuclear Power Economics -Analysis and Comments-1967," pp. 2 20, Joint Committee on Atomic Energy (U.S.) Report Nuclear Power Economics-1962 thru 1968, February 1968.

V-11. "Cost Benefit Analysis of the U.S. Breeder Reactor Program," WASH 1126, U.S.A.E.C. Division of Reactor Development and Technology, April 1969.

REFERENCES

Basic References on Forecasts of Nuclear Power Requirements (Continued)

V-12. "An Energy Model for the U.S.," Bureau of Mines, U.S. Department of Interior, Report IC 8384, July 1968. (EMUS)

V-13. Michael C. Cook, "Energy in the U.S., 1960-1985," Sartorius & Company, September 1967. (EUS)

V-14. "Projection of the Consumption of Commodities Producible on Public Lands of the U.S., 1980-2000," Robert J. Nathan Assoc., May 1968. (PCCP)

V-15. "Forecast of Growth of Nuclear Power," WASH 1084, U.S.A.E.C. Division of Reactor Development and Technology, December 1967. (FGNP)

Additional and More Recent References Giving Nuclear Power Forecasts

V-16. "A Review and Comparison of Selected United States Energy Forecasts," Prepared for the Office of Science and Technology by Pacific Northwest Laboratories of the Battelle Memorial Institute, December 1969.

V-17. "United States Petroleum Through 1980," United States Department of Interior, Office of Oil and Gas, July 1968. (USP)

V-18. Vogely, W.A., "Patterns of Energy Consumption in the U.S.," United States Department of Interior, Division of Economic Analysis of the Bureau of Mines, 1962. (PEC)

V-19. "Competition and Growth in American Energy Markets, 1947-1985," Texas Eastern Transmission Corporation, 1968. (CGAEM)

V-20. Gambs, G.C., "The Electric Utility Industry: Future Fuel Requirements 1970-1990," Mechanical Engineering, April 1970, pp. 42-48.

REFERENCES

Additional and More Recent References
Giving Nuclear Power Forecasts (Continued)

V-21. Gambs, G.C. and A.A. Routh, "The Energy Crisis," Chemical Engineering, May 31, 1971, pp. 56-68.

V-22. Earhart, J. P., "Basis for Projected Air Pollution from Combustion of Fossil Fuels in the U.S.," Air Pollution Control Office of the Environmental Protection Agency, June 1970.

V-23. Yarosh, M.M., "Changing Emphasis in the Siting of Steam Electric Power Plants," A.S.M.E. Paper No. 70-WA/Ener-12, November 1970.

V-24. "Potential Nuclear Power Growth Patterns," WASH 1098, U.S. A.E.C. Division of Reactor Development and Technology, December 1970.

VI. SAFETY CONSIDERATIONS, SPECIAL LICENSING REQUIREMENTS, DELAYS

Due to the potential hazards of radiation, the safeguards which are engineered into nuclear plant systems are multiple redundant, fail-safe, and subject to rigorous exhaustive and frequent testing, making the operation of a nuclear power plant among the safest and most reliable of power generating systems in terms of safety considerations. During the preliminary and final licensing procedures of nuclear power plants, these safety systems are reviewed several times for assurance of operating safety. However, because of the engineering design and lengthy evaluation of these systems, a significantly larger amount of engineering time is required to plan such systems and, because of the multiplicity of safety measures, often interlocked, substantially more time is required to construct and test a nuclear plant prior to placing it in service.

A nuclear-fueled steam electric plant, because of the nature of its fuel and the by-products of the fission reaction, comes under the licensing control and regulatory authority of the Atomic Energy Commission (Ref. VI-1). In order to protect the public and the environment, these regulations and operating guidelines are rigorously enforced and observed.

Fossil-fueled steam electric plants release to the environment the products of combustion, including particulates, carbon monoxide, carbon dioxide, nitrogen, and sulfur oxides, and others. A nuclear-fueled plant releases none of the above contaminants to the environment, but does release very minor amounts, carefully monitored and regulated, of the products of fission, including tritium, iodine-131, krypton-85, xenon-133, and other noble gases and their daughter products. The need to monitor, regulate, and control these radioactive materials within the nuclear plant and upon release to the environment predicates a highly sophisticated waste-product handling and monitoring system unlike any safety system found in fossil-fueled plants and invariably results in substantially greater initial capital investment and in greater construction complexities and time periods.

Presently, the construction of a large fossil generating unit may take from four to five and one-half years, depending on several factors such as the need for additional capacity, the region of the country, and the fuel to be burned. The construction of a nuclear power plant may take as long as seven years to construct, partially due to stringent licensing procedures and complexity of construction, but also attributable in several instances to the intervention of groups or individuals intent on the fullest measure of protection to the environment. Because of the predictable delays due to various causes and because of the increasing frequency of hearing and review delays through interventions, the placing of a nuclear unit on-line may take up to seven years or more. Further, because of the growing backlog of review necessary to license a nuclear plant, and because of the Atomic Energy Commission's new responsibility in the enforcement of the National Environmental Protection Act, it can be anticipated that even longer periods of time will be required in the future to place a nuclear plant in service. Utilities have recently adopted the practice of including a delay time in their planning schedule and, in several recent cases, have allowed as much as 10 years for the construction and licensing of a nuclear unit.

REFERENCE

VI-1. Section 10, Code of Federal Regulations, Parts 10, 20, 50.

VII. ADVANTAGES AND DISADVANTAGES OF NUCLEAR POWER

The advantages and disadvantages of nuclear power generation versus other types of generation are presented. Then the advantages and disadvantages of the various nuclear reactor types are discussed. The types of nuclear power reactors were listed in Table III-1 and described in Section III.

In Section V, it was shown that the forecast of power supplied by nuclear reactors in the future will be an increasing function if nuclear reactor technology is allowed to develop in an optimum manner. Thus a major advantage is that adequate sources of uranium and thorium will be utilized as fuel. The demand on the fossil fuel resources of coal, oil, and gas would be significantly alleviated. Ideally, by the year 2000 fossil fuel plants could possibly supply under 20 percent of the cumulative electric generating capacity installed after 1970. In addition, the class of breeder reactors would be breeding plutonium fuel for use in other nuclear power reactors.

Fewer fossil-fueled plants installed in the future would also reduce the air pollutants emitted: sulfur, nitrogen oxides, and ash. Nuclear power reactors are not contributors to the air pollution problem. However, there are the problems of nuclear waste disposal and thermal pollution.

The relative merits between the types of nuclear reactors are presented by considering each class in a separate section. The advantages and disadvantages of the particular class is considered with respect to any other type.

A. Light Water Reactors

The competitive marketability of LWR's depends strongly on the continued availability of uranium fuel at low cost. If the cost of uranium rises in future years, LWR's will be in a disadvantageous situation.

The loss of coolant accident resulting from a primary system rupture has been, and continues to be, a subject for accident analysis in LWR licensing and is a design basis for emergency systems to maintain core cooling.

B. Advanced Converters

As a class, these reactors have two main potential advantages over present LWR designs in that:

(1) Less uranium ore is required to produce the same amount of energy

(2) Estimated power costs are lower

Most of the advanced converter concepts have some close similarity with the light water concept in their reactor and fuel designs and in fuel processing requirements. However, the HTGR concept is an entirely different reactor and fuel concept that will require an almost entirely new and separate base of supporting industry (fabrication, reprocessing, and component manufacture) to sustain it. Some recognition of the difficulty of establishing the HTGR as a viable concept because of this unique problem would be appropriate. However, the HTGR appears to offer the most cost and fuel resource incentives. It also has the earliest date of commercial application for the advanced converters.

Heavy water availability, price, and risk of a major heavy water loss during reactor operation are major factors affecting the marketability of heavy water reactors. With present known and developed processes, the capital and operating costs of heavy water production plants can be predicted fairly well. Large plants operating at capacity for their useful life would be required to realize the assumed low D_2O price of $17.50/lb.

The major D_2O requirement for a heavy water reactor is in supplying the initial inventory. A high D_2O production rate and low cost thus implies a high rate of construction of new heavy water reactors. This results in a situation in which large D_2O plants cannot be justified without sound evidence that a high rate of heavy water reactor construction will prevail, and such construction commitments cannot be justified without firm low price commitments for D_2O production, implying large-capacity D_2O plants. Thus, a kind of mutually exclusive closed cycle is established and, because of this, the assumed cost of $17.50/lb for heavy water is thought to be optimistic.

The risk of a large heavy water loss and the insurance cost to cover the risk is another factor that appears to have a strong deterrent effect today on acceptance of the heavy water reactor concepts. Insurance costs as high as 20 percent per year have been mentioned and some recognition of this difficulty would seem appropriate.

The enriched uranium fueled HWOCR has economic advantage over the natural uranium fueled one. The use of thorium in the HWOCR has less economic potential than for the other advanced converters and LWR's. The HWOCR designs are quite sensitive to coolant temperature. For example, a decrease in outlet temperature of 25°F results in a 0.2 mill/Kw-hr increase in the energy cost. This illustrates how sensitive the designs are to engineering judgments and points out the difficulty of comparing such reactor plants. Another instance of design sensitivity is illustrated by the mixer-strippers for the HWBLWR plant. Without these mixer-strippers, which yet remain to be demonstrated, the reactor is economically unattractive.

C. Thorium Systems

The transition from the relatively crude MSRE to a much more complex full-scale breeder reactor requires an extensive R&D program including scaleup of components. The MSBR pump design flow rates and power density would be considerably greater than those in the MSRE. While individual facets of the technology may have been investigated in the MSRE as well as other reactor systems, e.g., HFIR, EBR-II, and Dounreay Fast Reactor, it is only by integrating all the various components and systems in an adequately sized reactor experiment under conditions similar to those existing in the actual breeder that the true operating characteristics and potential of the molten-salt reactors will be determined. To achieve this it would be necessary to construct a power-producing reactor which would furnish data on fuel processing, breeding ratio, and secondary coolant behavior that must be known before the MSBR can be built commercially with confidence. At the present time the single fluid breeder concept is in the very early design stage. Thus development of a finalized detailed design of the concept is necessary before a thorough evaluation can be completed.

While there are no indications that dynamic instabilities will occur, the dynamic behavior of the system is very complicated, and further accurate and detailed analysis and experimental work are needed for designing a self-regulating system that is stable for constant power and also for transient and load-following behavior.

Pumps and heat exchangers appear to be critical components. While the MSRE and experimental salt pumps have successfully logged thousands of hours of molten-salt operation and the MSRE heat exchangers have also operated successfully for thousands of hours, scale-up to MSBR size and modifications in design required for the MSBR operating conditions will have to be demonstrated. The presence of radioactivity, the need for adequate pressure relief against high-pressure steam, and salt cleanup problems in case of tube leakage appear to be some of the design and maintenance complications.

Remote maintenance of a molten-salt fluid-fuel reactor is required due to the presence of intense gamma radiation in the equipment outside the reactor caused by activation of sodium and fluorine in the salt, the presence of fission products, and activation of the structural material by delayed neutrons in the circulating salt. Pumps and heat exchangers will have to be capable of long maintenance-free life, as no practical reactor system could tolerate too many shutdowns due to failure of large components.

D. Breeder Reactors

A big advantage of the breeder reactor is that it produces fissionable material to refuel itself and in addition to fuel another reactor. The economic potential of fast breeder reactors lies mainly, but not entirely, in the fact that they would conserve resources of nuclear fuel. Also, because fast breeder reactors will operate at far higher temperatures than are encountered in contemporary water reactors, they will have greater thermodynamic efficiency. Today's light water reactors operate at an overall efficiency of about 32 percent. Modern fossil-fueled plants operate at about 39 percent efficiency. Hence, light water reactors add more waste heat to the environment per unit of electrical energy produced than fossil-fueled plants do. Fast breeder reactors will probably attain efficiences equal to that of the most modern fossil-fueled plant, thereby reducing the nuclear waste-heat problem.

In the LMFBR, the sodium coolant has excellent heat transfer characteristics. Moreover, it can be used at a fairly low pressure even though it emerges from the reactor at a temperature (about 500 degrees Celsius) that with water would give rise to high pressures. Indeed, the sodium pressure arises solely from the force required to maintain the high rate of flow through the maze of tubes in the core and the blanket. Compared with coolants such as water and gas, sodium requires low pumping power. It is not particularly corrosive to the reactor. Because of the inherently low pressure of the sodium coolant, the reactor vessel and its associated piping need be designed to withstand only moderate operating stresses, in marked distinction to the pressure vessels and other primary-system components of a pressurized-water reactor, a boiling-water reactor or a gas-cooled fast reactor.

Sodium does have certain disadvantages that markedly influence the design of a reactor. Since sodium is opaque, provision must be made for the maintenance and refueling of the reactor without benefit of visual observation. Sodium is, of course, highly reactive chemically, and it becomes intensely radioactive when it is exposed to neutrons, even though its "cross section," or neutron-absorption capacity, is relatively low. Hence, the sodium must be kept out of contact with air or water, and radiation shielding must be used to protect workers who are near sodium that has been through the core and blanket of an operating reactor.

Fast breeder reactors cooled by gas or steam are technically feasible in that no research and development breakthroughs are required. However, GCFR performance is strongly dependent upon the behavior of the fuel cladding. Thus, a demonstration of the satisfactory operation of some type of pressure-equalized cladding is necessary prior to any more comprehensive effort to develop the GCFR.

Both steam- and gas-cooled reactors are capable of operating with breeding ratios of interest. The gas-cooled breeder reactor, due to its harder neutron spectrum, can achieve the highest breeding ratio. Breeding ratios of low pressure steam-cooled reactors can be attractive, if somewhat lower. The conversion ratios attainable in supercritical pressure steam-cooled reactors, on the other hand, can be compared with those achieved in the advanced converter reactors. The loss-of-coolant accident or the loss-of-flow accident (more properly, depressurization) is a major consideration in the safety analysis of reactors operating with pressurized coolant in that forced circulation of coolant must be guaranteed. Nevertheless, it appears that engineered safeguards could cope with realistic depressurization rates in the gas- and steam-cooled reactors.

On the basis of the designs evaluated (Table III-6) and the combined criteria of low power costs and good breeding capability, GCFR's have the most potential of the concepts considered. Steam-cooled reactors suffer either from higher power costs (1250 and 2680 psi SCBR's) or low breeding ratio (3700 psi SCBR).

There is interest in the development of SCFR's because of the possibility that breeding ratios comparable with those of liquid-metal-cooled fast breeder reactors (LMFBR) can be obtained with a system utilizing the large background of light water reactor technology and utility experience with steam systems. The steam cooling technology is simpler than that for sodium. Since steam-cooled reactors operate on a direct cycle, primary heat exchangers are not required, and a simpler steam supply system results. Additional incentives include the elimination of chemical reactions with the coolant, the possibility of flooding the reactor with a transparent medium (water) during refueling or maintenance operations, and the possibility of using water spray systems for emergency cooling.

In the GCFR, helium gas at a pressure of from 70 to 100 atm is used to transport the heat from the reactor core to the steam generators. Since the gas does not become radioactive and cannot react chemically with the water in the steam generator, there is no need for an intermediate heat exchanger. The resulting simplification of the system is a helpful offset against the need to design for a higher coolant pressure with gas.

The use of helium as a coolant has other special advantages for a fast breeder reactor. Helium does not interact with the fast neutrons in the reactor core, resulting in both simplified control of the reactor and enhanced breeding of new fissionable fuel from fertile material. In addition, helium is transparent and chemically inert, providing visibility during refueling and maintenance operations, a simpler engineering design and freedom from corrosion problems.

VIII. EFFECT OF NUCLEAR POWER ON NATIONAL AIR QUALITY

From the information used and developed in this report, it is clear that an increasing preponderance of new steam generating stations will be nuclear-fueled. During the next three decades, nuclear power plants will increase from three percent to well over 40 percent of generating capacity, while fossil units will increase on an absolute basis and decrease on a percent basis. Fossil-powered plants supplied approximately 260,000 megawatts of capacity in 1970 and are forecasted to supply approximately 400,000 megawatts in 1980, 570,000 in 1990, and 740,000 in 2000. Typically, a large, relatively new fossil plant operated at about a 64 percent load factor in 1970 (Ref. VIII-1). This typical load factor can be expected to remain reasonably stable through the next several decades although some fluctuations due to improved transmission and dispatching system, greater plant reliability, and other technological and management developments may be expected. These factors, which would generally tend to increase load factors, would be counterbalanced by the increased use of nuclear plants for base-load and relegation of fossil stations to load-following service.

From relatively scattered sources (Ref. VIII-2), nuclear plants may be expected to operate generally at load factors in excess of 80 percent as base-loaded facilities. With increasing numbers of nuclear plants coming into service over the next 30 years, the nationwide average load factor may demonstrate a gradual trend upward.

The effect of the use of nuclear power on fossil steam generating emissions to the environment is of primary importance in the results of this study effort. Therefore, extensive efforts were extended to derive reasonable forecasts of nuclear and fossil steam capacities over the period of interest. Data from Figures V-2 and V-3 were used as a basis for the forecasts of total, nuclear, and fossil steam capacities. The following points were considered:

- With minor exceptions, the forecasts of total electrical power requirements through about 2020 were in quite good agreement. Therefore, it was assumed that the midline of

Figure V-3 and the extrapolation of it represented total electrical power capacity requirements. The values thus derived are shown in Table VIII-1.

- As shown in Figure V-2, there are substantial differences in nuclear generating capacity forecasts, largely depending on the year the projection was made. To provide some basis for comparative purposes, it was assumed that the extreme upper and lower values were too far off the midline curve to be truly meaningful and representative and could be disregarded.
- High, middle, and low values of nuclear capacity and the extrapolations of these curves were then derived, excluding extreme values. The results of this derivation are also presented below in Table VIII-1.
- It was assumed that fossil fuel usage never falls below 500,000 megawatts of capacity after 2000, since these plants may be expected to be used continuously as peaking units and in isolated areas with too low power demand for economical nuclear generation.
- From forecasted total capacity, and the high, middle, and low values of nuclear power, the corresponding low, middle, and high fossil capacities were derived. These values are included in Table VII-1.
- Based on proposed national performance standards for stationary sources promulgated by EPA (Ref. VIII-3), it was assumed that the average fossil-fueled station will emit approximately 10 pounds of sulfur dioxide per megawatt-hour, corresponding to a one percent sulfur content of fuel (i.e., coal and oil) and a mix of approximately 80 percent coal, 10 percent oil, and 10 percent gas as fossil fuel throughout the time period. The above emission value is approximately equivalent to 0.8 pounds of sulfur dioxide per million Btu of liquid fuel and 1.2 pounds of sulfur dioxide per million Btu of solid fuel. This assumption does not allow for the fact that methods of

TABLE VIII-1. CAPACITY FORECASTS BY FUEL

	1970	1980	1990	2000	2010	2020	2030	2040	2050
Total forecast steam capacity (x10^3 Mwe)	270	500	930	1410	2310	3300	4500	5900	6900
Nuclear capacity forecasts (x 10^3 Mwe)									
High (Case I)	10	190	650	910	1810	2800	4000	5400	6400
Middle (Case II)	10	105	350	700	1020	2000	3000	4200	5000
Low (Case III	10	50	120	240	470	650	900	1100	1400
Fossil capacity forecasts (x10^3 Mwe)									
Low (Case I)	260	310	555	500	500	500	500	500	500
Middle (Case II)	260	395	580	710	1290	1300	1500	1700	1900
High (Case III)	260	450	810	1170	1890	2650	3600	4800	5500

TABLE VIII-2. TONS OF SO_2 EMITTED PER YEAR (x10^6)

Case I (low fossil use)	8.8	10.8	14.3	14.3	14.3	14.3	14.3	14.3
Case II (middle fossil use)	11.3	16.5	20.2	36.8	37.1	42.8	48.5	54.2
Case III (high fossil use)	12.8	23.1	33.4	53.9	75.5	102.6	136.8	156.8

reducing sulfur content of fuels (e.g., coal gasification) may substantially reduce sulfur dioxide emissions. Using such a method, emissions would be reduced proportionally to the amount of sulfur removed. The important point is in the relative differences caused by varying amounts of installed nuclear capacity.

- A constant plant load factor of 65 percent was assumed throughout.

Based on the primary data and the above-listed assumptions and premises, three cases of fuel distribution were formulated. Case I assumes maximum nuclear capacity and minimum fossil capacity. Case II assumes middle values of both nuclear and fossil. Case III assumes minimum nuclear and maximum fossil capacities. In the table, the sum of Case I values equals the total forecasted capacity requirement. Similarly, the sums of Case II and Case III values also equal total capacities. Finally, values of sulfur dioxide emissions were derived, based on the assumptions outlined above. This information for the three cases is shown in Table VIII-2 and, more graphically, in Figure VIII-1.

It should be again noted that the results shown in Table VIII-2 presume no change in allowable sulfur dioxide emission rate through the period of interest. Throughout the 80 years considered, sulfur dioxide emissions were assumed to remain constant at 10 pounds per megawatt-hour. Any technological or legislative changes which could affect fossil fuel sulfur dioxide emissions would simply change the total emissions by a proportional amount. For example, halving the allowable equivalent fuel sulfur content from one percent to one-half percent would correspondingly reduce total emissions in Table VIII-2 by one-half.

From the results of the above analysis, it is significant to note the effects of differences in nuclear power derived from the forecasts noted. For example, in as short a time as 10 years, in 1980, the national emissions from fossil steam generation could be slightly less than nine million tons per year if the more optimistic forecasts of nuclear power supply are realized. On the other hand, if the pessimistic (low) expectations of nuclear power supply are met, the national sulfur dioxide emissions could be as high as 13 million tons per year. This trend continues through the forecast period until the emission

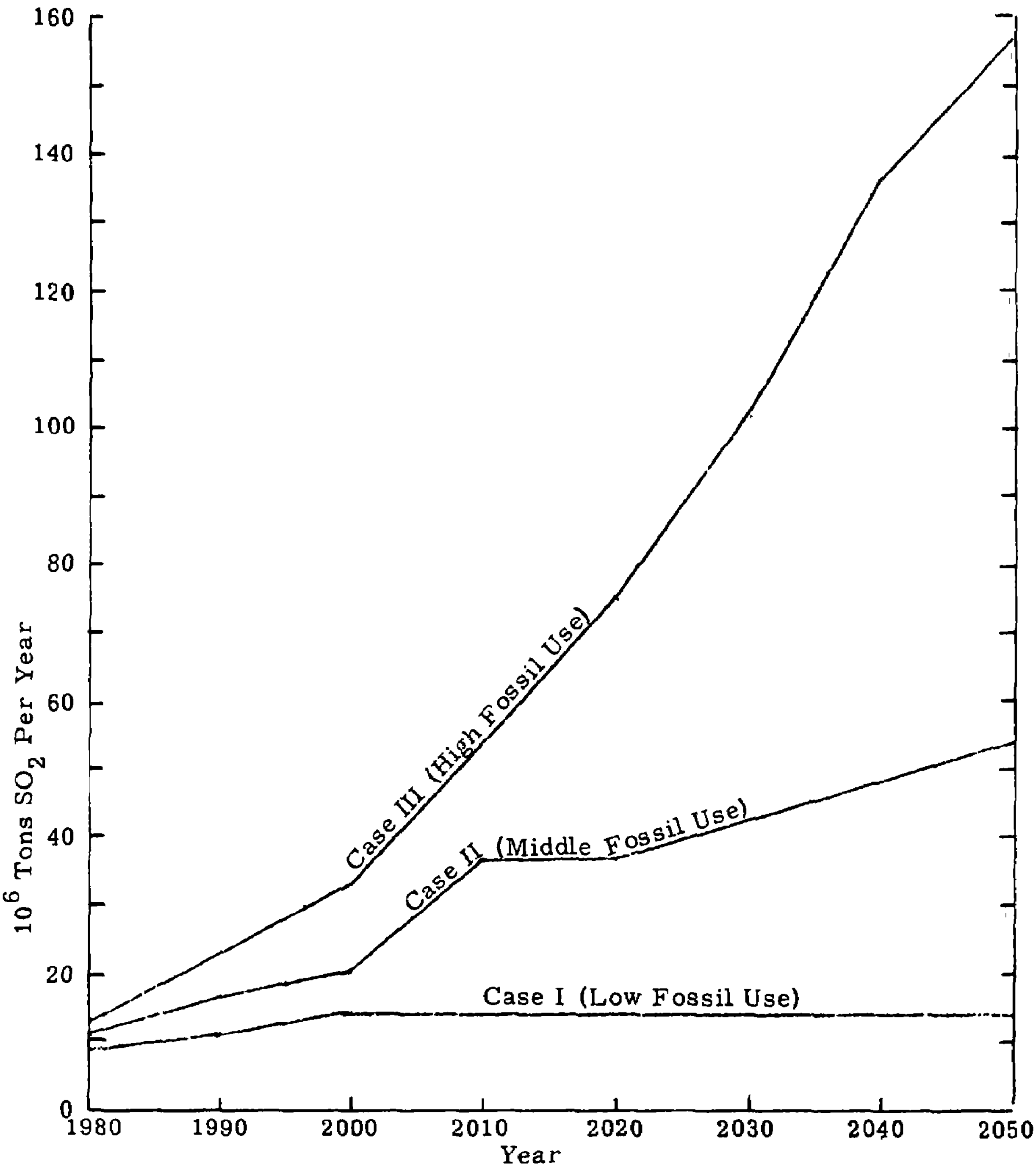


Figure VIII-1. Potential Sulfur Dioxide Emissions From Fossil Steam Electric Plants difference in 2050 has reached over a factor of 10. Thus, the impact of the successful development and use of the various forms of nuclear energy for electrical power generation over the next 80 years plays a significant part in air pollution control planning. At different development and use rates over that time period, very large differences in sulfur dioxide emissions to the atmosphere could occur.

REFERENCES

VIII-1. 49th Semi-Annual Electric Power Survey, Edison Electric Institute, New York, April 1971.

VIII-2. Derived from Chapter V references.

VIII-3. "Standards of Performance for New Stationary Sources," 42 CFR 466, Federal Register 36, 109, August 17, 1971.