CIVILIAN POWER PROGRAM

PART I

☆ SUMMARY ☆

CURRENT STATUS of REACTOR CONCEPTS

1959

Prepared By
EVALUATION and PLANNING BRANCH
DIVISION of REACTOR DEVELOPMENT

THE UNITED STATES ATOMIC ENERGY COMMISSION

DISCLAIMER

CIVILIAN POWER REACTOR PROGRAM

1959

PART I

TECHNICAL AND ECONOMIC STATUS

SUMMARY

Prepared by

Evaluation and Planning Branch

Division of Reactor Development

September 1, 1959

DISCLAIMER

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Preface

I. Preface

On June 1, 1959, the Division of Reactor Development's Evaluation and Planning Branch formed a special study group to undertake the task of establishing a ten-year program for civilian power reactor development.

The scope of the study encompasses the following areas:

1. Delineation of the specific objectives of the over-all Atomic Energy Commission civilian power reactor program.

2. Delineation of the specific technical objectives of each reactor concept.

3. Preparation of a chronological development program for each reactor concept.

4. Evaluation of the economic potential of each reactor type.

5. Delineation of a program to encourage the development of promising new concepts and reactor types, as well as major modifications of existing concepts.

6. Establishment of yardsticks for measuring the rate and degree of development on each reactor type.

The results of this work are to be used as a basis for the following:

1. Policy review by the Commission.

2. Program direction from the Division of Reactor Development to the Operations Offices.

3. Authorization and appropriation requests (FY 1961 and subsequent budgets).

4. Evaluation of new concepts with respect to program objectives and their integration into existing programs.

5. Establishing criteria for measuring and evaluating the progress of reactor programs.

6. Determining the relative degree of emphasis to place on various reactor systems.

This evaluation encompassed civilian power reactors rated at 25 MWE or larger and the experimental facilities and research and development related to advancement of the technology for civilian power reactors in this size range. Non-electric power generation reactors or military reactors were not included in this evaluation.

The full time members of the Atomic Energy Commission evaluation study group were:

H. E. Vann, Evaluation and Planning Branch, Division of Reactor Development
E. D. Jordan, "
I. E. Jackson, "
J. J. King, "
D. H. Kuhn, Office of Operations Analysis and Forecasting
R. B. Stuart, Division of Finance

The results of the evaluation are presented in a series of four reports. This report, (Part I) of the series, summarizes the significant results of the comprehensive effort to determine the current technical and economic status for each reactor concept in the Civilian Power Reactor program. Part II is a separate report entitled "Economic Potential and Development Program." Part III is the series of reports entitled "Technical Status" for each reactor type (8 separate reports). Part IV will be specific program recommendations for each reactor type.

This summary report (Part I) is based upon, and summarizes the detailed information in the status report (Part III).

The individual technical status reports (Part III) are to be edited and published by the Atomic Energy Commission for distribution (TID-4500).

The purpose and objectives of the civilian power program are based

primarily on previous Commission policy. However, these objectives have now been modified or expanded to quantitatively define such general statements as high fuel cost areas, short range and long range objectives, etc. In addition, the reactor types have been grouped into one of three different categories or classifications. Each category of reactors is associated with a specific program objective or objectives.

The Division of Reactor Development requested the assistance of the national laboratories and industrial organizations to accumulate the technical and economic status of each reactor type. This work was carried out under contract to the Atomic Energy Commission and under the supervision of Atomic Energy Commission personnel. The names and affiliation of the key contractor personnel assisting in this work are listed in Appendix I.

The economic data initially reported by the contractors reflected different design philosophy, different pricing policy and in some cases, a technology not consistent with the Atomic Energy Commission ground rules. The Division of Reactor Development therefore requested the architect engineering firm of Sargent and Lundy to undertake, under Commission contract and supervision, the task of modifying these costs, where necessary, to a consistent cost basis. This included the use of comparative cost for similar equipment, such as "like turbines" and components, the use of the same percentages for overhead cost and indirects, fuel cost consistent with Atomic Energy Commission specified fabrication cost and irradiation levels. Sargent and Lundy provided the Atomic Energy Commission with a revised set of cost curves for each reactor type that reflected the cost differences between plants, due to the inherent technical characteristics peculiar to the concept and the stage of technological development. The modified cost

curves are shown in this report. They represent the Division of Reactor Development estimate of the cost associated with current technological status and do not necessarily coincide with those presented by the contractor. The procedure used and the cost basis are shown in Appendix II.

Sargent and Lundy was asked by the Commission to prepare a curve of power cost for fossil fuel plants over the range of 25,000 to 325,000 KWE. They were also asked to escalate the cost of the fossil fuel through 1975. The results of Sargent and Lundy's work are represented by the fossil fuel curves in this report. The details of this work are provided in Sargent and Lundy Report No. 1564, entitled "Projections of Cost for Conventional Power Plants".

The summary material presented in this report on the general engineering and development programs that support the power reactor development, but which are not directly related to plant construction, were provided by the Nuclear Technology Branch, Division of Reactor Development.

The total civilian power reactor program costs have been obtained from the budget and cost records of the Division of Reactor Development. These totals include the cost of specific research and development and construction for each reactor type, as well as the cost of the general nuclear technology program. Also included are the Atomic Energy Commission and industrial costs associated with each cooperative arrangement project. No cost figures are included in this report showing the investment of private industry in non-cooperative arrangements projects.

This report is based on demonstrated technology. This somewhat penalizes a relatively new concept in a comparison of current status but

this "relative position in time" disadvantage should be offset when the comparison is made on the future potential. However, it must be recognized that the data on systems backed by operation and construction experience are on a sounder basis than those systems that have not yet reached this period in the development sequence.

This volume covers all major reactor concepts that are currently being developed under the civilian reactor program. A number of other concepts such as fluidized beds, spectural shift, etc. are not included because they are assumed to be "new concepts" with insufficient data presently available to qualify as current technology.

II

List of Illustrations

II. List of Illustrations

Summary of Current Status

III

Program Objectives

III. Objectives of Civilian Power Reactor Program

A. Purpose of Program

The basic purpose of the Atomic Energy Commission's Civilian Power Reactor Program is to establish and maintain the United States in a position of world leadership in the development of nuclear technology for civilian power application.

The objectives to achieve this purpose are delineated in the following paragraphs.

B. Objectives of Program*

The specific program objectives are listed below:

1. The achievement of nuclear power generation costs that are comparative with power cost of fossil fueled plant on an equivalent station design basis, in high energy cost areas in this country by 1968. High energy cost areas are defined as those where the cost of fossil fuel is 35 cents/million - BTU's (on 1959 cost).

2. The achievement of nuclear power generation costs that are comparative with fossil fueled plant cost on an equivalent station design basis in all but the low energy cost areas by 1975. Low energy cost areas are defined as those where the cost of fossil fuel is 25 cents/million BTU's or less (on 1959 cost).

3. The successful development of breeder type reactors that will serve as a source of plutonium and uranium 233. This type of reactor offers the potential of full utilization of the latent energy in both uranium and thorium. The primary objective is to develop those systems that offer the best potential of breeding. A second objective is to develop a breeder system that will also produce competitive power. The significance of stockpiling, or the potential value of plutonium and U-233, are not yet clear enough to

*See Appendix III for explanation of terms used in objectives.

evaluate this concept strictly on the basis of its potential as a producer of economic power.

4. The successful development of a reactor type that can operate independent of a continuous feed of enriched fuel, i.e. diffusion plants or separate breeding reactors. The prime objective for developing this reactor type is to obtain low power generation cost; however, there are two other basic objectives, i.e., the conservation of fuel reserves by burning natural uranium; and to assist friendly foreign nations who desire to be independent of this country as a source of enriched fuel, in the development of nuclear power technology. This class of reactors would utilize natural uranium as a fuel or operate on a self-sufficient (after initial spiking) recycle basis.

A fundamental role of the Atomic Energy Commission, in pursuit of the above objectives, is to develop a broad technology that can be used by the nuclear industry and the electric power supplier, in the achievement of economic power generation with nuclear reactor plants.

There are several different reactor types that offer the potential of achieving one or more of the program objectives. The various reactor types are divided into broad categories in the following paragraphs so that they can be identified and compared in perspective to the specific objective for which they offer the most potential.

C. Objectives of Specific Reactor Types

The Commission is actively engaged in research and development and plant construction on the following reactor types: Boiling Water; Pressurized Water; Organic Cooled; Sodium Graphite; Gas Cooled; Fast Breeders; Thermal Breeders; Heavy Water Moderated; and recycle reactors. These reactor types can be divided into three distinct categories and the basic objectives

delineated for each category. The categories are:

1. Thermal Converters. The primary objective of this class of reactors is to achieve economic nuclear power in high energy cost areas by 1968. The reactors in this category should be suitable for central station application in a size range of 25 MWE to 200 MWE or larger. The reactor types that fall into this category are:

   a. Pressurized water reactors

   b. Boiling water reactors

   c. Organic cooled reactors

   d. Sodium graphite reactors

   e. Gas cooled reactors, utilizing enriched fuel

2. Breeder Reactors. One objective of this class of reactors is to obtain a breeding ratio greater than unity as well as having a reasonable doubling time. A second objective is to obtain a power cost competitive with fossil fueled plants. The reactor types falling into this category are:

   a. Liquid metal cooled, fast reactors

   b. Fluid fueled reactors

   c. Heterogeneous - thorium fueled reactors

3. Natural Uranium Fueled and Recycle Reactors. The primary objective of this class of reactors is to achieve a power reactor capable of self sustained operation independent of a separate source of enriched fuel. A second objective is to obtain power cost competitive with fossil fueled plants. The reactor types falling into this category are:

   a. D_2O moderated, natural uranium reactors

   b. Gas cooled, natural uranium reactors

   c. Reactors capable of self sustained recycle

Each reactor type has been assigned to a category that has an objective which is consistent with one or more of the program objectives. This will enable a specific reactor type to be evaluated against a fundamental program objective and in perspective with other reactor types competing for the same objectives.

D. Technical Objectives

At this stage of the technology it is difficult to delineate the specific technical objectives or goals for each reactor type that, if reached, would lead to competitive power. This is complicated by the fact that the technical objectives are in many cases significantly different for each reactor type. It is possible, however, to state the broad technical objectives common to all reactor types which offer the potential of lower power generation cost. The manner in which these objectives will be approached is dependent upon the unique characteristics of each reactor type. More specific technical objectives are delineated in detail in the program for each specific reactor type. The significant technical objectives that offer promise of meeting the program objectives are:

1. To Increase the Thermal Efficiency of Nuclear Plants. Higher plant efficiency will reduce the reactor thermal power required and the amount of fissionable material burnup required for a given power output. Achievement of this objective offers the potential of reducing capital and fuel cycle cost. Higher plant efficiencies require higher coolant exit temperatures and higher operating temperatures for the fuel elements. There is an optimum temperature, beyond which additional increases will reduce fuel life and increase the fuel enrichment (due to the use of high

temperature structural material in the clad and core) to the point where these factors may offset the economic gains due to the higher cycle efficiency. This optimum temperature will vary with reactor type and coolant.

2. To Obtain Lower Fuel Cycle Cost. A fundamental method of reducing fuel cycle cost is to increase the irradiation period for the fuel. This requires a large amount of excess reactivity in the reactor at beginning of core life and presents basic reactivity control problems for systems with relatively low average conversion ratios. Long irradiation periods also require more enrichment in the initial fuel load. The realization of this objective depends upon developing more effective control methods and fuel management programs. Longer fuel exposure also demands the development of fuel materials and claddings that will maintain mechanical and nuclear integrity over the anticipated fuel lifetime.

Lower fuel inventories, lower cost fuel fabrication methods and cheaper chemical reprocessing will also lead to lower fuel cycle costs.

3. To Obtain High Specific Power in the Fuel and Higher Power Density. The successful achievement of higher specific power in the fuel and high power density in the core offer the potential of reduction in power cost of nuclear plants. The optimum value of specific power that is economically desirable, is a compromise between such factors as further subdivision of the fuel which results in higher fabrication cost and the economic gain due to a reduced core volume. The optimum economic power density is a function of reactor type, fuel material, core power distribution, fabrication cost, heat transfer considerations, and other variables.

4. To Obtain Better Neutron Economy. The cost of power from nuclear reactors can be reduced by obtaining a better neutron economy

in the core. This objective centers around the obtainment of a higher conversion ratio, i.e., an increase in the number of atoms of fissionable material (plutonium or U-233) produced per atom of fissionable material destroyed. The principal gain will be the extension in reactivity lifetime.

5. To Obtain Simplicity in Nuclear Plant Design. The capital cost of nuclear plants can be decreased in the following manner:

a. Use of conventional or less expensive materials in construction wherever practicable.

b. Simplification of plant design

c. Improved and less expensive methods of containment

d. Reduction in number and complexity of auxiliary system requirements

e. Development of reliable components that do not require undue rigid specifications

f. Improved understanding of reactor hazards

The successful accomplishment of the above technical objectives will be instruments in reducing power generation cost from nuclear reactors.

IV

Status
Civilian Power Reactors

A. Thermal Converters

IV. Status of Civilian Power Reactors

A. Thermal Converters

The reactor types that are classified as thermal converters are:

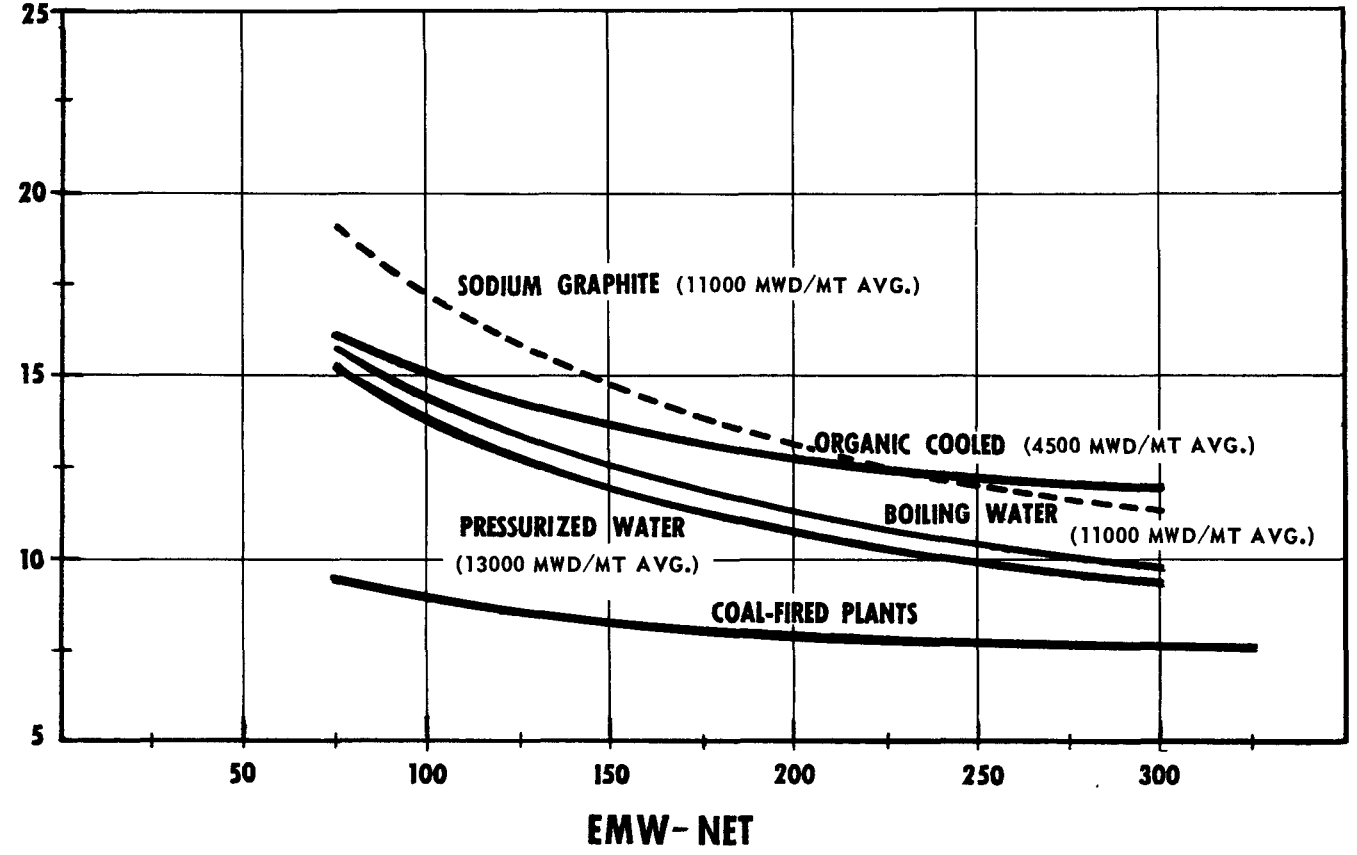
1. Pressurized water
2. Boiling water
3. Organic cooled
4. Sodium graphite
5. Gas cooled (enriched fuel)

The primary objective of the thermal converter class of reactors is to develop a plant capable of producing electric power at a cost that is competitive with fossil fueled central station plants. A composite curve showing the power generation cost of thermal converter reactors vs. coal fired plants is presented in Fig. 1. The cost data is based upon a "reference design". This reference design is representative of one that might be built without additional research and development based upon current technology. Plant construction would require approximately $3\frac{1}{2}$ years. It would require an additional 3 years to reach equilibrium on the fuel cycle. Therefore the cost associated with current technological status (as defined for this report) would be realizable in 1964 or 1965. The technical status of each reactor type included in the thermal converter class is summarized in the following pages of this section.

POWER GENERATION COSTS

THERMAL CONVERTER REACTORS
VS
COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.



NOTES :

NUCLEAR -
BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/GM.

COAL - FIRED -
BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%
FUEL COST	35¢/10^6 BTU

FIG. 1 SEPT 1959

1. PRESSURIZED WATER REACTORS

a. Description. The pressurized water reactor is a heterogeneous fueled, thermal reactor that uses light water as moderator and coolant. The system utilizes an intermediate coolant loop between the reactor and turbine. The water in the primary coolant loop is maintained under high pressure to keep the bulk temperature of the coolant leaving the reactor below the saturation temperature during normal operating conditions. A simplified flow diagram of the pressurized water reactor plant is shown in Fig. 2.

b. Technical Status. The basic technology of the pressurized water concept was developed for Naval Reactor Plants. Much of this technology in fuels and materials has also served as the basis for the rapid advance of other concepts such as the boiling water reactor. No other reactor concept has a comparative depth in technology and operating experience as the pressurized water reactor. Sufficient research and development and operating experience has been performed to arrive at the following conclusions on technical status:

(1) Physics. The basic physics of pressurized water reactors is well known. The theory of neutron slowing down and transport has been developed. Absorption cross sections in the thermal neutron energy range have been measured for all materials of interest. Measurements have been made over the entire energy range of interest for fissionable and major moderator and structural materials. However, accuracy and resolution leave much to be desired. Considerable effort is now being made to improving the accuracy and precision of already measured values. A large number of one and two dimensional multigroup diffusion theory computer programs have been developed at many sites within this country. Several transport theory

computer programs have also been developed. Simplified fuel depletion studies have been developed and are in use.

The major reactor experiments applicable to pressurized water power reactors have been done on single region, water moderated, rod lattices.

At present, reactor physics experimental data exist on:

(a) Aluminum clad UO_2 of 1.3% enrichment.

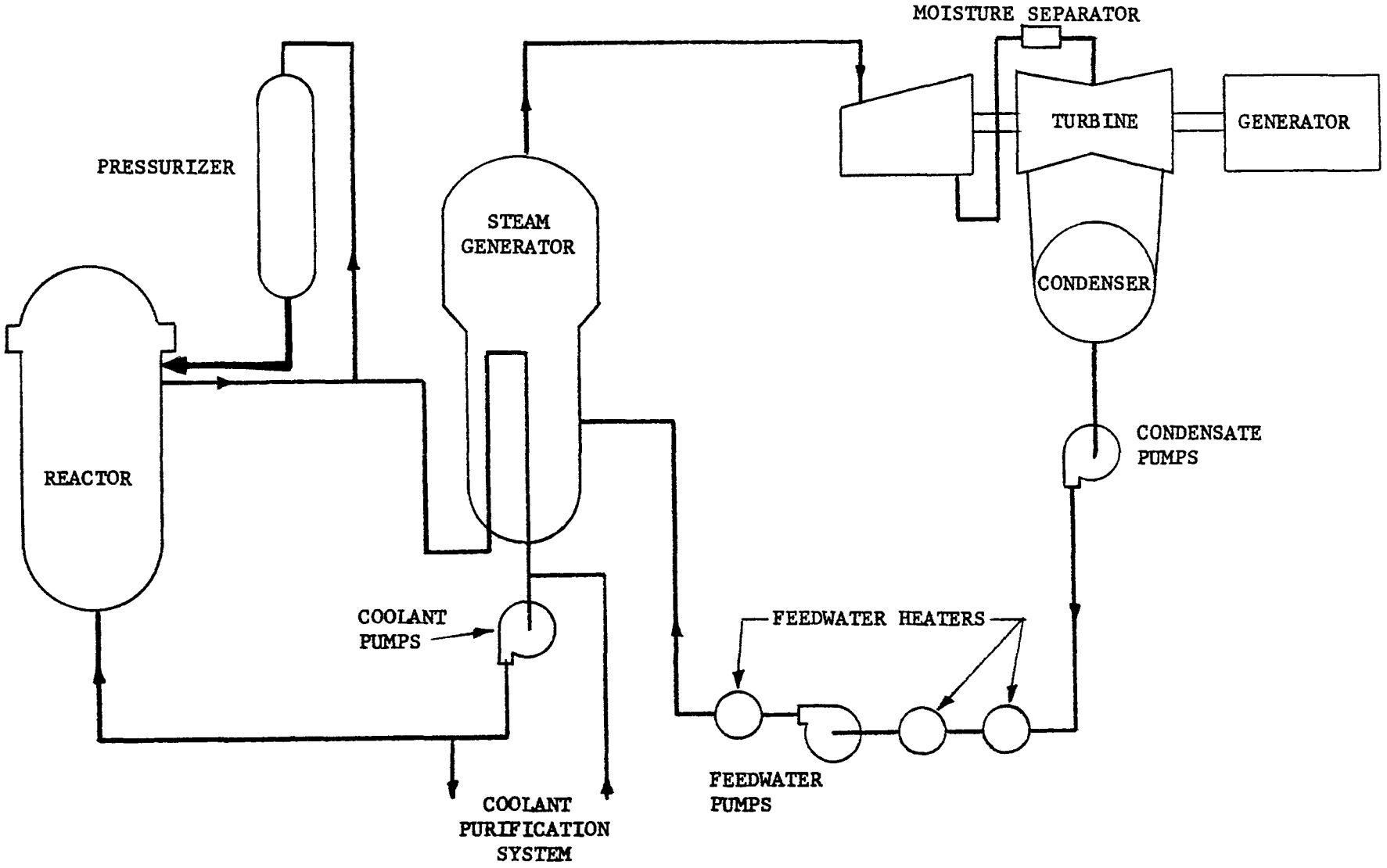
(b) Stainless steel clad UO_2 of 2.7, 4.0, and 4.43% enrichment with water to uranium volume ratios of 2.19, 2.93 and 3.87.

(c) Two region critical experiments with stainless steel clad and UO_2 fuel of 4.43% and 2.7% enrichment with a water to uranium volume ratio of 2.93.

(d) U-235 O_2-ThO_2 lattice with U-233 substituted for U-235 in selected elements.

Work is currently under way on three region critical experiments with stainless steel clad UO_2 of 1.6%, 2.7% and 4.43% fuel enrichment. These experiments on multiregion lattices will be of great assistance in studying flat power distribution reactors.

Although there is a large and growing amount of data on uniform UO_2-water lattices, this information is often insufficient to optimize precisely a practical reactor design. Practical reactors contain many more inhomogeneities (usually of a three-dimensional nature) than have been present in most of the UO_2-water experiments. The ability to handle inhomogeneity effects is important in predicting criticality, control of rod worth, temperature coefficient and power distribution. Adequate methods for dealing with this problem are being developed, but much work



SEPT 1959
FIG. 2

SIMPLIFIED FLOW DIAGRAM

PRESSURIZED WATER REACTOR PLANT

is still to be done.

To date, control concepts such as use of chemical control and burnable poisons have been confined to small, high enrichment reactors. Some information is available and applicable to the design of low enrichment pressurized water power reactors. Methods of analysis for fuel cycling studies have been developed, but in small detail.

The precise conditions for xenon oscillations are not thoroughly understood. How this instability can be avoided or what operational methods are best to handle it are questions that are only beginning to be answered by building, operating, and experimenting with large reactors which contain such instabilities.

Simplified methods of analyzing operational transients and reactor accidents have been developed at most reactor facilities. Experimental work has been performed by Phillips Petroleum in the SPERT experiments and by Argonne with the BORAX experiments. Detailed explanations of the various phenomena from the theoretical standpoint are not yet available.

It is now apparent that a cold critical experiment may leave an uncertainty of as much as $\pm$ 3% $\Delta k/k$ in the hot reactivity. This is due in part to the inability to handle 2 and 3 dimensional inhomogeneities as a function of reactor temperature and in part to the uncertainty in the detailed dependence of the resonance integral of U-238 on temperature and surface to mass ratio. The fuel contribution to the temperature coefficient is further complicated by the lack of precise information on temperature distribution throughout a reactor core.

(2) Fuel and Materials. Extensive development work has been completed on UO_2 fuels. This work has demonstrated UO_2 to be dimensionally

stable, easily fabricated, chemically compatible for water systems, and to possess acceptable retention properties for fission gases. These properties have been investigated under severe radiation conditions both as reactor fuels and in in-pile test loops. Small sample tests at burnups of greater than 27,500 MWD/MT have been performed on UO_2 rods at various heat fluxes. Recent loop experiments (CHVM) at Chalk River have demonstrated 60,000 MWD/MT irradiation of UO_2 platelets. Fission gas retention has been shown to be a function of UO_2 density. For high density material (93% - 95% theoretical) experimental evidence shows adequate retention to prevent development of excessive internal pressure.

The physical properties of UO_2 have been reasonably well established for the types and grades of UO_2 currently manufactured. However, a good correlation of measured physical characteristics with fabricability has not yet been established. The preparation of UO_2 powders and the technology for fabrication of high density (97% theoretical) sintered pellets and platelets has been demonstrated although it is not well developed.

The major undesirable property of UO_2 for fuel use is its low thermal conductivity. Work is under way to investigate other compounds, uranium silicides for example, that show some promise of considerably better thermal conductivities and perhaps adequate other properties.

The development of cladding materials for pressurized water reactors in the past has been primarily limited to the evaluation and development of either stainless steels or zirconium alloys. The feasibility of stainless steels from the standpoint of fabrication, corrosion and chemical compatibility, and resistance to radiation damage has been established. The major disadvantage of stainless steel is its relatively

high neutron absorption cross section (about 2.7 barns).

The desire to develop a lower cross section material for reactor use led to the extensive development program for zirconium and the development of the Zircaloys. Zirconium technology has been adequately developed to the point of its practical utilization as a fuel cladding material. At the temperatures of the pressurized water reactor its further utilization is hindered primarily by economic considerations and to a lesser degree by the progressive nature of the corrosion that can occur in case of a cladding defect in a water environment. Some work is under way to develop new zirconium alloys, such as Zr-Nb or Zr-Nb-Sn, that promise better strength characteristics at elevated temperatures.

Development work is under way to establish aluminum alloys that may be satisfactory for the environmental conditions of the pressurized water reactor systems. Alloys of Al with some Ni show promise of improved corrosion properties as do alloys containing trace additives of Zr, Be, or Si. Suitable strength characteristics at system temperatures have not been attained for this type of material. Work on sintered aluminum powder alloys currently in progress shows promise of meeting strength requirements, but corrosion resistance in high temperature water environments is poor relative to Zircaloy or stainless steels.

The potential of Fe-Al alloys as a substitute for stainless steel is under study. Strength characteristics at elevated temperatures are good, corrosion rates in water systems may be acceptable, and compatibility with UO_2 is probably satisfactory below 1500°F. The cross section is lower than for stainless steel (being 1.8 to 2.2 barn), but weldability is poorer. More work is necessary to establish radiation

damage characteristics and methods of fabrication prior to its use in a power reactor.

One way to indirectly extend core life and to operate at higher burnups is to introduce burnable poisons into the reactor core. This helps to alleviate the control problem due to the large excess of fuel. A number of development programs are under way to investigate the feasibility of incorporating the burnable poisons in the cladding material. A typical system being considered is a dispersion of B_4C in zirconium alloys.

Hafnium has been developed as a control rod material. The desirability of hafnium for pressurized water reactor control rods is well established and it will probably remain the preferred material from a technical standpoint. The continued use of hafnium, however, is limited by its availability for civilian use and its relatively high cost. These items have led to investigations of alternate materials.

Alternate control rod materials being studied include boron dispersions, solid boron-carbide in sealed containers, rare earth dispersions, and Ag-In-Cd alloys. Experience has indicated that Ag-In-Cd must be cladd. In general, the dispersion of boron in corrosion resistant materials has led to embrittlement. Solid boron-carbide in sealed or compartmented containers shows promise and is actively being investigated. Rare earth oxides have been used for APPR-I and are being considered for use in the SM-2 reactor. These dispersions, however, are expected to be expensive and will require cladding to obtain corrosion resistance. The creep strength of these alloys is also poor. Studies on Ag-In-Cd have progressed to the stage where it has been specified as the control rod material for reactors going into operation in 1960.

The primary system of pressurized water reactors consists of the reactor vessel, steam generator, pressurizer, main loop piping, and valves. Past work has demonstrated the feasibility of constructing these components of corrosion resistant stainless steels. Difficulties in maintaining components in a radioactive environment and the desire to maintain a low level of corrosion products in the circulating system suggested the development of these stainless steel systems. Initially systems were constructed of the stabilized types (347 or 348), subsequent development has shown that non-stabilized grades (304) having better weldability are adequate for use in the pure water environment of pressurized water reactors.

Stress corrosion is a potentially serious problem in stainless steel systems. Metallurgical and chemical studies are in progress directed toward the reduction of stress corrosion, reduction of crack susceptibility on welding, and the development of high strength alloys that would permit the reduction of pipe wall thickness.

(3) Heat Transfer and Fluid Flow. A considerable body of analytical and experimental investigation has been performed on the thermal and hydraulic problems of the pressurized water reactor. The work completed is adequate to justify the following design limits:

(a) Center fuel temperature less than fuel melting point.

(b) No bulk boiling in the core during steady state operation.

(c) Fuel rod size limit such that the fuel rod will remain intact during a loss of flow accident.

Work is under way to extend the knowledge of thermal and

hydraulic behavior sufficiently to justify relaxation of the current design limitations. This work includes studies of transition boiling phenomena, the influence of flow instability and flow redistribution during bulk boiling on burnout, and improved effective thermal conductivity determinations for UO_2 under service conditions.

(4) Coolant Chemistry. A high degree of water purity is required to minimize corrosion, net radiolytic dissociation, and radioactivity resulting from transport of corrosion and fission products from the core. Past work has done much to define the nature of the chemical problems and has developed ion exchange methods utilizing synthetic resins capable of maintaining water purity of one million ohm-cm. in primary systems. Additional work has demonstrated the feasibility of the lithium-hydroxyl system for continuously maintaining pH control as well as for removing radioactive isotopes. Filtration and evaporation methods for removal of insoluble impurities have been investigated. The value of H_2 overpressure in reducing corrosion of stainless steel has been established.

The factors influencing net radiolysis of water have been established. The technology of external catalytic recombiners has been developed to the point that suitable and highly efficient (but not necessarily optimum) designs for the recovery of nominal flows of radiolytic gas are available.

Extensive research and development has been carried out on corrosion inhibition. Alkaline corrosion inhibitors have been developed that show considerable beneficial effect but not without accruing some disadvantages, especially in systems utilizing lithium.

The problem of decontamination and removal of corrosion products from metal surfaces has received considerable study. These investigations

have led to the multi-solution (alkaline permanganate-ammonium citrate) process for decontamination of stainless steel which produces excellent results.

Research and development efforts have shown that boric acid and possibly certain borate salts show promise for use as chemical poisons. Out-of-pile tests have shown no serious corrosion problem for stainless steel and Zircaloy under operating conditions of low dissolved oxygen. With high levels of boric acid and oxygen, as during refueling operations, the possibility of galvanic corrosion exists. Dissolved galvanic corrosion inhibitors have been found to be effective.

Continuing work on gas recombination systems is directed toward the development of internal recombination catalysts. The internal recombination system has potential advantages, especially for systems which evolve very large quantities of gas. Large gas evolutions might be expected in systems utilizing soluble poisons at high power.

Work continues on the effect of high temperature and radiation on corrosion inhibitors and on the possibility of a combined corrosion inhibitor and soluble poison.

Work on decontaminating agents is continuing in an effort to develop a single solution procedure that will permit reduction in solution waste volumes.

(5) Components and Auxiliary Systems. The development of reactor vessels for pressurized water reactors has been an evolutionary process since the origin of the concept in 1948. Developments of a manufacturing nature have permitted the construction of progressively larger and thicker walled vessels. Processes for cladding carbon steel and low alloy vessels with 300 series stainless steel have been improved.

Both weld bead deposit and spot welded plate cladding is in use in addition to metallurgical bonded plate.

Extensive work was done to determine the materials for the steam generator tubes. Stainless steel was ultimately selected because of its availability and ease of fabrication. Several different design configurations were developed by various suppliers. These include the straight-through tube design, the horizontal U-tube U-shell design, and the vertical U-tube configuration. All of these designs have performed successfully. Major development effort has been devoted to the tube to tube-sheet welding and the cladding of carbon steel water boxes and tube sheets with stainless steel.

The major development work on hermetically sealed canned motor pumps was performed on the (Navy) STR reactor project. Work completed on the canned pump includes the development of bearings capable of operating in high pressure, high temperature water, fabrication of thin stainless steel or Inconel liners for rotor and stator, cooling of motor windings, and high pressure electrical terminal seals. The size and efficiency of the canned pump has been steadily increased, culminating in the pressurized water reactor Shippingport pumps.

The development of a complete line of valves suitable for pressurized water reactor operation was performed during the (Navy) STR project. The main achievement was the construction of hermetically sealed, hydraulically operated stainless steel valves for all primary system services. Recent efforts have been devoted to the design of controlled leakage valves which permit conventional construction and conventional motor operators. A major construction development has been the acceptance of cast material for valve bodies and other pressure parts, thereby

effecting appreciable cost reductions.

Piping for pressurized water reactor systems has been of the seamless (extruded or drawn) or hollow forged and bored type. The welding of heavy wall stainless steel piping has been extensively developed. Weldability and weld cracking were found to be very sensitive to material composition. A recent development is centrifugally cast and hydroforged piping for large sizes. This manufacturing process has promise of reducing cost and increasing the pipe size (diameter) that could be fabricated.

Auxiliary systems for liquid and gaseous waste disposal, coolant purification, emergency cooling, fuel handling, ventilation, etc., are necessary to all reactor systems. Safe design criteria and feasible engineering techniques to meet these requirements have evolved from past work on both military and commercial reactors.

(6) Reactor Safety. The pressurized water reactor can be designed with a large negative temperature coefficient of reactivity which would provide a degree of inherent safety. The safety of the pressurized water reactor concept has been demonstrated by the considerable operating experience on naval vessels, Shippingport, APPR and others.

c. Operating Experience. There is more operating experience available on pressurized water reactors than any other reactor concept in the AEC program. A summary of this experience is given below.

STR Mark I, the full-scale land based prototype of the Nautilus, attained initial criticality on March 31, 1953, and operated

at a very low power until late May 1953 to obtain the initial necessary physics and radiation shielding information. On May 31, 1953, Mark I was placed in power operation and full design power was reached on June 25. The nuclear plant met all design specifications. It took about one month after first power generation for Mark I to operate smoothly at full power.

The APPR-1 was the first plant to be constructed under the Army Nuclear Power Program. The APPR-1 has experienced no instability problems. This has been demonstrated by results of operation during both planned and unplanned load changes.

The measured radiation levels have been well within design limits and in most cases they were conservative.

The Shippingport reactor went critical on December 2, 1957, and reached full power (60,000 KWE) on December 23. Testing has included two power runs of more than 1000 hours each as part of a Reactivity Lifetime Test, and the seed fuel has operated for approximately 5100 effective full power hours to date. The original design life was 3000 effective full power hours and is now estimated to be 6000 effective full power hours.

The stability of Shippingport on loss of load has been demonstrated.

Shippingport has had several mechanical component difficulties. The stator in one of the four main coolant pumps was cut circumferentially by the lower end of the rotor stack. It is believed that operation of the pump at low system pressure caused the stator can to collapse.

During the early days of power operation, spurious motions of some of the fail-as-is type hydraulic valves occurred. A test was formulated to determine the causes and the conclusions were that two types of valve drifts were encountered:

1. Valves drifted from the closed position when the water flask was vented. Air is normally directed into the top of the water flask to operate the valves.
2. Valves bounced from both the open and closed positions during the operation of other hydraulic valves.

During initial plant operation it was found that both of the self-actuated pressurizer steam relief valves were leaking. Tests at several pressures indicated that the leakage may have been caused by thermal distortion or warpage.

On February 3, 1958, leaks were discovered in several tubes of one of the four steam generators between the secondary face of the inlet end tube sheet and the first tube baffle. Examination of the removed tube sections revealed that failure was caused by stress corrosion probably resulting from a combination of steam blanketing and boiler water chemistry out-of-specification with respect to free hydroxide. During initial plant operation, excessive blowdown was required to reduce silica concentration and at this time the boiler water chemistry could not be maintained within limits. To prevent steam blanketing in the future, two additional risers were installed between the inlet tube sheet and the first existing riser. In addition, sampling lines were installed in the defective area to determine the presence of a steam bubble and chemical concentrations. Other changes included modifications.

in the existing boiler sampling connections, the addition of thermocouples, and changes in the boiler water chemistry control to eliminate any possibility of free hydroxide. The defective tubes were successfully plugged with blind nipples and no difficulty has been experienced since the steam generator went back into service on May 14.

Inspection of the turbine moisture separator in late November 1958 indicated that the internals had been completely destroyed. Further inspection at a later date revealed that pieces of the turbine moisture separator were lodged in the turbine low pressure blades. The possible explanation for this failure is excessive mechanical vibration.

In general, measurements of the radiation intensities within the reactor plant container at various power levels showed these to be somewhat lower than predicted. A gradual increase in proportion to length of plant operation was observed for the first few months, after which steady state levels were approached. The actual measurement during operation and shutdown are listed in Part III - Status of Pressurized Water Reactors.

The level of fission products in the primary coolant has been higher than was expected on the basis of observed levels of uranium contamination in structural materials, although the level is considerably below design levels. On the basis of experimental evidence, it is suspected that one or more defected UO_2 blanket rods exist.

The operating experience on pressurized water reactors has demonstrated the feasibility and operability of this concept as an integrated system. Most of the difficulties have been non-nuclear in origin and steps have been taken to prevent similar occurrences in the future. A graphic presentation of the operating experience on APPR and the Shippingport plant is shown in Figs. 3 and 4, respectively.

d. Plants Under Construction. The pressurized water reactors under construction that will provide valuable data to advance the technology are shown in Fig. 5.

e. Economics. A reference design has been made for the pressurized water reactor based on the technical status summarized in this report. This design was used as the bases for estimating the cost at different plant ratings. The base parameters for the reference design are as follows:

SUMMARY OF PLANT CHARACTERISTICS
PRESSURIZED WATER REACTOR PLANT

A. Heat Balance	
1. Total Reactor Power, MW(t)	810
2. Gross Turbine Power, MW(e)	213
3. Net Plant Power, MW(e)	200
4. Net Plant Efficiency, %	24.8
B. Turbine Cycle Conditions	
1. Throttle Temperature, F	480
2. Throttle Pressure, psig	555
3. Steam Flow, lbs/hr	3.03×10^6
4. Condenser Back-Pressure, in Hg. A	1.5
5. Final Feedwater Temperature, F.	340
C. Reactor Description	
1. Reactor Vessel	
a. Inside Diameter, ft.	11.75
b. Overall Height, ft.	31.0
c. Wall thickness, in.	7.0
d. Material	SS Clad CS
e. Design Pressure, psia	2500
2. Reactor Core	
a. Active Diameter, ft.	8.6
b. Active Height, ft.	9.0
c. Active Core Volume, $ft.^3$	523
d. Lattice Arrangement	square
e. Lattice Spacing, in.	7.475
3. Reflector or Blanket	
a. Material	H_2O
b. Radial Thickness, ft.	1.5

4.	Fuel Elements	
	a. Fuel Material	UO_2
	b. Clad Material	SS
	c. Fuel Enrichment, %	3.34
	d. Fuel Element Geometry	rods
	e. Cladding Thickness, in.	0.029
5.	Material Inventories	
	a. Fuel, Metric tons	52
	b. Uranium, Metric tons	41.7
	c. U-235, initial-kg.	1390
6.	Reactor Control	
	a. Method of Control	rods
	b. No. of Control Elements	36
D.	Plant Performance Data	
1.	Primary Coolant Outlet Temp., F.	574
2.	Primary Coolant Inlet Temp., F.	533
3.	Reactor Temp. Drop., F.	41
4.	Primary System Operating Pressure, psia.	2200
5.	Primary Coolant Flow Rate, lbs/hr.	53.1×10^6
6.	Avg. Core Heat Flux, Btu/hr.-ft^2	861,800
7.	Max. Core Heat Flux, Btu/hr.-ft^2	395,000
8.	Max. Cladding Surface Temp., F.	636
9.	Max. Fuel Temp., F.	4500
10.	Core Coolant Velocity, ft/sec.	14.1
11.	Peak To Avg. Power Ratio	4.55
12.	Core Power Density kw/ft^3	1550
13.	Core Specific Power kwt/metric ton-U	19,500
14.	Fuel Burn-up MWD/metric ton-U (average)	13,000

The power cost for a 300 MWE plant extrapolated from the above design is as follows:

Total capital cost ----------$73,400,000	
Mills/kwh ----------	4.90
Fuel Cycle Cost - M/Kwh ----------	3.81
Operation and maintenance cost - M/Kwh ----------	.59
Nuclear Insurance Cost - M/Kwh ----------	.26
Total Power Cost - M/Kwh ----------	9.56

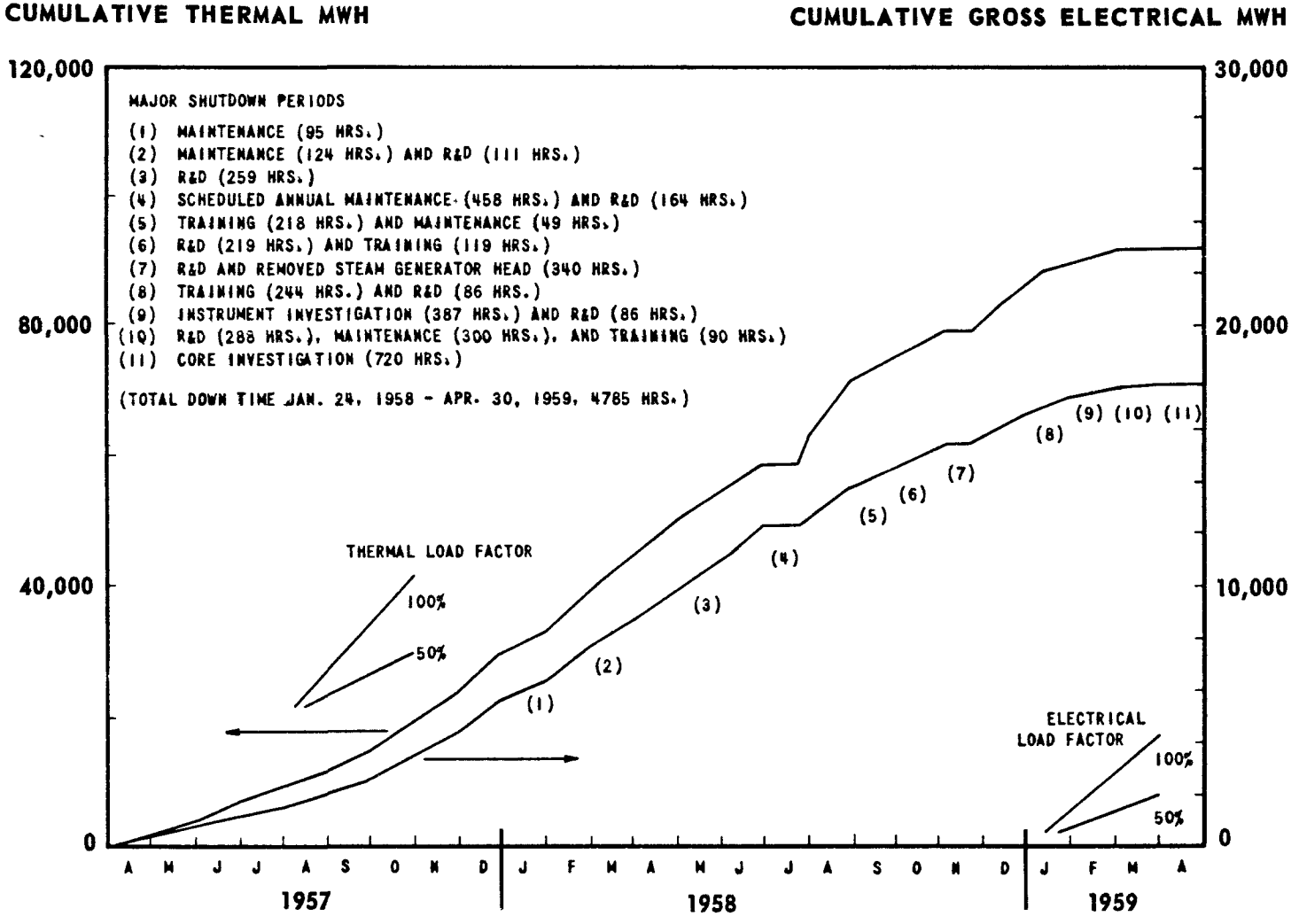
The relationship of power cost vs. size is shown in Fig. 6

The above costs are representative of plants that could be constructed with the current technological status. However, these costs could only be achieved after the period of time required for plant construction

and fuel cycle equilibrium. The design parameters used are close to optimum for a current status pressurized water reactor plant. A check point can be obtained by a comparison with the Yankee plant currently nearing completion of construction. The estimated cost of Yankee is $52,000,000 and the expected output is 140,000 KWE (2nd core). The expected capital cost is 371 $/kw or 7.43 m/Kwh for fixed charges.

The layouts and other design data for determining the economics is included in report S & L 1674 and Appendix II of this report. A moisture separator efficiency of 95% and 4-35" last stage buckets on the turbine were used in the reference design. This resulted in a 25% efficiency for the PWR plant. The plant could be designed with difference size last stage buckets on the turbine, a more effective moisture separator, more feed water heating, and obtain a 27.8% efficiency. This would reduce fuel cycle cost and increase turbine and plant cost. If the plant is designed for 28% efficiency the fuel cycle cost would be reduced approximately .4 M/Kwh and the capital cost would be increased by approximately .2 M/Kwh resulting in a net decrease in power cost of .2 M/Kwh under the cost shown.

PRESSURIZED WATER REACTORS
OPERATING EXPERIENCE
APPR - 1



SEPT. 1, 1959
FIG. 3

PRESSURIZED WATER REACTORS
OPERATING EXPERIENCE
SHIPPINGPORT ATOMIC POWER STATION

CUMULATIVE GROSS ELECTRICAL MWH

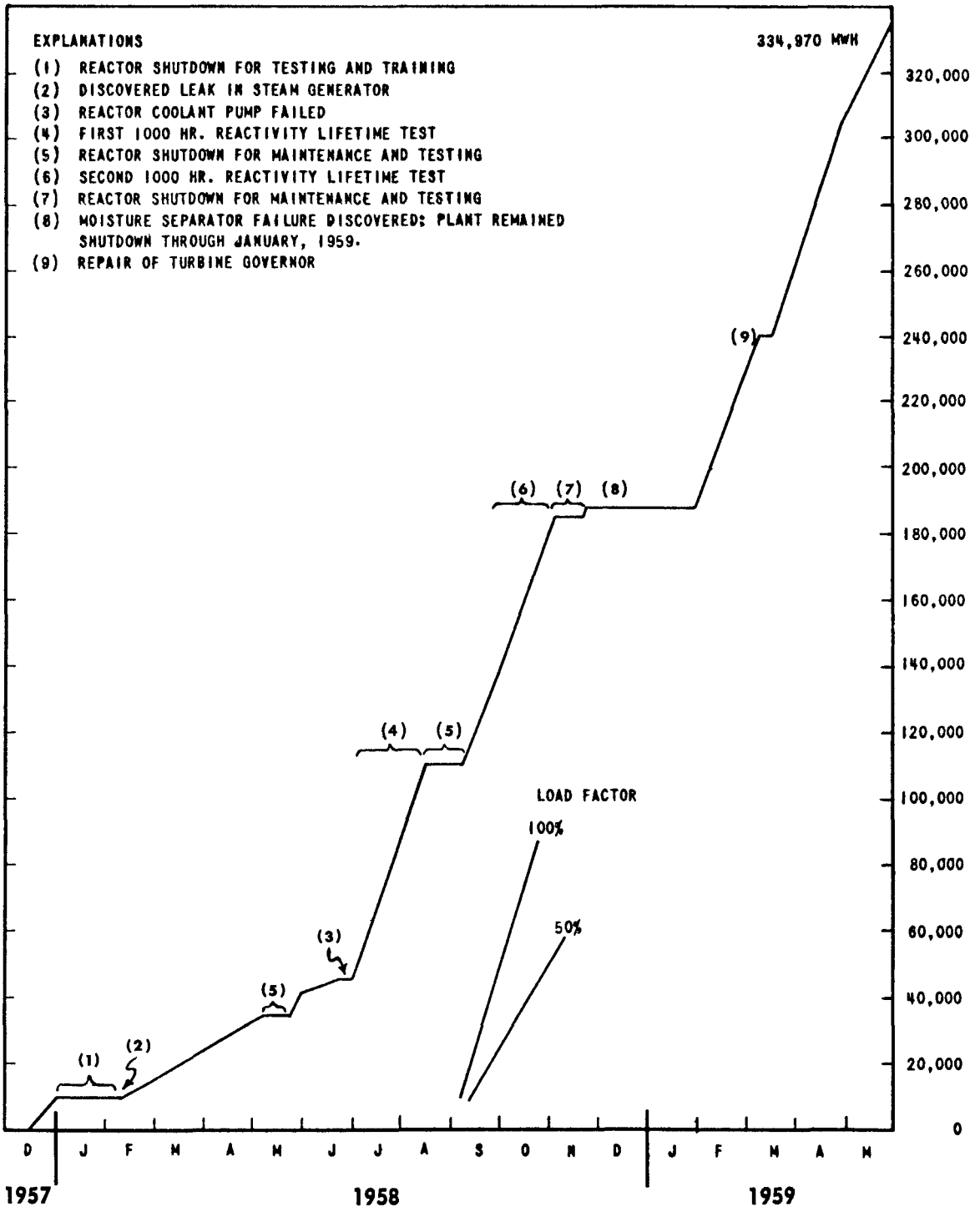


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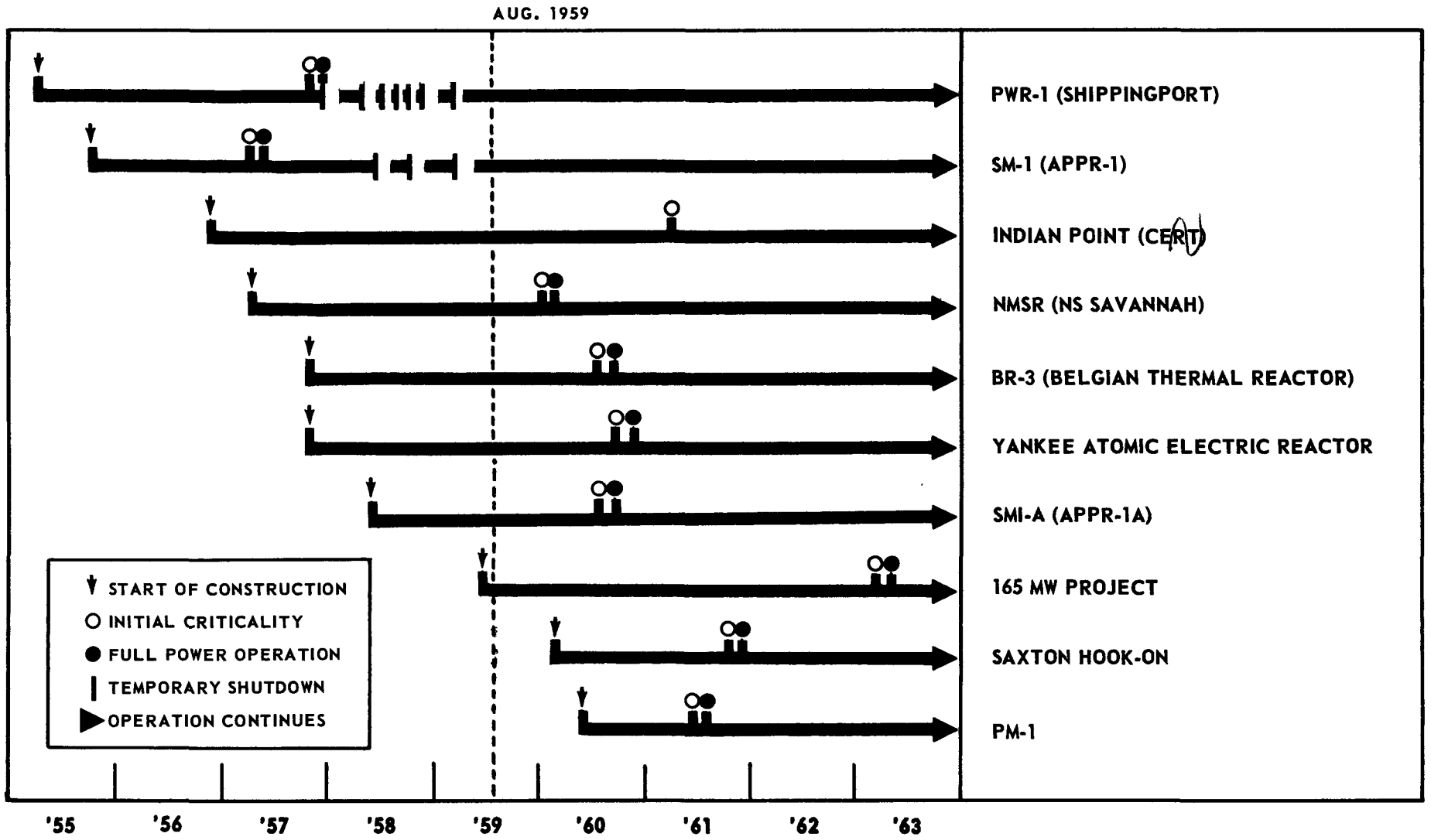
FIG. 4

PRESSURIZED WATER REACTORS
CONSTRUCTION SCHEDULE



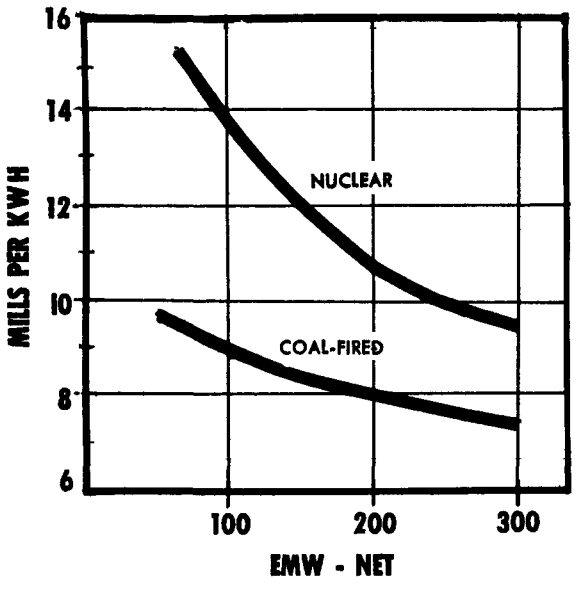
SEPT. 1, 1959
FIG. 5

POWER GENERATION COSTS

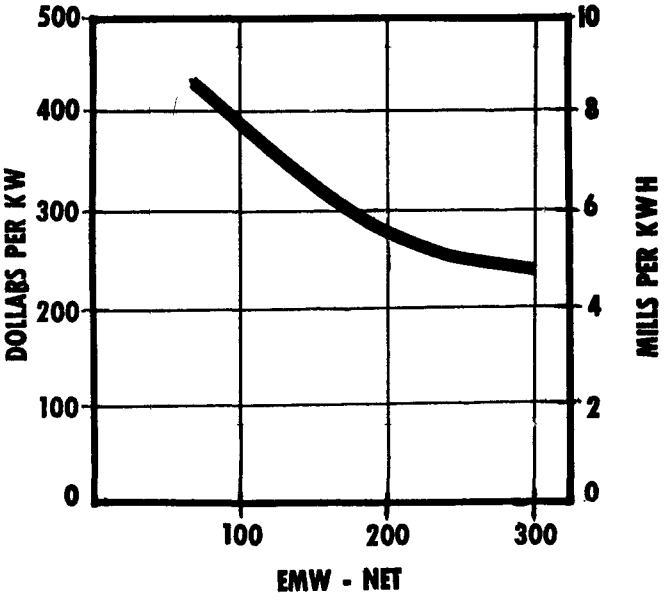
PRESSURIZED WATER REACTORS
VS
COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.

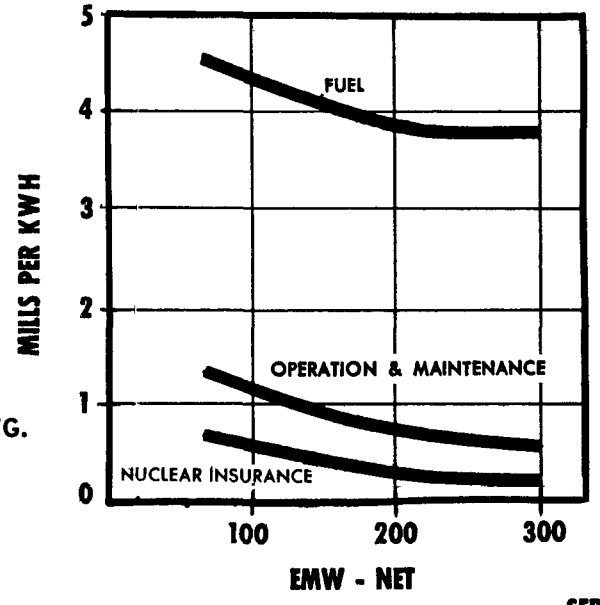
POWER GENERATION COSTS



CAPITAL COSTS



OPERATING COSTS



NOTES :

NUCLEAR -
BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/GM.
FUEL FABRICATION COST	
UO_2 - SS	$110/KG.
FUEL EXPOSURE	13000 MWD/MT AVG.

COAL - FIRED -
BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%
FUEL COST	35¢/10^6 BTU

SEPT 1959

FIG. 6

2. BOILING WATER REACTOR

a. Description. The boiling water reactor is a heterogeneous fueled, thermal reactor that utilizes light water as moderator and coolant. There are several variations of the boiling water reactor concept. These are: (1), the direct cycle system, in which the water is boiled in the reactor pressure vessel and the steam is bled directly to the turbine, (2), the dual cycle, in which part of the energy from the reactor forms steam which goes directly to the turbine and part of the energy is transmitted by hot water to a steam generator, where additional steam is formed to help supply the turbine demand, (3), the indirect cycle which utilizes a steam generator between the reactor and the turbine, and (4), the boiling water reactor utilizing nuclear superheat. The boiling water reactor can utilize natural circulation (in some size ranges), forced circulation, or a combination of natural and forced circulation. Steam can be separated either inside the reactor (smaller sizes) or externally. A simplified flow diagram of the dual cycle boiling water reactor plant is shown in Fig. 7.

b. Technological Status. The boiling water reactor has drawn heavily on the technology developed for pressurized water reactor systems. This is true especially in the areas of fuel and materials development. Sufficient research and development has been performed and operating experience been obtained to arrive at the following technical status.

(1) Physics. The physics of boiling water reactors is similar to that of pressurized water reactors although the system is complicated by the presence of steam voids.

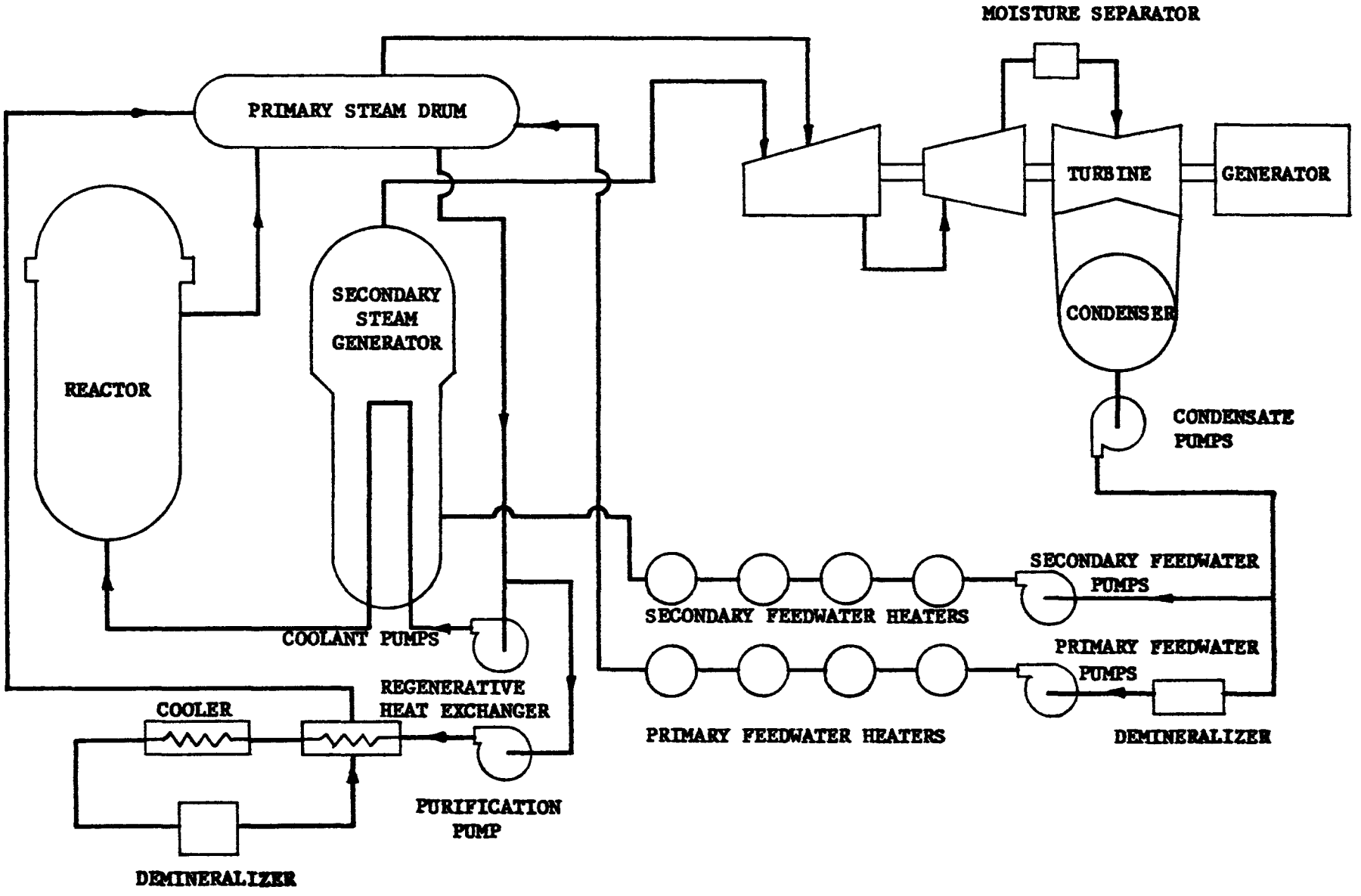
Most of the physics data has been largely concerned with the over-all behavior of boiling water reactors and has been obtained

from the BORAX experiments, SPERT, and EBWR. These reactors have used either metallic or ThO_2 fuels and therefore did not provide data directly applicable to reactors loaded with low enrichment UO_2. Although several reactors which will be fueled with UO_2 are under construction, no reactor completely loaded with this type of fuel has operated to date.

The presence of steam in a boiling reactor introduces a variable in the analysis which is not found in other reactor types. However, by using the proper water to fuel volume ratio, the desired void coefficient at a particular operating condition can be obtained. For large reactors, using oxide fuel, a water to UO_2 volume ratio of about two appears to be optimum. For smaller reactors, a larger water to fuel ratio is required to compensate for the effect of greater neutron leakage on the void coefficient.

The presence of voids also affects the power distribution by peaking the reactor power toward the bottom of the reactor. However, the voids also tend to flatten the power in the radial direction. Strategic radial positioning of the control rods will reduce the overall peaking factor, and the effect of the control rods in flattening the power is a very important parameter in steady state operation.

Control requirements of a light water boiling reactor are somewhat different from a non-boiling light water reactor of the same size. The non-boiling reactor will have a lower water to fuel ratio; and as such, will have a larger reactivity change with temperature. The lower reactivity change in the boiling reactor is offset by the additional reactivity requirements due to the steam coefficient of reactivity. However, the conversion ratio in the non-boiling reactor will tend to be higher, and the reactivity loss with burnout will be less. The net



SEPT 1959
FIG. 7

SIMPLIFIED FLOW DIAGRAM
BOILING WATER REACTOR PLANT

result is that boiling reactors usually require slightly more total control than non-boiling reactors, although the required control will depend very strongly on the reactor design. Control of the boiling water type reactor is made difficult by the fact that the void coefficient tends to oppose the steam demand changes.

During the past several years, an extensive program has been undertaken at Bettis and Brookhaven to measure the properties of slightly enriched uranium metal and UO_2 light water moderated lattices. These data, which are room temperature measurements, permit evaluation of theoretical methods used to determine lattice parameters.

Experimental measurement of reactivity effects and reactor control characteristics of operating power reactors have been reported only on the EBWR and BORAX-IV reactors. Since EBWR uses plate-type, metallic uranium alloy fuel elements and BORAX-IV uses a ThO_2-UO_2 mixture with a very high water to fuel ratio, it is not possible to make direct use of the experimental information in a rod-type UO_2 core.

Despite the apparent lack of information, initial criticality can be predicted to within 2 to 3% Δk for simple fuel elements, such as rod clusters with a water to fuel ratio of about 2 or slightly greater. The reported error in EBWR criticality at startup, based on calculations (no critical experiments) was 3.39% Δk. A similar degree of calculational error was attained for the total worth of control rods. This indicates the necessity of critical experiments for large central station boiling water reactors.

The advent of two-dimensional, multi-group diffusion theory has aided in removing a good measure of the geometric uncertainties. Therefore, for rod-type fuel arranged in a similar fashion to the cold

critical experiments used in determining lattice parameters, it is reasonable not to expect a large deviation for the initial criticality.

Although there is a large and growing amount of data on uniform UO_2-water lattices, this information is often insufficient to optimize precisely a practical reactor design. Practical reactors contain many more inhomogeneities (usually of a three-dimensional nature_ than have been present in most of the UO_2-water experiments. The ability to handle inhomogeneity effects is important in predicting criticality, control rod worth, temperature coefficient and power distribution. Adequate methods for dealing with this problem are being developed, but much work is still to be done.

The calculation of void and temperature coefficients suffers from lack of direct experimental measurement. However, cold critical mock-ups of void coefficient are in good agreement with theory (about 10%) for EBWR experiments. The temperature coefficient calculations for BORAX-IV are a factor of 2 lower than the measured values in the temperature range of 65°F. - 207°F. and at 420°F.

The experimental measurements of flux and power distribution in EBWR and BORAX-IV indicate a close correlation with calculations. On the other hand, there is no empirical knowledge of the operating void distributions.

The EBWR and BORAX-IV data indicate the type of power distribution expected in a boiling reactor. However, the EBWR is relatively small (4 ft. diameter and height) and the BORAX-IV was even smaller, so that a direct extrapolation of the power distribution to a large core is not possible.

No boiling reactor has operated to a long burnup, although

some of the fuel in EBWR has been irradiated to a maximum of 3300 MWD/MT. No theoretical calculations of reactivity changes occurring in an operating reactor have been compared with the measured changes.

The BORAX, SPERT, and EBWR experiments have furnished valuable information on the effect of increasing operating pressure and decreasing the thermal conductivity of the fuel elements. For example, until very recently, based on the early metallic fuel experiments, a maximum of 2% Δk in voids has been used as a design criterion to avoid instabilities. However, in the BORAX-IV experiment, using oxide fuel, 6.9% Δk in voids was achieved with no instability.

(2) Fuels and Materials. The most widely accepted fuel material for boiling water reactors is UO_2. This is due to its excellent corrosion resistance in water as well as its radiation damage resistance. In-pile tests (samples) indicate that low enrichment UO_2 fuel is radiation damage resistant to burnups greater than 27,500 MWD/T. It is relatively inert in hot water, retains fission products to a large degree, and is compatible with zirconium, stainless steel, and aluminum cladding. UO_2 pellets can be fabricated in various forms with densities of approximately 95% of the theoretical maximum. The thermal conductivity of UO_2 is low; typical data have indicated an average value of approximately 1.0 BTU/hr-ft-^{o}F. at operating temperatures.

Three cladding materials are being used in boiling water reactors: aluminum, zirconium, and stainless steel. Aluminum-nickel alloys have shown increased water corrosion resistance, but at present their long term integrity is questionable at temperatures exceeding 400^{o}F. Zirconium and zirconium alloys have been developed to the point where these materials are being used as fuel cladding and core structural material

in boiling water reactors. Good strength and water corrosion resistance properties have been reported for temperatures as high as 660°F. The low neutron absorption cross section of zirconium offsets its high cost relative to stainless steel. Stainless steel has been used extensively in reactors and has excellent corrosion and strength properties. In direct cycle boiling reactors, there has been little evidence of radioactive cobalt buildup in the steam loop because of corrosion of stainless steel structural components. In forced circulation loops and where an indirect cycle is used, reactors using stainless steel for fuel cladding may show significant accumulation of radioactivity.

Three types of control rod materials have been developed and are being used in water reactors - boron-stainless steel, hafnium, and rare earth oxides. These materials show excellent resistance to hot water corrosion. Boron-stainless steel is being used in most civilian power reactors because of the high cost and limited supply of hafnium; otherwise, hafnium would be preferable inasmuch as it does not suffer isotopic depletion or radiation damage.

(3) Heat Transfer and Fluid Flow. Most of the research and development to date on heat transfer and fluid flow has been centered on natural circulation boiling systems. Data are adequate to establish reactor design criteria but are not sufficient for obtaining the optimum performance necessary to yield the most economic design.

Neglecting stability requirements, heat transfer performance is limited by the center melting temperature of the fuel (5000°F. and UO_2) and by the maximum surface heat flux allowable at a particular flow velocity and subcooling, before burnout occurs. Present economic studies have indicated that a UO_2 fuel pellet size of approximately 0.4 to

0.5-inch O. D. should be used. A typical Zircaloy-2 clad, 0.45-inch O. D. pellet could operate at a maximum heat flux of approximately 400,000 BTU/hr-ft^2 with a centerline temperature of 4900°F; this is based on an average UO_2 thermal conductivity of 1.0 BTU/hr-ft-°F. This value of heat flux is well below established burnout heat fluxes for typical boiling reactor operating conditions.

One of the unknown areas in boiling water reactor technology exists in steam volume fractions and more specifically the velocity of steam relative to the velocity of water. However, sufficient work has been done to predict core performance.

The knowledge of the mechanism and factors influencing the separation of steam from steam/water mixtures is incomplete. The need for more information on steam/water separation becomes more acute as the sizes, pressure, and power density of boiling reactors are increased.

Natural circulation velocities up to 10 ft/sec have been obtained in non-nuclear laboratory experiments using a 4 ft.heated length of 1 inch diameter pipe having an 8 ft long, 2 inch diameter, riser above it. Geometries, more nearly like those used in actual reactors, give velocities which are much lower but still adequate for small cores.

Hydraulic stability has been shown to be a function of exit steam volume fraction, pressure and system geometry. Analytical methods have been developed to predict the hydraulic stability of a boiling water system. These have been checked by transfer function measurements on small reactors; however, more work is required in this area.

A considerable amount of experimental data has been

accumulated on burnout in water-cooled reactors. This data has been developed primarily on pressurized water systems but is applicable to the boiling water reactor. The current typical design limit is approximately 1.0×10^6 BTU/hr^2-hr for boiling water reactors.

(4) Coolant Chemistry. The areas of principal concern to date have been moderator decomposition and effects of "crud" produced by corrosion. Most of the experimental data available has come from the operation of the EBWR.

Virtually no experimental work has been done on the effects of oxygen in the steam of EBWR, but two years of operation indicate that the presence of 20 cc (STP) of oxygen per Kg of steam has no gross deleterious effect on construction materials in steam piping, turbine, or condenser.

The net decomposition observed is due to liquid phase recombination of H_2 and O_2 being hindered by the stripping of these gases from the liquid to the vapor phase by the steam bubbles. Argonne National Laboratory has reported that by bubbling CO_2 through water under gamma-irradiation, the recombination is completely inhibited and the water decomposes at the maximum theoretical rate.

In the EBWR at 20 Mw (thermal) and 600 psi, O_2 is produced at a rate of about 9 liters/min. The calculated rate of formation, assuming no recombination, would be three times this amount. Since experimental evidence indicates that very little recombination of H_2 and O_2 occurs in the steam phase, it must be concluded that only about one-third of the radiolytic gases are stripped from the water by the steam.

It has been found that if H_2 is added to the reactor feedwater at a rate such that the H_2 concentration in the core is 45 cc/kg,

the radiolysis is inhibited to such an extent that O_2 concentrations in the steam fall to less than 0.4 cc/Kg of steam at 20 Mw and 600 psi.

In systems containing only steels and Zircaloy, it is known that operation at pH 10-11 is effective in preventing crud deposition. In systems containing Al, this high pH could not be tolerated and either Al must be removed or some other suitable inhibitor be found.

(5) Components and Auxiliary Systems. In general, boiling water reactors have been able to use components and auxiliaries already developed for conventional steam plants or for non-boiling water reactors. However, the "conventional equipment" is often the major cause of operational delays and plant malfunctions. Specific developments have been required because of the water activity, water decomposition, and corrosion of materials in saturated steam.

Materials and technology are available for the design and construction of pressure vessels for 700°F. and 1500 psi operation in diameters of up to 15 feet. Corrosion of stainless clad vessels has not presented a problem as yet.

In the field of control rod drives, many types of drives have been developed -- screw jack, magnetic jack, rack and pinion, hydraulic -- all of which can be made to operate satisfactorily. Some bottom-mounted drives appear to suffer from buildup of radioactive crud which settles to the bottom of the thimbles, making maintenance more difficult. Methods of minimizing this problem in future reactor designs are being evolved.

Instrumentation for reactors has been developed rapidly by the electronics industry, and commercial units are readily available. More work is needed to increase the reliability of these instruments, since many scrams have been caused by false signals or inadequate

equipment. The greatest need for instrument development is in the in-pile measurement field where the reactor environmental conditions have seriously limited the type of materials which can be used.

Detection of even a minor fuel failure can be accomplished in a boiling reactor by monitoring the exhaust gases leaving the condenser for the fission product, xenon-135. Location of the particular fuel element which has failed is difficult. More development effort is required toward this end.

(6) Reactor Safety. Boiling water reactors which have large steam expansion volumes above the core have been shown to be inherently safe against plausible reactor excursions. Possibilities of metal-water reactions have been virtually eliminated by the use of oxide fuel, which does not undergo an exothermic reaction. The fuel clad, if zirconium, aluminum, or stainless steel, can enter into a metal-water reaction. However, the possibility of such a reaction occurring is remote, since conditions normally required for such a reaction (fine dispersal of the molten material into the water) are not readily attainable.

c. Operating Experience. In the short life span of BORAX-I, a large number of excursion type tests and a small number of steady-state boiling experiments were run. These tests demonstrated that boiling water reactors could operate stably and had a high degree of inherent safety due to shutdown by void formation. Fluctuations in power of the order of 15% were noted, but no instability tendency was detected. Beyond approximately 1.5% k_{eff} in voids, a pattern of oscillations was established which led to unstable operation or "chugging" when about 2.5% k_{eff} was held in voids at atmospheric pressure. Operation at higher pressures increased stability. The reactor operated stably at about 1200 kw, 135 psig.

A final experiment, which resulted in the self-destruction of BORAX-I, was made by adding approximately 4.0% k_{eff} in 0.2 seconds with the reactor water initially at room temperature. The power during this test increased on a period of 0.0026 seconds and reached a maximum of 18,000 megawatts. It is significant to note that relatively little damage to the equipment outside the pressure vessel or to the control rod drive mechanisms above the vessel occurred.

Examination of the debris around the reactor showed that a large fraction of the fuel plate material had been melted. However, it is not known whether an explosive chemical reaction occurred between the molten metal (Al and U) and the water. If such a reaction did occur, it was far from complete, as evidenced by the large quantity of unoxidized aluminum fuel fragments.

BORAX-II was similar to BORAX-I but had a core approximately twice the size of BORAX-I and a pressure vessel for operation of the reactor at 300 psig.

This reactor was only operated for a short time in 1954 and 1955, during which time tests similar to those conducted on BORAX-I were made at atmospheric and at 300 psig pressures. The tests served to further indicate that stable steady-state boiling operation at power densities suitable for power reactors could be obtained with pressurized, boiling water reactors. Stable operation under steady-state boiling was obtained in BORAX-II, pressurized to 300 psig, with a power density of $\approx$27 kw/liter of core (765 kw/ft^3) with up to 3.2% k_{eff} in voids. The stable reactor behavior during the tests led to the installation of a turbine generator plant with the reactor.

BORAX-III was essentially the same as BORAX-II with new fuel elements and a turbine generator added. The BORAX-III power plant was successfully operated at maximum power levels of 16 MWT and 3.5 MWE and represented the first boiling water reactor to generate electric power. The reactor plant operated stably at all pressures from atmospheric to 300 psig.

BORAX-IV was the BORAX-III reactor plant with a new core substituted for the BORAX-III core. The new core consisted of fuel elements using ThO_2-UO_2 fuel canned in aluminum and bonded with lead. Primary emphasis in the experimental program of this reactor was given to obtaining more detailed data on the stability and safety of boiling water reactor plants. Transient and steady-state power runs similar to the previous BORAX experiments were made and reactor transfer function measurements were made using oscillator techniques. At atmospheric pressure, this reactor was found to be unstable at a power level of 4.57 MWT with 1.5% k_{eff} held in voids. As the pressure was increased to 322 psig, stable operation was achieved up to 20.5 MWT with 6.9% k_{eff} held in steam voids.

Maximum power densities were: 19.6 KW/liter of core (555 KW/ft^3) at atmospheric pressure with 1.5% k_{eff} held in voids and 45 KW/liter (1275 KW/ft^3) at 300 psig with 6.9% k_{eff} held in voids.

Besides the steady-state and transient operating data, BORAX-IV provided information on radiolytic decomposition and experimental data on the contamination of the reactor coolant loop due to fuel clad failures. These fuel clad failures were detected in February

1958 by the sudden increase in activity in the turbine room to 70 mr/hr. Activity levels of 400 mr/hr were noted in the turbine room and 8,000 mr/hr in the hot well following further reactor operation with the failed elements. Examination of the Borax-IV fuel elements later indicated that at least 17 elements had failed. These failures were attributed to leaks caused by the collapse of the aluminum into the void space above the oxide pellets after repeated thermal cycling in the reactor. It is significant that operation with the failed fuel elements, although causing a relatively high activity level in the plant during operation, caused no significant contamination of the turbine and little increase in the gross activity of the water. The radioactivity in the turbine was very low when the reactor was shut down. This observation would indicate that mostly gaseous fission products were released by the oxide elements.

Surface activities at various points in the BORAX-IV plant were measured while the reactor operated with the failed fuel elements. The results of these measurements indicated no serious activity problem existed.

The Experimental Boiling Water Reactor at Argonne National Laboratory is a complete experimental nuclear power generating station designed to provide operating and design data on boiling water reactors suitable for power generation.

The radiation levels in the EBWR plant have indicated that a direct cycle boiling water reactor could be operated without undue activity at the turbine and that normal maintenance was not a problem.

On three occasions failure of the bearing in the feed-water pump occurred. This is a problem of pump design rather an a difficulty peculiar to a nuclear plant.

Frequent difficulty with "sticking" of the control rods was experienced during the early operation of the EBWR. This problem is associated with "crud" accumulation in the rod seal housing at the bottom of the reactor vessel. It has been largely corrected by substituting bushings made of softer material.

This crud accumulation in the control rod thimbles results also in a high activity level in the immediate vicinity of the control rod thimbles. Activity levels of the order of 65-70 r/hr have been reported. This would probably not be a major problem if provisions had been made for flushing the control rod thimbles.

Difficulties were experienced with the H_2- O_2 catalytic recombiner during the early operation of the EBWR. Frequent explosions of the hydrogen-oxygen mixture have been experienced. This system is now operating satisfactorily.

Failure of the stem of the turbine trip throttle valve occurred. A combination of improper operating procedures and incorrect design of the stem were given as reasons for the failure.

A turbine blade failure, not associated in any way with nuclear phenomena was experienced. Servicing operations were carried out on the slightly radioactive (about 2 mr/hr) machine in a conventional manner.

A leak was detected in a tube of the main condenser. At reduced power level no difficulty with activity was involved in plugging the defective tube.

The majority of operating problems that have been experienced, except for the control rods, were on the so-called conventional equipment in the plant. There have been no significant problems that were nuclear in origin.

With the pressure control by-pass valve system there has been no difficulty in synchronizing the power outputs of the reactor and the load of the turbine generator. Operation has also indicated that manual operation of the reactor without the by-pass valve in service can be accomplished to handle slow load changes on the turbine.

Three sets of planned defective fuel element (holes in the cladding) experiments have been run in EBWR on ThO_2 ceramic fuel enriched with 10% fully enriched UO_2. The results from these tests were:

Only fission product gases were evolved to any significant extent, and the rate of release was much less than 1% of the production rate of fission gas.

No increase over normal readings was registered by any radiation monitor measuring gross activity in the vicinity of plant components except during power increases. The fact that a cladding defect existed was readily detected, however, by scintillation counters which selectively measured Xe-135 activity in off-gases from the main condenser. Short bursts of activity were also registered on gamma monitors in the vicinity of the air ejectors when reactor power level

increases were made. The bursts were evidently due to expulsion of fission gas from the oxide element due to volumetric expansion of gas as a function of temperature. At constant power level, there is no such expulsion tendency, and fission gases escape by a much slower diffusion process. The amount of fission product activity released in this way is small compared with the N-16 normally present in the steam and therefore does not contribute significantly to the gross radioactivity.

Samples of reactor water taken from above the element containing the defected specimen showed no increased counting rate either due to radioiodines or to delayed neutrons, when compared to similar samples taken from above fuel elements containing no defected specimen.

One defected sample was placed close to No. 3 control rod. When this rod was moved to increase the flux in the specimen, a burst of activity was recorded on the gamma monitor at the air ejectors. Movement of any of the other eight rods did not affect monitor readings. This indicates that defects in the close vicinity of a control rod could be detected by proper manipulation of that rod.

During almost four months' steady operation during Test #2, no accumulation of any fission product could be detected in the turbine or steam pipe. The details of these tests are given in the Boiling Water Reactor Status Report. The Vallecitos Boiling Water Reactor (VBWR), which is owned and operated by the General Electric Company, is a boiling water reactor generating steam which is utilized directly in a turbine generator. The reactor operates at a nominal pressure of 1000 psig. The plant is designed as a test plant.

The reactor is licensed to operate at 30 MWT and the turbine generator is rated at 5200 KW. Excess steam is bypassed to the condenser.

Because of the high operating pressure (1000 psig), a very high power density was achieved in this reactor without excessive steam voids. An average power density of about 50 KW/liter of core was obtained at 30 MWT operation with about 1.75% k_{eff} held in steam voids. Because of the thin fuel elements used in this reactor (0.025 inch) and the large separation of fuel plates (0.49 inch), this corresponds to a power density of approximately 58 KW/liter of coolant. The possibility of obtaining stable operation at power densities of greater than 100 KW/liter of core with natural circulation is anticipated with this thin, fully enriched fuel element.

Radiolytic decomposition of the water decreased with the higher operating pressure, as was found in comparing the decomposition rates of BORAX-IV at 300 psig and EBWR at 600 psig.

At the highest power operation, radiation levels in the turbine building were 6 mr/hr or less except in the vicinity of the condenser, and 15 mr/hr at the high pressure end of the turbine. This again substantiates the previous data obtained from EBWR and BORAX-IV which indicated that although some activity is present in the direct steam system, little if any shielding is required.

Measurements of the steam void content from the hottest element indicated that at 20 MWT approximately 35% by volume of steam was contained in the coolant. Assuming stable operation at 80 MWT, it was calculated that about 72% by volume of steam would be in the coolant from the hottest channel.

The fuel element in this reactor has a thermal time constant of about 0.06 sec., which is much faster than that obtained with oxide fuel. It is expected that more stable steady state operation will result when an oxide fuel is used in combination with the higher operating pressure.

The Argonne Low Power Reactor (ALPR) is a natural circulation, light-water-moderated and -cooled boiling reactor which operates at 300 psig, 420°F. and produces 3000 KWT of heat. The plant generates a gross electrical output of 300 KW and provides 400 KW of space heat.

The reactor has been in operation for approximately six months and has operated satisfactorily with one notable exception. The failure of an oil seal in the deep well pump supplying the reactor make-up water resulted in the introduction of a significant quantity of oil into the reactor primary system. As a consequence, the demineralizers were rendered useless and extensive contamination of the fuel plates, core structure, reactor vessel and piping occurred. The oil-contaminated water has been removed and the system cleaned to the greatest extent possible. The detrimental effects of this incident upon overall plant performance, if any, will not be known for some time.

In general, because of the special design aspects of this reactor and its use of highly enriched fuel, very little data will be forthcoming of a nature suitable for extrapolation to more economic power reactors. However, operating history of the components will be of value.

The operating experience on boiling water reactor has been mostly on small experimental plants. The results of this experience appear satisfactory but experience is needed with large boiling water reactor

plants. The operating experience of the BORAX reactors and EBWR are shown in Fig. 8 and Fig. 9, respectively.

d. Schedule. The boiling water reactors under construction that will provide valuable data to advance the technology are shown in Fig. 10.

e. Economics. A reference design has been made for a boiling water reactor based on the technical status summarized in this report. This design was used as the basis for extrapolating the cost to different plant ratings. The plant characteristics for the reference design are as follows:

SUMMARY OF PLANT CHARACTERISTICS
BOILING WATER REACTOR PLANT

A.	Heat Balance			
	1. Total Reactor Power, MW(t)		690	
	2. Gross Turbine Power, MW(e)		212	
	3. Net Plant Power, Mw(e)		200	
	4. Net Plant Efficiency, %		29.0	
B.	Turbine Cycle Conditions	primary		secondary
	1. Throttle Temperature, F.	540		460
	2. Throttle Pressure, psig	950		460
	3. Steam Flow, lbs/hr	1.44×10^6		1.21×10^6
	4. Condenser Back-Pressure, in. Hg A.		1.5	
	5. Final Feed Water Temperature, F.	565		405
C.	Reactor Description			
	1. Reactor Vessel			
	a. Inside Diameter, ft.		12.25	
	b. Overall Height, ft.		44.0	
	c. Wall Thickness, in.		5.675	
	d. Material		SS clad CS.	
	e. Design Pressure, psia		1500	
	2. Reactor Core			
	a. Active Diameter, ft.		10.5	
	b. Active Height, ft.		9.75	
	c. Active Core Volume, ft^3		845	
	d. Lattice Arrangement		square	
	3. Reflector or Blanket			
	a. Material		H_2O	
	b. Radial Thickness, ft.		~1	

4.	Fuel Elements	
	a. Fuel Material	UO_2
	b. Clad Material	ZR-2
	c. Fuel Enrichment	1.5
	d. Fuel Element Geometry	rods
	e. Cladding Thickness	0.030
5.	Material Inventories	
	a. Fuel, metric tons	66.5
	b. Uranium, metric tons	52.3
	c. U-235, initial-kg	785
6.	Reactor Control	
	a. Method of Control	rods
	b. No. of Control Elements	84
D.	Plant Performance Data	
1.	Primary Coolant Outlet Temp., F.	545.3
2.	Primary Coolant Inlet Temp., F.	505
3.	Reactor Temp. Drop., F.	40.3
4.	Primary System Operating Pressure, psia	1015
5.	Primary coolant Flow Rate, lbs/hr.	1.43×10^6
6.	Avg. Core Heat Flux, Btu/hr.-ft^2	97,700
7.	Max. Core Heat Flux, Btu/hr.-ft^2	277,000
8.	Max. Cladding Surface Temp., F.	585
9.	Max. Fuel Temp., F.	4500
10.	Core Coolant Velocity, ft/sec.	--
11.	Peak to Avg. Power Ratio	2.92
12.	Core Power Density kw/ft^2	817
13.	Core Specific Power kwt/metric ton-U	13,200
14.	Fuel Burn-up MWD/metric ton-U (average)	11,000

The power cost of a 300 MWE boiling water reactor plant extrapolated from the above design is as follows:

Total Capital Cost ---- $78,900,000	
M/KWH ----------	5.26
Fuel Cycle Cost - M/KWH ----------	3.47
Operation and Maintenance Cost M/KWH ------	.61
Nuclear Insurance Cost - M/KWH ----------	.27
Total Power Cost - M/KWH ------	9.61

The relationship of Power Cost vs. size is shown in Fig. 11.

The above costs are representative of plants that could be constructed with the current technological status: However, these costs could only be achieved after the period of time required for plant construction and fuel cycle equilibrium. The layouts and other design data for determining the economics is included in report S & L 1674 and Appendix II of this report.

The cost data is based on a boiling water reactor utilizing a dual cycle arrangement. This type plant was extrapolated to the smaller sizes in order to show a consistent cost curve. In all likelihood a natural circulation - direct cycle plant would be used in smaller sizes at better economics than those indicated on the curve. It is currently thought that the capital cost could be reduced if a direct cycle was practical in the large plant sizes. No advantage was taken for this potential cost improvement although it could probably be done with current technology.

A reference cost point would be the published figures on the Senn Reactor which are $48,000,000 or 320 $/KW (6.4 M/Kwh).

OPERATING HISTORY OF

BORAX III and BORAX IV

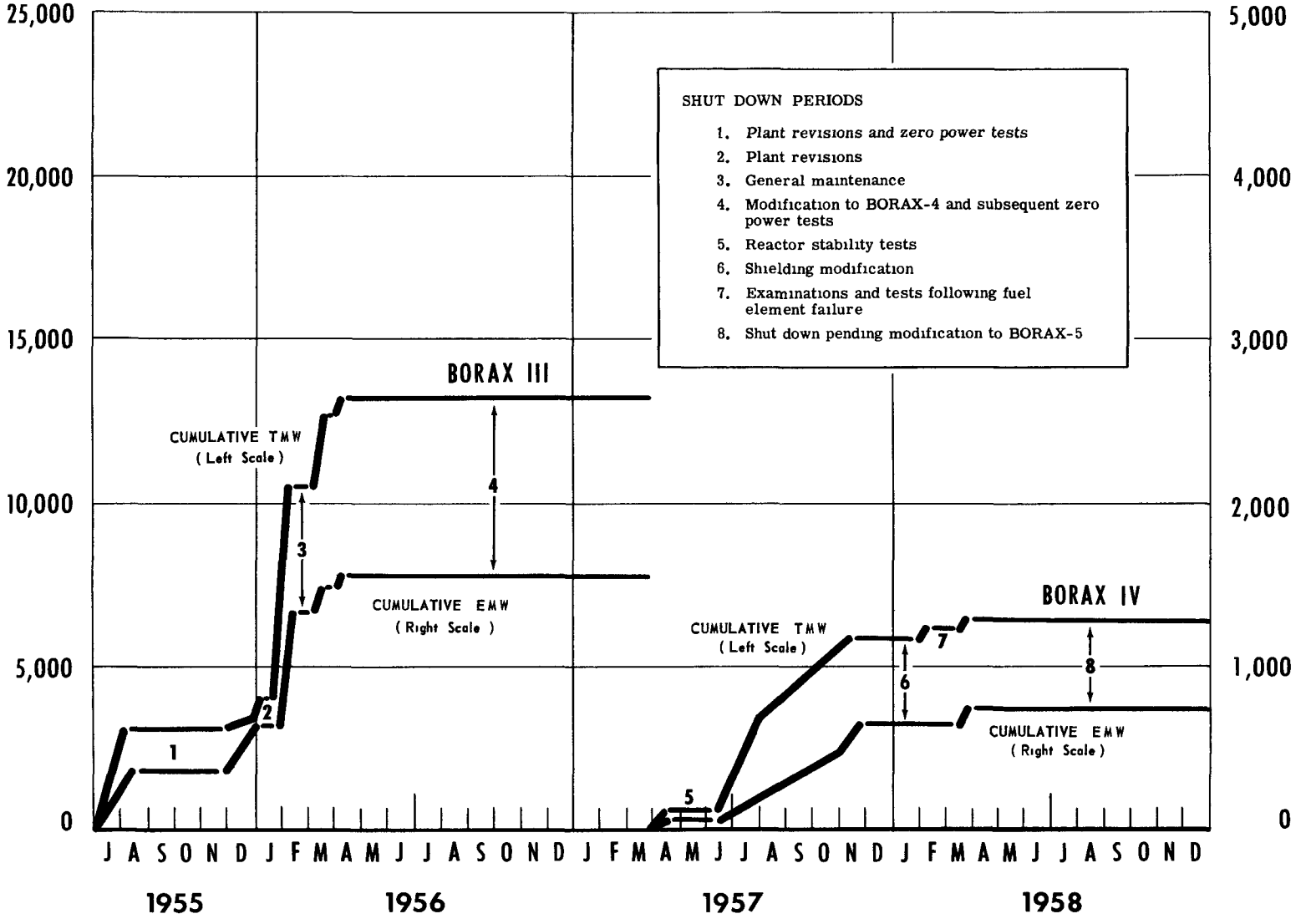


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FIG. 8

OPERATING HISTORY OF

EXPERIMENTAL BOILING WATER REACTOR

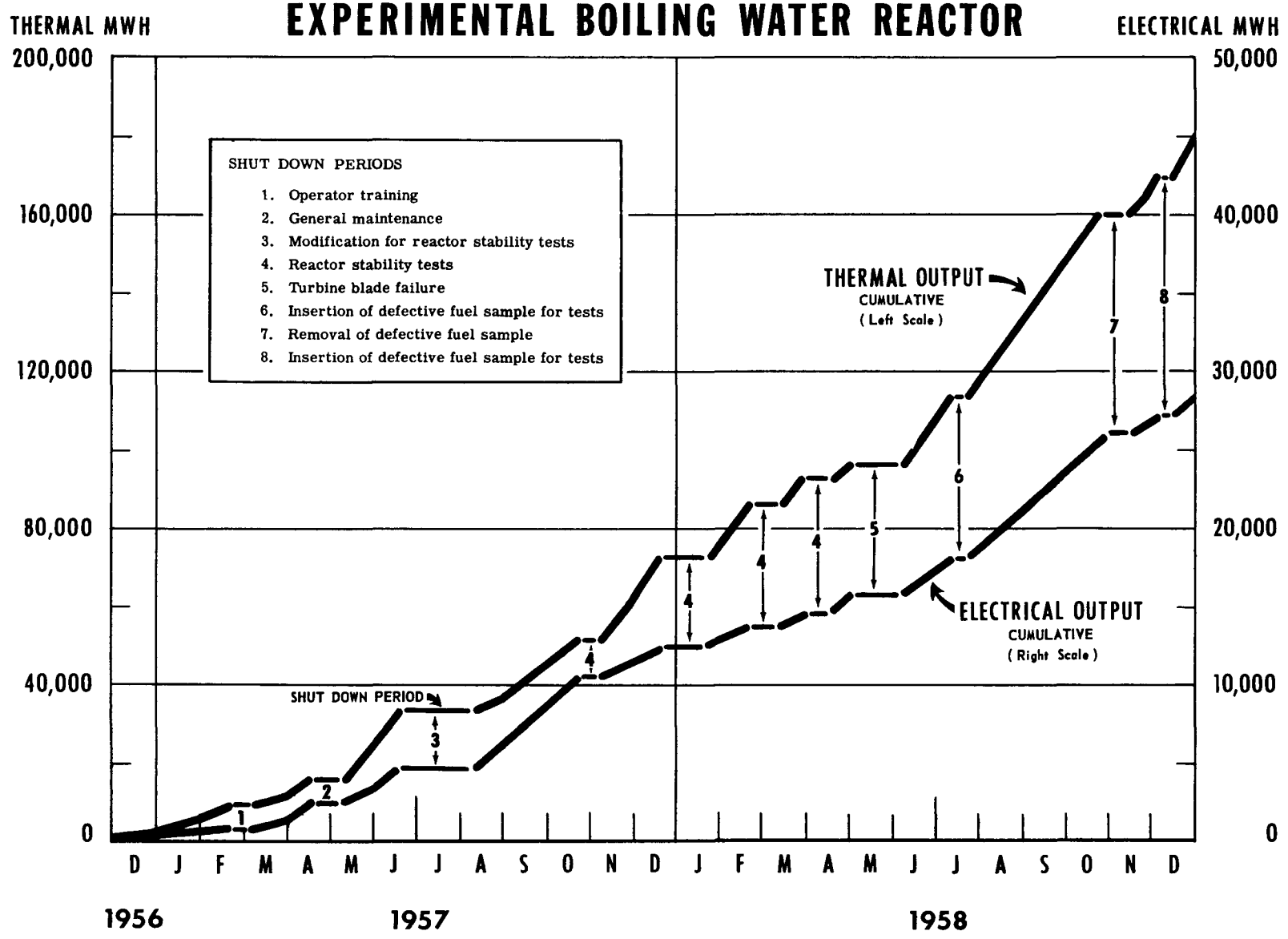

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FIG. 9

BOILING WATER REACTORS
CONSTRUCTION SCHEDULE

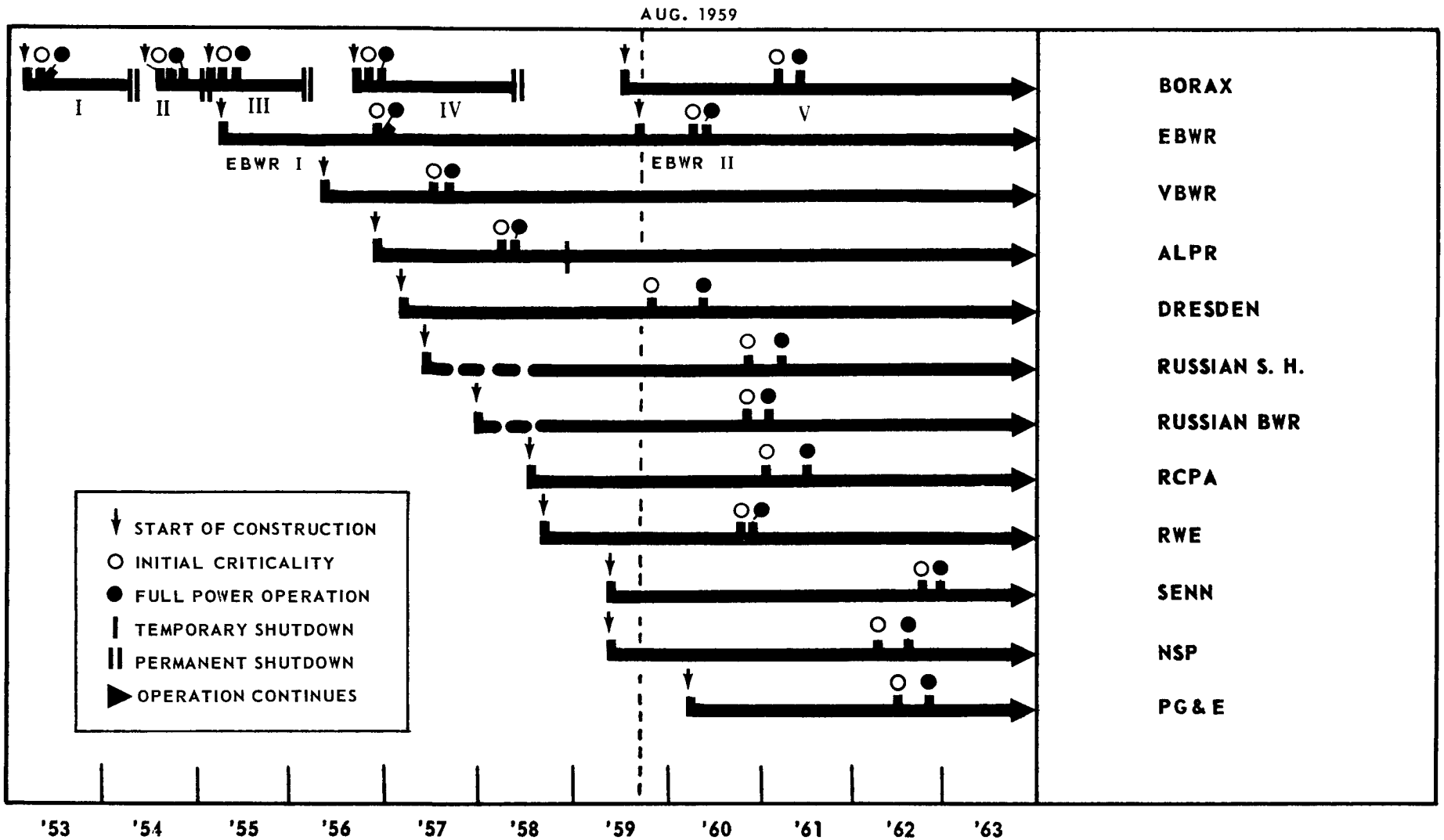

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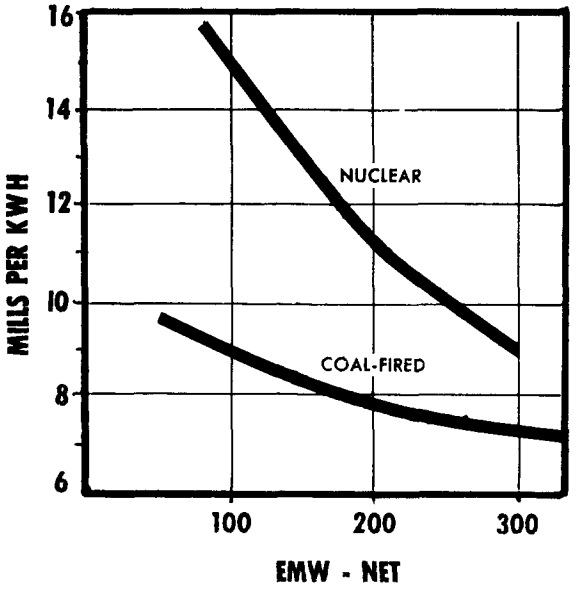
FIG. 10

POWER GENERATION COSTS

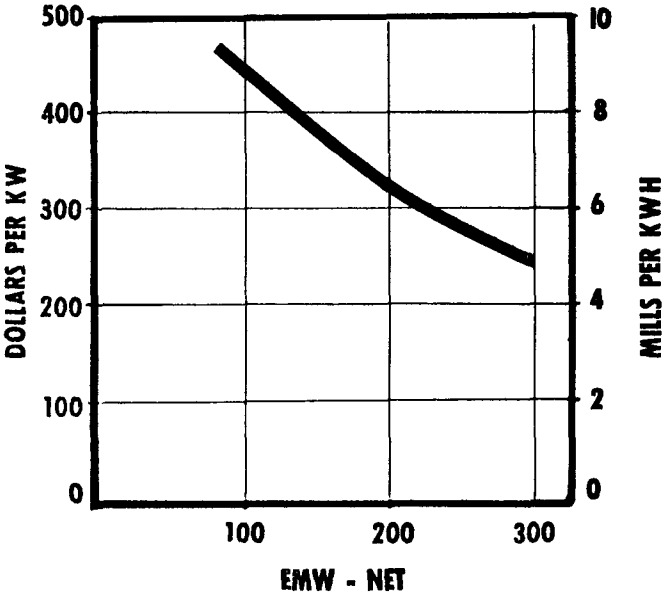
BOILING WATER REACTORS
VS
COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.

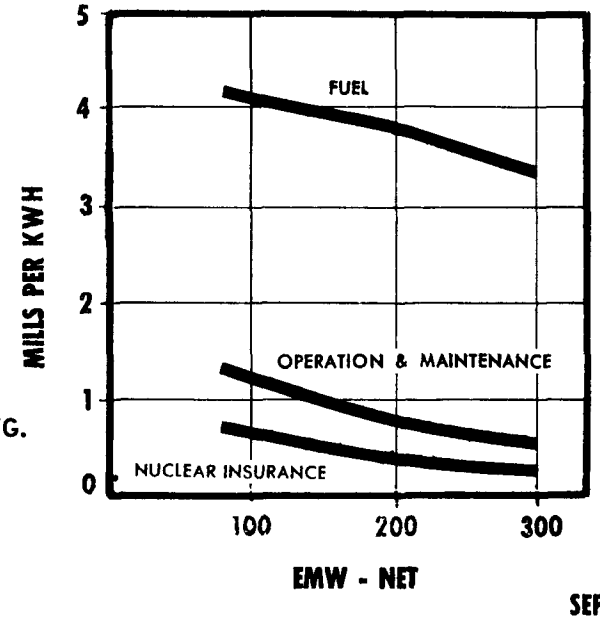
POWER GENERATION COSTS



CAPITAL COSTS



OPERATING COSTS



NOTES :

NUCLEAR -

BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/GM.
FUEL FABRICATION COST	
UO_2 - ZR-2	$140/KG.
FUEL EXPOSURE	11000 MWD/MT AVG.

COAL - FIRED -

BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%
FUEL COST	35¢/10^6 BTU

SEPT 1959
FIG. 11

3. ORGANIC COOLED REACTOR

a. Description. The organic cooled reactor is a heterogeneous fueled, thermal reactor utilizing an organic material as the coolant. The reactor is normally moderated by the same fluid utilized as a coolant, however other moderator materials can be used. The heat is removed from the fuel by the organic coolant and transmitted to a steam boiler, where steam is formed to drive the turbine. A simplified flow diagram of the organic cooled reactor plant is shown in Figure 12.

b. Technical Status. Sufficient research and development work has been done to arrive at the following technical status:

(1) Physics. The physics of organic moderated reactors is similar to that of water reactors since the principle means of neutron energy degradation is by scattering collisions with hydrogen. Sufficient information is known from data on water critical experiments, the OMRE, on prototype water reactors, and from the organic experimental work to extrapolate data that would be representative of a large power reactor. However, with any large hydrogenous moderated power system, xenon stability may be a problem. The precise conditions for xenon oscillations are not thoroughly understood. How this instability can be avoided or what operational methods are best to handle it are questions that are only beginning to be answered by building, operating, and experimenting with large reactors which contain such instabilities.

The physics of more sophisticated multi-region cores and cores in which power flattening techniques are used are not as advanced and additional physics work is required to confirm the designs.

(2) Fuel and Material Properties. Initial fuel work was directed toward the development of low alloy metal fuel which would exhibit radiation

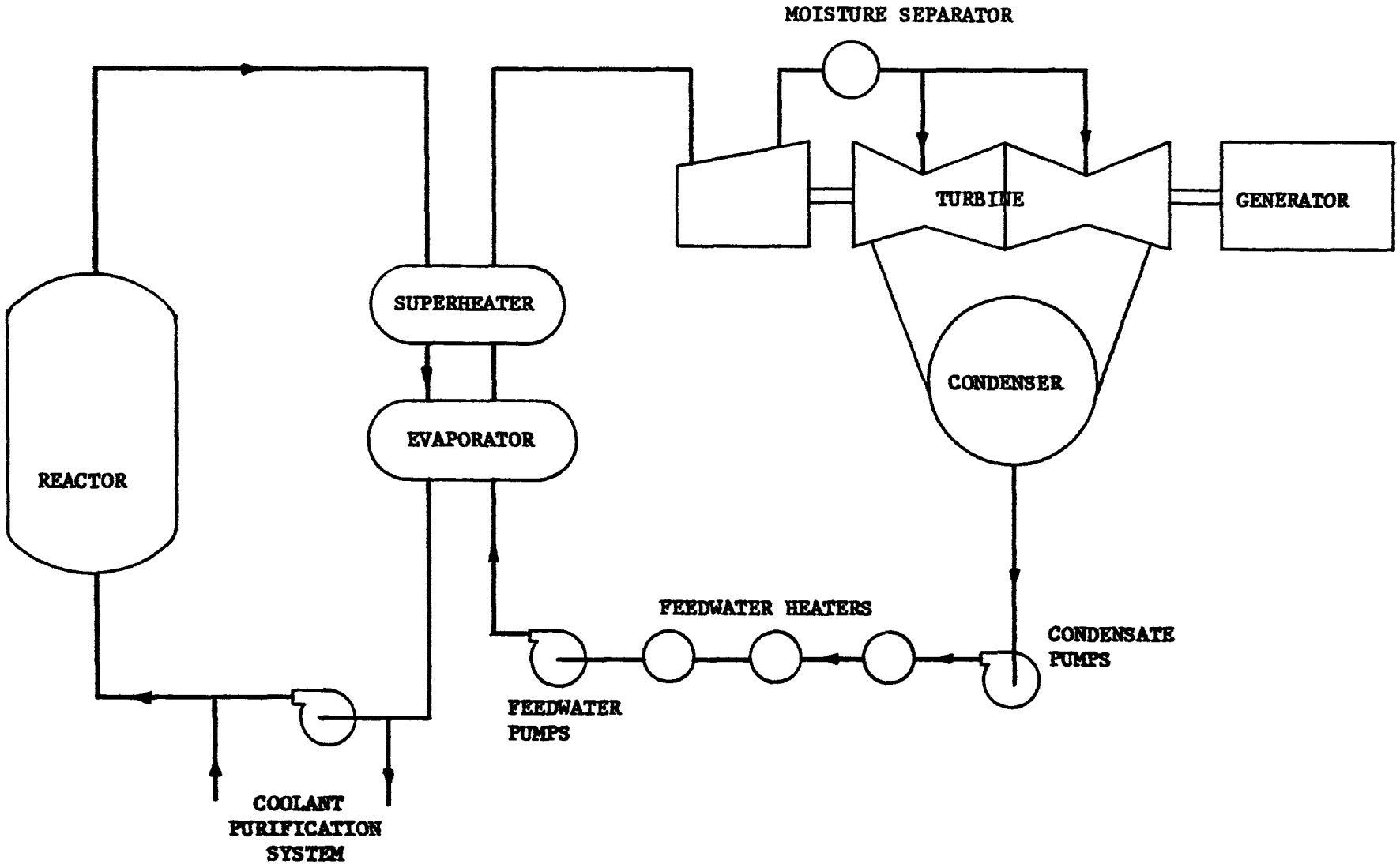
stability for reasonable exposures. From these studies and developments has evolved the U-3.5 wt% Mo + 0.1% Al alloy. These alloys have been operated up to 3,300 MWD/MT without distortion. The problems of predicting the upper limits of burnup for U-Mo fuel is difficult because of lack of test experience. It may be possible to achieve a maximum of 11,000 MWD/MT in an organic reactor at the operating temperature compatible with aluminum cladding.

Recent work on organic reactor fuel elements has been devoted to an investigation of UO_2 clad with aluminum powder metallurgy (APM) materials. Tests to date indicate that these cladding materials exhibit significantly higher strength than aluminum alloys at elevated temperature, exhibit satisfactory radiation stability, are compatible with UO_2, do not suffer from porosity, can be fabricated into the desired shapes, and can be joined satisfactorily utilizing welding techniques. More work needs to be performed on end closures and in general on aluminum powder metallurgy. As of this date, full-size prototype fuel elements of UO_2 APM materials have not been fabricated nor has full-scale in-pile testing been started.

(3) Heat Transfer and Fluid Flow. Limited tests on heat transfer in the organic cooled reactor indicated that the current limits are $\approx$ 500,000 BTU/ft^2-hr for burnout. This results in a limit of $\approx$235,000 BTU/ft^2-hr maximum design value and $\approx$ 60,000 BTU/ft^2 average for forced connection systems.

Correlations of heat transfer data have been obtained from limited duration tests over the range of surface temperatures from 600° to 900°F. These tests have indicated that under suitable conditions of forced convection heat transfer, the surface condition and temperature of the organic is stable.

The current status dictates aluminum cladding material whose physical properties limit the bulk coolant temperature to approximately 600°F.



SEPT 1959
FIG. 12

SIMPLIFIED FLOW DIAGRAM
ORGANIC COOLED REACTOR PLANT

Based on a maximum high boiler content of 30%, non-boiling, the organic makeup would be 25 lb/TMWD based on 4% gamma energy absorbed in the moderator.

The determination of the true burnout heat flux is complicated by the fact that the organic, under reactor operating conditions, is a completely miscible variable mixture of organic compounds.

Limited nucleate boiling data has been obtained with one run of 1,500 hours which indicates there is an initial decrease in heat transfer coefficient but that after 900 to 1,000 hours, the coefficient stabilizes at a value approximately 10 to 12% below the initial value.

Additional work in all phases of forced convection and boiling heat transfer, including the accurate determination of burnout heat fluxes, must be completed prior to the operation of large organic cooled power reactors. Forced convection appears to prevent fouling or deposition of carboniferous material on the fuel elements.

Heat transfer is low compared to water (about 1/5) and sodium systems; hence, a high surface to volume fuel element design is required. The system has a low specific power and a low power density.

(4) Coolant Chemistry. Considerable information has been developed in recent years on the physical and chemical properties of organic compounds. To date the investigations have been limited primarily to the polyphenyls, benzine derivities. Information is available for both unirradiated and irradiated material. The compound currently favored is Santowax R. This has a liquid density of 0.83 gm/cm^3 at 700°F and a thermal conductivity of 0.064 BTU/hr-ft-°F at the same temperature.

Organics have low vapor pressures, low induced activities, and low corrosion rates with standard materials of construction. An excessive radiolytic decomposition rate may result in high coolant makeup costs. More irradiation tests at high temperature and high reactor power levels are needed to determine long-term effects.

Decomposition may result in fouling of the material especially with nucleate boiling. Tests indicate that the coolant decomposition rate -- and thus the coolant makeup cost -- decreases with increasing high boiler (pyrolytic decomposition products) concentration.

(5) Component and Auxiliary Systems. Many components utilized in the organic system are of standard manufacture and have been utilized for several years in the petroleum industry. These items require very limited or no modification to be acceptable. However, it should be noted that in a nuclear plant it is often the "conventional equipment" which is the major cause of operational delays and plant malfunctions.

Process system design results to date have been obtained primarily in laboratory scale mockups. Mockups of the Piqua systems are now being completed for testing and evaluation. A magnetic jack control rod has been developed and successfully tested.

(6) Safety. Organic moderated reactors have large negative temperature coefficients, negative void coefficients, and negative power coefficients.

The physical properties of the coolant are such that there is negligible corrosion of either reactor or fuel materials if water and other deleterious impurities are kept out; there are no exothermic

chemical reactions between the coolant, fuel, or water; there is low induced radioactivity; and the system operates at low pressures due to the low coolant vapor pressure. However, the presence of organic compounds is a potential fire hazard.

Disposal of the high boiler residue is a problem.

c. Operating Experience. The only operating organic cooled reactor is the Organic Moderated Reactor Experiment which achieved criticality in September 1957. Initial criticality was followed by a few months of low power tests and physics measurements. Full power operation began in February 1958. The operation of the reactor has demonstrated the organic radiation decomposition rates that can be expected, the low induced activity, the low corrosion rates, and the technical feasibility of designing and operating organic cooled reactor plants.

Sustained power operation began February 1, 1958. Between February and September 15, 1958, 765 MWD of power operation was accumulated during a series of tests to determine the radiolytic and thermal decomposition rates and the heat transfer properties of the coolant moderator under a variety of operating conditions. During this period, tests were conducted at high boiler concentrations (HBC) of approximately 12, 30, and 40 percent, and a bulk coolant temperatures of 600°F (316°C) and 700°F (399°C). The fuel-plate surface hot-spot temperature was maintained at 750°F (399°C). during the majority of this period. Decontamination of the coolant and new buildup of decomposition product concentration to 30 percent were continued from September 15 through November 24, 1958. The accumulated reactor power increased to 950 MWD during this period. The reactor was then shut down for complete cleanup and removal of inorganic particulate

matter which had accumulated in the system, as well as installation of the second core loading. The operating experience of OMRE is shown on Figure 13.

d. Plants Under Construction. At present the only organic cooled reactor under construction is the Piqua plant, a unit rated at 11.5 MWE. The Piqua plant is scheduled to go into operation the latter part of 1961. Steam is to be produced at 450 psia and 550°F with a reactor coolant outlet temperature of 575°F and a maximum fuel element surface temperature of 750°F. The operation of the Piqua plant will supply valuable information on process system performance, plant control and stability characteristics, and further information on coolant and fuel element performance.

An Experimental Organic Cooled Reactor is scheduled for initiation of construction in FY 1960. The Experimental Organic Cooled Reactor will have test loops to enable simultaneous evaluation of several different organic coolants with and without inhibitors. Facilities will also be included to allow for extensive tests and optimization of design of fuel elements and components which are found to limit power level of organic cooled power demonstration reactors. The size and power level will be sufficient to allow irradiation of new fuel elements in quantities large enough to provide a statistical evaluation of the performance of the fuel elements.

A schedule for the Organic Moderated Reactor Experiment, Piqua, and the Experimental Organic Cooled Reactor are shown on Figure 14.

e. Economics. A reference design has been made for the organic cooled reactor based on the technical status summarized in this report.

This design was used as the basis for estimating the cost at different plant ratings. The base parameters for the reference design are as follows:

SUMMARY OF PLANT CHARACTERISTICS
ORGANIC COOLED REACTOR PLANT

A.	Heat Balance	
	1. Total Reactor Power, MW(t)	260
	2. Gross Turbine Power, MW(e)	79
	3. Net Plant Power, MW(e)	75
	4. Net Plant Efficiency, %	28.5
B.	Turbine Cycle Conditions	
	1. Throttle Temperature, F.	550
	2. Throttle Pressure, psig	585
	3. Steam Flow, lbs/hr.	9.36×10^5
	4. Condenser Back-Pressure, in. Hg A.	1.5
	5. Final Feed Water Temperature, F.	360
C.	Reactor Description	
	1. Reactor Vessel	
	a. Inside Diameter, ft.	11.0
	b. Overall Height, ft.	51.0
	c. Wall Thickness, in.	3.5
	d. Material	CS
	e. Design Pressure, psia	300
	2. Reactor Core	
	a. Active Diameter, ft.	9.5
	b. Active Height, ft.	9.5
	c. Active Core Volume, ft^3	674
	d. Lattice Arrangement	triangular
	e. Lattice Spacing, in.	6
	3. Reflector or Blanket	
	a. Material	Santowax R
	b. Axial Thickness, ft.	-
	c. Radial Thickness, ft.	0.75
	4. Fuel Elements	
	a. Fuel Material	U.3.5 w/o Mo
	b. Clad Material	AL.
	c. Fuel Enrichment	1.6
	d. Fuel Element Geometry	cylinder
	e. Cladding Thickness	0.035

5.	Material Inventories	
	a. Fuel, metric tons	41.4
	b. Uranium, metric tons	41
	c. U-235, initial-kg.	656
6.	Reactor Control	
	a. Method of control	rods
	b. No. of Control Elements	19
D.	Plant Performance Data	
1.	Primary Coolant Outlet Temp., F.	575
2.	Primary Coolant Inlet Temp., F.	490
3.	Reactor Temp. Drop., F.	85
4.	Primary System Operating Pressure, psia	150
5.	Primary Coolant Flow Rate, lbs/hr.	19.8×10^6
6.	Avg. Core Heat Flux, Btu/hr. - ft^2	28,000
7.	Max. Core Heat Flux, Btu/hr. - ft^2	112,000
8.	Max. Cladding Surface Temp., F.	750
9.	Max. Fuel Temp., F.	-
10.	Core Coolant Velocity, ft/sec.	15 (max.)
11.	Peak to Avg. Power Ratio	4.0
12.	Core Power Density kw/ft^3	386
13.	Core Specific Power kwt/metric ton - U	6340
14.	Fuel Burn-up MWD/metric ton - U (average)	4500

The reference design for the organic cooled reactor is at 75 MWE due to the fact that more design data was available for the smaller plant sizes. This will probably have some effect on the relative scaling factor (less accurate in larger sizes) used for the organic reactor cost curves as compared to the slope of the cost curves on the other reactor which used a larger reference design.

The power cost for a 300 MWE plant extrapolated from the above design is as follows:

Total Capital Cost - $66,122,561

----------	4.31	M/Kwh
Fuel Cycle Cost (U-3½ w/o Mo. ---------	5.72	M/Kwh
* Operation and Maintenance ----------	1.06	M/Kwh
Nuclear Insurance Cost ----------	.26	M/Kwh
Total ----------	11.35	M/Kwh

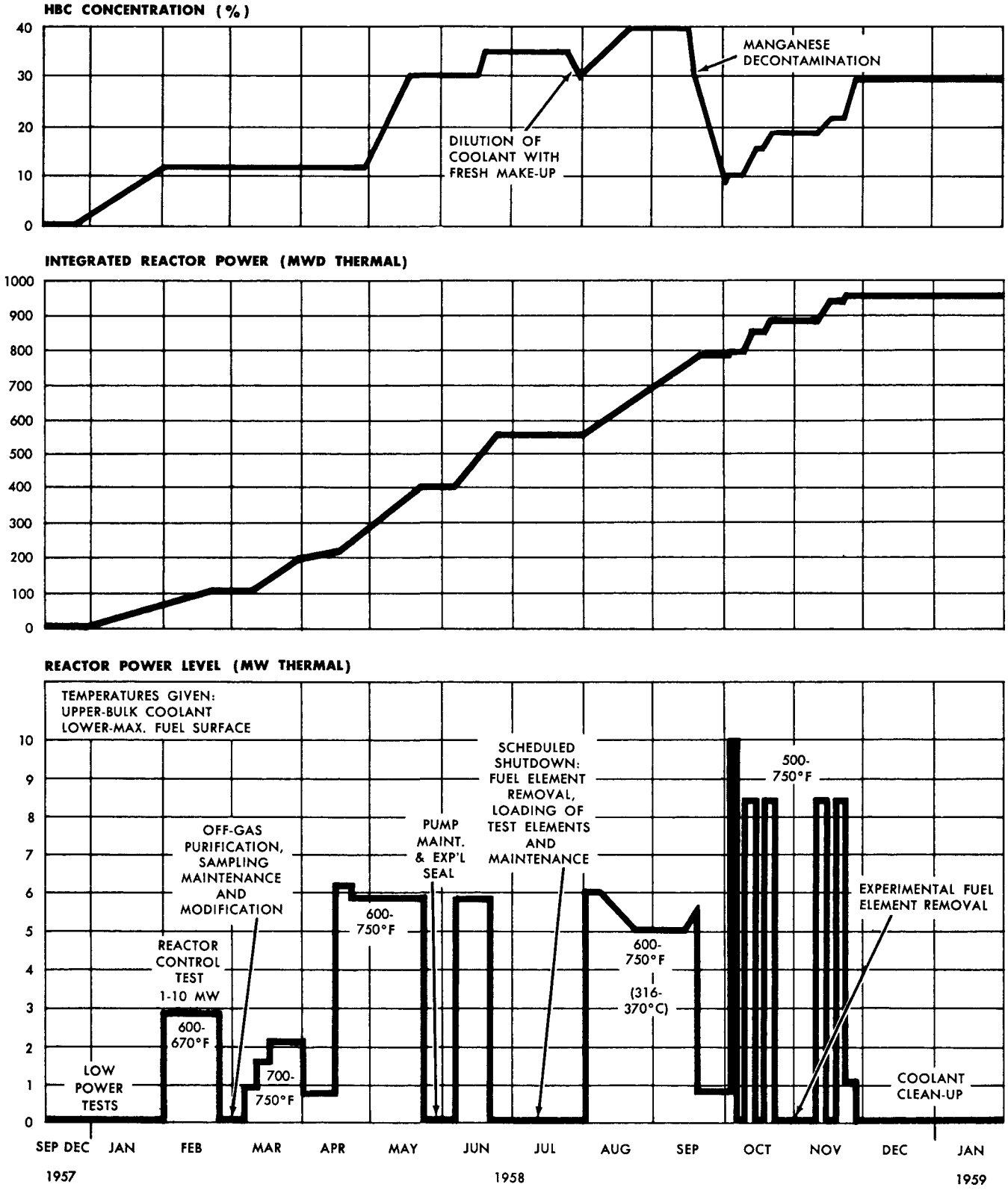
* includes organic make up cost based on 13½ ¢/#.

The relationship of power cost vs. size is shown in Fig. 15.

The above costs are representative of plants that could be constructed with the current technological status. However, these costs could be obtained only after the time required for construction of a large plant. The only technology currently available for the organic cooled reactor in the area of fuel is the U-3½ w/o Mo. with aluminum cladding. The limits on this fuel exposure in the reactor environment required are such that a large 300 MWE plant of this type would probably not be constructed. There are no large organic-cooled reactors of this type under construction to use as a check point on the estimated cost in this report.

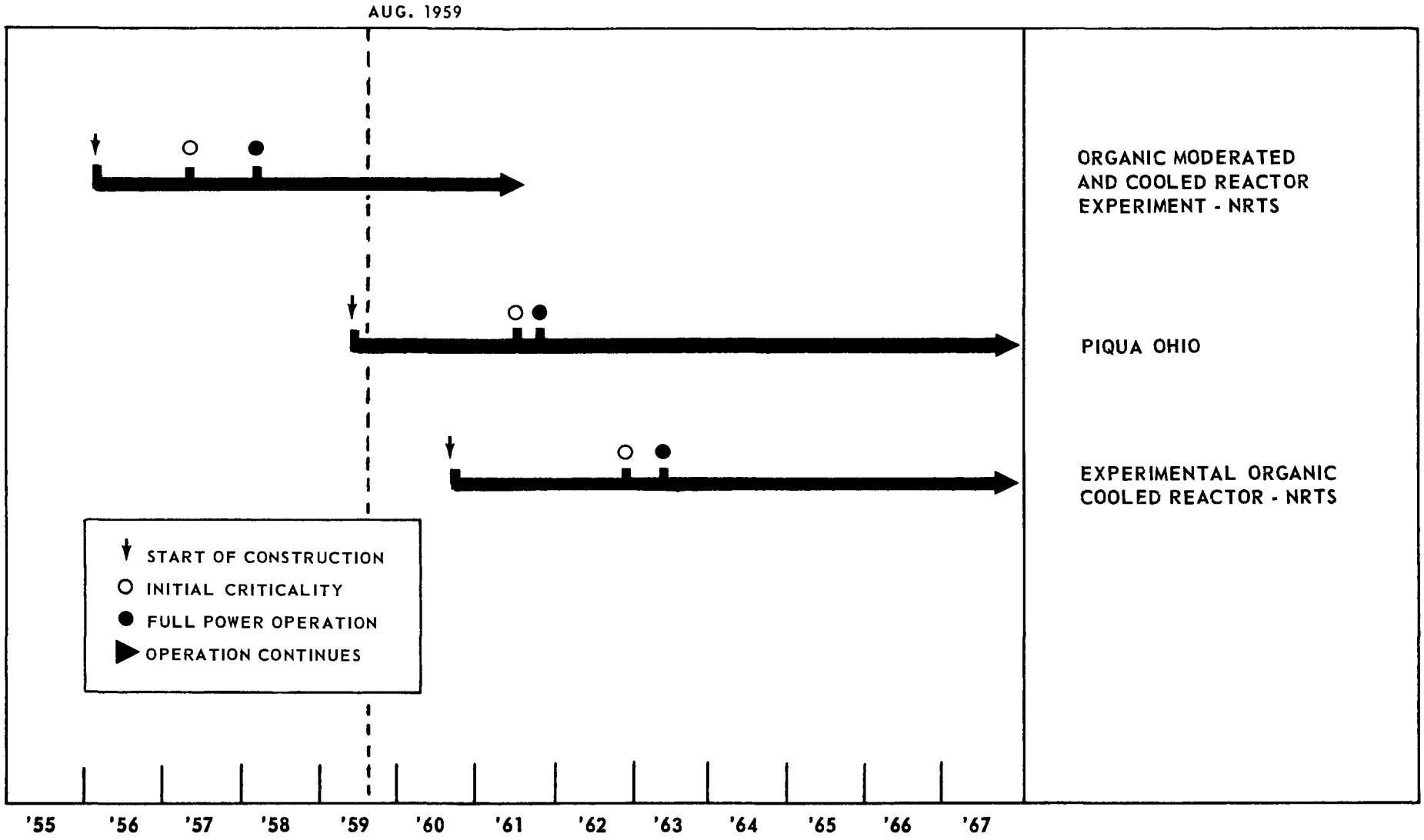
The plant arrangements and other data for determining the economics is included in Report S & L 1674 and Appendix II of this report.

ORGANIC COOLED REACTORS
OPERATING EXPERIENCE
OMRE



SEPT. 1959 FIG. 13

ORGANIC COOLED REACTORS
CONSTRUCTION SCHEDULE



SEPT. 1, 1959

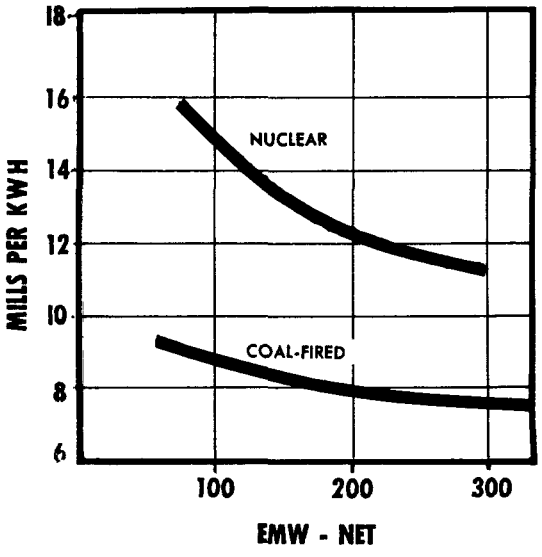
FIG. 14

POWER GENERATION COSTS

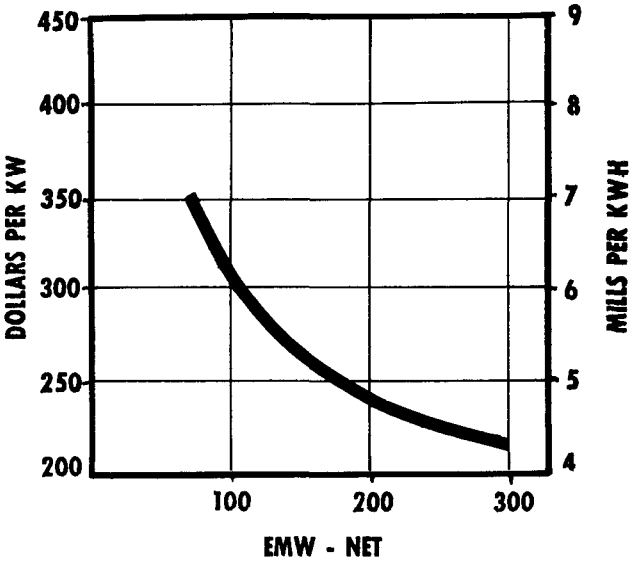
ORGANIC COOLED REACTORS
VS
COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.

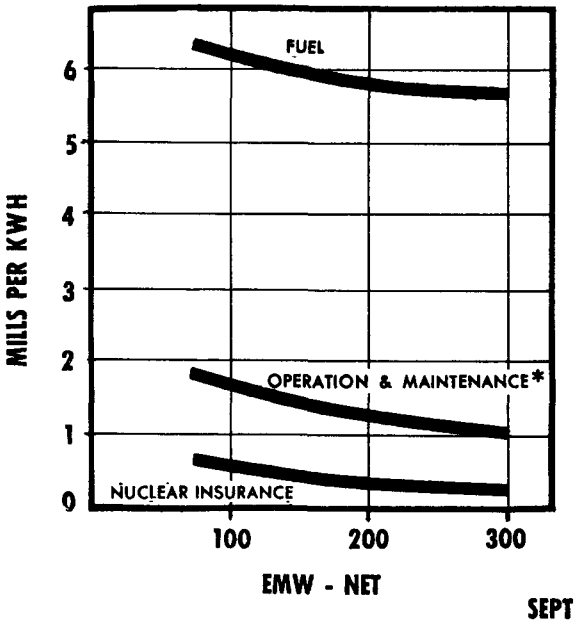
POWER GENERATION COSTS



CAPITAL COSTS



OPERATING COSTS



NOTES :

NUCLEAR -
BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/ GM.
FUEL FABRICATION COST	
U-3.5% MO - AL	$60/KG.
FUEL EXOPSURE	4500 MWD/MT AVG.

COAL - FIRED -
BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%
FUEL COST	35¢/10^6 BTU

*Includes organic makeup.

SEPT 1959
FIG. 15

4. SODIUM GRAPHITE REACTORS

a. Description. The sodium graphite reactor is a heterogeneous fueled, graphite moderated, sodium cooled thermal reactor. The reactor system has two coolant loops between the reactor and the turbine. Radioactive sodium circulates in the first loop, removing the heat from the fuel and transmitting the energy through a heat exchanger to the secondary loop. The nonradioactive secondary sodium transmits the heat to a steam generator where superheated steam is formed to drive a turbine. A simplified flow diagram of the sodium graphite reactor plant is shown in Figure 16.

b. Technical Status. Sufficient research and development work has been done to arrive at the following conclusions on technical status:

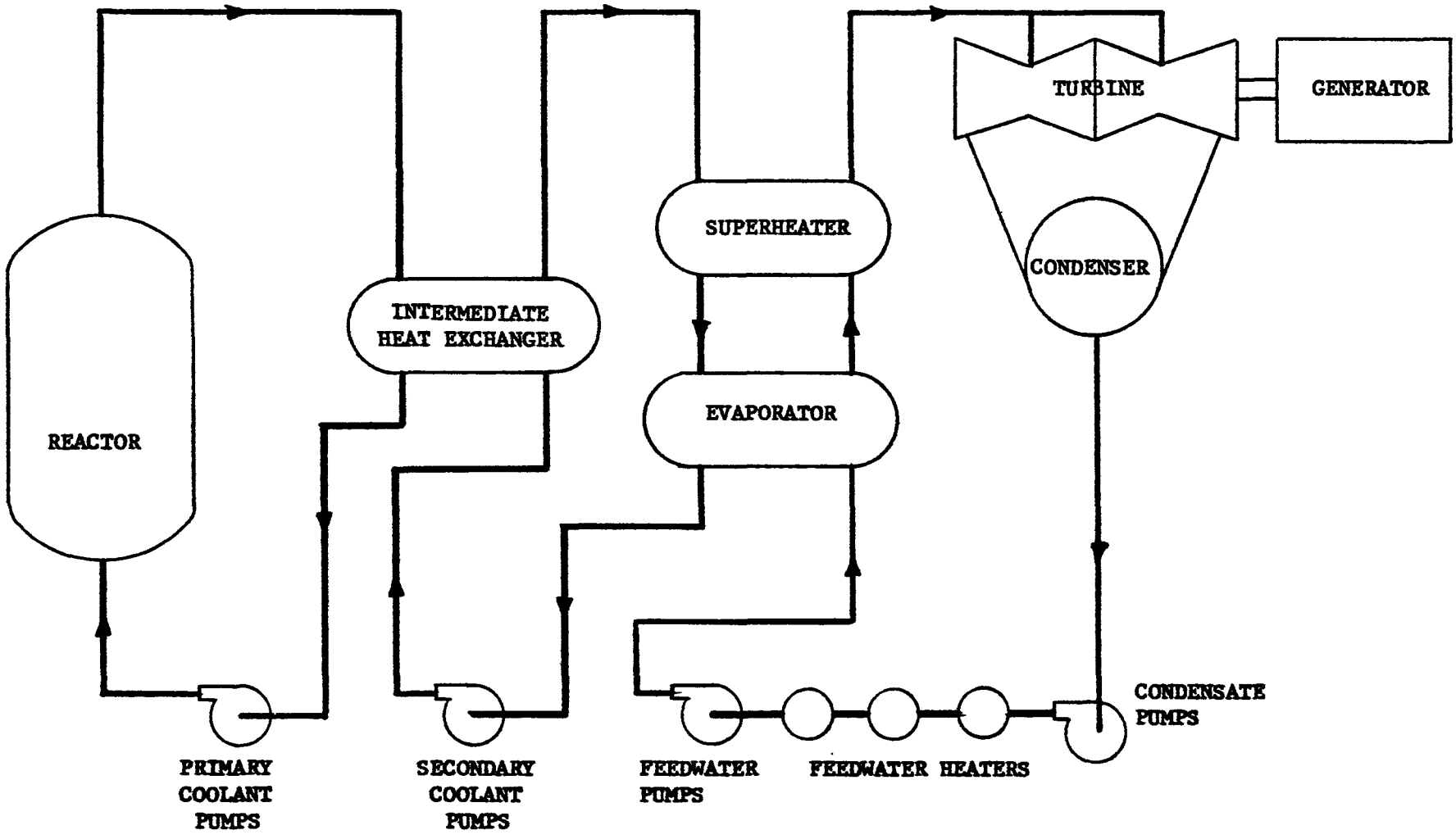
(1) Physics. Physics experiments have been made on static measurements on subcritical lattices and static and dynamic experiments on the SRE. The fuel geometry on the SGR, consisting of multi-rod clusters, makes theoretical predictions of the flux distribution difficult. A considerable amount of information on intra-cell flux distributions has been obtained from a series of exponential experiments on lattices containing both Th-U alloy and slightly enriched uranium fuel (NAA-SR 3096). The results of these measurements are useful in predicting critical masses, conversion ratio, and fuel disadvantage factors in sodium graphite reactors.

A related effort in subcritical experiments consists of measurements on single fuel elements in a graphite matrix, in the thermal column of a research reactor. These experiments yield essentially the same data as can be obtained from a full exponential experiment but with somewhat reduced accuracy. They are useful as an

inexpensive means of screening possible fuel geometries. A critical experiment capable of mocking up large (18" diameter) assemblies will be operational in late 1959.

The important SGR experimental physics information obtained to date has been acquired with the SRE; the extrapolation to larger areas have yet to be demonstrated. Experimental measurements of the isothermal temperature coefficient, control rod worths, and low power flux patterns have been made. The reactor oscillator technique has been utilized to obtain the reactor transfer functions at power levels ranging from zero to full power. From these data, the steady-state power coefficient and its associated time constants have been determined. In addition, the reactor oscillator and noise analysis techniques have been utilized to obtain the value for the prompt neutron lifetime. The results of these experiments show the prompt neutron lifetime to be 0.525 milliseconds, the time constant of the graphite to be 600 seconds, the time constant of the fuel to be 9 seconds, and the apparent over-all heat transfer coefficient between the moderator and sodium to be 180 BTU/hr-ft^2-^{o}F.

The reactor oscillator has shown the SRE to be an inherently stable reactor at all power levels and at all frequencies above 5×10^{-4} cps. No measurements were performed below this frequency; any instability below 5×10^{-4} cps has no practical significance to a power reactor because of its very long period. The determination of the power coefficient provides experimental information which will be of great use in predicting the stability of future sodium graphite reactors. The general agreement between the measured parameters, while oscillating at power, and those determined by theoretical methods give evidence of the ability to create a mathematical model upon which a theoretical analysis of reactor stability may be effected.



SIMPLIFIED FLOW DIAGRAM

SODIUM COOLED - GRAPHITE MODERATED
REACTOR PLANT

SEPT 1959
FIG. 16

Measurements to determine the long-term reactivity changes have been performed and an empirical relation has been developed for the prediction of these changes. The measurements have been made using fuel elements whose exposures range from 100 - 1,000 MWD/T.

(2) Fuel and Materials. A number of low alloy metallic uranium elements have been irradiated in the SRE. Even at the relatively low center temperature reached in the SRE (1100°F) these materials have generally been demonstrated to be unsatisfactory for extended exposure. The only exception to date is the U-1½ Mo cast alloy, which has shown good dimensional stability under SRE conditions at low burnup.

Additional tests of metallic uranium and thorium-uranium alloys have been made in the MTR. In the MTR tests, the temperature range on the uranium specimens was 900 to 1300°F at low burnup. The thorium-uranium alloy specimens (Th-10 w/o U irradiated at 1000°F) have indicated that good dimensional stability may be expected up to 11,000 MWD/T. With the possible exception of thorium-uranium alloy, tests have generally proven metal fuels to be unsatisfactory for the advanced SGR in which center temperatures above 1400°F are required.

UO_2 has been examined for application to sodium cooled reactors, and an experimental 19-rod element has been inserted into the SRE. Extensive effort on UO_2 at other sites has indicated good radiation stability at high burnup at low surface temperatures. Because of the very low thermal conductivity ($\approx$1 BTU/hr-ft-°F) of UO_2, it is of limited interest to high performance sodium systems. In order to take advantage of the heat transfer characteristics of sodium, a fuel element should be

capable of providing heat fluxes of 1 to 3 million BTU/hr-ft^2 to high temperature sodium. The surface of the element should operate at temperatures in excess of 1200°F.

Uranium mono-carbide with a melting point of 4300°F and a thermal conductivity of about 14 BTU/hr-ft-°F is being intensively investigated for use in sodium cooled reactors. Only limited information is available at the present time on the stability of the carbide under high temperature irradiation. Two capsule tests in the MTR have been completed. In the first of these, the carbide was irradiated to about 1,000 MWD/T at 1600°F central and 1200°F surface temperatures with no apparent change in the specimen. The second specimen was irradiated to 6,000 MWD/T with a center temperature of 1600°F (with peaks to 1900°F). This specimen showed some cracking and spalling but no severe density change. This specimen contained slightly more carbon (5%) than stoichiometric (4.8%). Fission product retention was excellent in both specimens. The uranium carbide development program has compiled preliminary data on compatibility with Na and NaK, thermal shock, some physical properties, and considerable arc-melting fabrication experience.

Reactivity considerations limit the amount of sodium in the reactor core and thermal stress and thermal efficiency considerations tend to limit the sodium temperature rise through the core. In order, then, to operate at a high specific power, sodium velocities of approx. 25 ft/sec through the core are required. Therefore, the conditions that a sodium-cooled fuel element must withstand are those of high surface temperature, high heat flux, and high sodium velocity. Carbide fuel offers promise of meeting these requirements. Problems of irradiation damage, irradiation induced outgassing, thermal stress, and moderator canning require more consideration.

(3) Heat Transfer and Fluid Flow. Heat transfer experiments with liquid metals have been actively prosecuted since 1948, with the inception of the Navy sponsored GENIE program at KAPL.

Simultaneously, a general liquid metal heat transfer research program was initiated at both KAPL and MSA. This information was developed to provide analytical information for core heat transfer purposes and liquid-metal-to-liquid-metal heat exchanger designs.

Concurrent with these efforts sponsored by military programs, heat transfer experiments aimed specifically at EBR-I conditions and requirements were in progress at the Argonne National Laboratory.

Heat transfer work at Atomics International commenced in 1954 as a part of the research and development program supporting the SRE construction. Actual heat transfer work was limited to specific engineering applications, such as heat removal from frozen sodium seals, both static and rotating.

Liquid metal system hydraulic experiments were undertaken as a part of the SIR development program, which extended from 1948 to 1956. Dimensionless parameter correlations of liquid metal behavior up to temperatures on the order of 850° have been derived. It is considered that sufficient hydraulic and fluid flow information has been developed to support routine design of liquid metal heat transfer systems and equipment. A large area of uncertainty still exists, however, in the field of natural convection under low flow conditions. "Stagnant" liquid metal, unfortunately, does not remain stagnant. Internal convection circulation loops are very easily set up, because of the high thermal conductivity, high film coefficients, and moderate density change of liquid metal with temperature. Effects which are both unexpected and

startling frequently crop up when even carefully designed liquid metal systems are operating. Because the behavior of these fluids is not as yet completely understood, there is a continuing need to perform both scale model and full scale experiments. Loop-type experiments on model and full scale components have been in progress at Argonne National Laboratory since 1949, and at Atomics International since 1954, to develop and demonstrate satisfactory performance of system components in liquid metal cooled reactor systems. During the period from 1950 to 1956, similar experiments were undertaken by Oak Ridge National Laboratory, as a portion of the ARE was liquid metal cooled.

(4) Coolant Chemistry. Sodium exhibits excellent compatibility with most metals and ceramics, including both cladding and fuel materials. The corrosion rates of steels, zirconium, uranium and uranium alloys, thorium and thorium alloys, uranium carbide, etc., are extremely low in pure sodium below 1200°F. Data above 1200°F is lacking.

No decomposition products need be considered, as sodium is an elemental substance and transmutation to magnesium 24 by neutron absorption is extremely slow. Difficulties with sodium chemistry stem from impurities rather than the metal itself. Oxygen, carbon, lithium, and calcium are representative impurities which exist in small quantities. Development of analytical techniques to determine quantitatively the amount of these impurities present is in its infancy. The difficulties stem from the chemically active nature of liquid metals, which require that they be handled under an inert atmosphere which may itself not be pure.

High temperature liquid metal chemistry and technology was initiated in 1948 by the Knolls Atomic Power Laboratory, Oak Ridge

National Laboratory, and Argonne National Laboratory. Handling of molten liquid metals at temperatures up to 1200°F was investigated, with the bulk of interest at that time lying around 850°F. Means of removing oxygen from sodium and sodium-potassium alloys were developed. These early developments involved "cold traps" to precipitate the oxide at lower temperatures than existed in the process system. "Hot traps" for final system oxygen cleanup were developed by Atomics International in 1956.

Analysis for oxygen in sodium has always been difficult due to lack of confidence that the sodium sample was not contaminated during removal. Laborious techniques, chiefly mercury amalgamation schemes and high temperature distillation methods, have been developed for quantitative analysis of oxygen as sodium oxide. Other impurities in sodium have not proved troublesome at temperatures up to 1000°F with helium and argon inert gases as cover. Nitrogen apparently does not react with sodium, nor does it dissolve in any significant amount in pure sodium. Evidence of nitriding in nitrogen cover systems has been occasionally reported. Tentative information indicates that, quite possibly, transport of nitrogen in liquid metal systems is due to impurities such as carbon, lithium, and calcium. The exact role of these impurities is not yet understood.

Experiments in all of these laboratories indicate that, up to about 1000°F, mass transfer is not a problem. Above these temperatures, mass transfer seems to appear. Transport of radioactive atoms may eventually prove a nuisance in accessibility to liquid metal reactor primary systems after shutdown. No effects such as plugging of small pipes and orifices by deposition of mass transfer products is expected on the basis of the evidence now at hand.

Sodium appears to be a noncorrosive substance. The product of a sodium water reaction, sodium hydroxide, is, of course, extremely corrosive when in a water solution. This is of importance in the secondary system where possible contact with boiler water may occur due to a steam generator failure. All the evidence accumulated at Knolls Atomic Power Laboratory, Argonne National Laboratory, Oak Ridge National Laboratory, and Atomics International since 1952 indicates that, if sodium is reasonably free of oxide, it is essentially inert to the usual materials of construction in high temperature systems. These materials are the austenitic stainless steels and most high strength ferritic alloys. Some decarbonization of ferritic steels has been detected by Atomics International; this tendency to carbonize or decarbonize can be effectively combatted by adjusting the chromium content of the metal.

(5) Components and Auxiliary Systems. The development of liquid metal components had its start in the SIR (Navy) program. Most of this information is still classified, and therefore cannot be properly discussed in this publication. The unclassified work which has been performed at Argonne National Laboratory in support of the experimental breeder reactor program, and at Atomics International, in support of the sodium graphite reactor program, has provided the following backlog of experience in the design and manufacture of liquid metal heat transfer components.

(a) Pumps - Liquid metals lend themselves to the use of electromagnetic pumps, and these have been developed to a rather high degree. These pumps are, however, generally expensive and their efficiencies leave something to be desired (44%). Argonne National Laboratory and Atomic Power Development Associates, in cooperation with pump

vendors, have developed and tested so-called liquid-metal-bearing pumps, in which the liquid metal being pumped provides a "pressure pad" type of bearing lubrication at the impeller. This type of pump has performed well on test, and is specified for the Fermi reactor installation and part of the EBR-II plant. Freeze-seal pumps with organic (tetralon) cooling are incompatible with sodium systems.

(b) Valves. Liquid metal valves are unique in their requirements for extremely high temperature operation. Some form of stem sealing is necessary. Two current methods have been developed to a point of feasibility. One is the bellows seal which takes the place of the stuffing box at the operating shaft. A second type of stem sealing is the freeze seal. Each of these types is in use, in the sodium reactor experiment and in EBR-I. Across-the-seat leakage remains a problem.

(c) Heat Exchange Equipment. The heat exchanger information is meager and sometimes inappropriate for the design of liquid metal heat exchangers, both liquid-metal-to-liquid-metal and liquid-metal-to-water. Continuous development of liquid metal heat exchangers on widely fluctuating scales of effort has been under way since 1948. Sudden transients are rapidly reflected in the materials which make up the heat transfer apparatus in a sodium system. Thermal stresses result and conventional equipment, with less attention paid to thermal transients, often suffers when subjected to liquid metal service. A family of heat exchanger concepts has been developed over the past few years, but no heat exchanger equipment has yet been built, which is truly oriented to liquid metals, although some has been conceived and designed. The Fermi plant embodies possibly the most advanced liquid metal-to-liquid-metal heat exchangers yet devised, and these appear to reflect most of the criteria

believed necessary for successful fabrication of this type of equipment. A cheap reliable sodium steam generator is not presently available.

(d) Piping and Component Heating. Because of the elevated melting point of most liquid metals (208°F in the case of sodium) it is necessary to preheat the system and components prior to initially filling, or refilling after draining. The schemes for doing this are simple and require no further development, only design for a particular application.

(e) Controls and Control Mechanisms. Use of "thimbles" is possible in low pressure reactors. This permits the use of control mechanisms which are not exposed directly to flowing sodium. Upon withdrawal of the rod, no poison is introduced into the thimble, and the control rod effect is maximized. A major disadvantage associated with thimble type control rods is dissipation of the heat generated in the rod by neutron capture.

Control and safety rod drive mechanisms are simple and can be devised satisfactorily in various ways. The low pressure of the system permits the use of a simple mechanical seal when necessary, and elaborate means for containment of a high pressure, radioactive system are unnecessary.

(f) Instrumentation. Reliable sodium process information, such as level, flow, temperature, and pressure is available for systems at temperatures below about 900°F. Above this temperature difficulties associated with high temperature insulation begin to appear. It appears that there is no obstacle to the use of electromagnetic flow meters at 1200°F in the near future, but considerable development

remains before level indicators and pressure gages can be expected to perform reliably in this temperature range. Possibly the most pressing immediate need in this regard is an accurate pressure gage, which could also be used as a flow indicator in connection with a differential pressure cell. No such instrument is currently available.

(g) Inert Gas Handling. The chemically reactive nature of most liquid metals in oxygen creates a demand for an inert gas atmosphere whenever a free surface exists in the reactor system. These inert gas atmospheres are low pressure systems, generally less than 10 psi gage, and present a problem only insofar as gas makeup is concerned. Noble gases, such as helium, have been used to date. Fragmentary evidence garnered from the SIR program indicates that nitrogen is a satisfactory substitute for the more expensive and rarer helium at temperatures up to 1000°F. Experiments are currently in progress at Atomics International to verify the satisfactory performance of nitrogen at temperatures above 1000°F. Other users of sodium systems are planning to use argon as a cover gas.

(h) Maintenance of System Components. The chemically reactive nature of liquid metals makes removal and maintenance of mechanican components which have been immersed in coolant somewhat more difficult than if the coolant were not a chemically reactive substance. Techniques of removing sodium-coated components into plastic bags purged with an inert atmosphere have been developed. Primary and secondary mechanical pumps are handled this way in a routine fashion when radioactivity has decayed away. Some improved system concepts and components which lend themselves to direct maintenance have been demonstrated in mockups. An example is the "overflow" sodium-cooled reactor system under study by Atomics International.

(6) Reactor Safety. The only operating sodium graphite reactor, the SRE, has been demonstrated to be extremely stable. Operating experience has shown that very little control rod movement is required when operating at full power under automatic control. The small, over-all temperature coefficient of reactivity renders unnecessary large amounts of control for the temperature rise to operating conditions. This reduces the excess reactivity and the number of control rods required. The negative, prompt, fuel temperature coefficient produces the stability of the reactor. The total steady-state power coefficient is negative at all power levels for the case of a constant temperature difference across the core, and is negative at power levels above 8 MW for the case of constant flow of 1,400 gpm. Below 8 MW, the steady-state power coefficient is positive because of the moderator. However, the long time constant of the moderator (600 seconds) implies that any instability would be of a very low frequency and presents no control problem of any consequence. The analyses of the reactor transfer function obtained with the pile oscillator have firmly established that the SRE is completely stable in the entire range of frequencies investigated (0.0005 to 20 cps) at all power levels.

Hazards analyses of SGR systems have revealed that this reactor type is particularly insensitive to the "runaway" excursion. In this situation the worst result is partial melting of the fuel elements and rapid shutdown of the reactor. There are no materials in the reactor which provide a source of chemical energy for explosive accidents and no large storage of enemy as in a high pressure coolant.

A more hazardous accident is the improbable one in which a small amount of excess reactivity is inserted over an extended period of

time with no corrective action taken, an excursion properly termed a "walkaway" incident. The hazard in this type of accident arises from the possiblity of boiling the sodium contained in the upper coolant plenum to the extent of reaching sodium vapor pressures which could rupture the primary cooling system. A considerable amount of information is required on heat transfer to sodium vapor, Doppler coefficient as a function of temperature and other areas of uncertainty before a completely realistic analysis of the "walkaway" accident can be made.

On the basis of information obtained to date from operation of and experiments on SRE, the sodium graphite reactor is unusually stable and easy to operate in a safe manner.

c. Operating Experience. The operating experience of a sodium-graphite reactor has been limited to the SRE. There is, however, operating data on the SIR (Sea Wolf) and the EBR-I that has provided data on sodium cooled systems though the reactors were of a different type, i.e., intermediate and fast spectrum reactors respectively. The SRE began initial power runs in July 1957. The pertinent experience obtained with SRE is as follows:

The intermediate heat exchanger has not performed according to specifications. An LMTD of 90°F at full power is required rather than the design value of 60°F. Thermal circulation of sodium within the shell has been observed, leading to after-scram stresses. Difficulties have been experienced with the freeze seal pumps. Some bellows-sealed valves have failed at the bellows. Initial cold trap design was of insufficient capacity; redesigned equipment is now installed and is capable of removing

any amount of oxide so far introduced into the system. The most significant equipment problem was a high stress condition which existed around core tank nozzles after scram. This condition has been corrected by installation of eddy-current brakes which limit post-scram convective sodium flow and reduce thermal transients in the nozzle vicinity to an acceptable value. No other equipment problems of a nonroutine nature have been experienced.

Load following. The SRE has been operated as a load forcing, rather than a load following machine. However, test transients of up to 20% power increase per minute have been induced and successfully followed manually with the reactor.

The transient convective sodium flow following scram has been the principal problem encountered to date. Thermal convective flow is most difficult to compute accurately during design of a liquid metal system, and a means for controlling the post-scram flow was not initially incorporated into the SRE system. Later installation of this equipment has eliminated this major transient problem. The reactor itself has proven exceptionally stable and easy to control under transient conditions.

Fission product gases have been detected in the inert gas atmosphere above the top pool of the SRE. Quantitative measurements of this activity have so far proved impossible due to the low level of activity; detection has been by gamma ray spectrometer only.

Expansion of the unalloyed uranium fuel has apparently occurred in the most highly irradiated (center) sections of the central fuel elements. The ten mil clearance between fuel slug and cladding has been closed by

fuel expansion. A maximum of four mils distortion of the cladding has been observed after 1100 MWD/MT irradiation. No rupture of the fuel cladding has occurred; observation of reference fuel elements is continuing in order that irradiation may be stopped before reaching cladding failure.

The life of the SRE core was terminated by failure of the fuel cladding of a number of elements. The cladding failure was discovered at the end of power run 14 on July 26, 1959, and has been attributed to local overheating of the fuel element rods caused by obstruction or blockage of coolant passages by foreign matter in the primary coolant system. Based on preliminary information, the mechanism of the clad failure was formation of an iron-uranium eulectic.

The main obstacle to quick repair of the sodium system is awaiting decay of the Sodium 24 activity. By advance scheduling of down-periods, it has been possible to minimize the outages necessary to permit decay. Other work requiring the reactor to be subcritical or just critical is scheduled for the decay period. Access to the primary system may be expedited by draining the radioactive sodium back to the shielded fill and drain tanks. If urgently required, access can be had to the piping galleries about five days after shutdown. Normally, a decay period of about 10 days is taken.

System and component designs based on improving those areas which have been somewhat troublesome can be expected to further simplify maintenance procedures in future sodium cooled reactor systems. The operating experience of the SRE is shown on Figure 17.

d. Plants Under Construction. A sodium graphite reactor plant of 75 MWE is presently under construction at Hallam, Nebraska. The Hallam Nuclear Power Facility (HNPF) consists of a 254 MWT sodium graphite reactor and the associated equipment for generating and delivering 800 psig, 825° steam to a turbogenerator.

The reactor is sodium cooled, graphite moderated, and will use slightly enriched Uranium 10 w/o molybdenum alloy as a fuel for the first core. Structurally, the arrangement of the reactor, similar to the SRE, is known as a "tank-type", "canned moderator" to distinguish it from other proposed SGR configurations.

The schedule for the SRE and the Hallam plant is shown on Figure 18.

e. Economics. A reference design has been made for sodium cooled graphite moderated reactor based on the technical status summarized in this report. This design was used as the basis for estimating the cost at different plant ratings. The base parameters for the reference design are as follows:

SUMMARY OF PLANT CHARACTERISTICS
SODIUM GRAPHITE REACTOR PLANT

A. Heat Balance	
1. Total Reactor Power, MW(t)	240
2. Gross Turbine Power, MW(e)	80
3. Net Plant Power, MW(e)	75
4. Net Plant Efficiency, %	30.8
B. Turbine Cycle Conditions	
1. Throttle Temperature, F.	850
2. Throttle Pressure, psig	785
3. Steam Flow, lbs./hr.	7.19×10^6
4. Condenser Back-Pressure, in. Hg A.	1.5
5. Final Feed Water Temperature, F.	300
C. Reactor Description	
1. Reactor Vessel	
a. Inside Diameter, ft.	19.0
b. Overall Height, ft.	34.0
c. Wall Thickness, in.	0.75
d. Material	SS
e. Design Pressure, psia	50

2. Reactor Core	
a. Active Diameter, ft.	13.4
b. Active Height, ft.	13.5
c. Active Core Volume, ft^3	1930
d. Lattice Arrangement	hex.
3. Reflector or Blanket	
a. Material	Graphite
b. Axial Thickness; ft.	2
c. Radial Thickness, ft.	2
4. Fuel Elements	
a. Fuel Material	U-10 w/o Mo
b. Clad Material	SS
c. Fuel Enrichment	2.85
d. Fuel Element Geometry	rods
5. Material Inventories	
a. Fuel, metric tons	36.7
b. Uranium, metric tons	33
c. U-235, initial-kg.	940
6. Reactor Control	
a. Method of Control	rods
b. No. of Control Elements	31
D. Plant Performance Data	
1. Primary Coolant Outlet Temp., F.	945
2. Primary Coolant Inlet Temp., F.	607
3. Reactor Temp. Drop., F.	338
4. Primary System Operating Pressure, psia.	atmos.
5. Primary Coolant Flow Rate, lbs./hr.	8.44×10^6
6. Avg. Core Heat Flux, Btu/hr. $-ft^2$	121,000
7. Max. Core Heat Flux, Btu/hr. $-ft^2$	302,000
8. Max. Cladding Surface Temp., F.	1000
9. Max. Fuel Temp., F.	1260
10. Core Coolant Velocity, ft./sec.	11.4 max.
11. Peak to Avg. Power Ratio	2.5
12. Core Power Density kw/ft^3	124
13. Core Specific Power kwt/metric ton-U	7270
14. Secondary Sodium Outlet Temp., F.	895
15. Secondary Sodium Inlet Temp., F.	557
16. Secondary Sodium Flow Rate, lbs./hr.	8.01×10^6
17. Fuel Burn-up MWD/metric ton-U (average)	3000

The power cost for a 300 MWE plant extrapolated from the above design is as follows:

Total Capital cost is - $90,900,000 ----------	6.05	M/Kwh
Fuel cycle cost (300MWD/MT) ----------	7.90	M/Kwh
Operation and Maintenance ----------	.70	M/Kwh
Nuclear Insurance ----------	.29	M/Kwh
Total Power Cost ----------	14.94	M/Kwh

The fuel material for the sodium-graphite reactor used in the reference design is U-10 w/o Mo. The fuel exposure is limited by the swelling effect which is a function of the % atom burnup and fuel temperature. An analysis of the swelling rate vs. % atom burnup and temperature for the U-10 w/o Mo. indicates a sharp increase in the swelling rate at approximately 1130°F. The reference plant (basically the Hallam Reactor) for current status was designed with a fuel center line temperature of approximately 1260°F. This temperature placed a severe restriction on the % atom burnup obtainable. This resulted in a maximum fuelexposure at 1260°F. of 7000 MWD/MT Max. and 3000 MWD/MT Ave.

As a result of the analysis on fuel exposure at different temperatures on the U-10 w/o Mo. and its relationship to economics, a new preliminary design was made for the sodium graphite plant.

The centerline temperature of the fuel was reduced to 1130°F. The specific power of the core was decreased by the addition of more fuel and the enrichment increased to provide more reactivity for burnup. The fuel rod size selected was .33". Hot channel factors were calculated for the new core design. The available data on U-10 w/o Mo. indicated that a fuel burnup of 1.75 atom percent ($\approx$20,000 MWD/MT) Max. can be obtained at 1130°F.

The base parameters for the new design are as follows:

Significant Design Parameters - 300 MW SGR

Thermal Power	MW	884
Electrical Power		
Gross	MW	315
Net	MW	300
Reactor Coolant		
Sodium Inlet Temperature	°F	607
Sodium Outlet Temperature	°F	945
Reactor Core		
Height	Ft	15
Diameter	Ft	17¼
No. of Elements		
Fuel		276
Control		31
Source		1
Fuel Element		
Rods per element		61
Fuel Slug Diameter	in.	0.33
Radial Sodium Bond	in.	0.01
SS Cladding Thickness	in.	0.01
Fuel Area	$in.^2$	5.2
Process tube I.D.	in.	4.52
Sodium Flow Area	$in.^2$	9.49
Max. Fuel Temperature	°F.	1130

Core Data		
Fuel Loading	Kg of U	65,300
Pu production	gm Pu/Kg of U	4.55
Initial enrichment	a/o U-235	3.89
Final enrichment (at discharge)	a/o U-235	2.67
Fuel exposure (average)	MWD/MT	11,000

The power cost for a 300 MWE sodium plant based on the modification above are as follows:

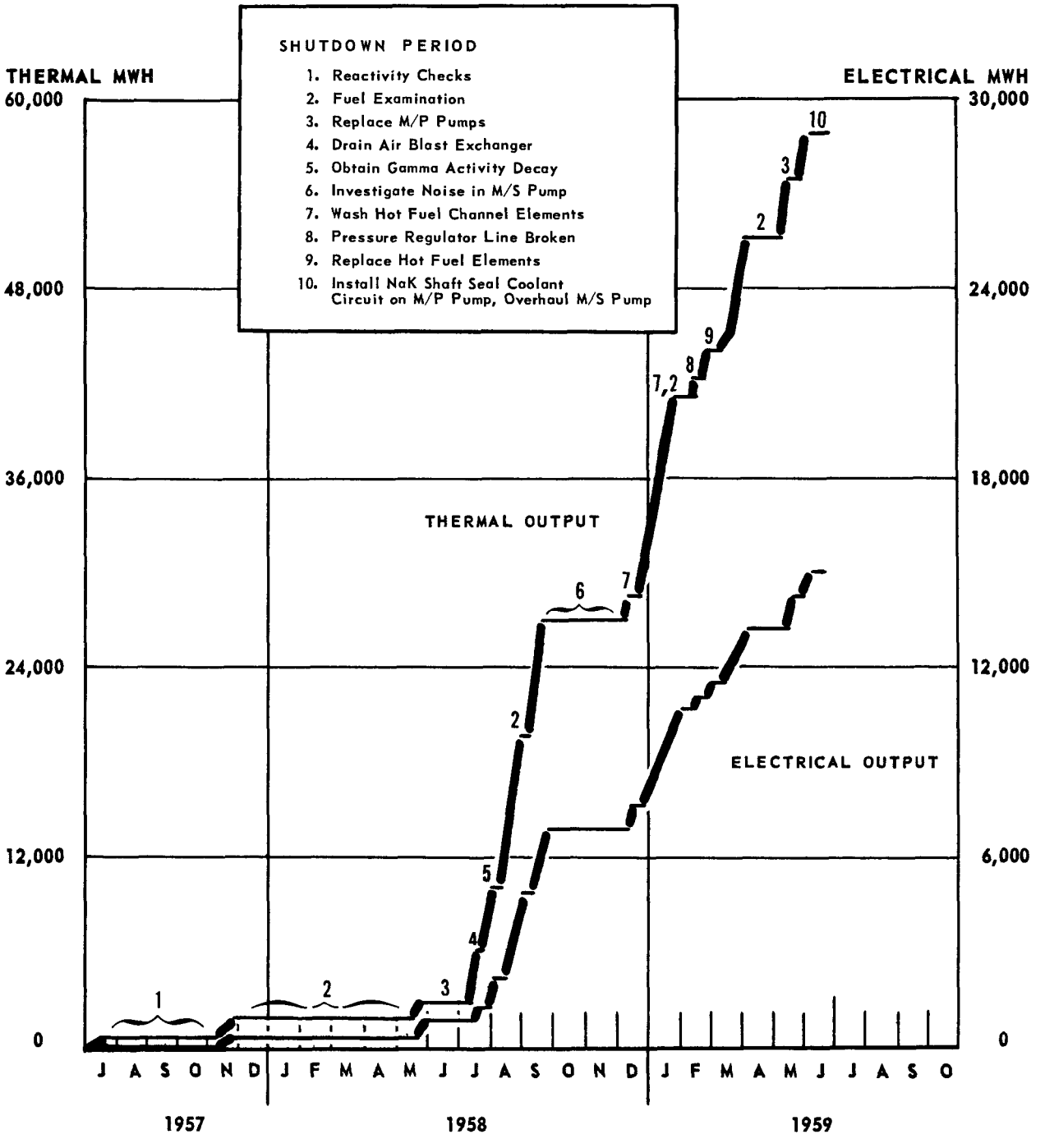
Capital	6.15	M/Kwh
Fuel Cycle Cost	4.12	M/Kwh
Operation and Maintenance	.70	M/Kwh
Nuclear Insurance	.29	M/Kwh
Total Power Cost	11.26	M/Kwh

The power costs vs. size for the sodium graphite plant is shown in Figure 19. This power cost is based on a more optimum (still consistent

with current status) core design than was used for the Hallam Reactor.

The plant layouts and other design data for determining the economics is included in report S & L 1674 and Appendix III of this report.

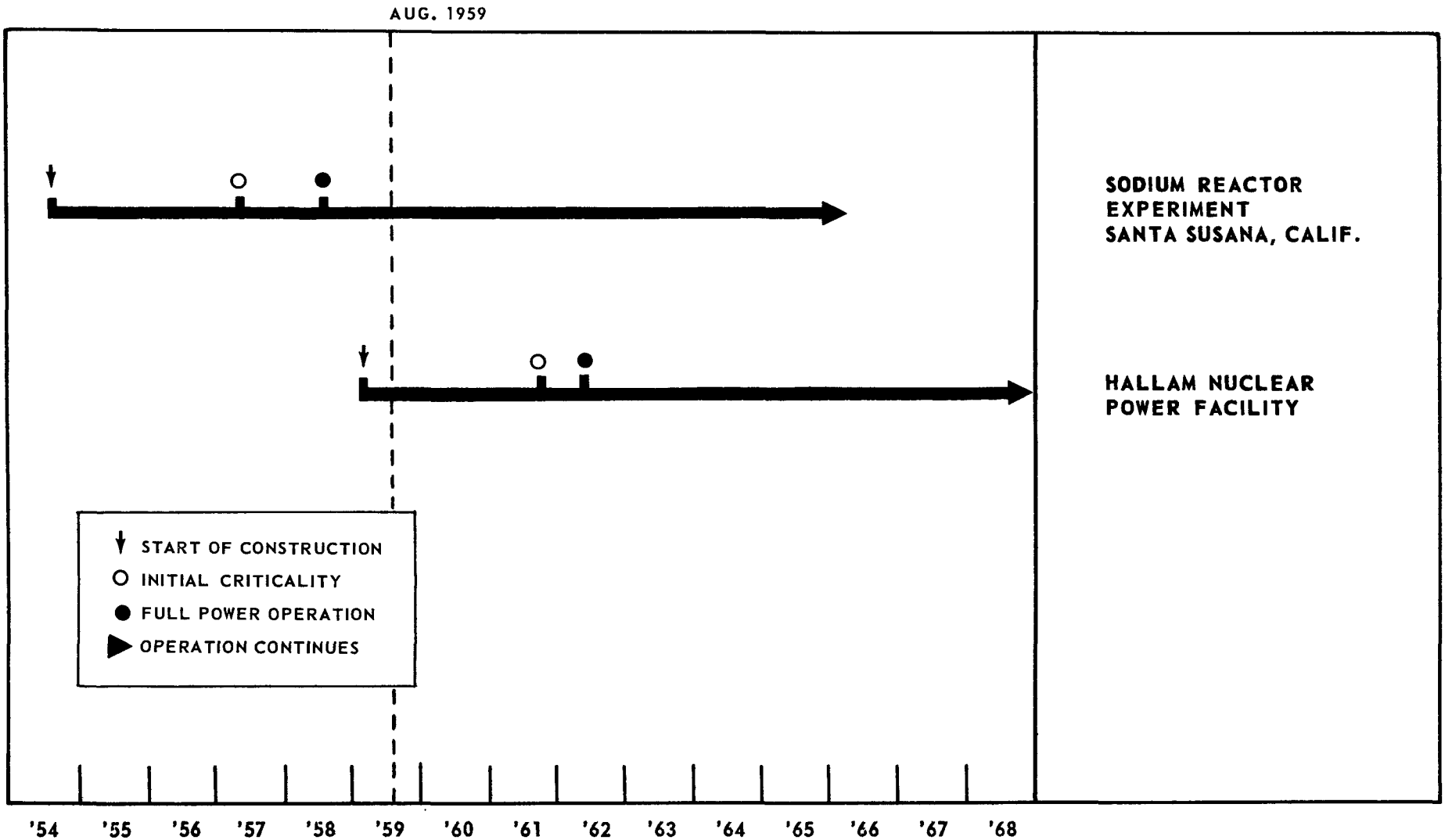
SODIUM GRAPHITE REACTORS
OPERATING HISTORY
SRE



SEPT. 1, 1959

FIG. 17

SODIUM GRAPHITE REACTORS
CONSTRUCTION SCHEDULE



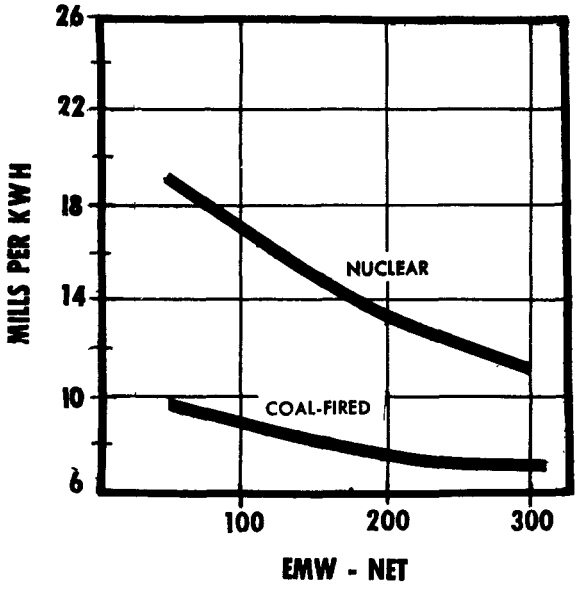
SEPT. 1, 1959
FIG. 18

POWER GENERATION COSTS

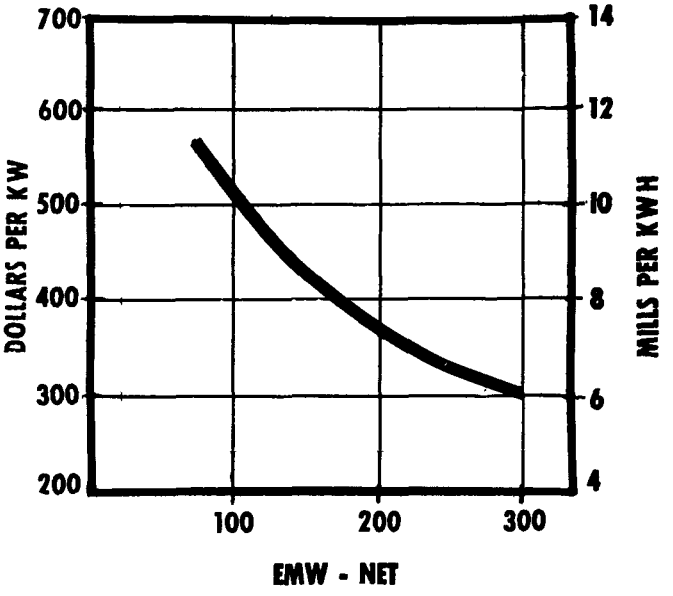
SODIUM GRAPHITE REACTORS
VS
COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.

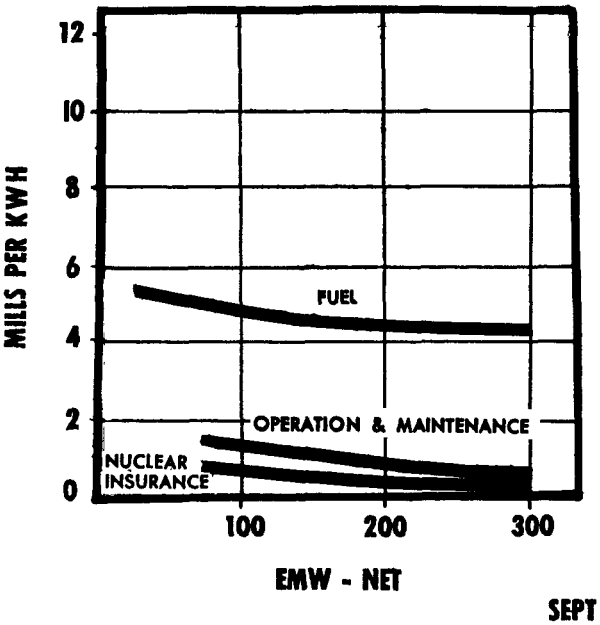
POWER GENERATION COSTS



CAPITAL COSTS



OPERATING COSTS



NOTES :

NUCLEAR -
BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/GM.,
FUEL EXPOSURE	11000 MWD/MT AVG.

COAL - FIRED -
BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%
FUEL COST	35¢/10^6 BTU

SEPT 1959
FIG. 19

5. GAS-COOLED REACTORS - ENRICHED FUEL

a. Description. The enriched uranium gas-cooled reactor currently under construction in this country is a graphite moderated slightly enriched UO_2 fueled stainless steel clad reactor. Helium is used as the coolant. The reactor heat is removed from the fuel by the helium coolant and is transmitted to a heat exchanger where steam is formed to drive the turbine. A simplified flow diagram of the gas-cooled reactor is shown in Figure 20.

b. Technical Status. The major effort in the United States on civilian gas-cooled reactor has been on partially enriched fueled reactors. The technology of enriched fuel, gas reactors has not yet been demonstrated. However, the technology on the natural uranium fueled reactors in the United Kingdom plus the work on enriched fuel in this country is sufficient to establish the following conclusions on technical status:

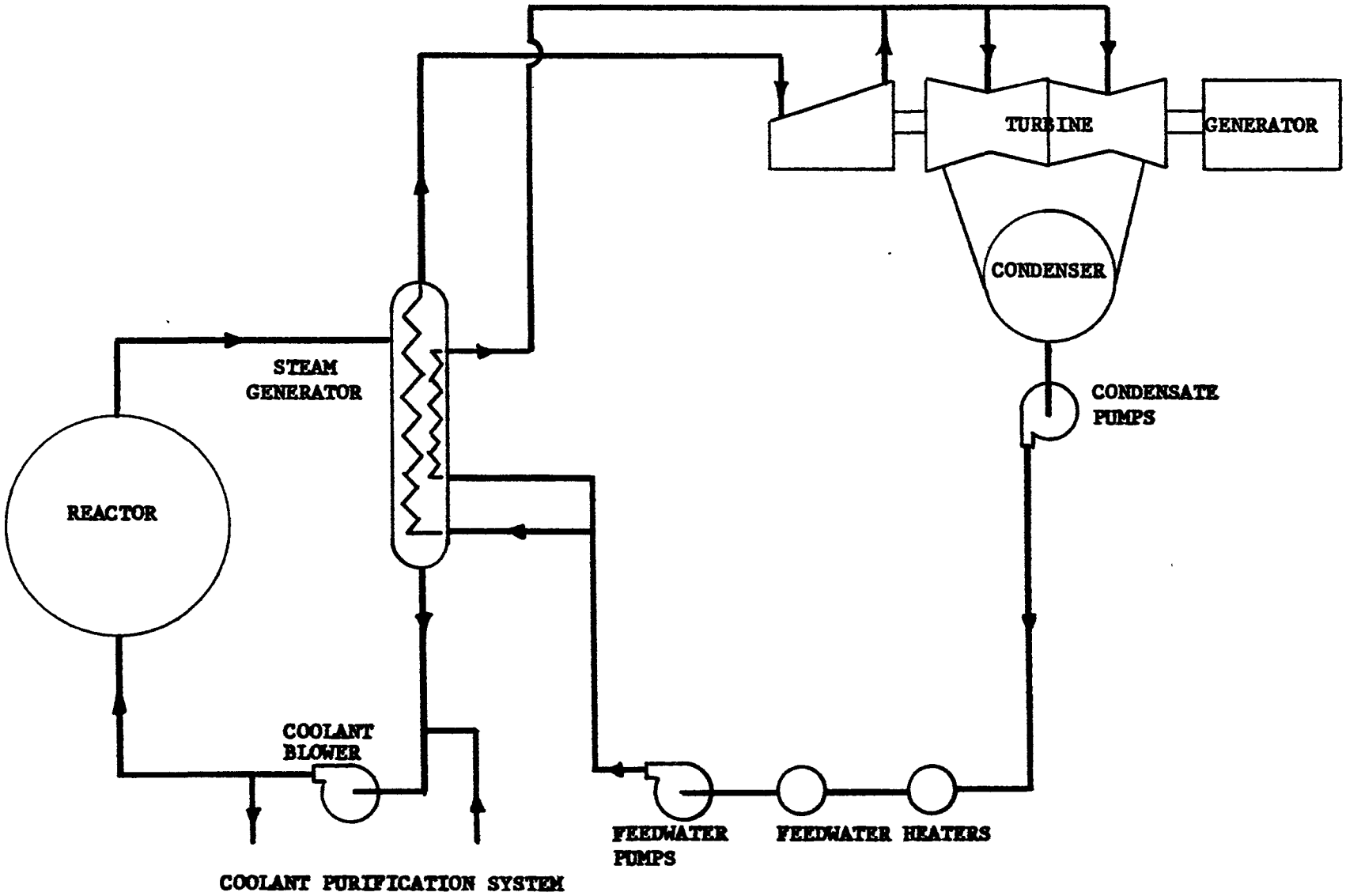
(1) Physics. The physics of gas-cooled reactors is based upon experience obtained with natural uranium systems and critical experiment data. Normalization to enriched system has been made through experiments conducted at the Hanford plant. Sufficient information is available to calculate these systems with a good degree of reliability.

Enrichment up to about 2 per cent permits compactness of the fuel lattice and results in higher power densities. Reactivity lifetimes for two percent enriched systems are in excess of 15,000 MWD/MT. Economic estimates have been made based on 7300 to 10,000 MWD/MT.

Less experimental data are available on temperature coefficients than for natural uranium systems. However, the coefficients are easier to calculate and of less importance in enriched systems.

Fuel elements tend to be strongly divided leading to cluster-type geometries. Considerable progress has been made in calculating the fine structure of the heat production and removal within these clusters. These computations are normalized against experiments which have been made in the United States, United Kingdom, France and Sweden.

(2) Fuel and Materials. All present designs contemplate using UO_2 since it is the best known ceramic fuel. Most of the experimental data available in this country on UO_2 performance is for water cooled systems however extensive data on UO_2 for gas reactor is available in other countries. The environment, most particularly the surfaces temperatures and central temperature of the UO_2, strongly influences its performance. Accordingly, experimental data on gas-cooled reactor fuel elements is lacking. The EGCR is designed to operate a UO_2 stainless steel clad (20 mills) element from 7500 to 10,000 MWD/MT at a gas temperature of 1050°F. Typical maximum clad surface temperatures are in the range 1400-1800°F. While no swelling occurs in UO_2 under irradiation even at burnups in excess of 50,000 MWD/MT, gaseous fission products diffuse from the lattice at rates sensitive to temperature. This is particularly important in gas-reactors where the typical fuel element failure is postulated to result from excess fission product gas pressure. The temperature at which fission product gas production catalyzes fuel



SIMPLIFIED FLOW DIAGRAM

GAS COOLED - GRAPHITE MODERATED REACTOR PLANT

SEPT 1959
FIG. 20

element failure is not well known. This is a function of many variables including the fuel element design.

(3) Coolant Chemistry. Helium is used as a coolant rather than carbon dioxide in order to avoid the carbon dioxide graphite chemical problem on reactors where the graphite operation temperature would be 500 to 600°C.

Helium is chemically inert and compatible with the fuel and moderator materials throughout the temperature range of interest. Gaseous impurities may be present, however, whose concentrations must be kept below specified limits to prevent corrosive attack on the graphite, fuel cladding and other structural materials.

With gas cleanup practices now available, it is possible to operate helium in contact with graphite with temperatures of 1100°F.

(4) Components and Auxiliary Systems. It is in this area that the greatest unknowns exist for helium cooled systems. Components of adequate integrity and performance exist for CO_2. Adequate components for helium have yet to be developed. A realist method of determining helium leakage is not known. Full-scale temperature and pressure experiments on blowers, valves and piping are not yet completed. A charge-discharge machine for helium cooled reactors would have problems similar to those for CO_2 cooled systems except greater leak tightness. Lubrication at temperatures of 500-600°F and in a high radiation field may prove very difficult.

(5) Safety. The containment requirement for the gas-reactor has not yet been determined.

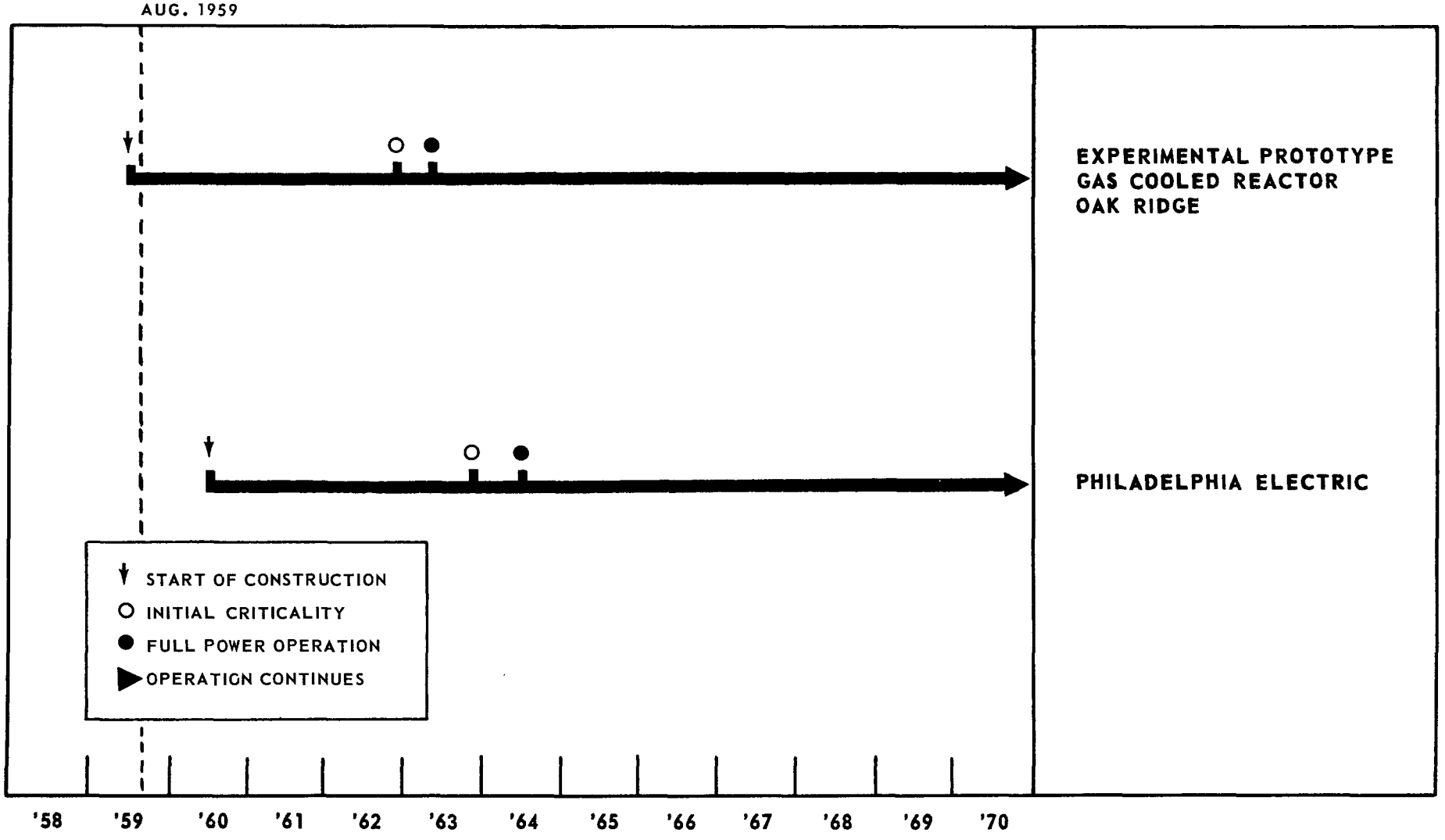
c. Current Plant Design Parameters. The technology of this reactor concept is not sufficiently developed to warrant construction of a large central station power plant at this time.. Therefore no design parameters are given in this status report. An experimental prototype that is currently in a stage of advanced design will develop information upon which a plant design may be based.

d. Operating Experience. No operating experience is currently available with slightly enriched gas cooled power reactors for civilian power application. There is considerable operating experience available on natural uranium fueled system, which is applicable to the enriched system.

e. Schedule. The anticipated construction schedule of the enriched fuel gas-cooled reactor is shown in Figure 21.

f. Economics. No cost data is given in this report on enriched fuel, gas-reactors.

GAS COOLED REACTORS
ENRICHED FUEL
CONSTRUCTION SCHEDULE



SEPT. 1, 1959
FIG. 21

IV

Status
Civilian Power Reactors

B. Breeders

B. Breeders

The reactor types that are classified as breeders are:

1. Fast (Plutonium Production)

2. Thermal (Uranium-233 Production) Aqueous Homogeneous

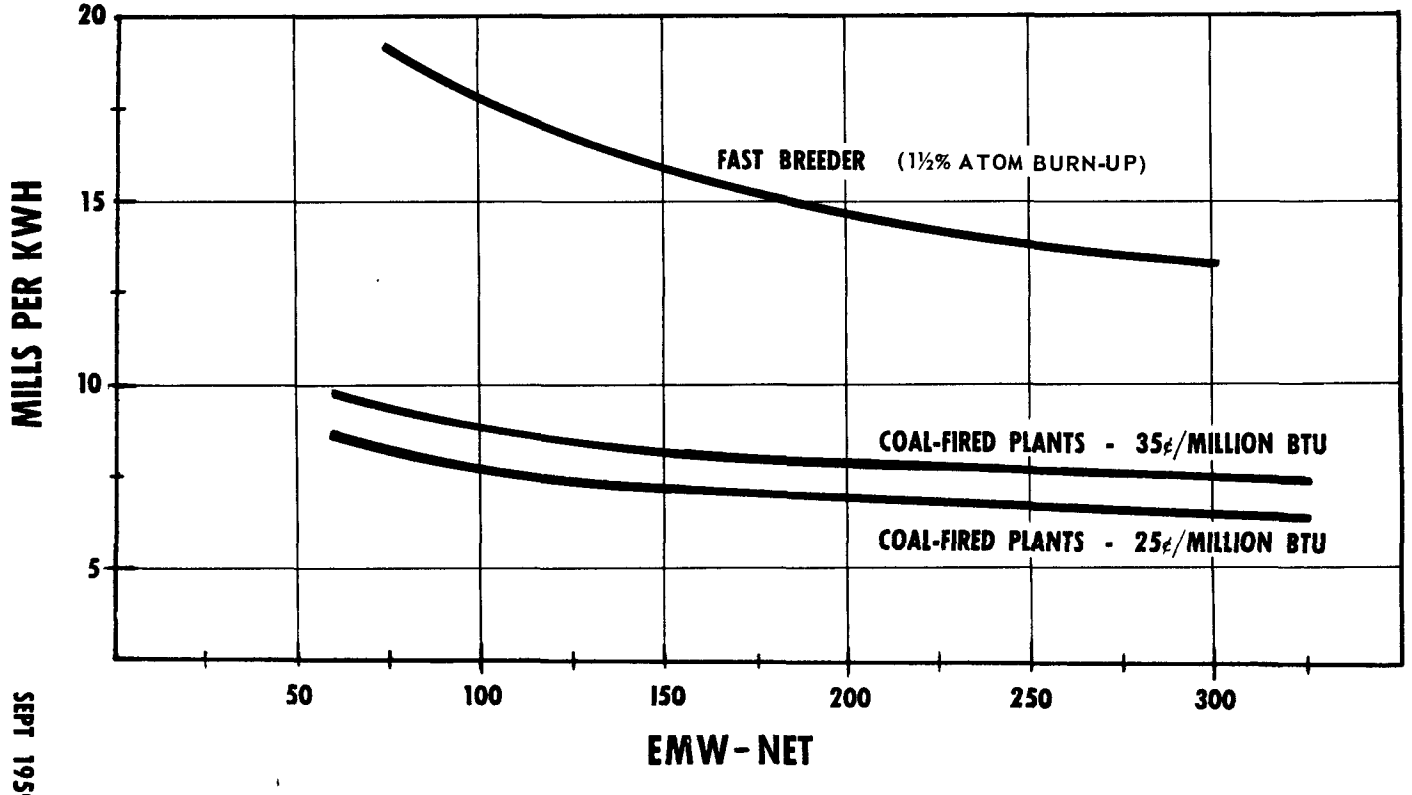
The primary objective of the breeder reactor program is to develop a power plant capable of achieving a net breeding ratio greater than unity. A secondary objective is to reduce the cost of power to levels that are economically attractive, without sacrificing the breeding objective.

A composite curve showing the reference design power generation cost of breeder reactors vs. coal fired plants is presented in Fig. 22. The technical status of each reactor concept included in the breeder reactor class is summarized in the following pages of this section.

POWER GENERATION COSTS

BREEDER REACTORS
VS
COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.



NOTES :

NUCLEAR -
BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/GM.

COAL - FIRED -
BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%

SEPT 1959
FIG. 22

1. FAST BREEDER REACTOR

a. Description. In a fast reactor, no deliberate attempt is made to moderate the fission neutrons, and most fissions are caused by neutrons with energy greater than 0.1 MEV. This results in the use of high density fissionable material in small cores, a large number of fissions in fertile materials, conversion ratios greater than unity and low flexible reactivity requirements. Present reactors utilize two intermediate loops between the reactor and the turbine. The heat is removed from the fuel by the liquid metal coolant in the first loop and transmitted through a heat exchanger to the second closed loop. The circulating secondary fluid in the second loop transmits the heat to a boiler where steam is formed to drive the turbine. A simplified flow diagram of the fast breeder reactor plant is shown in Figure 23.

b. Technical Status. Sufficient research and development work has been done to arrive at the following conclusion on technical status:

(1) Physics. Critical experiments have been run for moderate size U-235 fueled reactors and calculated critical masses have been determined within a few per cent. Reactivity coefficients have been verified by differential experiments.

The worth of control materials for EBR-II and Enrico Fermi have been determined.

These data are sufficient for reactors under construction.

The following important types of deficiencies currently exist in the physics area:

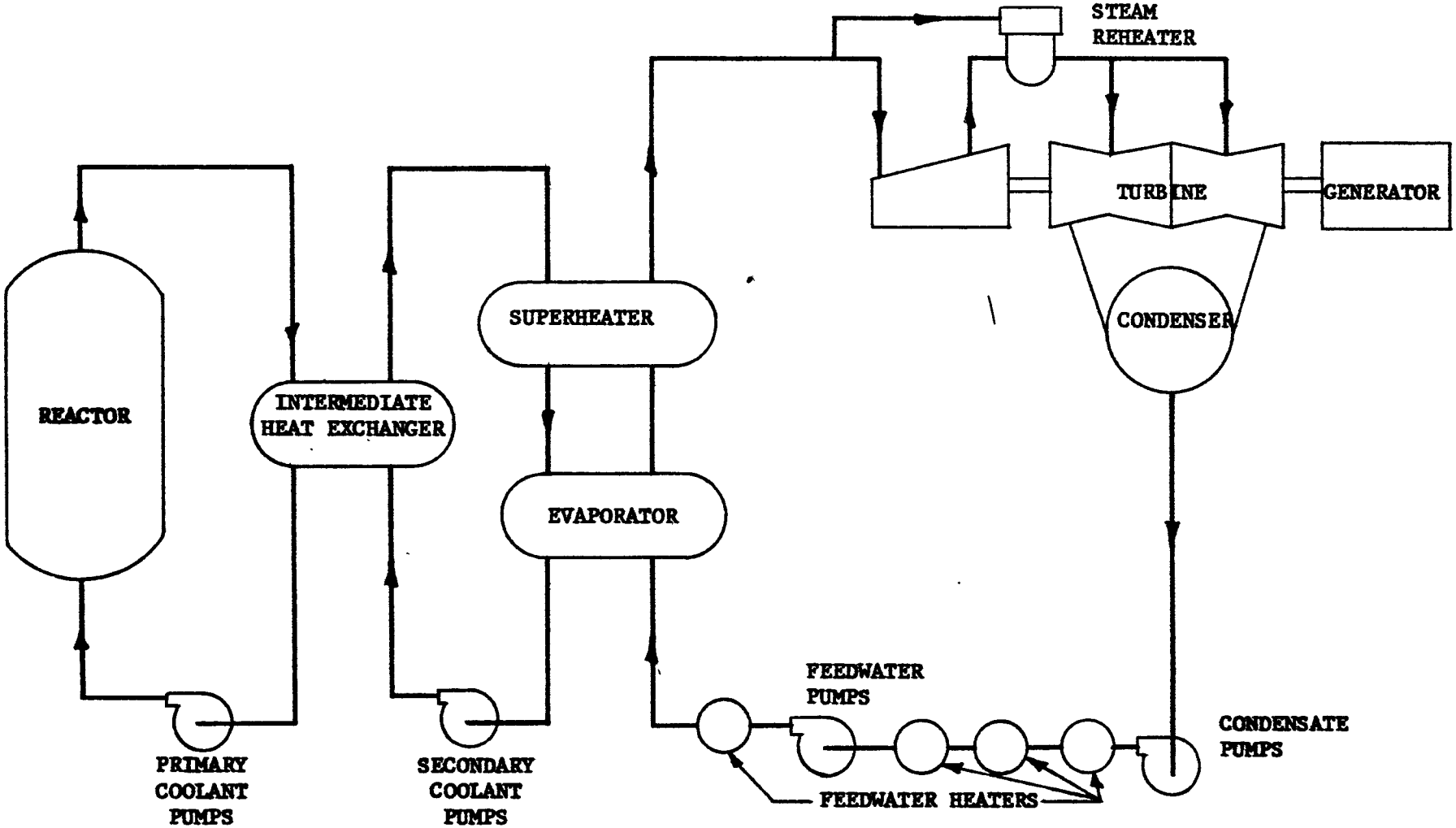
(a) Alpha for plutonium in the 0.1 MEV. range is not known to within a factor of two. This is important to both breeding ratio and control.

(b) The cross sections of Pu-240 and Pu-241 are not well known.

(c) The reactivity coefficients for large Pu fast reactors will not be well known until certain special critical experiments are run. This applies to the coolant temperature coefficient in particular. Additional codes may be necessary to achieve satisfactory methods of calculating these coefficients.

(2) Fuels and Materials. The area which requires the most development for fast reactors is the fuel cycle. In this area, the fast reactor has good potential but the present costs are high. High burnup or cheap and simple fuel cycle is desired.

Data are available on the irradiation stability of U-10 w/o molybdenum and fission metallic fuel alloys. The data indicates a current limit of 2 per cent total atom burnup at a maximum fuel temperature of 1100°F. Extensive work has been completed on cermet fuel elements with successful achievement of 25 per cent uranium burnup in the dispersed phase. This type of fuel element with fertile material as the matrix can give a high breeding ratio of about 1.5. However, with an inert matrix, such as stainless steel, the breeding ratio is around 1.0. Unfortunately, the only extensive data available to date are on the cermet with stainless steel matrix. There is also meager but encouraging information on ceramic elements. Work is required on plutonium fuels, methods of accommodating growth, and cladding materials for high temperature operation.



SIMPLIFIED FLOW DIAGRAM

FAST REACTOR - SODIUM COOLED REACTOR PLANT

SEPT 1959
FIG. 23

Low burn-up of the fuel is the immediate and major problem area. This adversely affects the breeding ratio because of the quicker through-put. The high fuel inventory results in a low specific power.

(3) Heat Transfer and Fluid Flow. The basic heat transfer characteristics of liquid metal coolants, particularly sodium, are excellent. Due to the excellent heat transfer film coefficients, heat transfer is not a limiting factor in the fast reactor system. More data are needed in some specific areas such as the effect of thermal shock on fatigue strength of materials and the effect of oxide and other films on the behavior of some fuel elements.

Because the fluid flow behavior can be correlated with that for other fluids, such as water, the general information available is adequate. Special problems need further investigation. Some of them are: (a) The nature of fog formation in the cover gas, (b) removal of vapor in vapor traps to prevent plugging of gas lines, and (c) flow distribution for unique geometries. For a fuller discussion of the heat transfer properties of sodium see the discussion on the status of sodium graphite reactors.

(4) Coolant Chemistry. As a result of the operation of large systems, such as the SeaWolf, the general behavior of sodium coolant systems has been determined and is considered to be excellent. This plant operated with no sign of sodium corrosion or mass transport in the sodium system, and it is felt that operation below 1000°F. has been satisfactorily demonstrated. Further information is desired on some special materials and on operation with a cover gas such as nitrogen.

Mass transport of carbon is of concern above 1000°F if certain materials like 2½ per cent Cr, 1¼ per cent Moly steel are used. This is a problem associated with diffusion of carbon out of the Cr-Molly steel and absorption by stainless steel.

More economical methods of purifying sodium, particularly of oxygen, is required. Improved purity inspection is required. An AEC program is currently under way to develop the technique for oxygen removal. The problem of detecting ruptured fuel elements has not been solved and requires immediate attention.

In general, the basic behavior of sodium has been excellent. It is a rather new coolant; therefore, the myriad of chemical analyses which have been developed for water have not yet been developed for sodium. Although excellent work has been done on Na-H_2O and Na-Air reactions, further tests should be carried to a more quantitative conclusion.

(5) Component and Auxiliary Systems. Operation of EBR-I, the Sea Wolf, the 1/10 scale model of EBR-II, and individual component tests have demonstrated that high temperature, compact reactors cooled with a coolant which reacts with air can be successfully operated. It has been demonstrated that reactor mechanisms can be operated in sodium and sodium vapor and that heat exchangers can be designed for radioactive sodium service.

For the Enrico Fermi plant, APDA developed a single walled, once-through-type steam generator. Long-term performance tests are desired. Economically reliable steam generators are not currently proven technology.

The maximum operating temperature to which sodium can be successfully handled without corrosion in standard materials of

construction has not been established: it is believed to be about 1200-1300°F. Reactor systems which have been operated or which are construction have been limited to 900°F., except for the short time SRE demonstration at 1000°F.

For a fuller discussion of sodium components see the sodium graphite reactor current status section.

(6) Reactor Safety. The inherent characteristics of fast reactors which are of peculiar importance to their safety are as follows:

(a) The lack of any moderator results in a short prompt neutron lifetime. This increases the damage potential of a prompt excursion if enough reactivity were suddenly introduced to make the reactor prompt critical. However, this type of reactor does not require large available excess reactivity.

(b) The short time constant for heat transfer gives a close coupling of reactivity response which prevents the reactor from getting on short periods.

(c) The strong reactivity effect of core distortion and the large temperature gradients commonly encountered make possible prompt positive power coefficients and tendencies toward power oscillations unless core distortion is controlled. This is being done in reactors being built.

(d) Owing to the large amount of fissionable material present, an agglomeration of the core due to melting could result in a critical reassembly and subsequent energy release. This is prevented by design of a so-called melt-down section.

The safety problems of the fast reactor were highlighted due to the behavior and the meltdown of EBR-I during special physics tests.

The early demonstration of a prompt positive power coefficient in the EBR-I led to speculation that some inherent characteristic of fast reactors was at fault. Although not as yet conclusively proven, recent experiments have yielded circumstantial evidence at least that core distortion was at fault. If this is the cause, then there are no inherent fast reactor characteristics which make it less safe than other power reactors.

Extensive work is being carried out on reactor safety associated with the characteristics attributed to fast reactors. The problem receiving most attention is melt-down and supercriticality and consequent energy release associated with non-moderated systems with high concentration of fissionable material. The results to date have been encouraging, but analyses have not been extended to plutonium systems nor to larger reactors.

c. Operating Experience. The operating experience on fast reactor is limited to EBR-I.

EBR-I - Operation of this reactor, which was the first reactor to produce power, began in December 1951. As a result of its operation, the principles of operation of a fast reactor cooled with sodium were successfully demonstrated. Considerable information on reactor physics was obtained. Plans are now under way to load the reactor with plutonium.

EBR-I was placed into operation in 1951. During the lifetime of the first core (Mark I) the major emphasis was on steady operation to obtain breeding information as quickly as possible as well as information on the engineering behavior of the system. Under certain start-up conditions certain operational anomalies were observed. These occurred

only under extreme operating conditions and were totally absent during routine operation.

The introduction of the second core (Mark II) was accompanied with an experimental program that increased operational anomalies. These consisted of oscillations in power level when the sodium flow rate was drastically reduced, and the appearance of a positive temperature coefficient under start-up conditions, with very short flux doubling periods and with zero or no coolant flow. In one experiment in which the doubling period had been reduced to approximately 0.3 seconds, a delay in scramming the pile permitted the temperature to overshoot so that the uranium became heated above the temperature (roughly 725°C.), at which the uranium-iron eutectic forms and the center of the core melted, forming this eutectic. The core was allowed to cool from the time of the incident in November until June of the following year before an attempt was made to remove it from the reactor. From an analysis of the core it appeared that in a meltdown of a sodium-cooled fast core, the core tends to disassemble itself, reducing the reactivity and power level.

In conclusion, the EBR-I Mark II core had an instability at low coolant flow rates and a positive temperature coefficient for zero flow and for start-up periods.

The third EBR-I core, Mark III, has a more rigid core as well as a modified coolant flow pattern. The operating experience of EBR-I is shown in Figure 24.

d. Plants Under Construction. The following plants are currently under construction:

EBR-II - This reactor is a 60 MWT (20 MWE) fast reactor with an integrated pyrometallurgical reprocessing plant. It is now being constructed at MRTS and is expected to go critical in the summer of 1960. The fuel elements are 1/6 inch in diameter fissium alloy which is sodium bonded to stainless steel tubes. Control is by fuel movement. The major components and the fuel elements have been demonstrated by separate tests. This plant will be operated on a recycle basis and ultimately with plutonium fuel.

Enrico Fermi Reactor - This developmental reactor is designed for initial operation at 300 MWT (100 MWE). The sodium temperatures are 550°F entering and 800°F. leaving the reactor for initial operation and 600 entering and 900°F. leaving the reactor for final operation at high power outputs. The fuel elements are 1/6 in. diameter 10 w/o Mo-U alloy which is metallurgically clad to zirconium. Aqueous offsite reprocessing will be used. The primary system of the plant is essentially complete, the preliminary operation of mechanisms has begun. The erection and preliminary operation has proven to be satisfactory. The primary system will be operated with dummy fuel elements as a non-nuclear test facility for 1 year with sodium at plant temperatures. This test is expected to confirm mechanical and hydraulic operation of the primary system and to demonstrate its integrity. The plant is expected to go critical in the fall of 1960. The operation of this plant and the EBR-II plant should demonstrate the feasibility and practicality of these plants operating on U-235.

Foreign Fast Reactors. The British are constructing a reactor of the same size as EBR-II. It is expected to go critical in August 1959. The Russians are operating one 5 MW reactor for fuel development and have started the design of a larger reactor.

The schedule for the fast breeders under construction is shown in Figure 25.

e. Economics. A reference design has been made for the sodium-cooled, fast breeder reactor, based on the technical status summarized in this report. This design was used as the basis for estimating the cost at different plant ratings. The base parameters for the reference design are as follows:

SUMMARY OF PLANT CHARACTERISTICS
FAST, SODIUM COOLED REACTOR PLANT

A. Heat Balance	
1. Total Reactor Power, MW(t)	440
2. Gross Turbine Power, MW(e)	160
3. Net Plant Power, MW(e)	150
4. Net Plant Efficiency, %	34.2
B. Turbine Cycle Conditions	
1. Throttle Temperature, F.	780
2. Throttle Pressure, psig	850
3. Steam Flow, lbs/hr.	1.46×10^6
4. Condenser Back-Pressure, in. Hg A.	1.5
5. Final Feed Water Temperature, F	380
C. Reactor Description	
1. Reactor Vessel	
a. Inside Diameter, ft.	14.5
b. Overall Height, ft.	36.2
c. Wall Thickness, in.	2
d. Material	SS
e. Design Pressure, psia	6000
2. Reactor Core	
a. Active Diameter, ft.	3.0
b. Active Height, ft.	2.54
c. Active Core Volume, $ft.^3$	17.75
d. Lattice Arrangement	square
e. Lattice Spacing, in.	2.693

3. Reflector or Blanket	
a. Material	dep. U-10 w/o Mo
b. Axial Thickness, ft.	1.17
c. Radial Thickness, ft.	2.4
4. Fuel Elements	
a. Fuel Material	U-10 w/o Mo
b. Clad Material	SS
c. Fuel Enrichment	93
d. Fuel Element Geometry	rods
5. Material Inventories	
a. Fuel, metric tons	0.509
b. Uranium, metric tons	0.463
c. U-235, initial-kg.	430
6. Reactor Control	
a. Method of Control	rods
b. No. of Control Elements	10
D. Plant Performance Data	
1. Primary Coolant Outlet Temp., F	900
2. Primary Coolant Inlet Temp., F	600
3. Reactor Temp. Drop, F	300
4. Primary System Operating Pressure, psia	atmos.
5. Primary Coolant Flow Rate, lbs/hr.	16.4×10^6
6. Avg. Core Heat Flux, Btu/hr.-ft^2	675,000
7. Max. Core Heat Flux, Btu/hr.-ft^2	1,167,000
8. Max. Cladding Surface Temp., F	1000
9. Max. Fuel Temp., F	1050
10. Core Coolant Velocity, ft/sec.	31.2
11. Peak to Avg. Power Ratio	1.44
12. Core Power Density, kw/ft^3	21,500
13. Core Specific Power, kwt/metric ton- U	947,000
14. Secondary Sodium Outlet Temp., F	850
15. Secondary Sodium Inlet Temp., F	550
16. Secondary Sodium Flow Rate, lbs/hr.	16.4×10^6
17. Fuel Burnup (average)	~~1½ w/o~~

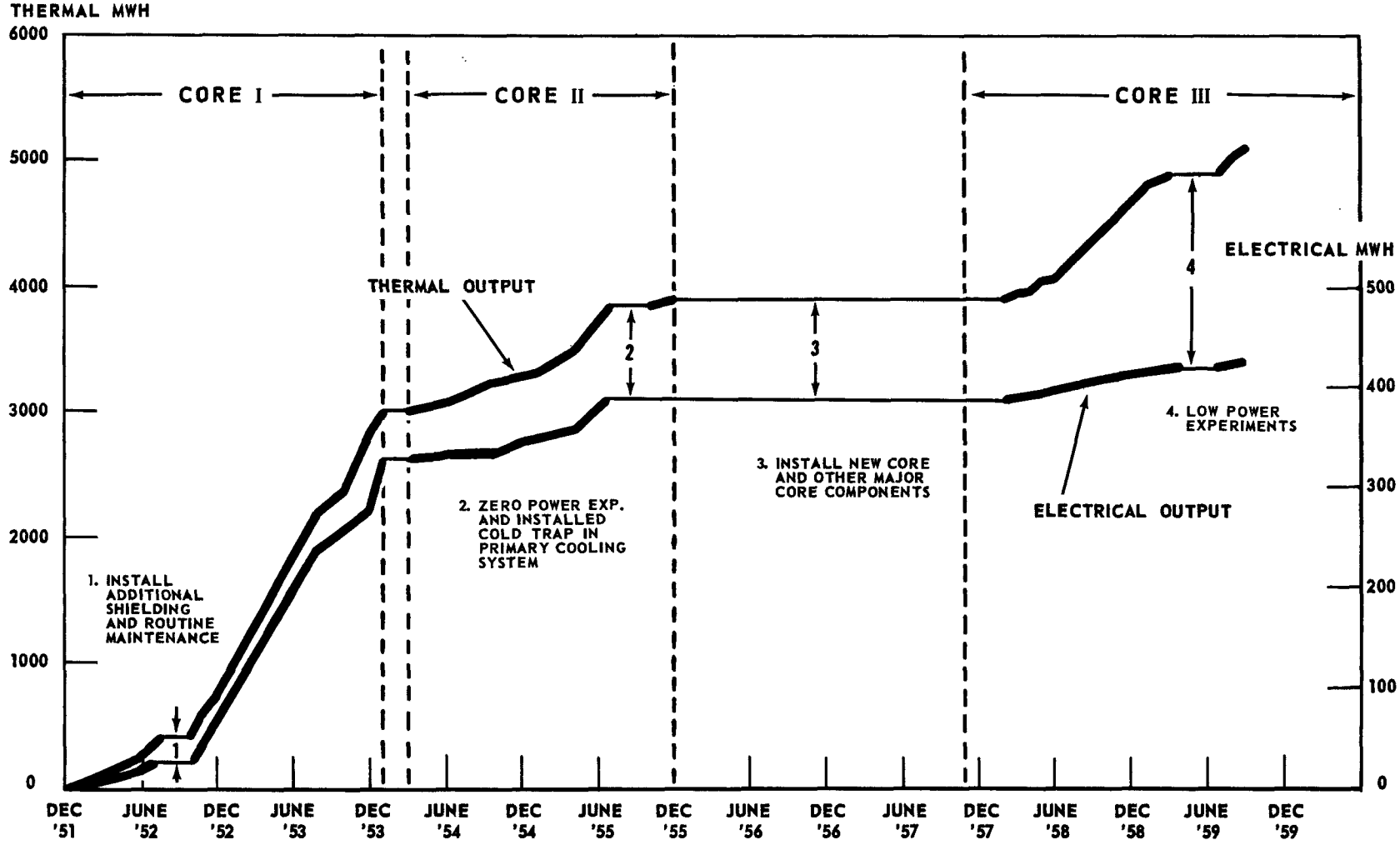
The power costs for a 300 MWE, sodium cooled plant extrapolated from the above design is as follows:

Total Capital Cost - $76,485,014 ---------	5.10	M/Kwh
Fuel Cycle Cost ----------	7.10	M/Kwh
Operation and Maintenance ----------	.79	M/Kwh
Nuclear Insurance ----------	.26	M/Kwh
Total ----------	13.25	M/Kwh

The relationship of power cost vs. size is shown on Fig. 26.

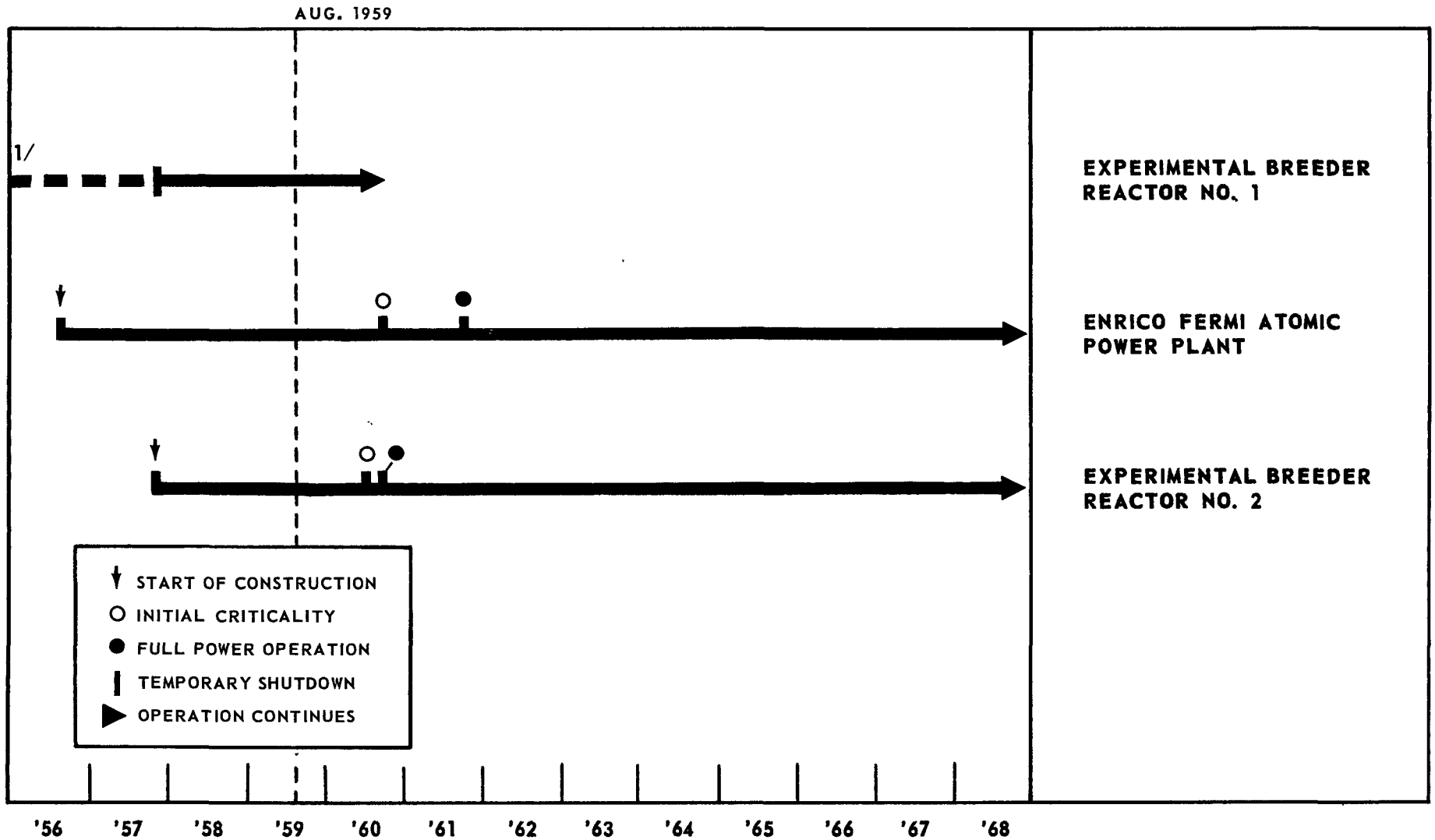
The above costs are representative of plants that could be constructed with the current technical status; however, these costs could only be achieved after the period of time required for construction of a large plant and equilibrium reached on the fuel cycle. These costs are based on a reactor designed for U-235 fuel (U-10 w/o Mo). A check point can be obtained by a comparison with the Enrico Fermi plant, currently nearing completion of construction. The estimated (AEC estimate) capital cost is $53,500,000 and the expected output is 143,000 KWE. The expected capital cost is $374/KWE or 7.49 M/KWH. for fixed charges. The layouts and other design data for determining the economics is included in report S and L 1674 and Appendix II of this report.

FAST BREEDER REACTOR
OPERATING EXPERIENCE
EBR-1



SEPT. 1, 1959
FIG. 24

FAST BREEDER REACTORS
CONSTRUCTION SCHEDULE



1/ CONSTRUCTION: Started January 1949
First Critical August 1951
Full Power February 1952

Temporary Shutdown
Dec. 1955 to Nov. 1957

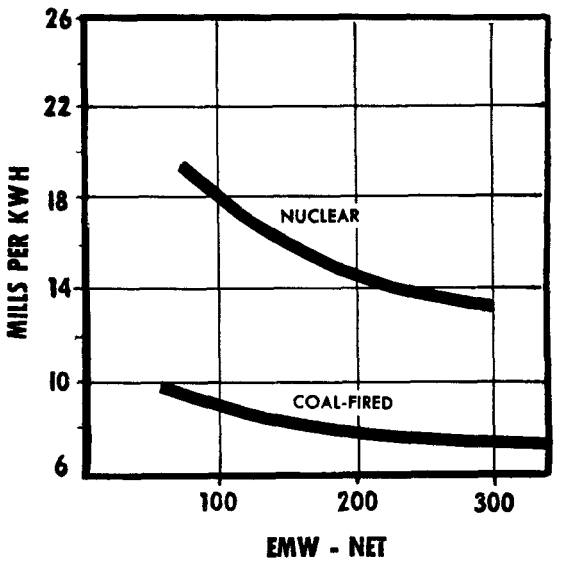
SEPT. 1, 1959
FIG. 25

POWER GENERATION COSTS

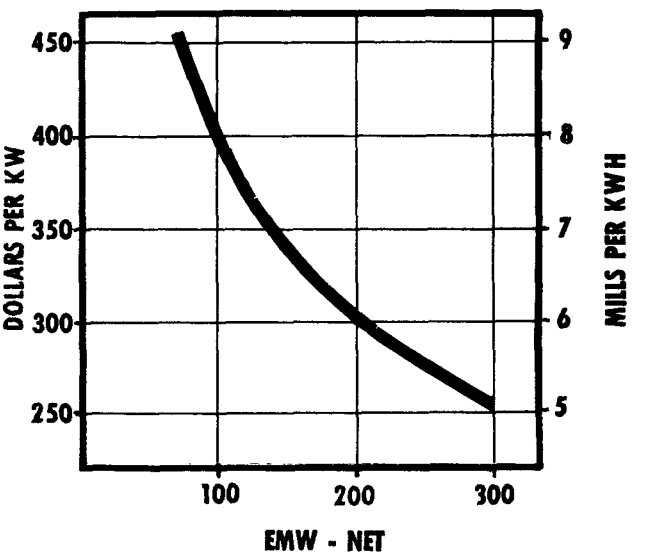
FAST BREEDER REACTORS
VS
COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.

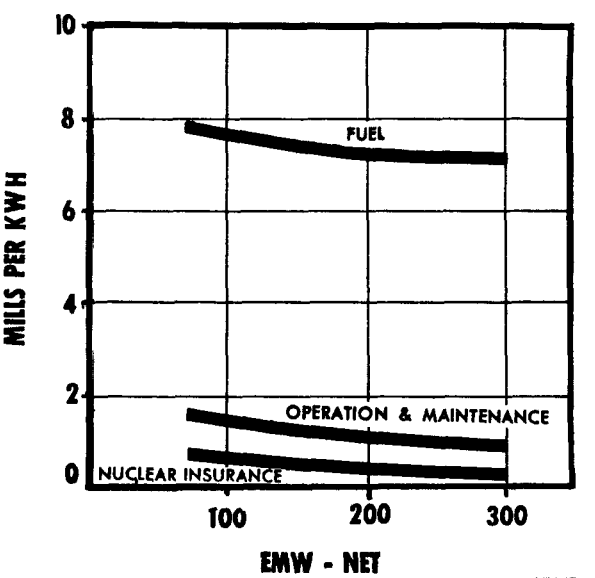
POWER GENERATION COSTS



CAPITAL COSTS



OPERATING COSTS



NOTES :

NUCLEAR -
BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/GM.
FUEL FABRICATION COST	
U-10% MO - ZR	480/KG.
FUEL EXPOSURE	1½% Atom Burn-Up

COAL - FIRED -
BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%
FUEL COST	35¢/10^6 BTU

SEPT 1959
FIG. 26

2. THERMAL BREEDER REACTOR - AQUEOUS HOMOGENEOUS

a. Description. Two-region circulating-fuel reactors are considered to be the most promising aqueous homogeneous thermal breeder power reactors. The preferred type has a dilute (less than 0.03m) solution of uranyl sulfate, predominately fissionable uranium, in D_2O in the core. The ThO_2 slurry is in the blanket. Preference is based on the greater potential of the solution-core reactor for combining high conversion ratio and high specific power to achieve an efficient breeder. Fuel solution and slurry are circulated through the reactor and steam generators, where nonradioactive steam is produced to drive a turbogenerator. The reactor plants require remote maintenance of primary systems and auxiliaries. "On-site" processing appears to be desirable for large power stations.

b. Technical Status. A central station plant of this concept could not be built based upon current technology. However, sufficient technical work has been done to arrive at the following conclusions on technical status:

(1) Physics. The physical and nuclear characteristics and the adjustment of fuel concentrations in homogeneous reactors are sufficiently simple that critical experiments have been limited to startup experiments on HRE-1, HRE-2, and the small research reactors. Reactor statics calculations are made by use of two-group methods in which the constants are based on data obtained from other programs. Agreement between calculated and measured critical concentrations has been good in small reactor experiments and should be even better as the size increases.

Neutron balances and breeding ratios have been calculated by both modified two-group and multigroup methods. An important uncertainty in the calculations has been in the value of η, the number of neutrons emitted per neutron absorbed in the fuel. The best value, based on

recent careful measurements, is about 2.29 + .02/-.01. This should allow a breeding gain of 0.1 and doubling times of about 15 years.

(2) Fuels and Materials. Investigation of fuels and materials both in- and out-of-pile, including the development of equipment for those studies, has constituted one-third to one-half of the research and development program. Studies of potential fuels have led to selection of uranyl sulfate solutions and thoria slurries as being the most promising. Most of the criteria have been established for use of stainless steel as the general container material and Zircalloy 2 as the core-tank material. Titanium has been established as a potentially superior container material.

(a) Solution Fuels. Solubility relationships have been investigated for uranyl sulfate in both H_2O and D_2O. The studies are being continued to establish details of the phase diagrams for the five-component reactor core fuel, UO_3-CuO-NiO-SO_3-$D_2O(H_2O)$, in the region 0 to 0.04 $\underline{m}$ in UO_3, CuO, and NiO and 0 to 0.20 $\underline{m}$ in SO_3. Copper is present in the fuel to catalyze the recombination of radiolytic gas, and nickel is a soluble corrosion product.

Hydrolysis occurs at high temperature and the hydrolysis products separate from the bulk solution as precipitates, which may or may not contain appreciable uranium (depending on the concentration region), or as a heavy liquid phase which is enriched in all metal constituents.

Dilute solutions in which the molar ratio $\frac{U_2O_5^{++} + Cu^{++} + Ni^{++}}{HSO_4}$ is 0.5 are stable against precipitation of solids to at least 300°C and the second liquid phase temperature is above 320°C. The temperature at which instability occurs is raised by adding sulfuric acid to decrease

the ratio. There is little difference between the solubilities in H_2O and in D_2O.

Decomposition of the water is the only observed effect of reactor radiations on dilute uranyl sulfate solutions. At low temperature D_2 and D_2O_2 are produced, and uranyl peroxide precipitates from solutions at power densities above 1 KW/liter at 100°C. At power reactor temperatures, peroxide decomposes very rapidly and the products are radiolytic gas, D_2 and O_2. Copper sulfate is an effective catalyst for recombining the gases in solution to reduce the hazard and the problems of handling large quantities of gas. Concentrated solutions have been irradiated at the power densities that would be produced in large reactors, but facilities have not been available for doing this with the dilute fuels.

(b) Thorium Oxide Suspensions. In the absence of a suitable thorium solution, suspensions or slurries of thorium oxide in water are being developed as reactor fuels.

Flow studies have established that the yield stress increases with increasing concentration and decreasing size of particles of a given shape. Current production material must be larger than two to three microns to obtain the desired 0.1-lb/ft^2 yield stress in blanket slurries. Slurries must be kept in turbulent flow to maintain homogeneity. Velocities of 2 to 8 ft/sec are necessary to obtain fully turbulent flow in pipes. Heat transfer and fluid flow behavior of slurries in turbulent flow are those of a Newtonian fluid with the same physical properties. In laminar flow slurries are non-Newtonian having properties similar to Bingham plastic.

Irradiation of slurries for a few hundred hours at power densities below 10 KW/liter, representative of the average power density in a blanket system, has shown no obvious effect on slurry

properties. Longer tests at higher power densities are required to provide definitive data.

Deuterium and oxygen are produced by irradiation of slurries also. Molybdenum oxide and palladium catalysts are being tested in-pile and out-of-pile and have recombined the gases satisfactorily in-pile at power densities of about 10 KW/liter. Considerable additional work is required to establish their suitability for use in a reactor.

(c) Corrosion. Corrosion of stainless steel, Zircalloy 2, and titanium in uranyl sulfate fuel solutions has been studied extensively. It has been established that ASA type 347 stainless steel is a satisfactory material for containing the dilute fuel solutions in the heat-removal and auxiliary systems of a homogeneous reactor if (1) the excess acid is less than 0.02 m or possibly 0.03 m, (2) the velocity is less than 25 ft/sec, (3) the temperature is below 150°C or above 250°C but below the boiling point, (4) there is enough oxygen in solution to stabilize the uranyl ion, and (5) the chloride concentration is less than 2 ppm.

Below 150°C the corrosion rate is less than 2 mpy (mils per year) and independent of velocity. Above 250°C and with flows below a critical velocity the metal corrodes less than 1 mil in forming a protective film, and then there is essentially no corrosion as long as the film is intact. Above the critical velocity the film does not form or once formed is eventually removed and rates of 100 mpy are common. The intermediate temperature range is a transition region in which heavy scales that are only partially protective form. Corrosion rates are 5 to 50 mpy. The rates and their dependence on velocity increase with rising temperature; critical velocities are low. Very few parts of a homogeneous reactor operate continuously in the intermediate temperature range.

Stainless steel is used if the velocity is low and a short service life is acceptable, but titanium is the preferred material for those parts.

Increasing the acidity lowers the critical velocity and increases the corrosiveness of the fuel. Oxygen depletion results in reduction of the uranyl ion, hydrolytic precipitation of the reduced uranium at high temperature, and an increase in acidity. A concentration of 50 ppm of dissolved oxygen is sufficient, but 500 ppm is maintained in engineering systems to provide against depletion in crevices and stagnant regions. Boiling depletes the fuel of oxygen. Chloride concentrations must be kept low to prevent stress-corrosion cracking of the austenitic stainless steel.

Experience in in-pile loops and in HRE-1 and HRE-2 indicates that fission product radiation in the fuel circulating systems and the auxiliaries has no important effect on corrosion. A few hours of high corrosion, tentatively attributed to oxygen depletion, during the first low-power operation of HRE-2 has been followed by several thousand hours at rates less than 0.5 mpy.

No important difference has been found between type 347 SS and other austenitic stainless steels. Titanium and zirconium are not corroded appreciably by uranyl sulfate solution fuels at much higher acidity, flow velocity, and temperature than can be used with stainless steels. Corrosion of zirconium is very severe in fuel solutions which are contaminated by small amounts of fluoride.

Corrosion in slurry systems is a combination of corrosion by water and erosion by slurry particles. Austenitic stainless steels appear to be satisfactory container materials for oxygenated slurries at velocities as high as 20 ft/sec and at all temperatures and slurry

concentrations of interest if the design of the equipment and size or shape of the slurry particles are properly controlled. Careful design to prevent flow separation is important. Irregularly shaped particles with sharp corners must be smaller than 5 microns. The upper limit on size of spherical particles is not known.

Stress-corrosion cracking has been troublesome in some parts of slurry systems. The tolerable chloride concentration has not been established. Inconel and low-alloy steels which are highly resistant to chloride-induced cracking are promising substitutes for stainless steel. Zircalloy 2 and titanium are highly resistant to stress-corrosion cracking and to corrosion and erosion by oxygenated slurries under severe flow conditions. Hydriding of Zircalloy has been experienced with hydrogenated slurries when exposed to highly turbulent flow of mixtures of slurry and vapor.

Zircalloy 2 is presently the only material which is sufficiently good in corrosion resistance, mechanical properties, and low neutron-absorption cross sections to be useful for the core tank of a breeder. Exposure to fissioning solutions has been shown to increase markedly the corrosion rates of Zircalloy 2 and other materials. Corrosion rates of 11 mpy at 250°C and 40 mpy at 300°C have been estimated for the core tank of the target breeder in fuel solution. The estimates are based on the correlation of results of many tests over wide ranges of conditions; facilities are not available for performing tests with dilute fuel solutions at the high power densities of a large breeder. Tests have shown that zirconium-niobium alloys may be more corrosion-resistant than Zircalloy 2.

Preliminary tests in autoclaves indicate that corrosion of Zircalloy 2 is increased by exposure in fissioning slurries. The effect appears to be less important than that found with solution fuels.

(d) Chemical Processing. Krypton, xenon, and iodine are removed from the core fuel continuously by boiling a portion that is circulated between the reactor high-pressure and low-pressure systems. Iodine can be separated from the vapor by reaction with silver in a packed column or absorption in water in a distillation column. The vapor is then condensed, and the gases that are left are absorbed on carbon until they decay. Processes involving silver and carbon beds have been demonstrated on HRE-2; efficiency of the iodine removal process is still being determined there.

Most of the fission and corrosion products have been shown to be only slightly soluble in fuel solutions under reactor conditions, and a method based on hydroclones (hydraulic cyclone separators) was developed for removing the solids. A hydroclone plant has been operated in conjunction with HRE-2. The low solubility of fission and corrosion products has been demonstrated, but only about 10% of the solids have been removed by the hydroclone. Most of the solids deposited on pipe walls and separated in regions of low flow in the reactor systems. Means are being sought for increasing the effectiveness of the hydroclone plant. Nickel and manganese, from the corrosion of stainless steel, and fission-produced cesium will not precipitate from fuel solution until concentrations have been reached which could result in fuel instability or significant loss of neutrons. It is expected that the processing rate will be determined by the rate of buildup of corrosion products. The amount of uranium processed daily should be so small that the solution can be combined with the blanket

slurry and processed in a Thorex plant. Uranyl peroxide precipitation and electrolytic methods for processing larger amounts of solutions rapidly in D_2O have been demonstrated in laboratory. In HRE-2, the first batch of fuel required processing after a few hundred hours; the second has been used for more than 5,000 hours.

Methods are being sought for rapid removal of uranium and fission products from thoria without destroying the slurry particles, but at present it is necessary to convert the oxide to a solution of thorium nitrate in H_2O and process by solvent extraction in a Thorex plant. Simple methods which should be easily adaptable to remote handling have been developed for converting the uranyl nitrate product to uranyl sulfate in D_2O.

Thoria is presently produced from thorium nitrate in a pilot plant by precipitating thorium oxalate, decomposing the oxalate in a furnace at 300°C, firing the resulting oxide in the range of 800 to 1600°C, and dispersing the dry product in water. Cubes or platelets with average size controlled in the range 1 to 5 microcuries are produced. Uranium has been incorporated in the thoria in U/Th ratios from 0.005 to 0.33 in the development of slurry fuels for the core of the alternative breeder. Methods of making spherical particles and of converting the nitrate directly to oxide are being developed. The processes must be adapted to remote handling for recycling long-irradiated thoria.

(4) Components and Auxiliary Systems. The component development work has been directed primarily toward equipment for use in the Homogeneous Reactor Experiment (HRE-1) and the Homogeneous Reactor Test (HRE-2). Although the HRE-2 has both a core system and a blanket system, most of the components in the two systems are identical and are designed for use with solutions rather than suspensions.

Since suspensions, or slurries, have not been used in either of the reactor experiments, the slurry equipment problems have received less attention than corresponding solution problems. Much of the solution technology can be applied to slurries, although additional difficulties such as the settling tendency of slurries, their less ideal fluid-flow behavior, and their erosiveness must be taken into consideration.

(a) Pressure Vessel. The HRE-2 reactor-vessel assembly presented a number of special design and fabrication problems. Since it was desired to test a model of a power breeder core, Zircalloy 2 was selected as material for the core tank, which is 32 inches in diameter and 5/16 inch thick. The main pressure vessel, 60 inches in inside diameter and 4.4 inches thick, was constructed of carbon steel with a cladding of type 347 stainless steel. Because of uncertainties in the long-term irradiation damage of carbon steel, the pressure vessel was surrounded by a stainless steel, water-cooled blast shield which will stop any possible missiles from the reactor vessel. Thermal radiation from the pressure vessel to the blast shield permits the pressure vessel to operate at close to an optimum temperature distribution from the thermal-stress standpoint.

(b) Solution Pumps. Pumps are required in aqueous homogeneous reactors to circulate solutions and slurries at 250 - 300°C and 2,000 psi pressure, at heads of up to 100 psi. The two main considerations for these pumps are that they must be absolutely leak-free and must have a long maintenance-free life. At this time the hermetically sealed canned-motor centrifugal type pumps are considered capable of meeting these requirements. They consist of a centrifugal pump of standard hydraulic design and an electric-drive motor, built in an integral unit.

(c) Solution Steam Generators. "The performance of steam generators required for homogeneous reactor service, measured in terms of absolute leaktightness during long-term operation, considerably exceeds that of similar units in conventional plants. Primary efforts in assuring leaktightness for an extended period of time have been expended in development of adequate fabrication techniques, inspection of raw materials and fabricated parts, and inspection and testing of the finished unit. No problems with leaks have been encountered in the small heat exchangers for HRE-1 and HRE-2 which were manufactured by use of those techniques. At the present time, failure of a steam generator in a homogeneous reactor would require that the entire unit be replaced. This would be expensive and would lead to excessive shutdown time if failures were frequent. Methods to detect and repair leaks in place are in a preliminary stage of development."

(d) Slurry Fuel Systems. The development of HRE-2 scale equipment for slurry reactors is less advanced. Slurries have been circulated in many loop tests under a variety of conditions. Two tests were for 3,800 hours each. Severe wear of Graphitar-Stellite bearings and stainless steel impellers in the circulating pumps was experienced in those tests. An improved pump with aluminum oxide bearings and a Zircalloy impeller has operated for 2,800 hours with no significant wear. No completely satisfactory solution has been found for the problem of erosion of throttling valves and check valves in high-pressure feed pumps. Improvements have been made in the hydrodynamic design of reactor vessels for slurries, but not enough testing has been done to establish whether a satisfactory design has been achieved.

(5) Safety. Effectiveness of the negative temperature coefficient in controlling the reactor has been demonstrated in HRE-1 and

HRE-2. It has been concluded that control rods are not required in the reactors. The kinetics of a variety of reactors have been investigated by calculation, and experiments have been done in HRE-1, HRE-2, and in the "water-boiler" reactors. Criteria have been established for stability against divergent oscillations in power produced by the interaction of the nuclear and mechanical systems of a circulating-fuel reactor. Methods have been developed for calculating the power, temperature, and pressure behavior of a circulating-fuel reactor in relation to a step increase or rate of increase of reactivity. The kinetics studies have been used to establish criteria for design of reactor pressurizer and pressure-relief systems.

c. Current Design Parameters. The technology is not sufficiently developed to currently construct a central station power plant; therefore no design parameters are given in this status report.

d. Operating Experience. There is no operating experience on a thermal breeder reactor as such. HRE-1 was not potentially capable of breeding, and the failure of the ZR-2 core tank on HRE-2 prevented any attempt at breeding with that system. However, the operating experience with both HRE-1 and HRE-2 is pertinent to the development of thermal breeder reactor technology. Both reactors employed uranyl sulfate solution in the core and D_2O in the blanket. No reactor has operated with slurry fuels in core or blanket.

(1) HRE-1. In 1950 construction of Homogeneous Reactor Experiment (HRE-1) was undertaken to investigate the nuclear and chemical characteristics of a circulating-fuel reactor at temperature and power sufficiently high for production of electricity from the thermal energy

released. The reactor was designed to deliver 1,000 KW of heat from fuel solution at 482°F and 1,000 psia, yielding, after heat exchange, steam at a saturation pressure of about 200 psia. Fuel was a solution of uranyl sulfate in H_2O. The reactor core was an 18-inch diameter stainless steel sphere, reflected with D_2O.

The reactor was operated for about 24 months beginning in April 1952. Liquid was circulated for about 4,500 hours; the reactor was critical a total of 1,950 hours and operated above 100 KW for 720 hours. The maximum power attained was 1,600 KW, and a total of 600 MWH of heat was produced. HRE-1 was dismantled in the Spring of 1954 to be replaced by HRE-2.

Operation of the reactor demonstrated:

(a) nuclear and chemical stability at average power densities as high as 32 KW/liter,

(b) safe handling of the highly radioactive fuels at high pressure and temperature,

(c) effectiveness of the negative temperature coefficient in regulating the normal reactivity fluctuations and in controlling power surges that result from large rapid additions of reactivity, making regulating and safety rods unnecessary,

(d) control by control of fuel concentration and release of steam to turbine, making shim rods unnecessary,

(e) safe handling of radiolytic gas and use of copper as a homogeneous catalyst to recombine the gas.

Although the reactor was not designed for maintenance, the fuel circulating pump and feed pump were replaced several times, and some other work was done in radiation fields as high as 2,000 r/hr by use of improvised shielding and long-handled tools and without exposure of personnel beyond accepted tolerances. The work was difficult and even simple operations took a long time. At the end of the operation the reactor was partially decontaminated, and the reactor and shield were dismantled by similar methods.

(2) HRE-2. During operation of HRE-1 it became evident that a second reactor experiment should be run before attempting to construct a full-scale homogeneous power reactor. HRE-2 was built to provide a facility for investigating the engineering problems of operating and maintaining fluid-fuel reactors and the longer term effects of operation on fuels and materials. Basic design criteria were: a 32-inch-diameter core contained in Zircalloy 2 and reflected by a D_2O blanket to permit operation at 250 to 300°C with a 10-g-U-per-liter solution of uranyl sulfate in D_2O as the fuel. Cost considerations led to the specification of 5,000 KW (thermal) for the power. Although the average specific power in the core was only 17 KW/liter, this was considered acceptable since operability at 30 KW/liter had been demonstrated in HRE-1.

The reactor was designed to operate at 2,000 psia. Equipment and systems were designed to take advantage of experience with HRE-1 and to test ideas that were being proposed for power reactors. Maintenance and containment were primary considerations in the design. Primary equipment and auxiliaries were housed in a steel tank which is welded closed and maintained at $\frac{1}{2}$ atm abs when the reactor is operating. Provision was made for submerging the equipment in water to provide shielding so that

most operations could be done with simple long-handled tools. Special shields and tools were provided so that some operations could be done dry.

First nuclear operation was in December 1957. Between then and August 1, 1959, fluids were circulated for a total of 5,750 hours, 3,420 hours critical, in generating 5,720 MWH of heat, and operation is continuing. A maximum power of 6.3 MW was reached during the first week of high-power operation. A hole was melted in the core tank after 30 minutes at 6.3 MW. Subsequent operation has been with fuel in core and blanket at power levels as high as 5.3 MW but mostly at 3.5 MW and below in investigation of fuel instability.

Experience with HRE-2 has shown that mechanical components can be developed which will permit long uninterrupted operation of a complete reactor system. Since the core-tank incident the reactor has been operated in four tests of 800 to 1,100 hours duration. Maintenance was required between tests, but in no instance was a test terminated by an equipment failure.

The ability to construct completely leaktight systems by care in fabrication and inspection and to keep the systems tight and the highly radioactive fluids contained have been demonstrated in $1\frac{1}{2}$ years of operation.

Major difficulties experienced with HRE-2 have resulted from an unsatisfactory hydrodynamic design of the core coupled with the high-temperature instability of the fuel. In addition to the hole in the core tank, damage to corrosion assemblies, thermocouples, and diffuser screens in the core has provided evidence of severe overheating of some metal surfaces. This experience has led to renewed emphasis on core design and hydrodynamics studies in the development programs.

The hydroclone plant has operated satisfactorily but has removed only 10 to 20 per cent of the fission - and corrosion - product solids which have been produced.

Determination of fuel inventory and analysis of abnormal behavior have been difficult. Additional and improved instruments for measuring flows and fuel concentrations are required to obtain completely satisfactory process control.

Experience with HRE-1 and HRE-2 has established that remote maintenance of the radioactive equipment is an extremely important consideration in the design of homogeneous reactors. Methods presently in use on HRE-2 are generally satisfactory for small experimental reactors. Advances in present techniques, such as inspection and repair of large components and possibly the development of remote cutting and welding methods, will be required to provide satisfactory methods for large power reactors.

e. Economics. No economic data is given in this report on the aqueous homogeneous reading concept.

C. Natural Uranium Reactors

The reactor types that are classified in this grouping are:

1. Natural Uranium - Heavy Water Moderated Reactors

2. Natural Uranium - Gas Cooled Reactors

3. Plutonium Recycle Reactors

The primary objective of the natural uranium and recycle class of reactors is to develop a power plant capable of sustained operation, independent of a continuing supply of enriched uranium fuel. A secondary objective is to reduce the cost of power to levels that are economically attractive without sacrificing the natural uranium feed objective.

A composite curve showing the power generation cost of natural uranium and recycle reactors vs. coal fired plants is presented in Fig. 27. The technical status of each reactor concept in this class of reactors is summarized in the following pages of this section.

IV

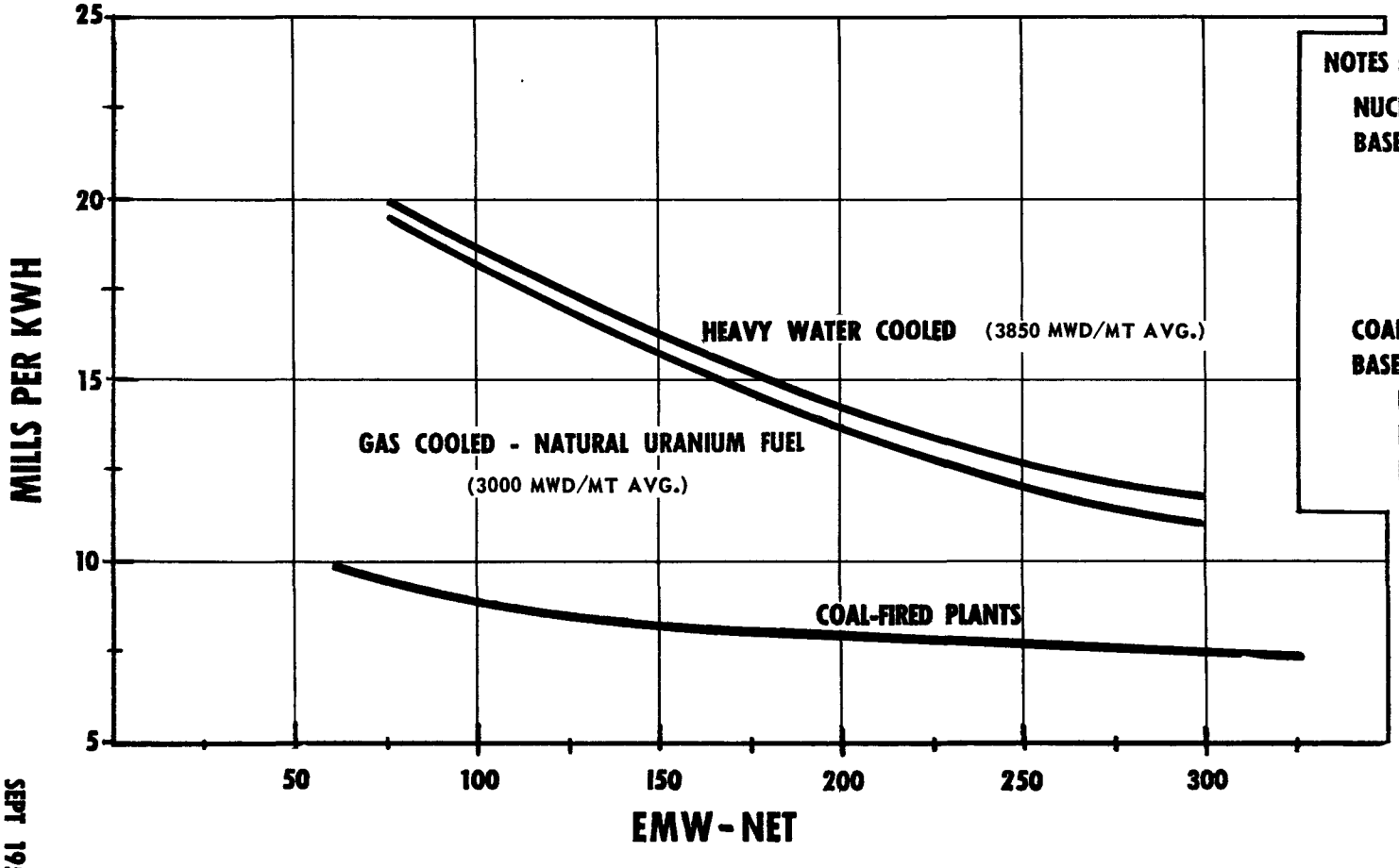
Status
Civilian Power Reactors

C. Natural Uranium
and
Pu Recycle Reactors

POWER GENERATION COSTS

NATURAL URANIUM FUELED AND RECYCLE REACTORS VS COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.



NOTES :

NUCLEAR -
BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/GM.

COAL - FIRED -
BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%
FUEL COST	35¢/10^6 BTU

FIG. 27
SEPT 1959

1. D_2O MODERATED POWER REACTORS

a. Description. Heavy water moderated power reactors include a variety of concepts employing natural or partially enriched fuel, pressurized or boiling D_2O, and a pressure tube or pressure vessel type of construction. Alternative coolants include gas or D_2O.

The variation of the concept which has the most advanced current technology is the D_2O moderated and cooled, pressurized reactor of the pressure vessel type. This reactor utilizes solid, heterogeneous natural uranium metal fuel elements clad with Zircalloy-2 in a uniform single-region core. D_2O is circulated through the reactor to remove the heat from the fuel, the heat is transmitted through a heat exchanger to form steam to drive a steam turbine. This concept is based upon the technology of pressurized H_2O moderated and cooled reactors and the D_2O moderated production reactors at Savannah River. A simplified flow diagram of the heavy water moderated reactor plant is shown in Figure 28.

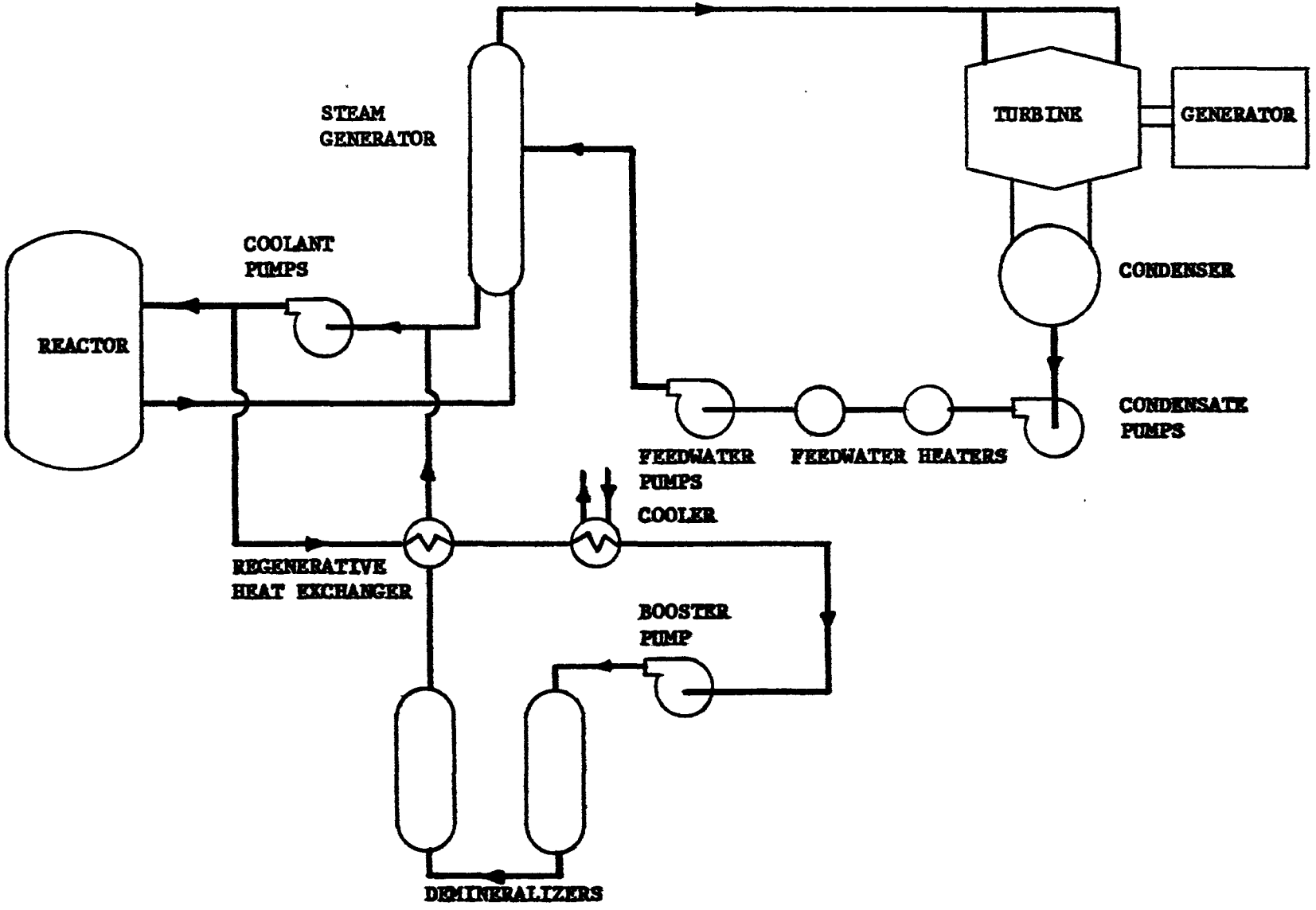
b. Technical Status. No large central station power plants of this reactor type have been constructed. The minimum size limitation on natural uranium fueled power reactors is set by neutron economy considerations. This leads to the consideration of slightly enriched fueled prototypes of concepts potentially capable of operation on natural uranium fuel in a larger plant size. At present it is not possible to determine what version of the heavy water reactor, pressurized or boiling, pressure tube or pressure vessel is technically and economically most attractive. Technical status consists primarily of development work completed and under way in support of various experimental and demonstration power reactor construction projects and applicable production reactor technology. The following technical

facts may be drawn from this work:

(1) Physics. Nuclear properties of a wide variety of cold-clean natural uranium lattices in a D_2O moderator have been measured. The type of core lattices for which bucklings had been measured include D_2O-cooled assemblies of uranium metal plates, uranium metal rods, uranium metal tubes, and uranium oxide rods, as well as gas-cooled assemblies of uranium metal plates. However, most of these experiments were made with moderator temperatures below 95°C. Other important physics parameters of typical power reactor lattices of natural uranium in heavy water have been measured. These measurements included such items as the worth of control rods, the effect of H_2O contamination of the D_2O moderator, the fast fission factor, the neutron age, flux distribution, etc. In addition, correlations were derived from the experimental data for calculating resonance integrals and resonance escape probabilities for fuel elements of uranium oxide as well as of uranium metal. All of the measurements were made in either the Process Development Pile (PDP) or the subcritical experimental facility (SE) at the Savannah River Laboratory.

Most of the measurements discussed in the preceding paragraph were made at temperatures below 95°C. In order to extend the range of the experimental physics work to moderator temperatures as high as 215°C, the pressurized exponential facility (PSE) was built. The unit was put into operation during the first half of 1959.

Additional physics information is required to advance the technology of D_2O moderated reactors sufficiently to permit construction of the more promising concepts. This work would include:



SIMPLIFIED FLOW DIAGRAM

HEAVY WATER MODERATED AND COOLED
NATURAL URANIUM REACTOR PLANT

SEPT 1959
FIG. 28

(a) A thorough survey of the properties of oxide-fueled lattices.

(b) Measurement of the nuclear properties of lattices moderated with hot D_2O.

(c) Measurement of the nuclear properties of the more promising conceptual designs by fairly accurate mockups of the reactor lattices.

(d) Careful analysis of the transient response of each reactor concept to sudden changes in temperature, load, or reactivity.

(2) Fuel and Materials. A major area in which research must be done before any heavy water reactor concept can be successfully constructed is in fuel element fabrication and the determination of irradiation behavior.

Although extensive testing of prototype fuel elements has been under way and much testing of Zircalloy has been done, the following list summarizes the major areas in which considerable research and development effort may be fruitful:

(a) Irradiation testing of prototype metallic fuel elements is necessary in order to prove their adequacy to withstand long periods of exposure to neutrons under conditions comparable to those expected in the power reactor designs.

(b) The irradiation testing of pressure tubes and their joints with other materials is highly essential for reliable design of all pressure tube reactors. The mechanical and hydraulic testing of these items, out-of-pile, would also be of much worth. In addition, development of a low-cost high-quality fabrication process

for the pressure tubes and their joints would assist significantly in reducing the cost of several promising reactors. Thermal stress conditions are a particularly important consideration in a pressure tube design.

(c) Corrosion testing of fuel elements, pressure tubes, housing tubes, and other important components of the reactor should be done in order to assure the safe operation of a given design.

An extensive program has been in progress to fabricate low-cost fuel elements and components suitable for use in D_2O-cooled power reactors. Primary interest has been on uranium metal fuel elements, but oxide elements are also being developed as an alternative to the metal fuel. The metal was selected in preference to the oxide as the fuel for initial development, in spite of better irradiate stability and corrosion resistance of the oxide, because of the generally higher reactivity, lower cost and better known fabrication process of the metallic fuels.

The potentially low cost coextrusion process was applied by Nuclear Metals, Inc. to the manufacture of Zircalloy-2 clad tubes of U-2 w/o Zr. A "production" run of about a dozen tubes was successfully carried out. Numerous corrosion tests in hot water have been made of this fuel, withand without intentional defects. The fuel is deemed to be suitable from a corrosion viewpoint for a D_2O-cooled power reactor.

Extruded zirconium and Zircalloy components are being fabricated by Harvey Aluminum Company. Very promising results were obtained in the manufacture of thin-walled, ribbed housing tubes of Zircalloy-2.

One of the most important phases of the research and development program on fuels for D_2O-moderated power reactors involves the irradiation testing of prototype reactor fuel elements. The ability of uranium metal to adequately withstand long exposures to neutron irradiation was tentatively verified at low temperature. In order to irradiate significant numbers of full-size fuel elements at high temperature, the design and construction of the Heavy Water Components Test Reactor (HWCTR) is underway.

(3) Heat Transfer and Fluid Flow. The major experimental work in this field to date has been with subcooled liquid coolant. In order to determine the safe limits of operation, of various reactor fuel elements that are cooled by a boiling fluid, it will be necessary to undertake an experimental program on boiling heat transfer burnout.

An experimental study of heat transfer burnout of water-cooled tubular assemblies of the type designed for the du Pont concepts of liquid D_2O-cooled power reactors was completed at the Heat Transfer Research Facility of Columbia University. The range of variables of this study was limited to the particular range that is applicable to the du Pont power reactors. A study was also conducted of the vibration phenomenon that was encountered in some of the burnout experiments. The results of the experimental study were used to compute the safety factor on heat transfer burnout for the liquid-water-cooled assemblies of the power reactor designs and of the HWCTR.

Full-scale flow tests of the liquid-cooled fuel elements of the du Pont power reactors have been run at the Savannah River Laboratory. The corrosion, erosion and vibration characteristics, as

well as the customary pressure-drop-versus-flow behavior, of these fuel elements were and are still being determined.

(4) Coolant Chemistry. The chemistry of heavy water is in an advanced state of development due to the extensive work done in this area at Savannah River. Studies of resin lifetime at High Temperature and impurity-removal effectiveness are needed.

(5) Component and Auxiliary Systems. The most important work in this area will involve the construction of prototype refueling machines.

Development of high-integrity mechanical components such as pressure tubes, and heat exchangers is required. Special precautions must be taken to make the entire system leakproof, including the turbine in a boiling water system, and to prevent the loss of heavy water during refueling operations.

(6) Safety. Considerable theoretical study of the transient behavior and the responses of the reactor to unusual incidents (i.e., changes in reactivity, temperature, etc.) were studied, particularly for the hot moderator pressure vessel reactor and the HWCTR. It was shown that the hot moderator reactor and the HWCTR are to a great extent self-regulating and that such unusual incidents would cause no difficulty in control. A thorough analysis of the safety of the experimental reactor, the HWCTR, was completed.

Experiments which were carried out to confirm the safety aspects of the Canadian power reactor schemes included a series of tests in which pressure tubes were ruptured to determine the effect on the enclosing calandria tubes and adjacent tubes.

Reaction of heavy water with fuel and/or structural materia is a potential and significant hazard.

c. Operating Experience. To date, construction has begun on only one of the D_2O-moderated power reactors, the AECL Nuclear Power Demonstration, NPD-2. No experience has been gained as yet in the operation of any D_2O-moderated power reactor. However, there is considerable experience available on the operation of D_2O-moderated production and experimental reactors, such as the Canadian NRU and NRX reactors and the U. S. Savannah River production reactors. A short history of the experience with heavy-water-moderated reactors follows.

The production reactors have given operating experience that should be helpful to the designers of a D_2O-moderated power reactor. For example, these reactors circulate large quantities of D_2O at moderate temperatures and pressures through heat exchangers, valves, pumps, etc., and the annual loss of D_2O has been only about 3% per year, based on the total D_2O in the system.

The D_2O isotopic purity of the moderator at Savannah River has been maintained at above 99.7% via a distillation procedure. The inleakage rate of H_2O has been in the range of 6 lbs/day requiring 6000 lbs/hr of low pressure steam for its removal. Because the pressure in a D_2O-moderated power reactor will be higher than in a production reactor, the H_2O inleakage would be expected to be smaller. On the other hand, the higher D_2O pressure will probably affect D_2O losses adversely. Other moderator problems, including radiolytic decomposition and entry of ionic contaminants, have been easily kept under control. Tritium in the moderator presents a minor health hazard when this material is spilled, but the usual protective devices,

such as gloves and aprons, are generally sufficient to allow safe working in the area.

In addition to this experience with heavy water moderator, some experience has been obtained with each of the cooling mechanisms that are being suggested for the D_2O-moderated power reactors. Several liquid-water-cooled and boiling-water-cooled reactors have been built and operated successfully in the United States and Canada. Among the water-cooled reactors are the Hanford, Savannah River, PWR, MTR, NRU and NRX; among the boiling-water reactors are the EBWR and VBWR. Experience in the United States with gas cooling has been gained with the X-10 and Brookhaven piles. Experience with sodium cooling has been gained in the operation of the graphite-moderated SRE. The greatest amount of experience, however, has been with liquid-water cooling.

Thus, much of the experience with heavy-water-moderated production reactors is applicable to power reactors moderated with the same fluid. However, there are still sufficient differences between the production and power reactors, especially in the fuel elements and materials of construction, to require a significant amount of research and development effort.

The chief problem is related to the higher temperatures required for economic power reactors. The Savannah River reactors have the moderator tank at atmospheric pressure and therefore can heat the water that discharges into this space to only about 210°F. Power reactors require a temperature of at least 400°F and preferably, considerably higher. If the reactors are cooled with D_2O, this high

temperature requirement forces the designers to employ pressures of 500-1500 lbs/sq in - a not too difficult item in itself, but when coupled with the fact that zirconium is almost the only material of construction that can be used for the pressure tubes entering the reactors, the problems are increased several fold.

Another major difference is in the cladding of the fuel. All production reactors in the Western group of nations currently use either aluminum or magnesium for this service. Neither of these materials is very corrosion resistant in 200°C water. Zirconium is being used as the cladding in most of the power reactors that are cooled with either D_2O or H_2O. Again new technology must be and is being developed.

d. Plants Under Construction. At present there is no D_2O moderated, natural uranium fueled power reactor under construction in this country.

Canada is now building a 20 MWE power reactor that will be moderated with heavy water and is planning to build a 200-eMW reactor. Several pilot and test reactors of a similar type are under construction, both in this country and abroad. Two of these plants are not for the purpose of producing electric power, but the experience gained by their construction and operation will be applicable to the advancement of the technology. A brief description of these plants is given below:

(1) Heavy Water Components Test Reactor (HWCTR). The HWCTR is a pressurized D_2O moderated and cooled reactor facility at 60 MWT with no electrical output. The reactor will utilize U-Zr fuel.

The reactor is rated at 1500 psig and 290°C maximum. The plant is being designed by E. I. du Pont for the AEC. Construction has been started.

(2) Plutonium Recycle Test Reactor (PRTR). The PRTR is a D_2O moderated and cooled pressure tube type reactor facility rated at 70 MWT with no electrical output. Plant construction started in March, 1959.

(3) Carolinas-Virginia Nuclear Power Associates (CVNPA). The CVNPA reactor is a D_2O moderated and cooled pressure tube type reactor rated at 60.5 MWT (17 MWE). The fuel for this reactor will be slightly enriched UO_2. The steam conditions are 605 psia and 480°F. The plant is being designed and constructed by Westinghouse Electric Corporation and Stone and Webster. Construction is scheduled to start in March, 1960. A schedule for the plants and facilities under construction is shown in Figure 29.

(4) Nuclear Power Demonstration Reactor (NPD-2) The NPD-2 is a D_2O moderated and cooled horizontal pressure tube prototype power reactor rated at 20 MWe gross, being built in Canada. It will use natural UO_2 fuel pellets clad with Zr-2. The reactor is controlled, to meet the requirements of the turbine, by adjusting the level of the moderator. The D_2O coolant will leave the reactor at 530°F and pressure of 1050 psi, just sufficient to keep it from boiling. Construction is underway and scheduled for completion in June, 1961.

e. Economics. A reference design has been made for the D_2O-moderated reactors based on the technical status summarized in this report. The design was used as the basis for estimating the costs at different plant ratings. The base parameters for the reference design are as follows:

SUMMARY OF PLANT CHARACTERISTICS
HEAVY WATER REACTOR PLANT

A.	**Heat Balance**	
	1. Total Reactor Power, MW(t)	860
	2. Gross Turbine Power, MW(e)	214
	3. Net Plant Power, MW(e)	200
	4. Net Plant Efficiency, %	23.2
B.	**Turbine Cycle Conditions**	
	1. Throttle Temperature, F.	366
	2. Throttle Pressure, psig	150
	3. Steam Flow, lbs./hr.	3.03×10^6
	4. Condenser Back-Pressure, in Hg A.	1.5
	5. Final Feed Water Temperature, F.	251
C.	**Reactor Description**	
	1. Reactor Vessel	
	a. Inside Diameter, ft.	14.7
	b. Overall Height, ft.	34.0
	c. Wall Thickness, in.	4.5
	d. Material	SS clad CS
	e. Design Pressure, psia	850
	2. Reactor Core	
	a. Active Diameter, ft.	12
	b. Active Height, ft.	15
	c. Active Core Volume, ft^3	1700
	d. Lattice Arrangement	Triangular
	e. Lattice Spacing, in.	6.5
	3. Reflector or Blanket	
	a. Material	D_2O
	b. Axial Thickness, ft.	1
	c. Radial Thickness, ft.	1
	4. Fuel Elements	
	a. Fuel Material	nat-U metal
	b. Clad Material	Zr-2
	c. Fuel Enrichment	nat.
	d. Fuel Element Geometry	cylinders
	e. Cladding Thickness	0.030
	5. Material Inventories	
	a. Fuel, metric tons	27.2
	b. Uranium, metric tons	27.2
	c. U-235, initial-kg.	188
	d. Moderator, lbs. (incl. coolant)	5.5×10^5

6.	Reactor Control	
	a. Method of control	rods
	b. No. of Control Elements	26
D.	Plant Performance Data	
1.	Primary Coolant Outlet Temp., F	480
2.	Primary Coolant Inlet Temp., F.	414
3.	Reactor Temp. Drop., F.	66
4.	Primary System Operating Pressure, psia.	750
5.	Primary Coolant Flow Rate, lbs/hr.	39.6×10^6
6.	Avg. Core Heat Flux, Btu/hr. - ft^2	335,000
7.	Max. Core Heat Flux, Btu/hr. - ft^2	777,000
8.	Max. Cladding Surface Temp., F.	575
9.	Max. Fuel Temp., F.	880
10.	Core Coolant Velocity, ft/sec.	15
11.	Peak to Avg. Power Ratio	2.32
12.	Core Power Density kw/ft^3	506
13.	Core Specific Power kwt/metric ton-U	31,600
14.	Fuel Burn-up MWD/metric ton-U	3960

The power cost for a 300 MWE plant extrapolated from the above design is as follows:

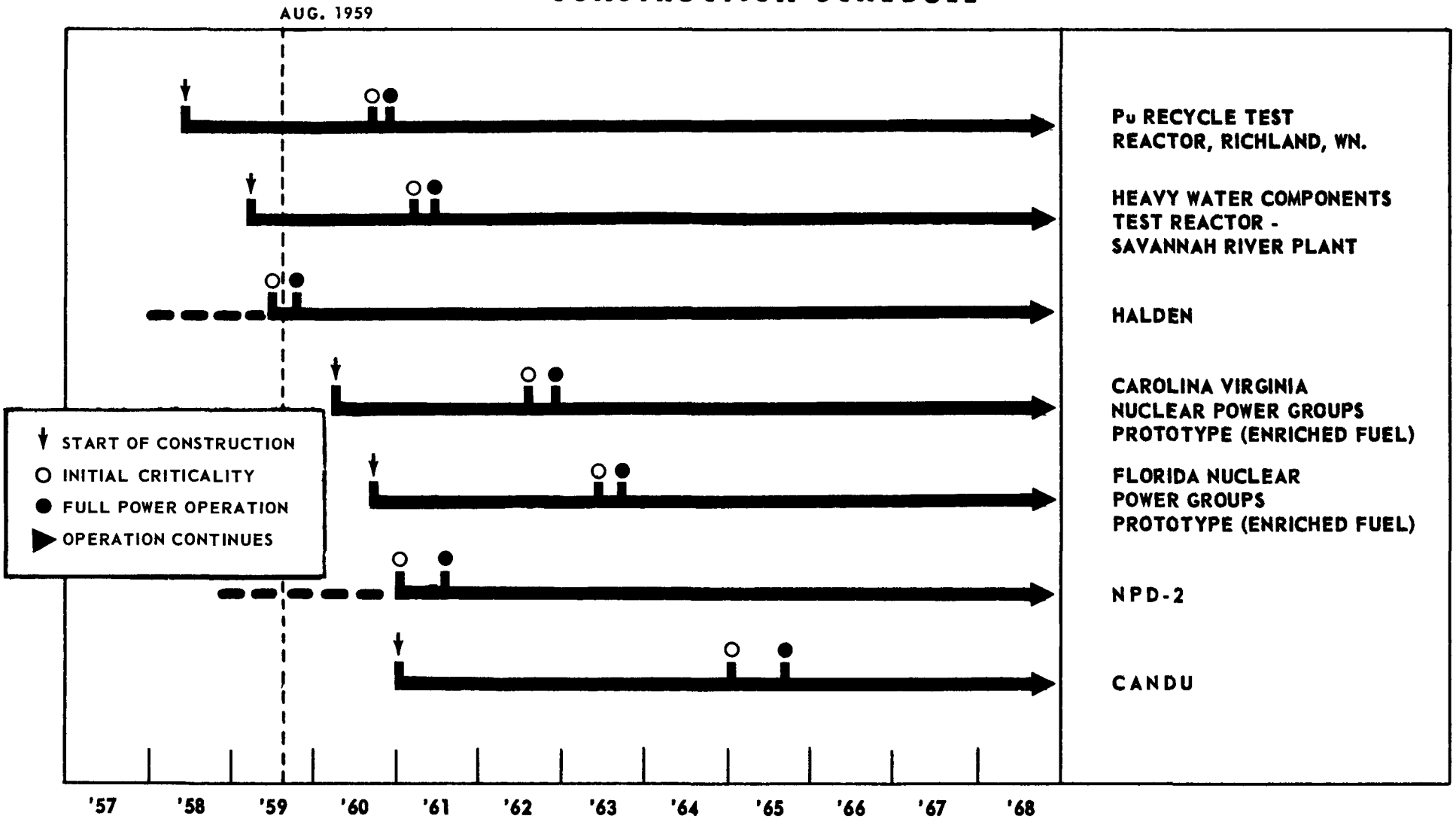
Total Capital Cost - $108,000,000 ----------	7.05	M/Kwh
Fuel Cycle Cost ----------	4.22	M/Kwh
Operation and Maintenance cost ----------	.91	M/Kwh
Nuclear Insurance cost ----------	.32	M/Kwh
Total power cost ----------	12.50	M/Kwh

The relationship of power cost vs. size is shown in Fig. 30.

The above costs are representative of plants that could be constructed with current status; however, they could only be achieved after the period of time required for the construction of a large plant. The low coolant pressure and the resulting low steam conditions, the pressure vessel type reactor construction, and the metallic fuel used in the reference design were selected to remain as consistent as practical with the definition of current status i.e., based on demonstrated technology.

The plant layouts and other design data for determining the economics are included in report S & L 1674 and Appendix II of this report.

HEAVY WATER MODERATED
NATURAL URANIUM / RECYCLE REACTORS
CONSTRUCTION SCHEDULE



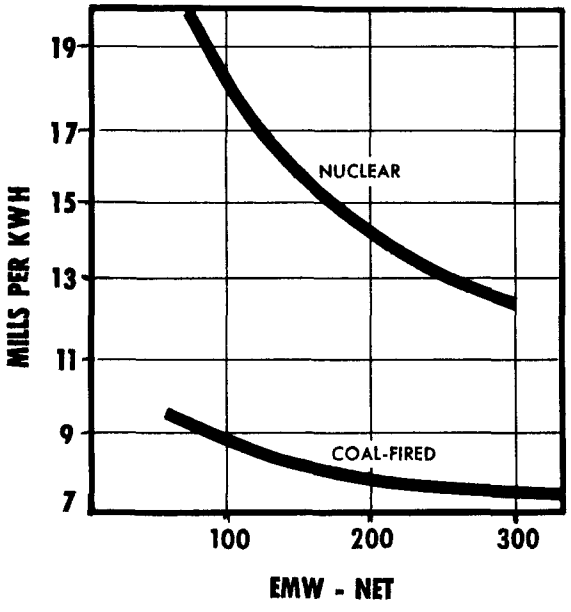
SEPT. 1, 1959
FIG. 29

POWER GENERATION COSTS

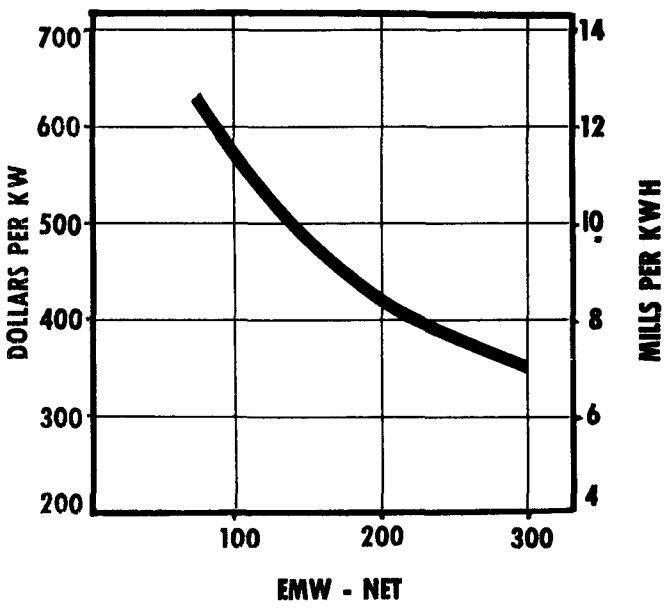
HEAVY WATER MODERATED NATURAL URANIUM REACTORS VS COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.

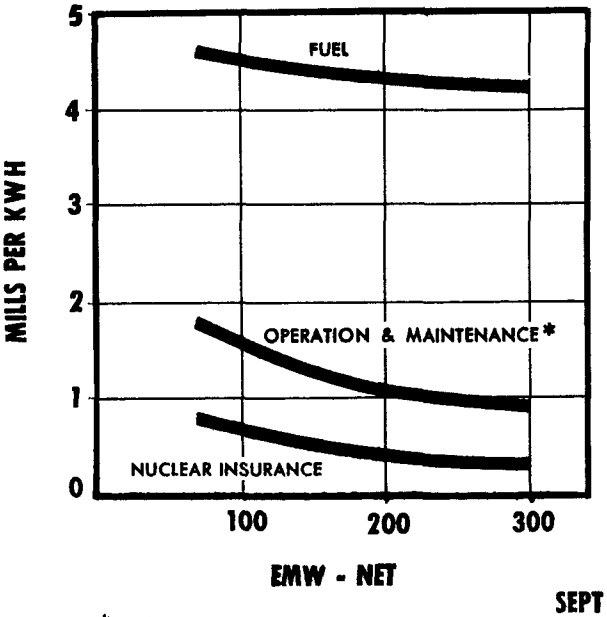
POWER GENERATION COSTS



CAPITAL COSTS



OPERATING COSTS



NOTES :

NUCLEAR -
BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/GM.
FUEL FABRICATION COST	
U-ZR - ZR-2	$50/KG.
FUEL EXPOSURE	3850 MWT/MT AVG.

COAL - FIRED -
BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%
FUEL COST	35¢/10^6 BTU

*Includes D_2O losses.

SEPT 1959
FIG. 30

2. GAS-COOLED REACTORS - NATURAL URANIUM

a. Description. Gas-cooled reactors employing natural uranium fuel may be graphite moderated or D_2O moderated. The technology of the graphite moderated version has been highly developed in the United Kingdom and in France. The British reactors employ natural U metal fuel elements clad with Magnox alloy, and CO_2 as coolant. The core is enclosed by a a pressure vessel, from which hot CO_2 is circulated to a second pressure vessel within which feed water heating, steam generation, and superheating are accomplished. The cooled CO_2 then returns to the reactor. A simplified flow diagram of the gas-cooled reactor plant is shown in Figure 20.

b. Technical Status

(1) Graphite Moderated. The most advanced technology for graphite moderated natural uranium fueled GCR's has been developed in the United Kingdom for the Calder Hall - CEGB series. There are no plants of this type under construction or planned in the United States, although designs are available. The technical status of this concept may be summarized as follows:

(a) Physics. Analytical and experimental physics, including critical experiments, were carried out in the U.K. development program sufficiently to establish the feasibility of operation. It is now possible to take adequate account of refinements required for core optimization. The reactivity lifetime is presently calculated to be 5500 MWD/MT. All economics to date have been premised on 3000 MWD/MT which has been demonstrated on a few rods and which is conservative

from the reactor physics point of view. The U.K. expects to achieve 3000 MWD/MT in the second cores of Berkeley and Bradwell, and in first core at Hinkley Point. Detailed lattice calculation procedures are well in hand to account for effects of burnup, changing neutron spectrum, growth of Pu and fission products up to 3000 MWD/MT and fine structure constants within fuel elements. There are very extensive experiments in the U.S., U.K., and France to normalize the calculational procedures. Moderator coefficients of such systems become positive after an exposure which is dependent on the moderator temperature. This positive coefficient represents an operating problem but not a hazard since it is a slow coefficient.

(b) Fuels and Materials. Unalloyed U metal is capable of exposures up to 3000 MWD/MT at temperatures below the alpha-beta transition in the U.K. reactors. Bowing of fuel elements, originally encountered in early Calder Hall operation, was eliminated by structural restraint. Problems of uranium swelling under thermal cycling and fission gas retention have not been completely eliminated, although fuel element failures in operation have been very low. Magnox as a cladding is well understood both with respect to production technology and performance. It is compatible with CO_2 up to 645°C. The newest magnox has the necessary ductility to accommodate the necessary U expansion which seems to result from thermal cycling. It should be noted that even the cast U which the U.K. produces does suffer growth under thermal cycling due to strains.

Thin-walled stainless steel which might be considered for such reactors is available on a custom basis in various countries, notably U.S., U.K., and France, in wall thickness down to 2.5 mils. It is not a commercial product, and its technology and practice are not well enough established at this time to make it a useable cladding. The French gas cooled reactors use magnesium. The temperature limitation on the system arises from the need to hold down the central U temperature, the grain growth and cavitation problems.

Graphite is well understood for service in such reactors.

(c) Heat Transfer and Fluid Flow. In order to maintain a gas temperature as high as possible, consistent with a specified limit on center line temperature, the temperature drop between the fuel and the gas must be minimized. Extended fuel surface areas (fins) are used on the fuel cladding for gas reactors. Temperature rises of 350-400°F are employed in the British reactors, based on mixed mean inlet and outlet coolant temperatures. Current experimental work is concerned only with refinement of details. Pressure drops have been optimized with respect to heat transfer by full-scale and double-scale channel tests. Such tests permit making detailed measurements to develop correlations between types of extended surface pressure drop and Reynolds number. Orificing, flow shaping via fuel element hangers and spacers, and varying the channel size as a function of radial position are all well understood experimentally.

(d) Coolant Chemistry. Either CO_2 or He can be used as the coolant in such a system. The major limitation of CO_2, i.e.

compatibility with graphite, is not a problem below 950°F. Since the system temperature is set considerablęy below this figure by the U metal alpha-beta transition temperature inside the fuel, the low cost, wide availability of CO_2 makes its use preferable to He for this application.

The purity specifications on the CO_2 are fairly stringent but can be met in practice as has been amply demonstrated. Additions of the order of one per cent CO to the CO_2 will minimize the burnout or oxidation of the graphite in the hot regions of the reactor. It would also minimize the effect of water impurities in CO_2 on the oxidation problem of both the graphite and the cladding materials. It has been found that this small addition of CO does not increase the deposition rate of carbon in the cold spots of the system.

(e) Components and Auxiliary Systems. There is such a large body of practical experience in these areas that all the necessary design and operational problems appear to have been solved. There are still many design choices and matters of taste which for a given reactor will influence the final cost, but the reactors can be made to operate satisfactorily on the basis of available technology. An exception may be that charging and servicing under load have not been completely demonstrated. There are now four such machines under construction, each differing from the others in detail. Each appears to represent a satisfactory solution to the problems. The first machine will be completely tested in late 1959.

An apparent limitation on the maximum size of these reactors due to the reactor vessel should be noted. Pressure vessel

materials currently in use in the U.K. are low alloy steel, mild steel, and boiler plate from 2 to 3-1/2 inches thick on diameters up to 70 feet. An upper limit on vessel thickness in the U.K. is imposed by the requirements of field welding, inspection, and stress relief. Maximum thickness of economically-sized sections which can be formed on existing industrial equipment is also a limiting factor. This limit, originally 2 inches, is now set at 3-1/2 inches. In France, gas cooled reactor containment vessels have been fabricated of pre-stressed concrete with a steel liner as well as steel pressure vessels. Since natural uranium fueled reactors appear to have reached the upper limit of power density for a given coolant pressure, the vessel fabrication limits also limit the generating capacity of a single reactor plant.

(f) Reactor Safety. Natural uranium fueled graphite moderated gas cooled reactors are characterized by low power densities and specific powers. The provision of adequate shutdown cooling and emergency coolant power supply in the event of normal power supply failure is adequate assurance against core meltdown. Stored (Wigner) graphite energy release is not a factor in these reactors due to the closed coolant system, the higher reactor operating temperature, and the extensive instrumentation employed. The Windscale incident therefore is not applicable to gas cooled power reactors. It has not been considered necessary to contain these systems in the United Kingdom.

(2) D_2O Moderated. The technical status of this concept is not sufficiently advanced that a plant could be constructed without pre-operating research and development. A sufficient amount of work has been done to justify the construction of a slightly enriched

experimental prototype plant of 50 EMW as a basis for a 325 EMW full scale natural U fuel plant. Technical status for this concept consists primarily of applicable graphite-moderated GCR technology, and development work completed or under way in support of the ECNG-FWCNG 50 EMW prototype by General Nuclear Engineering Corporation.

(a) Physics. The estimated performance of the Be-clad Zr pressure tube D_2O moderated system proposed by GNEC cannot be confirmed without additional development work.

Temperature coefficients of the heavy-water system are all negative.

Heavy water is a sufficiently better moderator than graphite from the neutron economy point of view that a possibility exists of designing a UO_2 fueled reactor for operation on natural uranium. The balance in favor of one or the other is presently dictated by the fiscal policy of charges levied by the AEC for D_2O on the one hand and enriched U on the other.

(b) Fuels and Materials. The behavior of Zr pressure tubes for period approaching 20 years in an irradiation field is unknown. Experiments now under way will give the required information. Present practice in the selection of a zirconium tube thickness is to make conservative allowance for the uncertainty in the long-term corrosion and irradiation effects on zirconium.

Beryllium metal appears to be an important material on which work in the last 18 months has provided new ideas and data. It is potentially a promising alternative but has not been completely

(f) Safety. Although higher specific powers can be obtained in these systems, the very strong negative coefficients provide an additional safety factor. There is less experience with these systems on which to evaluate safety aspects than in the graphite moderated concept, however they appear to be safe for the same reasons mentioned above.

c. Operating Experience. Operating experience has been obtained on graphite moderated natural uranium fuel gas cooled power reactors since May, 1956 in the United Kingdom. It should be noted that this experience is on the early designs (Calder Hall type) which were limited to lower specific power, fuel element surface and coolant temperature, and thermal efficiency than the current technology (CEGB) designs.

A significant amount of information on fuel element performance is available. Fuel element performance is limited by thermal stress, creep, and corrosion of the cladding to a greater extent in gas-cooled reactors than in lower temperature reactors using liquid coolants. The fuel elements are also exposed to more severe flux gradients. This, coupled with the lower heat transfer characteristics of the gas coolant, leads to asymmetrical temperature distributions in the fuel element and consequent distortion. Also, "hot spot" temperatures may be several hundred degrees higher than the bulk coolant temperatures. Proper design of the individual fuel elements and assemblies is the major factor in maximizing permissible exposure.

Operating experience on Calder Hall "A" Reactors 1 and 2, Calder Hall "B" Reactors 1 and 2, Chapel Cross "A" Reactor 1, and Marcoule Reactors G2 and G3, has shown a high degree of reliability. The only known fuel element ruptures have been easily managed in every case without excessive exposure or shutdown time. Fuel element failures

to date have not been due to causes usually attributed to the limitations of natural uranium fuel. The bowing problem noted under Fuels and Materials was caused by increased creep under irradiation. This has been remedied by long braces on the fuel element which minimize its unsupported length.

In the 80,000 fuel elements manufactured for the Calder Hall reactor as of July, 1958, approximately 40,000 had been operated with only 15-20 failures indicated during operation. Post operative examination of 40 additional elements indicated a failure rate of 1 to 4 in 5,000. Complete metallurgical examination of 12 elements showing abnormal radiation indicated "cold end" failures due to leak paths in the Magnox clad induced by creep strains resulting from uranium growth due to preferred orientation. Only 4 or 5 failures (1 in 10,000) were caused by manufacturing defects, which were primarily porosity or incipient weld cracks. There were no failures due to manufacturing defects in the first reactor loading at Calder Hall.

d. Plants Under Construction.

None in the United States. The following plants are under construction in the U.K.:

Station	No. of Reactors	No. of Turbogenerators	Net MWE
Berkeley	2	4	280
Bradwell	2	6	300
Hunterston	2	6	300
Hinkley Point	2	6	500
Trawsfynnydd	2	4	500

Berkeley and Bradwell, scheduled for completion in 1961, are in the most advanced construction stage. See Figure 29 for ECNG-FWCNG plant.

e. Economics. A reference design has been made for the graphite moderated-gas cooled reactor with natural Uranium fuel, based on the United Kingdom experience. This design was used as the basis for estimating the cost at different plant ratings. The base parameters for the reference design are as follows:

SUMMARY OF PLANT CHARACTERISTICS
GAS COOLED REACTOR PLANT

A.	Heat Balance		
	1. Total Reactor Power, MW(t)		830
	2. Gross Turbine Power, MW(e)		240
	3. Net Plant Power, MW(e)		200
	4. Net Plant Efficiency, %		24.1
		High-Pressure	Low-Pressure
B.	Turbine Cycle Conditions		
	1. Throttle Temperature, F.	650	450
	2. Throttle Pressure, psig	500	100
	3. Steam Flow, lbs/hr/	1.77×10^6	7.85×10^5
	4. Condenser Back-Pressure, in. HgA		1.5
	5. Final Feed Water Temperature, F		260
C.	Reactor Description		
	1. Reactor Vessel		
	a. Inside Diameter, ft.		70
	b. Overall Height, ft.		-
	c. Wall Thickness		3.0
	d. Material		c.s.
	e. Design Pressure, psia		500
	2. Reactor Core		
	a. Active Diameter, ft.		50
	b. Active Height, ft.		29
	c. Active Core Volume, ft.		57,000
	d. Lattice Arrangement		square
	e. Lattice Spacing, in.		8
	3. Reflector or Blanket		
	a. Material		Graphite

4. Fuel Elements

a. Fuel Material	U-metal
b. Clad Material	Magnox
c. Fuel Enrichment	nat.
d. Fuel Element Geometry	finned rods
e. Cladding Thickness	0.020

5. Material Inventories

a. Fuel, metric tons	274
b. Uranium, metric tons	274
c. U-235, initial-kg.	1970

6. Reactor Control

a. Method of Control	rods
b. No. of Control Elements	100

D. Plant Performance Data

1. Primary Coolant Outlet Temp., F	710
2. Primary Coolant Inlet Temp., F.	323
3. Reactor Temp. Drop., F.	387
4. Primary System Operating Pressure, psia	200
5. Primary Coolant Flow Rate, lbs/hr.	28.3×10^6
6. Avg. Core Heat Flux, Btu/hr - ft^2	56,000
7. Max. Core Heat Flux, Btu/hr. - ft^2	96,000
8. Max. Cladding Surface Temperature, F	730
9. Max. Fuel Temp., F.	1200
10. Core Coolant Velocity, ft/sec.	—
11. Peak to Avg. Power Ratio	1.72
12. Core Power Density kw/ft^3	14.6
13. Core Specific Power kwt/metric ton- U	3030
14. Fuel Burn-up MWD/metric ton - U	3000

The power cost for a 300 MWE plant is as follows:

Capital cost - $114,000,000	7.60	M/Kwh
Fuel cycle cost	3.35	M/Kwh
Operation and Maintenance	.61	M/Kwh
Nuclear Insurance	.33	M/Kwh
Total Power Cost	11.89	M/Kwh

The relationship of power cost vs. size is shown on Fig. 31.

The above costs are representative of current technology in the United Kingdom, converted to U. S. Economics. The fuel cycle costs are based on a package cost of 50 $/kg. of U as purchased from the United Kingdom. This price includes all fuel cycle cost with the

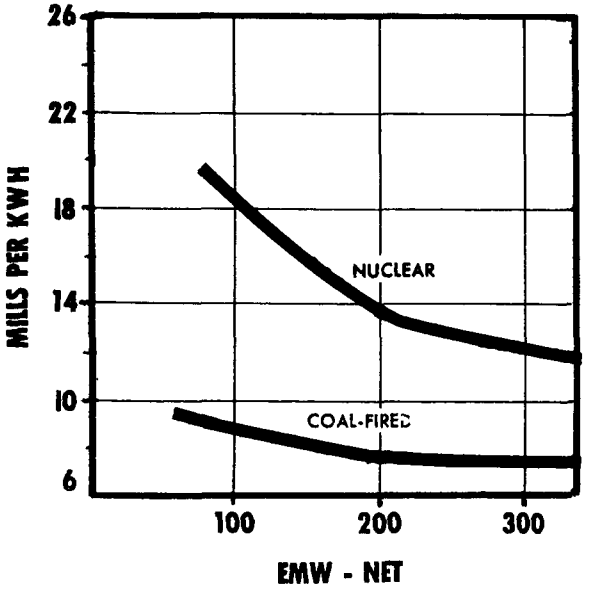
exception of shipping cost. The costs are quoted at 300 MWE to show the relationship with U. S. reactors in that size range. The economics of gas cooled reactors are more favorable in larger sizes. The technology is available for plants in the size range of 500 MWE. The plant layouts and other design data for determining the economics is included in report S & L 1674 and Appendix II of this report.

POWER GENERATION COSTS

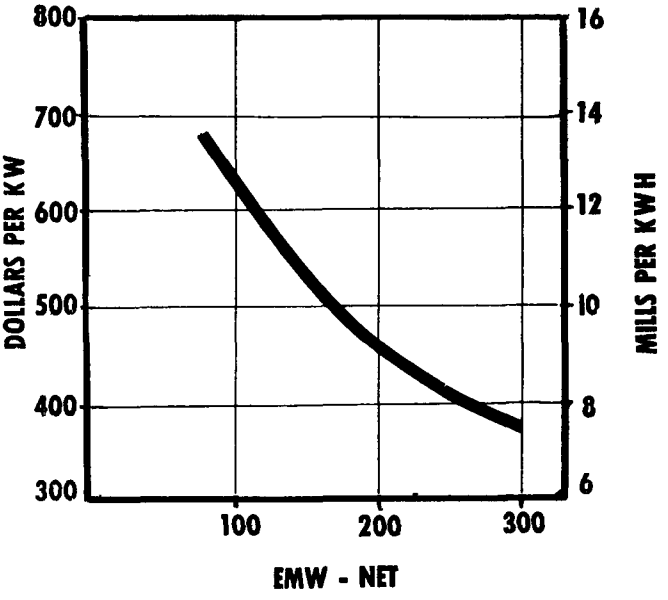
GAS COOLED REACTORS - NATURAL URANIUM FUELED VS COAL-FIRED PLANTS

SINGLE UNIT STATIONS - 1959 COSTS
RATING AT 1½" HG.

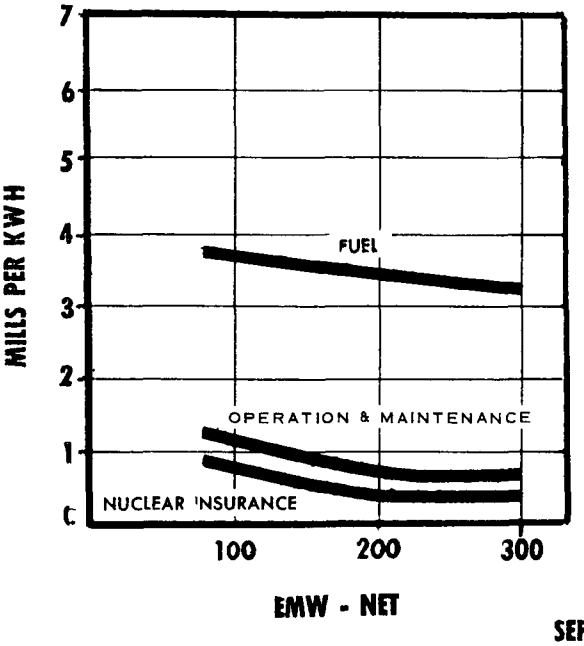
POWER GENERATION COSTS



CAPITAL COSTS



OPERATING COSTS



NOTES :

NUCLEAR -
BASED ON 1959 STATUS REPORT

LOAD FACTOR	80%
FIXED CHARGES	14%
URANIUM USE CHARGE	4%
PLUTONIUM CREDIT	$12/KG.
FUEL FABRICATION COST	
U - MAGNOX	$50/KG.
FUEL EXPOSURE	3000 MWT/MT AVG.

COAL - FIRED -
BASED ON SL - 1564 SUPP. 2

LOAD FACTOR	70%
FIXED CHARGES	14%
FUEL COST	35¢/10^6 BTU

SEPT 1959

FIG. 31

IV

Status
Civilian Power Reactors

D. Nuclear Technology

D. PLUTONIUM RECYCLE PROGRAM

1. Description. Plutonium fuel work is applicable to most of the thermal reactor concepts discussed in this report. Plutonium recycle work encompasses the following areas: (a) where plutonium is used as an alternate to U-235 as a means of fuel enrichment, especially in reactors or complexes of reactors for which the availability of U-235 is limited, and (b) where plutonium is used in conjunction with enriched or depleted uranium as reactor fuel. The Plutonium Recycle program was started late in 1956 with the objective of developing the technology necessary for the safe and economical recycling of plutonium. The primary emphasis is on fuel cycles for thermal, heterogeneous reactors, but with appropriate cognizance taken of the interaction between various thermal converter and breeder cycles. Current work is oriented toward accomplishing the following goals:

a. New and improved knowledge regarding the physics quantities necessary for the accurate prediction of performance and efficiency of plutonium bearing fuels.

b. Inexpensive, stable plutonium fuel elements.

c. Inexpensive uranium fuel elements that are compatible with plutonium enrichment.

d. Effective, inexpensive separations methods for recovery and recycle of plutonium.

e. Determination of optimum plutonium recycle applicability and economic feasibility.

Two major experimental facilities are under construction to facilitate development and demonstration of recycle technology. The Plutonium

Fabrication Pilot Plant (PFPP) is an experimental fuel fabrication facility, specially designed for working with plutonium. Completion is expected in 1960. The Plutonium Recycle Test Reactor (PRTR) is designed for irradiation of plutonium-bearing fuel and investigation of operating characteristics of recycle systems. The reactor is a heavy water moderated and cooled, pressure tube type rated at 70,000 kilowatts thermal power, to be placed in operation in 1960. The reactor, while operated at reasonably high temperatures, includes no power recovery facilities. It includes provisions for special pressurized loops for study of advance fuel concepts and is designed for accommodating changes in reflector material, reactor coolant, and periodic replacement of the entire reactor core structure.

A summary of the work completed to date in the existing facilities at Hanford is given below.

2. Fuel Cycle Analyses. Fuel cycle studies have been carried out to investigate the economics of recycling plutonium fuel in power reactors. The work involves correlation of reactor performance data with over-all fuel cycle costs to determine the value of plutonium in a variety of situations. An extensive series of computer programs, or codes, has been developed for the IBM-709 machine to enable rapid calculation using detailed physics parameters and economic factors. For recycle operation with natural uranium feed (self-sustaining recycle in a single reactor) a relationship was developed between fuel exposure level and the multiplication constant k_{oo} required to supply leakage, xenon and control losses. This information provides a quantitative measure of the value of plutonium in extending fuel lifetime.

Other analyses showed that plutonium enrichment may be more valuable when applied to reactor systems without the constraint of self-sustaining recycle. Thus, improved utilization of plutonium may be obtained in a combination of two reactors, one producing plutonium and feeding it to a separate reactor optimized for plutonium enrichment. This stems from the fact that for cases with low neutron leakage (geometrically large reactors) plutonium enrichment provides a higher fuel cycle burnup than does U-235. On the other hand, for small reactors, e.g., k_{oo} = 1.3, U-235 may be a more valuable fissionable material than plutonium.

A similar study of a reactor combination comprising a thermal reactor and a fast breeder indicated that it would be beneficial if the Pu-239 formed in the fast breeder blanket were to be supplied to a high-conversion-ratio thermal reactor. The mixture of plutonium isotopes resulting from irradiation of Pu-239 in the thermal reactor would be advantageous in fueling the core of the fast breeder. Shorter doubling times may result.

For some types of thermal reactors, the use of plutonium of selected isotopic composition has been shown to result in a very long lived fuel, due to the dual role of Pu-240 as a fertile material and a burnable poison.

3. Physics. In addition to the physics effort devoted to fuel cycle analysis and to PRTR design, basic physics work included the following items:

Low energy fission cross section measurements on Np-237 and Am-241 have been completed. High purity samples of Pu-240, U-234, and U-236,

and a sample of Pa-231 have been prepared for subthreshold fission cross section measurements. Fission cross sections have been measured in a sample of 96.6 per cent Pu-241 in the range of 0.1 ev to about 20 ev. A spectrometer for studying slow neutron differential scattering cross sections has been put into operation.

The Physical Constants Test Reactor has been used to measure lattice parameters of rod-clusters of UO_2 and of mixtures of plutonium-aluminum alloy (Pu-Al) and UO_2 rods. Calculations of epithermal multiplication of lattices containing only plutonium bearing fuel have been made as a function of the isotopic composition.

Nuclear safety studies have been conducted for the manufacture, storage, and transport of plutonium bearing fuel elements. Special plutonium bearing fuel elements have been fabricated preparatory to obtaining further critical-mass data.

4. Fuel Element Technology. Studies have been made on both ceramic and metallic core fuels. Improved and potentially low cost methods of core fabrication, core cladding, and fuel element assembly have been developed for a variety of fuel types. Efforts have concentrated on fuel elements of the rod cluster type as well as on three-component nested tubular designs.

Plutonium-aluminum (Pu-Al) alloys for metallic core fuel elements have been prepared by adding metallic plutonium to molten aluminum in pot furnaces. A cryolite process has been further developed in which Pu-Al alloy is made at lower cost by heating mixtures of PuO_2, Al, and cryolite.

Extrusion techniques have been developed to produce fuel element cores for use in rod cluster fuel element designs, and drawing and

swaging methods have been investigated for unbonded, Zircaloy-2 jacketing of these cores. Coextrusion methods have been developed for fabrication of aluminum alloy clad Pu-Al rods. Development is continuing on methods better suited for highly exposed plutonium, such as the direct casting of Pu-Al core materials into aluminum, stainless steel, or Zircaloy sheathing, using air or mechanical pressure injections. Initial work with aluminum as a stand-in for plutonium has produced satisfactory castings as long as eight feet with densities up to 96 per cent of theoretical.

Compositions of Al - 1.65 w/o Pu, Al-12 w/o Si - 1.65 w/o Pu, and Al - 8 w/o Pu clad in Zircaloy-2 have been irradiation tested to high exposure, with satisfactory performance demonstrated.

Basic studies on PuO_2 and PuO_2-UO_2 systems have included determination of sintering characteristics, melting points, and conditions of solid solution formation. Small-scale irradiations of Zr-2 clad PuO_2-UO_2 have been conducted without incident.

Extensive use is being made of piloting future work on plutonium-uranium ceramic fuels by studies of UO_2 systems. UO_2 powders for ceramic core fuel elements have been studied to determine the effects of powder pretreatment prior to core fabrication. It was found that the sinterability of UO_2 powder can be greatly enhanced by ball milling and air elutriation of the powders or by the use of additives such as TiO_2 or Nb_2O_5. Studies are in progress to determine the thermal conductivities of irradiated and unirradiated UO_2. Preliminary studies have already been made of compounds such as UC and UN for core materials.

Cold swage fabrication of UO_2 fuel rods has been under intensive development, mainly with Zr-2 cladding but to a lesser extent with stainless steel and aluminum cladding as well. Quality of Zr-2 tubing, UO_2 powder selection, and swaging techniques were defined as important variables. More than 30 irradiation tests of swaged specimens have been conducted without failure. Hot swaging processes have been developed. Preliminary results are promising.

Vibration compaction studies on graded UO_2 powders have shown that densities of 85-90% of the theoretical are attainable. This technique is being exploited directly for core fabrication and also as a simplifying, preliminary step in swage fabrication. It is especially attractive for eventual application to plutonium bearing fuels where a minimum of direct handling is desired.

In studies of the high temperature behavior of UO_2 systems, cylindrical specimens of unsintered UO_2 powder packed to a density of only 35 to 40 per cent of theoretical have been irradiated. During irradiation the UO_2 powder sintered and relocated to give a density of about 80 per cent of the theoretical and a granular structure suggestive of recrystallization from a melt. However, extensive ex-reactor experiments revealed that melting did not occur.

Several improved welding techniques have been developed for making high integrity closures on Zircaloy-2 cladding. One, of unusual interest, involves use of an electron beam heating technique.

5. Materials Development. The successful vendor fabrication of 95 Zircaloy pressure tubes culminated development of extrusion and tube reduction techniques over the past several years. Each tube

passed a detailed examination for dimensional tolerances and surface imperfections, plus being pressure tested. Approximately 30,000 feet of Zircaloy-2 fuel jacket tubing has been delivered and inspected. Specifications and test procedures have been developed to insure satisfactory performance under reactor operating conditions. A welding process for attaching spacer ribs to thin wall Zircaloy tubing has been developed and demonstrated to be feasible.

Techniques for etching and autoclaving Zircaloy process tubes and fuel jackets have been developed to determine corrosion resistance and establish fuel element integrity. The process found best thus far consists of etching in nitric-hydrofluoric acid solution, followed by an aluminum nitrate bath, water rinsing, and exposure to steam at 400°C and 115 psi.

Studies have been made to establish the conditions under which hydriding of Zircaloy occurs, and its effects on the properties of the metal. It was shown that hydrogen pickup by Zircaloy can occur in the corrosion process. The ductility of Zircaloy samples with known hydrogen content has been measured at low strain rates.

Corrosion testing of aluminum progressed satisfactorily, with emphasis on those alloys containing small amounts of nickel, iron, and titanium. Autoclave corrosion tests of Pu-Al alloy indicated good corrosion resistance at 350°C for the as-cast material; worked alloys were considerably less resistant.

6. Chemical Processing. Research and development studies have been conducted on several chemical processing schemes for recovery and recycling of plutonium and uranium from spent fuel.

The applicability of the Zirflex process (ammonium fluoride-ammonium nitrate) for dissolution of zircaloy fuel cladding has been confirmed. Pertinent process variables have been defined in pilot plant studies under tentative flow sheet conditions. Available radiometallurgy and chemical processing facilities have been shown to be capable of processing PRTR fuel by the Zirflex process at Hanford.

An anion exchange process has been developed for recovery of high purity plutonium from aqueous nitrate solutions of low enrichment or natural uranium. Two cycles of an anion exchange, incorporating fluoride washing, will give complete decontamination of plutonium from fission products. Permutit SK resin has been found to possess an adequate combination of radiation stability and exchange capacity for this application.

Studies on advanced pyrochemical processing methods have shown that a basis exists for separation of uranium and plutonium by chlorine or phosgene dissolution of mixed oxides in molten alkali chloride. Uranium oxide dissolves at a faster rate than does plutonium oxide. In addition, it was discovered that zinc reduces uranyl chloride in salt solution but apparently not plutonyl chloride.

The distribution of uranium and plutonium between metal and salt phases in the aluminum-aluminum chloride-potassium chloride system has been investigated. It was shown that a separations factor of at least 100 may be obtained per equilibrium stage. The behavior of other actinide elements in this metal-salt system has been studied and distribution coefficients have been measured at 725°C.

IV. STATUS OF CIVILIAN POWER REACTORS

E. GENERAL ENGINEERING R & D (NUCLEAR TECHNOLOGY)

The information in this section is a summary of the significant results of the engineering and development programs to date supporting the civilian power reactor program. This material does not include the results of the support work for the military programs. The subjects covered in this Section are:

1. Fuels
2. Cladding Materials
3. Fuel Fabrication
4. Irradiation Testing
5. Fuel Reprocessing
6. Components
7. Environmental Investigation and Effluent Control
8. Physics
9. Safety

1. FUELS

There are two basic types of fuel under development for civilian power reactors. These are metallic uranium and ceramic fuel materials. The metallic fuel is subdivided into two groups, i.e., unalloyed uranium and uranium containing alloys to improve the material properties. The ceramic fuel is also divided into two groups, i.e., oxides and non-oxides.

A summary of the current status of the above fuels and materials is given in the following paragraphs:

a. Metallic Fuel. The major limitation in the use of metallic fuels for civilian power reactors is irradiation induced swelling at the temperatures desired for central station plants. This swelling is not unique to any one fissionable or fertile metal; it occurs in uranium, thorium, plutonium, and their alloys. Until more is known of the swelling phenomenon, no theoretical predictions are possible concerning limiting burnups and operating temperatures.

Metallic fuel is highly corrosive in water and it is necessary to utilize large amounts of alloying materials in water reactors to achieve even moderate corrosion resistance. This affects the nuclear properties of the fuel material. Protective cladding is used to prevent corrosion of the metallic fuel in water reactors.

Because of their high uranium density, metal fuels are more desirable for fast reactors than ceramic fuels. However, present metallic fuels do not appear to have adequate resistance to high temperature swelling at desired burnups.

Metal fuels which will perform satisfactorily in gas-cooled reactors are not now available. Such materials would have desirable characteristics inherent in metals, but it appears unlikely that they would be as satisfactory as ceramic fuels of slightly enriched uranium.

Fission gas accumulation causes excessive swelling of uranium at temperatures over 750-930°F (400-500°C). This severe temperature limitation, coupled with the poor heat transfer characteristics of the gas coolant, is so limiting that all present long range plans for gas-cooled reactors using known technology contemplate the use of ceramic fuels.

Basic studies on irradiation induced swelling of fissionable metals and alloys are currently underway. Until more is known of the swelling phenomenon, no predictions are possible concerning fuel life.

Many binary and ternary uranium alloys have been investigated to determine if a nominal addition of some metal would markedly improve the properties thus permitting the alloy to be used under conditions much more severe than feasible with unalloyed uranium. Typical additives examined include silicon, chronium, molybdenum, zirconium, and niobium. These alloys are referred to as alpha uranium alloys.

Primary reasons for additives are: (1) to refine the grain size to minimize radiation induced surface bumping, (2) to make the uranium more amenable to heat treatment for removal of preferred

orientation, (3) to improve aqueous corrosion resistance, (4) to minimize growth and swelling during irradiation, and (5) to improve the elevated temperature mechanical properties, while not sacrificing too many neutrons due to poisoning and dilution.

None of the alpha uranium alloys investigated to date appears to sufficiently improve the corrosion properties so that it could be considered a logical candidate as a fuel in water-cooled power reactors.

Alloys receiving considerable attention are the U-2 w/o Zr and U-1 to 5 w/o Mo in the cast, wrought, and powder compacted conditions. Irradiation data on their resistance to swelling is contradictory, ranging from results indicating a great deal of resistance to other cases where the alloys were no better than unalloyed uranium. It is apparent that the irradiation behavior of these alloys is very dependent upon fabrication and heat treatment.

It is conceivable that one or more of the alloys currently being investigated could operate satisfactorily in an organic-cooled reactor, providing the resistance to swelling is sufficiently improved. Use of these alloys in water-cooled, sodium-cooled, or gas-cooled reactors appears less attractive, assuming high operating temperatures and substantial burnups.

The stable and metastable uranium classes of alloys were developed in an attempt to eliminate radiation induced dimensional instability due to preferred orientation, minimize radiation induced swelling, improve the corrosion resistance in aqueous or other media, and to utilize fuels of lower enrichment than have been used in military reactor cores. Under radiation, the metastable alloys convert slowly to the stable class.

The U-10 w/o Mo and some of the uranium-fissium alloys are among the materials of this group still receiving serious consideration as civilian power reactor fuels. Radiation enhanced changes in the metastable states, swelling, loss of corrosion resistance, difficulties in fabrication, and excessive dilution accompanied by loss of neutrons have been principal causes for discontinuing work on the other alloys of this group.

The U-10 w/o Mo alloy appears satisfactory as a fuel if an approximate upper operational limit of 1110°F (600°C) and 0.7 a/o burnup is established. The temperature limit is particularly important since there is evidence that cycling above this temperature results in gross swelling. There is uncertainty about the swelling behavior of this alloy arising largely because of insufficient data. The poisoning effect of Mo requires increased uranium enrichment. Metal fuels need peak temperatures greater than 1250°F (620°C) to reduce fuel costs. For sodium-cooled systems at temperatures of about 1200°F (650°C) there would be a compatibility problem with the steel cladding since the iron uranium eutectic forms at approximately 1350°F (735°C).

b. Ceramic Fuels. The major advantages of oxide fuels in water systems are (1) the potential for reaching very high burnups without excessive volume changes, (2) resistance to corrosion in a water reactor environment, (3) low parasitic neutron absorption cross section, and (4) adaptability to fabrication by relatively simple techniques; however, the oxide fuel is not as adaptable to simple fabrication as metal fuel.

The technology of UO_2 is fairly well known, and mixtures of UO_2 and a second oxide have been investigated with the view of improving one

or more of the properties. In contrast, many of the properties of PuO_2 are unknown, and its potential as a fuel material has not been established.

Oxide fuel materials can be used safely in reactor environments where corrosion by water would limit the use of other fuel materials. The high melting points of the oxide fuels permit use at high temperatures. Uranium dioxide, as an example, has a melting point of about 5000°F (2700°C), has no harmful phase transformations, and is chemically stable to most reactor environments.

Irradiation studies on UO_2 fuel samples have been made to burnups exceeding 50,000 MWD/T. Experimental studies of irradiation behavior of oxides to various burnups indicate that volume changes are negligible during irradiation. Thus, compared to metal fuels, oxides show an outstanding ability to maintain dimensional stability under extended exposure at high temperatures. Defected UO_2 and ThO_2-UO_2 mixed crystal fuel elements irradiated to heat fluxes of several hundred thousand BTU/hr-ft^2 in pressurized water were found to release solid fission products at rates far below similarly defected metal fuel elements.

UO_2 is a competent fuel material for water-cooled reactors and probably will not be superseded in the foreseeable future by another fuel. It is also a material on which a great deal of information has already been obtained.

One general feature of ceramic fuels is extreme brittleness, and low thermal conductivity. Both factors result in susceptibility to thermal stress failure when used as reactor fuel elements. For this reason, ceramic fuels except BeO and graphite semi-homogeneous elements require structural support from the cladding. The low thermal conductivity also tends to limit the use of these fuels to reactors which do not require

a fuel having a high specific power.

Although the technology of UO_2 is fairly well known, information is still lacking concerning the termal conductivity of the material under irradiation and the burnup limits to which UO_2 can be irradiated. The oxide fuels, having low thermal conductivity, operate at high central fuel temperatures, and central melting or void formation is thought to be the limiting factor. Also lacking are engineering data concerning the high temperature mechanical properties such as ductility, temperature limits due to volatility and center melting.

There is some uncertainty with the oxide fuel concerning the power output capability of the clad fuel element. In a few experiments in which very high heat fluxes were obtained, central melting of the fuel definitely occurred and were thought to be the cause of failure; in other cases, a zone of coarse-grained UO_2 was formed around a central void. This has been interpreted by some, but not all, experimenters as indicative of melting, and as representing a condition which must be avoided. This opinion is not universally held, and it is possible that the central temperature could be increased beyond present limits.

The burnup limits of oxide fuel in metal cladding are not yet defined. Although oxide fuel is known to release fission gas at greater rates than metal fuels, quantitative data on the release of gaseous fission products from UO_2 as a function of density, burnup, temperature, and time are meager and sometimes contradictory. Statistical data as to the buildup of fission gas pressure, the frequency of clad failures,

and the amount of activity released to the coolant is not available for long irradiations of full scale fuel elements under reactor operating conditions.

Non-oxide ceramic fuels are of interest in order to improve on some of the properties inherent to UO_2. These materials are in the early stages of scientific investigation. In many cases physical properties are unknown, and only limited laboratory quantities have been synthesized. The general characteristics being sought include high uranium density, high melting point, low neutron cross section, high thermal conductivity, radiation stability and fission gas retention at high temperature and long burnup. The largest current effort is on uranium monocarbide.

The carbides, nitrides, sulphides, phosphides, and some of the aluminides have a higher uranium atom density than UO_2. These materials also have high melting points, are free from harmful phase transformations, and have thermal conductivities intermediate between uranium metal and the dioxide. Uranium monocarbide is the only non-oxide ceramic fuel materials on which any irradiation data of consequence has been obtained. These data, although limited, indicate that at moderate irradiation temperatures uranium monocarbide decreases in density at a rate somewhat less than that observed in uranium metal fuels, and at elevated temperatures it appears to be superior to metal fuels in resisting swelling. The volume change under irradiation is not exaggerated at surface temperatures up to 1200°F (650°C); however, the quantitative extent of this volume change has not been determined, nor have there been enough tests, or tests at high burnup, to have proved the usefulness of UC as a fuel material.

The monocarbide of uranium has a melting point of 4000°F (2200°C) is chemically stable, and is compatible with some reactor coolants such as sodium. Its thermal conductivity is close to that of uranium metal. This would make it applicable as a fuel for high temperature reactors with high specific power. UC has been found to be pyrophoric in powder form, and will react with water. This factor could make the material unsuitable for use in water reactors.

The work in the field of non-oxide ceramic fuel materials, with the exception of UC, is primarily concerned with materials preparation and the evaluation of pertinent physical and mechanical properties. As this work is in an initial laboratory stage, there is no current application of these materials for fuels.

Uranium monocarbide appears to have properties that make it well suited for use in sodium-cooled reactors. Little work has been done on UC up to the present time; the principal data were developed in FY 1959. The development work included determination of the thermal conductivity, fabrication by arc melting and casting, corrosion in Na and NaK, and irradiation damage. Two irradiations have been made with burnups of 1000 MWD/T and 6000 MWD/T. The preliminary results were excellent.

2. CLADDING MATERIALS

Materials considered for fuel element cladding include aluminum, magnesium, zirconium alloys, stainless steels, niobium and some of its alloys, special materials such as tantalum, yttrium, and beryllium, and ceramic cladding.

a. Aluminum. Aluminum alloys have been extensively used in low temperature water cooled production reactors. Substantial development work has been completed to produce an aluminum alloy capable of operating in water in the range of 600-650°F (315-345°C). Out-of-pile tests in water, where the pH is held to 3 to 4 by addition of phosphoric acid, have given very good results in the temperature range of interest. However, in-pile data at the above temperatures with alloys, such as X-8001 (2S aluminum plus 1 per cent nickel) are of limited value due to insufficient data.

Limited work indicates that aluminum alloys can be used as cladding in organic cooled reactors at temperatures to 800°F (430°C), providing that strength is not a major factor. The aluminum powder metallurgy products containing Al_2O_3 have been cursorily investigated. Substantial work on end closures, the effect of welding on metal properties as well as corrosion characteristics during irradiation, must be completed before their worth can be assessed.

b. Magnesium

Magnesium alloys such as Magnox (magnesium plus 0.1 per cent beryllium) have been used in British gas cooled reactors. All known magnesium alloys have unsatisfactory high temperature properties and since the U.S. is not currently interested in relatively low temperature gas cooled reactors, there is no incentive for magnesium alloy development.

c. Zirconium. Zirconium and zirconium alloys, such as Zircaloy-2 (zirconium plus 1.5 per cent tin, 0.1 per cent iron, 0.1 per cent nickel, 0.05 per cent chromium) and Zircaloy-4 (Zircaloy-2 less nickel), are being used in water and sodium cooled reactors. Principal drawbacks of these materials are limited high temperature mechanical properties, inadequate corrosion resistance in high temperature steam above 680°F (360°C), high cost of fabricated shapes,

d. Stainless Steel. The stainless steels suffer from the major disadvantage of high thermal neutron absorption cross section. This necessitates additional enrichment or the use of very thin clads. Nevertheless, these materials are being used in water cooled, gas cooled, sodium cooled, and organic cooled reactors because of their strength, fabricability, weldability and reasonable cost. Potential problems include: (1) progressive deterioration of mechanical properties at extended neutron exposures; (2) intergranular oxidation in gases at high temperatures; (3) "green rot" in reducing carbon monoxide atmospheres; (4) possible mass transfer; (5) formation of reaction layers or a low melting uranium-iron eutectic, in case of localized overheating with metal fuels; and (6) compatibility with UC at elevated temperatures has not been well established.

e. Iron-Aluminum. The more ductile iron-aluminum alloys should be considered as cladding materials because of superior oxidation resistance, seemingly good aqueous corrosion resistance, and a slightly lower neutron absorption cross section than the stainless steels. The

iron-aluminum alloys should be possible replacements for stainless steel in water, organic, or sodium cooled reactors and in gas cooled reactors providing the temperatures do not exceed 1000-1200°F (540-650°C). Due to the limited data available, operational limits and applications cannot be well established. Little is known in this country regarding fabrication, irradiation behavior, and long time corrosion resistance.

3. FUEL FABRICATION

The experience that has been obtained on fuel fabrication technique for different materials is summarized below.

a. Unalloyed Uranium. The extensive use of unalloyed uranium in production reactors led to the development of the technology of uranium fabrication which can be compared to that for more common materials.

Most methods of fabricating uranium metal involve casting of ingots or large dingots, and working to final sizes by rolling, forging, or extrusion.

b. Alloys with a Uranium, Thorium, or Plutonium Base. Extensive fabrication experience is not available for most fuel alloys. An exception is the Zr-low U alloys used in a special application where cost is not as important a factor as for civilian power reactors.

Alloys most likely to be used in power reactors are high strength, high temperature materials such as U-Mo and U-Nb alloys. The relatively high strength of these alloys at elevated temperatures presents some problems in fabrication.

Thorium alloys have not been used extensively with the exception of the Th-U alloy used in the Sodium Reactor Experiment.

Plutonium alloys for civilian power reactors hold considerable promise. The first reactor fuel fabrication experience in this field is presently being acquired by HAPO. A U-1.8 w/o Pu alloy is being fabricated by casting, followed by extrusion. The problems connected with plutonium alloys are those of criticality and of toxicity, which require glove box operation.

There is considerable discussion about suitable types of fabrication methods for remote operation. Little question exists as to the

eventual importance of remote fabrication, but the time at which remote fabrication will become part of the fuel cycle is unresolved. Problems in remote fabrication of recycled fuels are considerably more complex than those associated with plutonium.

c. Ceramic Fuel Materials - UO_2, PuO_2, and ThO_2. Uranium dioxide is used more than any other ceramic material as fuel in civilian power reactors. Present costs of fabricating UO_2 fuels are higher than metallic fuels. Only the fabrication of small pellets by cold pressing and sintering has approached the production stage.

The production of UO_2 powders presents several problems such as:

(1) The influence of the powder production technique on the further fabrication steps such as pressing, sintering, and extruding of the finished fuel. Because of this problem, pronounced variations have occurred between different batches of powders made by the same process.

(2) Material enriched in U-235 presents special problems because of criticality. With enrichments up to $3\frac{1}{2}\%$ this problem is not too limiting and such material can be processed continuously. Material with an enrichment of $3\frac{1}{2}\%$ to 5% falls in an intermediate class with considerable increase in cost. Enrichments over 5% produce severe criticality problems and a large increase in processing costs.

(3) The fabrication of fuel element shapes from the UO_2 powders is presently quite expensive. The only large scale experience has been obtained in the production of small pellets, up to about 1/2" in diameter having a length to diameter ratio of slightly more than 2. These are made by a cold pressing operation followed by high temperature firing. The small diameter of these pellets is dictated by reactor considerations. The effect of diameter of pellets on cost has not been

well established. Larger pellets represent smaller numbers per given weight but need longer sintering times. The short length, which contributes to a large degree to the high cost, is a fabrication limitation peculiar to the cold pressing method. The sintered pellets have some variation in length and diameter.

(4) Other fabrication processes for UO_2 not having some of the limitations of the pellet process have been suggested. Of these only swaging (which will be discussed later) has received serious experimental efforts.

Work on fabrication of PuO_2 is very limited because facilities are only now becoming available.

d. Uranium Carbide. The fabrication of massive UC is presently the major problem facing eventual users. Uranium monocarbide can be produced by reaction of carbon with uranium metal, UO_2, UF_6, or UF_4 but problems exist in producing an exact stoichiometric product. Uranium monocarbide is pyrophoric in the powder form. However, this pyrophoricity diminishes with fabrication. Compacts formed from powdered material are usually in the range of 65 to 75% with possibly as high as 85 to 90% of theoretical density.

Melting and casting of UC appear to yield an acceptably dense product which behaves well under irradiation. The casting method presently used consists first of arc melting and remelting small buttons. Several buttons are then remelted together and bottom poured into a mold. At present the largest melts are 3/4" in diameter and 6" long. No casting methods are presently known which offer promise of rapid development into low cost, large scale production methods.

e. Bonding. Critical aspects of any clad are its method of attachment to the fuel core, the nature of the bond formed, and its behavior during irradiation. A partially bonded fuel element may be worse than one totally unbonded, due to localized overheating.

All present UO_2 fuels are unbonded to the cladding. Such fuels are used in the Pressurized Water Reactor blanket, the Con Edison, Dresden, and the Advanced Boiling Water Reactors.

The present technique for assembly consists of inserting UO_2 pellets into tubes of Zircalloy-2, stainless steel, or aluminum and welding on end caps. Because of the requirement for high integrity, the end cap welding has presented some difficulties.

The EBR-I, EBR-II, Fermi Blanket, SRE, and Hallam are examples of fuel elements having a liquid metal bond.

The main problem remaining is that of reproducible wetting of the fuel and cladding by the liquid metal where small clearances exist between fuel and cladding. Because of the higher temperatures needed for fabrication and inspection, sodium technology is more difficult than NaK technology. The main advantage of liquid bonding is that close dimensional tolerances for components are not necessary. Excess volume is also available for the swelling of metal fuel. Assembly and disassembly of fuels can be done easily. The first remotely operated fuel fabrication facility (EBR-II) will combine a simple melting step with liquid metal bonding.

Presently used fuels incorporating a diffusion bond and mechanical deformation include all roll bonded plate elements such as EBWR, PWR seed, and all extrusion clad rods and tubes such as Fermi and EBR-I.

Cast, wrought, or powder metallurgy cores are used. Cladding components are tubes, rods, or plates which can normally be obtained commercially. In some cases the fuel is cast directly into the cladding components.

The fabrication of diffusion bonded elements made by mechanical working consists of the assembly of the extrusion billet or roll pack followed by deformation at elevated temperature.

There are relatively limited irradiation information on the behavior of such bond layers at U-Ni-Al, U-Zr, and U-Zr-Nb. Available data are insufficient to determine the limits of such bonds at high temperatures and burnups.

Fuel fabrication by diffusion bonding and mechanical deformation has the advantage of relatively low cost and produces a high quality bond.

4. IRRADIATION TESTING

New and untried reactor fuel materials must be tested under conditions of radiation and temperature which simulate those it would experience as an operating fuel element. Other characteristics such as corrosion resistance, mechanical strength, and good thermal conductivity are of considerable importance, but the ability of the proposed fuel material to resist undesirable irradiation-induced changes will almost invariably form the final basis for its acceptance or rejection as a possible reactor fuel.

Evaluation of the irradiation characteristics of a proposed fuel is one of the most expensive materials tests in existence. For example, a relatively simple irradiation test involving the use of an uninstrumented capsule containing three or four fuel specimens may cost several thousand dollars. If heaters and thermocouples are added to the capsule for temperature control, the cost of the test may run from $20,000 to $40,000. For typical capsule tests the total time required to procure the experimental material, fabricate the specimens, perform the irradiation, and complete the post-irradiation examination may range from 8 months to 2 years. More complex experiments requiring loops may result in expenditures ranging up to $1,000,000 or more. The time required to design and fabricate the loop, procure the experimental fuel materials, and obtain the required experimental information, may take 2 years or longer.

The two most important parameters which can be controlled during irradiation are fuel temperature and burnup.

The present unsatisfactory status of the art of fuel burnup determination is a source of serious concern to groups engaged in

experimental fuel irradiations. Much of the significance in the results from such experiments is lost if uncertainties exist regarding the fuel burnup.

a. Loops. Fuel element testing loops containing the coolant of interest have been heavily relied upon to furnish information on the irradiation behavior of fuel element-coolant combinations. These tests can provide conditions closely simulating those to be experienced in reactor operation. Among the most important information supplied in a well-engineered loop test is the following: (1) irradiation behavior of a prototype or full-scale fuel element interest at the reference operating temperature, burnup level, and heat flux; (2) compatibility of the fuel cladding and the coolant at reference temperatures, flow rates and heat fluxes; and (3) behavior of the coolant in the expected radiation, pressure, and temperature environment.

b. In-Pile Tests. Capsules provide a relatively simple means for altering the test fuel environment from that of the reactor whose neutrons are utilized. Although somewhat restricted in application by lack of forced convection cooling, these devices have proven quite useful. Capsule tests, as well as loops, suffer in utility from lack of high temperature sensing devices and suitable structural materials applicable for ceramic and cermet fuels.

The main disadvantage associated with in-pile irradiation testing is that the test element is usually not cooled separately from the reactor core.

c. Test Reactor Facilities. The majority of fuel irradiation tests require thermal neutrons. Most tests in the U.S. at the present time are being conducted in the MTR or ETR, although several other

smaller test reactors are also being used. In addition, privately-financed test reactors are becoming available.

Almost all locations in MTR and ETR that are available for capsule irradiations are designed to accommodate capsules with an outside diameter of 1.125 in. with lengths up to approximately 30 in. This diameter is adequate for uninstrumented capsule work. However, it severely taxes the ingenuity of the experimenter wishing to instrument capsules, for example, to measure and control sample temperatures during irradiation. Lead wires from instrumented capsules require valuable space in each reactor that could otherwise be used by uninstrumented capsules.

ETR is designed to accommodate through-type loops. The design of MTR on the other hand is such that most of the loops in the reactor are restricted to beam hole locations, and hence must be of the bayonet type. In general, because of the more uniform flux field, most experimenters prefer locations which accommodate through loops. Demand for these locations exceeds the supply.

A major shortcoming of the available test reactors in the U.S. is the lack of experimental space in the 10^{14} range of thermal flux in which a flat flux field of several feet is available. The steep flux gradients in MTR and ETR are a severe handicap to the experimenter who wishes to irradiate a number of samples to identical burnups.

Essentially no facilities exist for irradiation of experimental fuel materials in a fast flux. Although EBR-I is being used for such work to a limited extent, it can be accomplished only be replacing standard fuel elements with experimental assemblies of identical shape and dimensions. Also, the flux levels are low and the core size is

small. At best, irradiations of fast reactor subassemblies or plutonium-rich specimens in thermal fluxes are make-shift experiments and frequently are of doubtful significance.

d. Hot Cells. The lack of adequate hot cell facilities continues today to be the rate determining step, in in-pile experimentation. Test samples can be produced at a much higher rate than they can be examined in the limited hot cell facilities available. More hot cells equipped to do ordinary analytical laboratory work are required.

e. Advanced Testing and Research Reactors. A program is under way on research and test reactor development, up to the point of critical experiments, if needed to establish concept feasibility. Further work on particular concepts is being determined on a specific project basis.

Continuing plans of the AEC for high flux research and testing reactors have required a well-integrated and rather specific R&D program on reactors of this type. A recent ANL study estimates that reactors of the MTR type can be built to achieve fluxes of 5×10^{15} neutrons/cm^2-sec for less than $15,000,000. ANL studies, including critical experiments, are continuing to determine the advantage of the flux trap reactor concept for reactors in the 10^{16} flux range.

The reactors emphazing the flux-trap concepts have been under study for some years. At present conceptual designs are complete and a number of detailed engineering studies are under way.

A specialized Beam Research Reactor has been under study at Brookhaven for about two years. This reactor will have an epithermal flux within the core of greater than 10^{15} neutrons/cm^2-sec, and will be

uniquely valuable for cross section research.

Development of BSR-type reactors with aluminum elements is complete. Some work is under way on a similar core with stainless steel elements.

f. New Concepts

New concepts are supported up to the point where technological feasibility is established and the work can be projectized with confidence that a reactor system can be built sucessfully. For instance, work is under way at LASL on the molten plutonium fueled, sodium cooled LAMPRE reactor. The molten plutonium reactor technology being established in the work on LAMPRE reactor will provide a first step toward a high burnup, high breeding ratio fast reactor. Within the limitations in present metallurgical knowledge it is possible to conclude that eutectic alloys of plutonium with iron, nickel, or cobalt can reduce the melting temperature of the fuel from that of plutonium-239 (640°C) to a value in the vicinity of 400 to 450°C, and that ternary and quaternary alloying agents will further lower these melting temperatures by a few per cent. A structural material must be fabricable and resistant to corrosion by the fuel alloy and the coolant. At present, tantalum appears to be one of the best container materials. LAMPRE-1, a 1 MW experimental reactor is presently under construction.

The high temperature gas cooled reactor technology being established in the work on the proposed 3 MW thermal TURRET reactor is a part of the AEC - Bureau of Mines cooperative program on using nuclear energy for coal gasification. However, the TURRET reactor

will also provide most useful information on the operating problems encountered in a contaminated reactor coolant loop. The TURRET development program is presently centered around a helium cooled reactor operating with outlet temperatures of 2,000 to 2,400°F. and at a static system pressure of about 500 psi.

5. CHEMICAL REPROCESSING

Chemical reprocessing is defined as the conversion of irradiated fuel elements into products suitable for refabrication of fuel, products returnable to the AEC for credit, and waste products which can be safely stored. Chemical reprocessing is justified whenever material can be recovered for a credit greater than the cost of reprocessing.

The present status of reprocessing development can best be presented by discussing the three general approaches currently being followed. They are: (1) Aqueous processes with multiple dissolution facilities or "headends"; (2) Uranium hexafluoride volatility processes; and (3) Integrated "closed cycle" processes in which a reactor and processing plant are built and operated as a single unit.

The first two processes, the aqueous and volatility, are essentially competitive, each offering certain advantages but both being capable of achieving decontamination factors of the order of 10^7. The third approach is specialized according to the particular requirements of the specific reactor with which it is associated. In general, the latter approach is not capable of attaining decontamination factors higher than 10 to 100, but due to the nature of the reactor design this is not a handicap, since direct (manual) fuel refabrication is not required. An equally important consideration is the possibility that certain fuel systems will, due to their very nature, require semi-remote or remote refabrication anyway. This circumstance arises in the Th-U-233 fuel cycle, and also with U-Pu when Pu recycle is considered. During irradiation Th, U, and Pu build up isotopes which are not separable in ordinary chemical reprocessing. These isotopes in turn give rise to daughter activities which remain as contaminants even in the purified

U, Th, and Pu and make the materials too hot for other than at least semi-remote refabrication. This gives rise to additional expense and also to the suspicion that completely remote refabrication methods may be no more costly, while also eliminating the need for decontamination other than that little required for maintenance of reactivity.

At present, the Commission has "proved" technology applicable to the reprocessing of only three basic fuel compositions: (1) Pure uranium clad with aluminum; (2) Uranium - aluminum alloy clad with aluminum; and (3) Uranium - zirconium fuels having less than 2% total uranium by weight (exemplified by the Submarine Thermal Reactor (STR) Mark I and II cores). Technology is not considered "proved" until it has been demonstrated on at least a pilot plant scale with irradiated fuel. The vast majority of reactor designs now under consideration for civilian power purposes involve fuel elements having different compositions from the above, and for these adequate processing technology is at best in an advanced stage of cold pilot plant development.

a. Aqueous Processing. Aqueous processes are by far the most advanced in development due to experience gained in U and Pu production programs. For this reason, they are being applied to reprocessing the civilian fuels which will be produced in the near future. Table D-I lists the current status of aqueous processes under development.

The basic concept involved in aqueous processes is the reduction of solid fuel elements to an aqueous solution followed by continuous solvent extraction separation of contained fissionable materials from each other and from dissolved fission products and structural materials. Long experience with solvent extraction techniques for aluminum-clad

TABLE D-I

Aqueous Reprocessing Methods

Process	Type	Reagent	Container	Application	Status	Remarks and Limitations
Purex	DJ	$NaOH, HNO_3$	309-Nb SS	Al clad U or UO_2	Production	Ni in X-8001 not dissolved by NaOH
Thorex	TD	HNO_3-F^--Hg	309-Nb SS	Al clad Th or ThO_2	Pilot Plant	
25	TD	HNO_3-Hg^{++}	309-Nb SS	Al clad U-Al or Pu-Al	Production	
STR	TD	HF-$Cr_2O_7^-$	Monel	Zr clad U-Zr	Pilot Plant	Limited to 2% U in Zr
SIR	TD	H_2SO_4, HNO_3	Carpenter-20	SS clad UO_2-MgO	Laboratory[a]	Corrosive waste
Darex	TD	HCl-HNO_3	Titanium	SS clad UO_2-SS or UO_2	Laboratory[b]	Clad enters waste
Darex	DJ	HCl-HNO_3, HNO_3-F^-	Titanium	SS clad ThO_2	Laboratory	
Zirflex	DJ	NH_4F-NH_4NO_3, HNO_3	309-Nb SS	Zr clad UO_2, U-Zr or U-Zr-Nb	Laboratory[b]	Fines and precipitates must be handled
Sulfex	DJ	H_2SO_4, HNO_3	Nionel or Hastelloy-F	SS clad UO_2 or ThO_2	Laboratory[b]	Initiation problem
Niflex	TD	HF-HNO_3	Hastelloy-F	SS or Zr clad UO_2	Laboratory	Corrosion problem
PRDC	TD	HF, HNO_3-$Al(NO_3)_3$	309-Nb SS	Zr clad U-Mo	Laboratory[b]	
Hallam	DJ	H_2SO_4, HNO_3-Fe^{+++}	Nionel or Hastelloy-F	SS clad U-Mo	Laboratory[b]	
Zircex	DJ	HCl gas, HNO_3	?	Zr clad UO_2	Laboratory	Corrosion
Zircex	TD	HCl gas	?	Zr clad U-Zr, U-Zr-Nb	Laboratory	Corrosion
Shear-Leach		HNO_3	309-Nb SS	SS, Zr or Nb clad, UO_2, ThO_2 or UC	Laboratory	

DJ = dejacketing
TD = total dissolution

a = plant built not yet operated
b = pilot plant under construction or planned

uranium and uranium-aluminum fuels has been gained at Hanford, Savannah River, Idaho Chemical Processing Plant, and Oak Ridge National Laboratory. Dissolution of these fuels in HNO_3 followed by solvent extraction in the nitrate system has allowed construction of process equipment from relatively inexpensive stainless steels. With the extreme diversity of fuel compositions (e.g., zirconium, niobium, chromium, nickel, molybdenum, graphite, carbide, etc.) being developed or considered for power reactors, aqueous nitrate systems alone do not suffice. This is due to the fact that the newer and more diverse fuel compositions require more corrosive dissolution reagents, and such reagents are likewise often highly corrosive to stainless steel equipment. The "multiple headend" concept then arose, in which it was envisioned that fuels could be combined into groups having similar dissolution chemistries, the dissolver solutions being converted to a pure nitrate system for the subsequent standard solvent extraction. Experience has shown that capital costs for single headend systems are high ($500,000 - $3,000,000). It has therefore been the goal of research and development to minimize the number of headends required to handle the gamut of proposed fuel types.

(1) Mechanical Dejacketing. There are may basic designs of reactor fuels, e. g., metallic clad pins, rods, plates, and ceramics. By appropriate mechanical methods, it is often possible to separate difficult soluble claddings from the more easily dissolved metallic or ceramic cores, thus avoiding the use of highly corrosive chemicals for cladding dissolution. This technique also allows separation of cladding materials from the highly radioactive

fission product wastes, potentially simplifying possible future specific fission product separations and minimizing high level waste handling problems.

Sectional wedging, abrasive disc sawing, fuel jacket derodding, and shearing are being studied as possible techniques for disassembling and decladding fuel elements. Results to date indicate that the mechanical approach is feasible and that the economics of the operation may be satisfactory. Several of the machines in advanced stages of development are currently being constructed for cold testing. If successful they will be tested with actual irradiated fuels. Problem areas requiring resolution are dependability of operation, ease of maintenance, and over-all economics.

(2) Uranium - Aluminum Fuels. Workable aqueous processes have been demonstrated for this class of fuels. The technology is well established, and though continued process improvements are under development, the major cost improvements to be expected will probably be realized from continuous large capacity plant operation and from improved waste treatment and storage methods.

(3) Thorium Fuels. A modified solvent extraction flowsheet has been developed for reprocessing thorium containing fuels. The technique, known as the "Thorex" process, involves dissolution of the fuel (after decladding) in concentrated HNO_3 catalyzed with NaF, followed by solvent extraction cycles for Th and U-233, plus an additional ion exchange cleanup cycle for U-233. Evaluation of the Thorex process has been completed with Th irradiated to levels as high as

3500 grams/ton using both long-cooled (400 day) and short-cooled (30 day) material. Decontamination specifications were met for both Th and U-233 when processing 400-day cooled fuel, but product losses in solvent extraction operations were not considered satisfactory. In the processing of 30-day cooled Th, decontamination specifications were met for U-233, but were not met for Th. Process improvement studies have been continued. It is currently planned to process Th containing fuels such as the Consolidated Edison and Elk River fuels by the Thorex process.

As discussed earlier, radioactive growth of daughter activities in Th and U-233 subsequent to high decontamination Thorex processing may well dictate closed cycle processing with remote refabrication for the Th-U233 fuel cycle over the longer range.

(4) Zirconium - Niobium Clad and Alloy Fuels. The Submarine Thermal Reactor (STR) Mark I and II cores of this general type have been successfully processed. The technology used was developed specifically for these fuels and is limited in application to fuels having a maximum over-all U composition of approximately 2%. The U composition limitation of the process seriously limits its applicability to power reactor fuels, and the highly corrosive dissolution medium required (concentrated HF) makes eventual development of a more optimum and economic process necessary.

The Zirflex process in which zirconium alloy materials are dissolved in NH_4F-NH_4NO_3 shows great promise. This process is of particular interest in that the system, though corrosive, can be contained in standard process equipment with tolerable corrosion rates. For this

reason it has been chosen for initial installation and pilot plant testing of Zr decladding. Construction of this headend in a Power Reactor Fuel Reprocessing Pilot Plant is now underway. Studies to date also indicate promise for Zirflex application to integral alloy dissolutions.

In the longer range "Zircex" process anhydrous HCl is reacted directly with metallic fuel constituents, leading to the separation of the volatile $ZrCl_4$ or $NbCl_4$. This process allows separation of cladding and/or alloy metals from fission products which, as noted above, is desirable from a waste storage viewpoint. Unit operations data have been obtained on both dejacketing of UO_2 fuels and integral attack on alloy fuels. The dejacketing step now appears to be technically feasible. Direct reaction of HCl in the gas phase with massive quantities of zirconium alloy has led to severe reaction heat removal problems. Also under study is the gas phase reaction of zirconium alloy material submerged in a chemically inert bed, fluidized with a gas containing anhydrous HCl. This approach is attractive in that it appears to alleviate the heat removal problem, though a number of engineering problems still require resolution. A "high HNO_3-low HF" process is under investigation as a dissolution procedure, especially for integral dissolution of highly alloyed fuels as, for example, the PWR core. It is hoped that fluoride concentration can be controlled to decrease severe corrosion.

To summarize, the processing picture relative to Zr alloy type fuels is much brighter than it was one to two years ago. Newer processes being developed tend to reduce the severity of the corrosion problem, and it now seems possible that a process can be developed

which will be compatible with some of the equipment used with other fuel processes. However, serious problems still remain to be resolved, particularly in the area of mixed alloys (Zr, Nb, Mo, etc.) and in the treatment and disposal of process wastes.

(5) Stainless Steel Clad and Alloy Fuels. The "Darex" process is of wide applicability to both military and civilian power reactor fuels. In this process, which is in an advanced stage of development, stainless steel fuels are dissolved in dilute aqua regia (HCl-HNO_3) followed by distillation separation of the highly corrosive chloride ion and conversion to the nitrate system. Continuous dissolution has been demonstrated on a cold engineering basis, and integrated loop testing involving dissolution, chloride stripping, rectification, and HNO_3 evaporation for feed preparation is in progress. A pilot plant demonstration of the Darex headend is now in the design stage and hot operation is scheduled for FY 1962. Titanium is the construction material.

An alternate to mechanical dejacketing or Darex dissolution of stainless steel clad fuels is the "Sulfex" process. Here, the stainless steel clad is dissolved in H_2SO_4 in a Nionel or Hastelloy-F vessel. The principle problems in this approach are passivation of stainless steel, maintenance of low fuel losses to the decladding solution, and proof of suitable Nionel or Hastelloy-F fabrication techniques. Design and construction of a hot pilot plant are currently underway.

Several years ago a process was developed, and a hot pilot plant headend was constructed, for reprocessing Submarine Intermediate Reactor (SIR) fuel. The process involved integral dissolution of the stainless steel clad - UO_2-stainless steel matrix fuel in H_2SO_4, followed by sufficient HNO_3 dilution to allow the use of standard solvent

extraction technology. Carpenter 20 steel was used for process vessels. This technique has only limited applicability for treatment of diverse fuel types in comparison with Darex, and is therefore felt to be of dubious long range value. The existing facilities will be operated, however, in FY 1960.

Use of mixed HNO_3-HF is also being investigated as a process for decladding UO_2 clad in stainless steel. Its major advantage is that several dissolver construction materials may be serviceable not only for this dissolution reagent but also for the Zirflex reagent used for Zr dissolution. This might for the first time permit dissolution and treatment of the two major classes of fuel materials in a common system.

To summarize, technology for the reprocessing of stainless steel fuels of the types currently under consideration has not been demonstrated in hot pilot plant operation, though it has reached the stage where processes are being engineered for hot testing.

(6) Uranium Carbide and Graphite Fuels. Development of reprocessing technology for these types of fuels is at a very preliminary stage.

b. Volatility Processing. In volatility processing, U fuels are fluorinated to produce UF_6 and the fluorides of the fission products and alloying metals. UF_6, being much more volatile than the other fluorides, can be readily separated by fractional distillation or adsorption-desorption processes. Volatility processes are capable of attaining fission product decontamination factors of 10^7, comparable to those of present aqueous methods. Potential cost advantages of volatility vs. aqueous techniques result from the reduced number of

costly process steps required to obtain the final product and also from the approximate 30-fold reduction in the volume of high level wastes produced.

The BrF_3 process has been studied extensively for application to low enriched fuels. A hot pilot plant has been operated, successfully processing fully-irradiated 75-day cooled Hanford material. Fission product decontamination was excellent. This method is not of great interest currently in that it is applicable only to bare U metal, and volatilization of Pu was not achieved. The work was of great significance, however, in that it demonstrated the feasibility of volatility separation.

Technical feasility of the fused salt volatility process has also been demonstrated in hot pilot plant operation by actual reprocessing of the irradiated fused salt fuel from the Aircraft Reactor Experiment. The process involved melting the salt mixture ($NaF-ZrF_4-UF_4$) and converting the UF_4 to UF_6 with gaseous F_2. U decontamination and recovery were again excellent. Fused salt studies have been extended to include dissolution of highly enriched metallic fuels (e. g., PWR core type). Pilot plants are currently being modified to allow process testing with irradiated civilian reactor fuels. The major effort is currently being concentrated on the testing of container materials for molten salt operations. The most promising materials appear to be low-carbon nickel and a "graphite-lined frozen-salt wall" vessel.

Extensive experience in fluidization techniques has led to the study of the "aqueous dissolution-fluorination" (ADF) concept. In this concept reactor fuels dissolved by conventional aqueous techniques are

fed directly to a fluidized bed fluorinating unit. It is possible that UF_6 can be distilled directly (leaving a calcined waste similar to that desired in aqueous waste processing), thus eliminating aqueous solvent extraction and subsequent steps. Work to date has been limited to U-Zr fuel of the PWR core type which has been moderately successful.

Development studies are currently under way to investigate the control of highly exothermic reactions between metallic fuel elements and reactant gases (e.g., HF, HCl, F_2, etc.) by means of a fluidized bed inert to the reactant gases. Laboratory studies to date have been highly successful in controlling the reaction of anhydrous HCl with U-Zr alloy fuel elements. This work is of potential significance as a possible multipurpose headend for volatility processes.

A method being investigated for reprocessing U-Zr alloy and other metal fuels involves dissolution of the metal in a liquid NO_2-HF system. To date, preliminary laboratory work has been done.

Laboratory work is being continued on the difficult problem of separating plutonium hexafluoride (PuF_6) from low enriched, volatility processed fuels. It has been established in the laboratory that PuF_6 can be produced at reasonable rates and at low enough temperatures to be of process significance and that the material can be maintained and handled for reasonable periods of time.

To summarize, certain volatility techniques have been proved in limited pilot plant operations to be competitive with aqueous solvent extraction methods in product recovery efficiency and fission product decontamination. The approach holds significant potential for reduced processing costs and as such is worthy of continued and expanded effort.

c. Closed Cycle Processing. "Closed cycle" processes, though frequently referred to as only another variety of chemical separations, represent in actuality an entirely new approach to the whole civilian reactor fuel cycle. In this approach it is recognized that the basic concept underlying much of today's fuel cycle development, namely high decontamination of spent fuels followed by manual refabrication of fuel elements, may not be as applicable in the civilian power program as in the military program from which it was derived. However, the closed cycle concept presents severe technical difficulties and unless reasonable economics can be shown it will have its greatest application to certain advanced reactor types in which fuel inventories are large (e.g., fast reactors) or which employ Pu recycle or the Th-U233 fuel system.

Closed cycle processes, more commonly known as "Pyrometallurgy" or "Low Decontamination" processes, are those in which all operations in the fuel cycle from decontamination through re-enrichment and refabrication are performed remotely in a facility closely integrated with a specific reactor. For closed cycle processes fission product decontamination factors of 10-100 are adequate (as compared with 10^7 for aqueous and volatility processes), since partial removal of fission product poisons is sufficient when manual handling is not required.

A processing facility integrated with the EBR-II reactor is now under construction at National Reactor Testing Station. The facility is being designed to reprocess EBR-II core material on a continuing basis and to demonstrate a process for the blanket. The sodium bonded (core) fuel pins will be mechanically removed from the assembly, mechanically declad, refined by melting and oxidative slagging, and after

fuel addition will be refabricated into fuel elements remotely. Recovery of Pu from the blanket using molten magnesium extraction or fractional crystallization from cadmium is also under study. The major process steps have been demonstrated at full EBR-II plant scale with 0.1% of the activity level anticipated and on a laboratory scale with high fission product activity. Major chemical problems under investigation include optimization of conditions for fission product drossing, development of suitable process container materials, and scrap and crucible recovery techniques. It should be emphasized, however, that the mechanical problems of transport, fabrication, inspection and self-maintenance are also major areas requiring development. This can be seen by visualizing the difficulties in handling materials much hotter than present reprocessing feeds, chemically treating and refabricating them, and then inspecting them to precise dimensions, all with machinery that must be maintained remotely with no possibility of any human re-entry into the operating areas. It is significant that the responsible development people are quite optimistic.

Conventional oxide drossing techniques are not capable of removing fission products less electropositive than U and Pu (e.g., Ru, Rh, and Mo). A new process originally aimed at removal of these less electropositive elements now appears to be a possible alternate processing approach. In this process (Pyrocadmium) the U-Pu alloy is dissolved in molten cadmium after which, by lowering the temperature, the intermetallic compound UCd_{11} is precipitated and filtered, leaving the fission products behind in the Cd solution. Pu can be precipitated or not as desired. This technique holds great promise as a closed cycle process, and considerable effort is being expended in this direction.

Closed cycle research is also being undertaken with Th-U fuels. Applicable processes under investigation include arc melting, electrorefining, Mg-Ag extraction, and Pyrozinc. Considerable more work will be required before it is possible to evaluate the relative merits of any of these concepts.

An extensive program is also being conducted to develop a closed cycle process applicable to UO_2-ceramic fuel systems. Cold and low level process studies have indicated that a relatively simple process involving cyclic oxidation-reduction operations yields an oxide product with satisfactory properties. This program is being continued with emphasis on hot cell studies to determine the effect of multiple and high fuel burnup irradiations on the process. This technology is of interest in the PRTR program and also has a significant potential for the UO_2-ceramic fueled power reactors. In many of these systems, Pu recycle is being considered and the closed cycle has certain potential advantages as stated above.

Very preliminary investigations are in progress on the development of closed cycle processes for UC and graphite type fuels.

In summary, the first full-level, closed cycle experiment is well underway. The EBR-II facility, scheduled for hot operation in 1961, is expected to allow resolution of mechanical problems to be expected in such complicated remote operations. Subsequent evaluation of closed cycle concepts should be easier in light of the experience gained.

d. Waste Treatment. The major waste disposal problem arising from reactor operation is associated with fission products in the irradiated fuels. Fission product effluents from chemical reprocessing

operations vary in volume and composition depending upon the characteristics of the fuel being processed and the type of process used. In general, aqueous reprocessing techniques result in relatively large waste volumes, which to date have been concentrated and stored in underground retention tanks. This method has been safe and adequate as an interim measure, though it is not suitable for ultimate disposal because the useful life to tanks is not known and must be assumed to be relatively short as compared with the 500 to 800 year hazard of the fission products. Thus the long term storage cost cannot be determined accurately.

The first approach in the development of ultimate disposal methods has been to minimize volumes by converting the wastes to solids while at the same time making them less mobile. Several methods have been investigated. Calcination of aqueous wastes to dry solids in a fluidized bed at 400° to 500°C is the technique most advanced at this time. A hot pilot plant is currently under construction which will allow demonstration of the process with aqueous aluminum nitrate wastes. This facility is scheduled for operation in 1960. A number of severe problems associated with the extremely high levels of radioactivity involved will require resolution. The treatment of off-gases from fluidized bed operations is a serious problem, involving not only de-entrainment of micron and sub-micron size particulates by factors of 10^8, but also the equivalent separation of volatilized and subsequently condensed radioactive ruthenium oxide particulates. Heat removal from the stored Al_2O_3 solid wastes is also a problem, since Al_2O_3 is a refractory insulating material. While experience to date in the operation of pilot plant equipment with solutions containing

spiked quantities of actual ICPP wastes indicates that these problems can be resolved, only significant scale operation on a hot basis with actual waste solutions will prove the technical feasibility of the process.

Handling of wastes containing corrosive ions such as F^-, Cl^-, and $SO_4^=$ presents severe problems both when considering tank storage or in the development of ultimate disposal techniques. Chemical complexing of fluoride ion with $Al(NO_3)_3$ has been employed in tank storage of STR (Zr containing) wastes, but the stainless steel tanks required are costly, and as of this date minimum estimated tank life is only ten years. Preliminary programs are being conducted to investigate and develop processes to separate and highly decontaminate bulk solids (Al, Zr, Nb, and stainless steel components) from fission products to aid ultimate utilization and/or disposal. It is anticipated that technology developed in the calcination program can be extrapolated or modified to complement work in this area. It should be noted again that development of the reprocessing schemes themselves is aimed at minimizing subsequent waste problems.

Little development effort has yet been expended on the disposal of volatility and closed cycle process wastes. These wastes are produced as solids and as such storage problems may be minimized. Techniques for disposal of contaminated equipment from closed cycle plants are now under investigation.

Gaseous radioactive fission products are released during dissolution of irradiated reactor fuels. A process has been developed and operated extensively for the removal of I^{131}. Though experience

has been gained in the removal of Xe and Kr isotopes, development of a new and potentially more efficient process is being conducted.

To summarize the status of waste treatment and disposal, current practice is to reduce aqueous waste volumes by evaporation, and to store the concentrates in underground tanks. A program to calcine aqueous aluminum nitrate waste solutions to dry oxides is in the late stages of development, with hot process demonstration scheduled to start in 1960. Alternate processes which complement and/or compete with the above methods are in early stages of development.

e. Economics. Recognizing that private industrial development of atomic power presupposed a method of estimating all significant fuel cycle costs, the Commission devised a "conceptual plant" capable of processing those industrial fuels then anticipated, thus filling in the last gap in the civilian fuel cycle. Inasmuch as none of these fuels had yet been prepared and little or no research had been done on the specific problems involved, the plant was designed on a purely conceptual basis. Recognizing the inexact nature of the total plant estimate it was not felt reasonable to attempt to differentiate conceptual cost differences that might arise from one fuel to another and, therefore, within the limits of uncertainty of the estimates, a standard daily charge was chosen for all fuels considered. This charge ($15,300/day with minor adjustments) is recognizably lower than present estimates of possible private industrial plant charges which would be required if conventional industrial pricing policies were used in the light of the small loads of widely diverse fuel types found in the civilian power industry today. The Commission,

however, which will actually process these fuels in existing facilities will be able to take advantage of its large capacity plants and recover full costs even at these prices. Additional capital funds are needed only to modify or add certain special equipment. With increasing nuclear power construction underway it is felt that within 5 to 15 years adequate fuel loads will be available to permit private industrial plant construction and industrial pricing policy to be compatible with the existing Atomic Energy Commission price structure. When this occurs, the AEC will discontinue its current commercial work in this field.

As presently provided, the costs of processing are determined primarily by size of batch to be processed, because this quantity determines the number of days that the conceptual plant, operating at a fixed capacity, must work on the batch. Fuel enrichment affects processing costs directly (on a unit weight basis), because higher enrichment fuels necessitate lower plant capacity if criticality control is to be maintained. To the extent to which inert alloying or cladding materials are introduced into the conceptual solvent extraction system, capacity for fissionable material is also reduced, thus increasing processing days and total batch costs. All of these influences on charges are reasonable in the light of potential industrial practice. Charges for new civilian fuels which were not considered in the original conceptual design will be determined as above with an incremental charge based on new or special equipment not initially provided. It is extremely difficult to project reprocessing costs more accurately at this time. It is well-known, of course, that increasing fuel burn-up will significantly decrease the affect of processing costs on over all fuel cycle mills/KWH.

6. COMPONENTS

The information included in this Section summarizes the status of the technology in reactor instrumentation and control and heat transfer reactor components.

a. Out-of-Core Instrumentation.

(1) Neutron-flux sensors for environments up to 570°F are commercially available.

(2) A neutron indicator scheme recently examined by Oak Ridge National Laboratory consists of a moving chamber and positioning servo. Flux values are read out in terms of chamber position. An advantage of this scheme is the ability to indicate flux values over a 10 decade range as compared to the 6 decade range of conventional sensors.

(3) Work is currently underway to develop and test a neutron sensitive microwave waveguide system. This sensor, if proven feasible, will permit the monitoring of the average flux around the periphery of a reactor under all flux-distribution conditions. It is anticipated that a sensor of this type will be useful in temperature environments above 1000°F.

(4) Several attempts have been made to develop a neutron-flux indicator based on thermometry. The advantages of this type of sensor are its ruggedness, low impedance output, and self-generating signal. Savannah River has successfully developed and operated a "neutron thermometer" employing a fissile material as heat source and reactor coolant as heat sink. The Savannah River work is being extended to develop a sensor that is fast enough to operate in a reactor control or protective circuit and is designed to operate in a 1000°F environment.

Battelle Memorial Institute has developed a similar neutron thermometer based on the heat balance principle. The merit of the Battelle Memorial Institute thermometer is its ability to compensate for gamma heating in the sensitive element; the disadvantage is the long time required for taking a measurement.

b. In-Core Instrumentation.

(1) Miniature neutron-flux monitoring sensors are commercially available for operation at 650°F and 1,000 psi. These are ion chambers $\frac{1}{4}$" diameter. Small in-core flux sensors employing both fission heating and gamma heating have been used at Savannah River in temperature environments up to 200°F for long periods of time. Savannah River has also had success with the use of activated wires which may be withdrawn at will, although this method is cumbersome and time consuming. Another development by Battelle Memorial Institute has been described under out-of-core instrumentation.

(2) Coolant flow meters employing turbines have been used extensively at Argonne National Laboratory. Currently available turbine flow meters are designed to operate in a 500°F environment. Argonne National Laboratory is investigating the hot-wire anemometer as a possible replacement for turbine flow meters. Turbine type coolant flow meters have been successfully run out-of-pile in NaK at 1500°F at Oak Ridge National Laboratory.

(3) Temperature indicators employing thermocouples have been used successfully up to 650°F in reactor cores. Failures due to mechanical breakage have been experienced where rigid fabrication specifications were not observed. Resistance thermometers have not been extensively used in cores. However, Argonne National Laboratory plans to

make investigations in this field. Thermocouples are not available to make required in-core measurements. Particular attention should be paid to the thermometry of UO_2 near its melting point but no reliable in-pile temperature measurements exist above approximately 3000°F.

(4) Pressure indicators for measuring transient pressure changes in the coolant channels between the fuel plates or fuel pins of boiling water reactors are not available; however, Spert uses instrumentation to measure the integrated pressure change within a single fuel element assembly. Argonne National Laboratory plans to investigate the development of improved in-core pressure sensing devices.

(5) Moderator void indicators have been tested in cold loops; however, little success has been shown. Argonne National Laboratory is considering sonic systems for this service. It has been demonstrated that coolant void fraction can be inferred from measurements made of the alternating current component of noise in ion chambers inserted near boiling coolants. G. E. Vallecitos, in conjunction with Knolls Atomic Power Laboratory, performed the demonstration.

(6) Coolant level and density indicators based on sonic systems are under development. These sensors are designed to operate in 1000°F environments and will be able to discriminate between froth and liquid.

(7) Fuel-element failure detectors have been under investigation for several years. No adequate fuel element detector system has yet been developed for water reactors.

c. Connecting Cables

(1) Special signal cables for flux monitors have been developed. These cables employ quartz fiber insulation and should be

capable of operation at 1000°F. For low temperature environments, polystyrene insulated cables are satisfactory. Mgo cables are believed to be adequate for 2000°F.

d. Control-Rod Actuators.

(1) Electric powered actuators of the magnetic-jack type, harmonic drive type, rack and pinion type, and ball and screw type have been successfully developed and tested extensively up to 600°F. Scram times reported for these actuators range from 50 to 750 milliseconds, depending on the amount of spring loading in the device. Commercial versions of all these types are available.

For higher temperature operation the magnetic-jack type appears to offer the greatest promise. In this type of actuator, the drive mechanism consists of a support tube, four sets of stationary electromagnet coils, an annular fixed armature, and an annular movable armature. The movable armature can be moved up or down a short distance (about 0.1 inch) between end stops by energizing either lift coils or pull-down coils. By energizing either the grip or hold coils, the rod segments are magnetically attracted and held against the inner bore of the movable or fixed armatures, respectively. In this manner, the rod extension is gripped by the movable armature to be moved and held stationary by the fixed armature while the movable armature is returned for another step. In addition to moving in discrete steps, the control rod may be released at any position in its travel by de-energizing the coils.

There are several advantages inherent in a magnetic-jack rod actuator. It is relatively simple in design and can be fabricated at low cost. Most parts are either simple coils or can be made of standard seamless tubing. Fabrication consists entirely of simple machining and

welding of these shapes. In addition, there are no parts to lubricate. Other important advantages of the magnetic jack are a clear loading face and the elimination of shaft seals. The actuator can be used in any water or organic cooled reactor, wet, or in a thimble.

Atomics International has designed and tested a magnetic-jack actuator for Piqua that can be placed in toto inside a core tank and operated immersed in a 700°F coolant. A prototype has been operated in air at room temperature at rod speeds up to 25 in/min lifting a 50 lb. load, and 14 in/min with a 300 lb. load. Maximum scram times during a prototype test in 550°F Santowax R has been 725 milliseconds. Endurance tests have run as long as 260 hours during which 2,020 scrams have been made.

The cost of magnetic jacks presently runs from $1,100 per unit (in quantities of 10) for the Argonne National Laboratory design lifting a 220 lb. load, to $20,000 per unit (in quantities of 10) for the Westinghouse design lifting a 400 lb. load.

(2) Hydraulic control-rod actuators are not in wide spread use at present. This type of design will be used in Dresden which will operate in a 600°F environment. It incorporates a piston stabilizer in the form of detents along path of travel.

e. Control System.

(1) Transfer Functions. Techniques for measuring reactor transfer functions for both thermal and fast reactors have been developed to a high degree of refinement by Argonne National Laboratory. From these measurements, mean neutron lifetimes and reactor stability under various operating conditions can be determined.

(2) Experimental neutron-flux distribution measurements employing oscillating control rod techniques made at Savannah River

indicate that this method shows promise for obtaining power flattening information in power reactors where in-core instrumentation is not possible. Further work in this area is being planned.

(3) Electronic systems for both control and protective functions have recently reached a high degree of refinement through the use of transistor circuitry and magnetic amplifier circuitry.

f. Heat Transfer.

(1) Gas Cooled Heat Transfer. Helium temperatures up to 2300°F at 250 psig have been maintained for an appreciable length of time, and temperatures up to 2500°F have been attained for a short time, in a simulated reactor loop with heat transfer to the gas taking place in a bed of unfueled graphite spheres. Helium purity has been maintained, but satisfactory operation of the helium compressor has been a problem.

A considerable amount of investigation of high temperature heat transfer in gas systems is under way. This includes studies of heat transfer and pressure drop phenomena at temperatures up to 4000°F in flow channels of various shapes, measurement of thermal conductivities for helium and other gases at high temperatures, and adequate means of high temperature measurement in gas streams.

Preliminary experimental work in methods of increasing the mass heat transport capability of gas by use of finely divided solids carried in suspension has resulted in attainment of high gas heat transfer coefficients, indicating that the gas-suspension cooled reactor concept may be promising.

(2) Liquid Cooled Heat Transfer. In the area of water coolants, investigations are underway with respect to (1) boiling heat transfer and burnout under various conditions, (2) core hydrodynamics,

and (3) means of stimulating increased heat transfer by vortex flow, ultrasonics, electric fields, etc.

Investigations are underway to determine heat transfer characteristics for (1) light and heavy liquid metals in heat exchangers, and (2) direct heat exchange between immiscible liquids such as water and mercury.

Work is underway to determine means of calculating heat flux in superheated liquids.

g. Components.

(1) Gas System Components. In the field of components for high temperature gas cooled reactors, a recent survey of the availability and specifications of current equipment indicates that there is little, if any, commercially available to meet the needs of helium cooled reactors. It is believed that a considerable amount of development must be undertaken to provide satisfactory mechanical components such as compressors, heat exchangers, valves, etc.

(2) Pressure Vessels. Investigations are underway with respect to materials, fabrication and welding techniques, stress corrosion phenomena, and fatigue in reactor pressure vessels, which are directed toward improved design and more economical manufacture of these components.

Investigations of thermal cycling at coolant vessel wall surfaces will be started soon to study the causes and means of avoiding deterioration of pressure vessel and heat exchanger surfaces and fuel element cladding caused by turbulent eddies.

There is a need to investigate techniques for remote inspection and testing of reactor pressure vessels after long service in a radiation environment.

7. ENVIRONMENTAL INVESTIGATION AND EFFLUENT CONTROL

The environmental investigation and effluent control program has as its major objectives the safe management and disposal of various types of radioactive wastes, the quantitative determination of the behavior of these radioactive effluents in the environment and the development of engineering criteria associated with the environmental aspects of nuclear technology operations.

Low-level wastes for which treatment and disposal systems have been developed are those generally in the microcuries per gallon concentration range and are those generally associated with laboratory and research activities, light-water coolants in reactors, and isotope use.

The handling and disposal of solid or packaged low and intermediate level wastes (up to the order of a curie per package) has also been successfully carried out by such operations as baling and land burial.

Storage systems for high-level wastes, primarily those evolved from the chemical reprocessing of irradiated fuels (concentrations up to hundreds of curies per gallon) have been in successful operation for more than 10 years, but are not equated with final disposal. Capital costs for such storage facilities range from about 30-40¢ per gallon to $2.00 per gallon depending on the chemical composition of the wastes. Storage costs for waste from Zr fuel may go up as high as $5.00 per gallon.

The current development projects included in the program are divided into four categories as follows:

(a) Environmental investigations

(b) Low and intermediate level radioactive waste studies

(c) High-level radioactive waste studies, and

(d) Gaseous and particulate waste studies.

a. Program Activities. Environmental and sanitary engineering activities involve work essentially in two major areas: (a) radioactive waste handling and ultimate disposal and (b) environmental studies as related to the safe handling and disposal of radioactive materials. An equally important phase of the sanitary engineering program is the establishment of technical working relationships with federal, state, and local agencies and private industry in the areas of radioactive waste disposal and related environmental problems.

These activities have resulted in the promulgation of an extensive research and development program in the above areas in order to assure that the disposal of radioactive materials from nuclear energy operations will not deleteriously affect the health and safety of the public and its environment.

b. Environmental Studies. The program in environmental studies is related to the controlled discharge and dispersal of radioactive wastes to nature. Because the wastes are evolved and discharged in all possible physical forms, i.e., liquid, solid, and gaseous, there is prime concern with safeguarding man and his resources from direct and indirect effects of the radioactivity via air, water, soil and food routes of transmission. Each individual waste discharge must be considered unique, because of differences either in the form or constituency of the wastes or in the receiving mass of air, water or soil. (As examples, the meteorological diffusibility at the National Reactor Testing Station differs markedly from that at the Shippingport reactor site (as do the gaseous effluents); the stream dilution capabilities of the Animas River in New Mexico differs from those of the Mohawk River in New York, etc.) This program is aimed

at obtaining the scientific data needed to assess the safe capacity for radioactive wastes of both local and regional environments.

Experience during the past several years has shown an ever increasing use of the air, ground, and surface water environments for disposal of certain low and intermediate level wastes. The quantitative assessment of specific environments to receive radioactive effluents is being investigated through cooperative agreements with the U.S. Public Health Service, U. S. Geological Survey, and the U. S. Weather Bureau and by contract with Harvard University, Knolls Atomic Power Laboratory, and the Oak Ridge National Laboratory. Hanford, the Savannah River Plant, Idaho and other Atomic Energy Commission installations have done similar work.

Assessing the environmental hazards associated with high level waste disposal in terms of the geologically significant periods of time during which they continue to be hazardous is a complex engineering problem. This is a particular problem in evaluating the feasibility of disposal into salt formations or deep wells and in evaluating the degree of fixation and non-leachability of radioactivity in sinters, ceramics, or glasses.

The U. S. Public Health Service has initiated studies to determine the fate of specific radionuclides in streams below KAPL, PWR and SRP operations. Data to be obtained will provide a quantitative assessment of environmental dilution or concentration factors available in receiving streams and aid in the establishment of engineering and operating criteria for the continuing, safe disposal of low-level liquid wastes. In these studies considerable effort has been devoted to the development of precise radiochemical techniques for the measurement of specific isotopes such as cesium, strontium, and tritium in stream concentrations less than drinking water standards. This analytical technology is required before an accurate assessment by field survey can be made of waste dispersal

operations from the sites mentioned above.

In another cooperative study, a comprehensive industrial waste survey of uranium milling operations on the Colorado plateau is being conducted by the U. S. Public Health Service. Extensive stream investigations and in-mill surveys have been carried out at several representative mills in order to determine the characteristics of wastes from different uranium processing methods and their effects on the receiving stream environment. An interim report has been received which indicates that the Durango, Colorado mill is causing a certain degree of pollution in the Animas River by the discharge of radium-bearing wastes in excess of maximum permissible concentrations. Results of the over-all Colorado Plateau survey will determine whether extensive engineering development work will be required to provide design criteria for a complete radium waste processing and disposal system for the uranium milling industry.

The fate of radioactivity in tidal estuarties in connection with the nuclear ship program is being investigated jointly by the U. S. Coast and Geodetic Survey and the Chesapeake Bay Institute (Johns Hopkins University). The purpose of this study is to determine the behavior of fission products in coastal environments as introduced by nuclear propelled ships, i.e., either from the controlled discharge of operational wastes or by an incident involving an accidental release of radioactivity. Properties of the marine environment which influence: (a) the movement and dispersal of radioactive material, (b) the geochemical exchange of the material with suspended inorganic material and bottom sediments and (c) the reconcentration of activity into the marine biota -- all these factors are being investigated to develop basic information which will allow the prediction of the safe rate of release of radioactive materials to any selected marine site as well as the consequences of a major accidental release.

New York Harbor was chosen as the site for the first field study and it is planned to utilize applicable data developed at this location in model studies of other harbors. The Coast and Geodetic Survey is conducting the field survey program involving sample collection and salinity, temperature, and current velocity measurements; Chesapeake Bay Institute is utilizing the data in calculations on the rates of diffusion in the Harbor as well as conducting laboratory studies on geochemical exchange and biological uptake of activity under simulated estuarine conditions.

Sea disposal is being seriously questioned by certain state agencies and many administrative problems involving licensee operations in this area are being encountered. A definite lack of quantitative information to assess these operations from an environmental standpoint has been a major deterrent. Preliminary investigations of both the Atlantic and Pacific disposal areas have revealed no readily detectable deleterious conditions. Because of increased interest by both waste disposers and regulatory agencies, the Committee on Oceanography of the National Academy of Sciences at the request of the Atomic Energy Commission has suggested a number of in-shore disposal areas off the Atlantic and Gulf Coasts. Oceanographic studies are being initiated to develop operating criteria and regulations for insuring safe disposal of low-level wastes at sea.

In conjunction with the sea disposal studies, investigations are being carried out by the Stanford Research Institute on the feasibility of establishing a land burial area to service the western states area. At the same time, the U. S. Geological Survey is obtaining detailed geohydrologic information on areas in New York and Pennsylvania and Oak Ridge as a step towards the establishment of a northeastern burial ground facility.

c. Low and Intermediate Level Waste Studies. These studies are directed towards the development of improved systems for the handling,

treatment and disposal of low and intermediate level liquid wastes.

Ground disposal of these wastes has been practiced at Hanford, Oak Ridge National Laboratory, and Savannah River Project. The efficiency and safety of this disposal method requires continuous analysis and evaluation. The rate of vertical and horizontal movements of waste liquids through different soil formations is being determined. In soil disposal operations at Oak Ridge the only radionuclides that have reached monitoring points are Ru-106 and the complexed ions of Co-60 and Sb-125 and these concentrations have been several orders of magnitude below acceptable limits. To obtain a better understanding of our ground disposal operations, fundamental studies have been initiated at the University of North Carolina on the physical-chemical reactions between radioactive ions and natural earth materials. At Hanford, studies are being conducted on the determination of ion exchange capacity and replacement reactions between minerals and certain radioisotopes. The application of specific minerals such as calcite for Sr-90 removal and clinoptilolite for Cs-137 is being investigated as a means of decontamination for low and intermediate level wastes.

Studies on the effectiveness of conventional water treatment processes, for the removal of long-lived isotopes (Sr-90 and Cs-137) from low-level liquid wastes are being conducted at the Oak Ridge process waste water treatment plant. The removal of strontium and the rare earths has been increased to 95% with increased dosages of lime and soda ash while the introduction of an illitic type of clay mineral has improved cesium removal efficiencies to 85%.

At the University of California an investigation of the dispersal of low-level wastes in shallow reverse wells is being made. The rate of

movement of radioactive fronts under straight injection conditions is being determined and similar observations are being made under the conditions of injection and pumping a well simultaneously.

d. High-Level Waste Studies. The Sanitary Engineering program in developing systems for ultimate disposal of high-level wastes is charted along two promising courses: (1) fixation or immobilization of the fission products in clay, glass or ceramic structures and (2) direct discharge to selected, geologic formations such as salt formations or deep, permeable strata which will isolate the wastes for their period of radioactive hazard. Feasibility studies (on paper) and laboratory experiments, especially in the fixation approach, have indicated promising chemical interactions to be used and have overcome many obstacles.

Currently, emphasis of the development program is directed towards expanding the feasibility of laboratory studies to pilot plant scale engineering studies on both the fixation and direct disposal approaches. Construction has started on a 60 gph conversion-to-oxide (aluminum nitrate wastes) plant at Idaho Chemical Processing Plant utilizing a fluidized bed. At Hanford, laboratory studies on radiant heat spray calcination on production type fuels has been promising. Work has now been initiated on power reactor fuel wastes. A larger scale experimental unit is being planned. Laboratory studies on the development of new ceramic mixtures with Purex wastes are being conducted for use in a self-sintering experiment at Oak Ridge National Laboratory. On the direct disposal approach more laboratory work has been accomplished on utilization of salt structures than on the other possible geologic formations. Structural capabilities of halite (salt) formations, chemical compatibility of Purex and acid aluminum nitrate type wastes with halite and formation impurities,

and calculations involving the dissipation of heat generated by decaying fission products within salt formation cavities have been carried out.

The engineering design for a field test to determine the feasibility of disposing of high-level liquid wastes into salt formations has been completed. A subcontract has been negotiated by Oak Ridge National Laboratory with the Carey Salt Company of Hutchinson, Kansas, to conduct the field experiment in an unoccupied section of their mine. Two cavities (7.5 x 7.5 x 10 feet) have been excavated and auxiliary test equipment, including electrolytic heaters for simulating fission product decay heat, has been installed. Synthetic chemical processing wastes with no radioactivity will be used in the first tests, which are expected to run for two years. Information on the structural integrity of salt cavities, ion migration, temperature gradients both inside and outside of the cavity, possible migration of the waste cavity itself and gas production rates will be obtained in this initial field test program. The first tests will begin in September 1959.

Development work on the other direct disposal approach, i.e., injection of high-level wastes into deep porous formations is being conducted along lines recommended by a special American Petroleum Institute study group. Laboratory studies on deep well problems such as plugging, formation fluid and reservoir compatibility with the wastes, heat dissipation and corrosion have been initiated at Oak Ridge. Methods of controlling adsorption and precipitation near an injection well were investigated by laboratory studies of the cesium and strontium uptake relations in clay slurries, interstitial precipitation, and sorption on feldspars, quartz, limestone, and clay minerals. The formation of precipitates on neutralization of acid wastes can be controlled with citric

or tartaric acid. Complexing the waste introduces some corrosion problems but preliminary tests suggest that this can be controlled with inhibitors. An associated complementary attack on the final disposal problem, involving either high efficiency removal of specific nuclides or separation of inert salts from fission product waste mixtures is being investigated on a laboratory scale.

e. Gaseous Effluent Studies. The program in treatment of gaseous effluents is planned to keep ahead of developments in the industry. At one time small particulate contaminants presented a difficult air cleaning problem. Equipment and facilities such as high efficiency filters, deep bed sand and fiber filters, and iodine and rare gas removal units have been developed and built as a result of concerted effort by plant operators, the Stack Gas Problem Working Group and the U. S. Weather Bureau. In short, a difficult problem was brought under control.

During the immediate future, research and development work will be directed toward improving efficiencies and capabilities of air cleaning systems, developing high capacity iodine removal equipment and equipment for filtration of particulates at high temperatures (above 1000°F). For example, the Harvard Air Cleaning Laboratory is developing an inexpensive method for the removal of iodine from reactors after an incident, or from the operation of chemical processing plants. The iodine collector consists of a copper gauze ("Chore Girl", a housewives' kitchen aid) which is silver plated. Both copper and silver are excellent media for fixing iodine. Since copper oxidizes in service, it is silver plated to provide additional protection against an iodine release. The silver iodide formed fixes the material at a low flow resistance, thereby making it possible to incorporate this simple device in existing air cleaning systems without

adding any appreciable fan or air mover load. These iodine reductors can provide suitable protection in reactor containment vessels and are being actively considered for installation in the existing Hanford production reactors.

The U. S. Weather Bureau continues to provide meteorological consultant services in connection with reactor site evaluation. Meteorological assistance is also provided to all Atomic Energy Commission installations on engineering and operating problems and basic research is being continued on the dispersal of atmospheric wastes and on turbulent diffusion. Research activities are directed toward a better understanding of the diffusive characteristics of the lower atmosphere, which may be translated into practical use for engineers and operators at atomic energy installations. For example, the useful application of meteorology to power reactor facilities is exemplified in the operation of the pressurized water reactor at Shippingport, Pennsylvania. Data collected to date on all standard meteorological parameters, as well as temperature gradient measurements, has provided a basis for quantitatively assessing the limitations of the site because of the high frequency of inversion conditions. In this connection, a three-dimensional wind flow study, using meteorological balloons and smoke as tracers, was conducted to measure vertical velocities and trajectories of the air currents in the area. This project, which was carried out during stable atmospheric conditions, furnished vertical motion data that was unattainable by other methods. Information obtained has been utilized in the establishment of emergency planning procedures for the pressurized water reactor.

8. PHYSICS

There are, in general, two kinds of physics activities which affect reactor work, (1) the determination of the values of fundamental contents which can strongly influence the course of reactor design, and (2) the improvement in the knowledge of the values of reactor constants which result in increasing accuracy in predictions of performance of specific reactor designs. An example of the former class is the determination of the eta of U-233 - unless this number is sufficiently larger than 2.00 there would be no point in attempting to design and construct a thermal breeder reactor. Typical of the second class is the precision measurement of the cross section of the fission product Xe-135 which permits a more exact specification of the performance requirement of the control system for a high powered thermal reactor.

The activities discussed in this section are: reactor statics studies, control rod studies, reactor kinetics studies, fundamental neutron physics, shielding and technical publications.

a. Reactor Statics. Static reactor physics studies utilize critical and sub-critical lattice assemblies to determine such things as the initial fuel loading and size of various reactor types, their neutron economy, neutron spatial distribution and energy distribution, the relative power distribution, the worth of control materials, and the variation of reactivity with temperature voids, depletion or breeding of fissionable materials, and fission product poison buildup.

Reactor lattice studies on low enrichment uranium metal systems have been under way on light water moderated systems for several years. Some work has been done on heavy water moderated systems, and more recently started. Some work is being done on organic moderated

systems.

Reactor lattice studies have recently started on low enrichment uranium oxide rods in light water and on thorium oxide, uranium-235 oxide rods in light water.

Criticality measurements on idealized geometries, such as bare spherical tanks of fuel solution, are performed to check out theoretical methods of analysis.

b. Control Rod Studies. The present situation in the theory of control is as follows: for a reactor not bearing a great degree of similarity to reactors already studied experimentally it is possible, with the use of elaborate computing machine programs and data from critical experiments on the reactor lattice under consideration, to calculate control system strength to an accuracy of a few percent. The use of simplified calculation methods with no critical experiments can at best give results accurate to ± 50 percent in the control strength. The use of elaborate computing machine programs and data from critical experiments on reactor lattices similar in some respects to the lattice under consideration leads to accuracies that are dependent on the degree of similarity of the experiments to the case at hand. For the past year an intensive program of theoretical and experimental studies has been under way. This program has as its objective the development of simplified calculational methods and certain input data which can be applied to a wide range of power reactors and produce results accurate to a few percent.

c. Reactor Kinetics. Controlled variations in neutron flux are discussed under this heading of Reactor Kinetics. Violent changes such as occur in potentially dangerous transients are studied under the

Atomic Energy Commission's reactor safety program.

Rapid flux transients are being produced by injecting pulses of neutrons from accelerators into reactor lattices and samples of moderator materials. These experiments are studying neutron thermalization, diffusion, and the energy distribution of neutrons leaking from moderators and reactor lattices.

Slugs of thorium metal are being irradiated in the MTR and removed periodically for testing. In tests made during such removals, the amounts of thorium converted to U-233, and the changes in the absorption cross section due to fission product buildup, etc., are being determined by reactivity measurements in a nearby critical facility, the Reactivity Measurement Facility.

d. Fundamental Neutron Physics. Experimental measurements are continuing on the fundamental physical constants of the fission process, and on the behavior of neutrons through their life cycle, from birth in fission to leakage, or moderation, thermal diffusion and absorption.

The theoretical capability of a thermal reactor to breed more fissionable material than it consumes is determined largely by eta, the ratio of the number of fast neutrons produced in the fissionable isotope. Experimental values of eta have been determined for the more common fissionable isotopes, including even plutonium-241. However, the fraction of the neutrons available for breeding is so small that any appreciable uncertainty in the experimental value of eta for a particular isotope has a strong effect on estimates of breeding rates. Therefore, an intensive effort has been made to reduce the uncertainty in eta for uranium-233, the most promising thermal breeder material. It is now possible to say with confidence that eta for U-233 is quite close to

2.28 and is being determined to an accuracy of within 1% and perhaps better.

An experimental program is under way and some results are available on the energy spectra of gammas from fission, including both prompt gammas emitted by the fission products at short times after fission.

As reported definitively at the June 1959 meeting of the American Nuclear Society, recent work on the slowing down of fission neutrons at Argonne National Laboratory has greatly improved the agreement between the experimental and theoretical neutron slowing down distributions in water.

The process of thermalization in which the neutron slows down from energies slightly above thermal to energies at which it is in thermal equilibrium with the moderator, is under intensive study. There are experiments now under way that will make direct measurements of the energy distribution of neutrons leading from moderator materials, and determine how such energy distributions vary with moderator temperature.

By injecting a pulse of neutrons into bodies of neutron moderator and observing the rate at which the leakage flux decreases with time, it is possible to determine, for a moderator, the thermal neutron diffusion coefficient. Studies utilizing this technique are under way in beryllium, graphite and water.

Neutron cross section measurements are supported largely by the Research Division of the United States Atomic Energy Commission. However, other Divisions, such as Military Application, Production, and Reactor Development also support programs intended to obtain measurements needed in their particular fields. For instance, the Reactor

Development program supports measurements needed for civilian or military power reactors, research reactors, and test reactors.

Two types of cross section measurements are being made: differential measurements and integral measurements. In the former, the cross section is measured as a function of neutron energy by exposing a sample to monoenergetic neutrons. In the latter, an average cross section is determined by exposing a sample to the energy distribution of neutrons in a reactor core.

Considerable emphasis is being given in the differential fission cross section measurements to the fissionable materials, including uranium-233 and plutonium-241. Where resonances are observed in the epithermal region, they are being analyzed to obtain resonance level parameters. Capture and inelastic scattering cross sections are being measured in the fast neutron region, and total, absorption, activation and scattering cross sections are being measured in the epithermal and resonance neutron reactions. Inelastic scattering at and near thermal energy is being studied because at these low neutron energies moderator atoms begin to show chemical binding effects.

In the high energy region of interest to the fast reactor program, work on uranium-234 and -236, and plutonium-241 and -242 will place particular emphasis on energy resolution in order to define the inelastic scattering levels in competition with fission.

Integral cross section measurements are being supported at Argonne National Laboratory, Phillips Petroleum Company, and Oak Ridge National Laboratory. At Argonne National Laboratory, a Van de Graaff accelerator is being used as a pulsed neutron source to make absorption cross section measurements in water solutions of absorber materials. ORNL is using a pulsed Van de Graaff for time-of-flight neutron studies for differential cross sections in the KEV region. A fast chopper is being

used for differential cross sections up to tens of ev.

The highly sensitive Reactivity Measurement Facility is being used in conjunction with the Materials Testing Reactor to make otherwise impossible measurements on intensely radioactive samples immediately after they are removed from the Testing Reactor.

e. Techniques of Reactor Analysis. Experimental data and advanced theoretical methods are being utilized to refine the techniques of reactor analyses. On occasion these analytical techniques are translated into codes for use on high speed computers. Typically, the CORNPONE code, which treats neutron slowing down quite rigorously, is being applied to various clean critical systems for which criticality conditions have been measured.

Theoretical work is under way on the energy distribution of neutrons diffusing between moderator regions at different temperatures. Theoretical analyses of resonance capture for various fuel element geometries is under way with due consideration to neutron scattering, energy losses in the fuel and interaction between adjacent fuel assemblies.

f. Shielding. Experimental measurements are supporting the development of techniques for shield analyses. Neutron and gamma energy and spatial distributions are being measured in shields as a function of distance from radiation sources.

g. Reactor Physics Constants Center. In addition to reporting individual experiments and calculations in AEC reports and in the technical literature, new reactor physics information and the pertinent nuclear data are being collected, digested and disseminated in an organized way by the Reactor Physics Constants Center at Argonne National Laboratory.

9. REACTOR SAFETY

The objective of the reactor safety program is to develop an understanding of the abnormal behavior of reactors, components, and supporting facilities through a systematic reasearch program.

The Reactor Safety Program includes experimental and theoretical research into known and postulated reactor hazards for all major reactor types that have been built or will be developed as part of the Commission's reactor development program. It includes studies and experiments in three major areas: Reactor Kinetics, Hazardous Chemical Reactions, and Reactor Containment. It also includes research into preventive safety measures and the development of safety features and devices intended to further assure protection of the public health and safety. The status of the Reactor Safety program is provided in the following paragraphs.

a. Reactor Kinetics. The program for the study of reactor kinetics has been carried out under the SPERT and KEWB projects. SPERT, which represents the major effort under this program, concerns itself with the experimental investigation of parameters which influence the transient behavior of heterogeneous reactors. KEWB work was directed to the study of homogeneous reactors. By carefully selecting experimental parameters which are of importance in dynamic behavior it became possible to derive mathematical models which generally describe the dynamic characteristics of all chain reacting systems. Largely as a result of the SPERT work, but also because of contributions from the kinetic studies with KEWB, BORAX, TRIGA and the GODIVA reactors, a satisfactory description of the general features of excursion behavior is now available when self-shutdown effects are taken into account. The general applicability of present theory to power reactors remains to be confirmed. This will require additional experimental work at

higher power levels and on different reactor systems. Such work will permit the development of detailed theories of the shutdown processes for each reactor system together with a means for determining the nature of the shutdown processes from static measurements and the performance of a limited number of mild transients. A more detailed account of the present status of reactor kinetic technology is provided in the following summary of work which takes into account experimental as well as theoretical effort.

SPERT I, an open tank nonpressurized water reactor, has been in operation since July 1955. The major contributions to the reactor kinetics program are derived from the analysis of the experiments conducted with it. SPERT II, which will be completed in October 1959, will permit transient tests under conditions of flow and pressurization to 300 psig. Its principal use will be to provide a comparison on the effects of neutron lifetime due to different moderator and reflector materials including heavy water. SPERT III, which is designed for kinetic tests to full pressurization (2,500 psi), achieved criticality in December 1958. Work done to date includes plant check-out, critical experiments, and engineering tests under pressurized conditions. Systematic transient testing is scheduled after the completion of the engineering tests. A pool type facility, SPERT IV, will begin operation in the Fall of 1960.

SPERT I experiments have been conducted on a total of five cores having widely differing void coefficients and thermal properties. Aluminum clad enriched uranium alloy plate type elements were used in the first four cores and variations in core characteristics were brought about by varying the plate spacing or by insulating the plates with a plastic coating. Present work, which is nearing completion, involves

tests with a stainless steel core which initially had been used in the APPR Program (Army Packaged Power Reactor). Pile oscillator studies have recently been initiated in SPERT I to permit a comparison between the information obtained as a result of transfer function analysis and the knowledge already available from the transient test work.

The experimental work on the first SPERT I aluminum core included step, ramp and stability experiments in sufficient detail to permit a general evaluation of the properties of the system for various types of accidents. The initial objectives of repeating and extending the BORAX experiments and the comparison of ramp and step accidents were essentially fulfilled by these tests. The extensive data on this core also served as a reference point in determining the influence of various parameters on dynamic behavior. Experiments on the five cores having widely differing void coefficients constituted the first phase of the general program of determining the effects of individual reactor parameters.

The principal analytical work performed during this period has been on the analysis of ramp rate and step transient tests conducted with the five SPERT I cores. The analysis of ramp rate transients has provided an accurate and generally applicable method for relating ramp accidents to step accidents so that the two situations can be treated as a single problem. It may soon be possible to place the treatment of accident initiation on a realistic basis by incorporating a limit on the reactivity addition rate explicitly into hazards analyses.

The analysis of step transients has provided simple mathematical models which not only correlate most of the features of the experimental data on the first five SPERT cores, but which also predict the behavior characteristics of reactors of many other types (KEWB, GODIVA, AND TRIGA).

Since these models reveal the influence of various reactor parameters explicitly (e.g., different reactivity coefficients and prompt neutron lifetimes) the general power burst properties of a system can be stated at the outset if these parameters are known. Conversely, the nature of the internal processes and the corresponding dynamic properties of a reactor can be inferred with considerable accuracy from the observed behavior of a few transient bursts. This means that the applicability of many of the experimental and analytical results should extend beyond the class of heterogeneous water moderated reactors.

Theoretical work has also been performed on oscillatory behavior and the characterization of the initial power burst. Work on oscillations resulted in a feedback equation with three different constants which are related to the thermal, hydrodynamic and other processes occurring within the reactor. The theory provides a means of predicting the oscillatory behavior of the power from the time constants of the detailed processes and a method of determining the values of the constants from the analysis of transient data. It has been shown that whenever the constants were less than a certain value, undamped oscillations could not occur.

Other work on the analysis of initial power burst behavior, has shown that some of the properties of a power burst in a reactor are independent of either the feedback mechanism or the pile kinetics, and may be described quantitatively with no assumption other than that the pile kinetic equations are nonlinear. The theory refers primarily to the shape of the burst and is applicable chiefly to the faster transients observed in SPERT, KEWB, AND BORAX.

KEWB, a 50 kilowatt homogeneous research reactor, was constructed to provide transient test data applicable to liquid fuel systems and has

been in operation since July 1956. The objectives of the experimental and analytical studies were to investigate and reach an understanding of the kinetic behavior of aqueous homogeneous reactors. Principal attention during the kinetic tests has been given to the following parameters: amount and rate of reactivity insertion (to a maximum of 4% delta K), initial core pressure and temperature, void volume in the core vessel, and initial power.

The dynamic response of the reactor to reactivity inputs which give rise to transients with stable reactor periods down to two milliseconds, has been examined and considerable data have been accumulated. Qualitatively, the behavior of this reactor is now quite well understood. The knowledge required to accurately predict the behavior of homogeneous reactor systems of different design is, however, not yet available. Although significant progress toward the latter objective has been made, particularly of late, the basic obstruction in the attempt to predict the dynamic behavior of this reactor type is the inability to account for the microscopic origin and subsequent growth of gaseous voids in the fuel solution. This phenomenon was shown, by analysis of the experimental results, to indisputably account for reactor shutdown in the most severe excursions, while ironically producing a major source of hazard. Shutdown of the reactor results from expansion of the fuel solution which enhances neutron leakage and thus lowers the multiplication of the assembly. The expansion required in transients resulting from large reactivity inputs is accomplished principally by the production of radiolytic gas within the solution. This expansion occurs so rapidly that a pressure wave is impressed upon the reactor core vessel. Since containment of fission products is the prime consideration in hazard evaluations this pressure

wave cannot be tolerated if the magnitude is such that the integrity of the core vessel may be compromised. Hence, it is necessary to understand thoroughly the details of the generation of these gaseous voids--their origin, their growth history, and their character at all times during their evolution. Without this information the analysis of potential hazards of homogeneous reactors must rest largely on empirical data, which allows virtually no latitude in application of the data to systems other than the equivalent of the test prototype.

It is expected that the solution of the radiolytic gas problem as a shutdown mechanism will be aided by completion of tests with the present spherical core and continued investigations with a core of cylindrical geometry. Tests with the cylindrical core are scheduled to begin in FY 1960. It is not anticipated that the work done with the present low power systems will be applicable to a full understanding of the kinetics of homogeneous power reactors which operate at high pressure and temperature. Plans for extending the KEWB program into the power reactor region were initiated. However, due to the stretchout and delays which are expected in the development of homogeneous reactor systems for power production, no such program is presently contemplated.

Regarding the status of reactor safety work outside of the SPERT and KEWB activities, the following statements can be made. Considerable work throughout the reactor field on the high power transfer function with feedback specifically included has brought this phase of work to an advanced state of development. For fast reactors, extensive theoretical studies have been carried out. Except for the GODIVA experiments, only a limited amount of experimental excursion work has been performed and this was with EBR-1. Extensive transfer function measurements and analyses have

been performed with this reactor. Many isolated experiments, such as those conducted with the TRIGA reactor, contribute to the general fund of knowledge but do not have specific application to presently important classes of reactors. For example, very little work has been done on organic and sodium graphite systems which are generally presumed to be "safer" than pressurized water systems.

An important part of the reactor kinetic studies is the understanding and assessment of the consequences of accidents. A correspondingly large theoretical effort has been expended in the industry on this problem but the experimental reactor program has been limited to the one BORAX destructive test and the EBR-1 core meltdown.

The non-reactor criticality problems that arise in the fabrication, transportation, storage, and chemical processing of fissile material have not been overlooked but the approach has been from a restricted point of view consisting mainly of theoretical calculations and measurements of static nuclear properties of various arrays of material. There has, in addition, been much experimental work on measurements of static nuclear properties of solutions which is particularly applicable to chemical processing problems. However, no dynamic studies have been made.

b. Hazardous Chemical Reactions.

(1) Metal-Oxidation - Ignition Studies. Many spontaneous ignitions of uranium, zirconium, thorium, and plutonium have occurred. The frequency of these explosions is high enough to constitute a serious hazard. A summary of the research work undertaken to study this phenomena is as follows:

The properties of zirconium were studied in an effort to determine those factors affecting pyrophoricity. The studies were

empirical, with emphasis on determination of the hazards involved in manufacture, shipping, storage, and handling. Results of impact, friction, and high velocity shock waves indicated that zirconium distillation residues have the same order of magnitude sensitivity level as TNT, which is fairly insensitive.

Present work is directed to a carefully controlled systematic study of the variables involved in metal-oxidation - ignition studies. The effects of temperature pressure, metal purity, metallurgical history, gas composition, type of oxide film, thermal conductivity, specific area (degree of subdivision), temperature gradient, and mass, represent some of the factors under experimental investigation.

Three basic types of experiments have been conducted on the materials of interest. The first is isothermal oxidation at various temperatures. The effect of each of these factors can then be investigated in turn to determine those which significantly speed up the oxidation process. The second is determination of the ignition temperature by programming a furnace for a certain temperature rise per minute and determining the temperature at which the sample ignites. This ignition temperature can also be correlated with the probability of spontaneous ignition and the effect of parameters studied. The third is the study of burning rates, or the rate at which an ignition front proceeds along a strip of metal after it is ignited at one end. The rate at which it proceeds can then be used to determine the effects of the parameters studied. To illustrate the thoroughness of these studies, the effects of 1, 2, and 4 a/o additives on zirconium isothermal oxidation have been studied for over 25 additives. The present status on the various materials may be

tabulated as follows:

Ignition and burning experiments on uranium are almost complete. Isothermal oxidation studies up to 300°C were completed. At this temperature thermal cycling becomes a problem.

Isothermal oxidation studies on zirconium have been completed. Ignition temperature studies are underway and the emphasis has been placed on the effect of additives to powders. Propagation rate studies are nearing completion.

Work on thorium and plutonium remains to be done. Studies of irradiated materials will begin after completion of the plutonium work.

The possibility that zirconium carbide and zirconium hydride are the initiating agents for zirconium explosions is also being investigated. The oxygen concentration required to spontaneously ignite zirconium at various temperatures constitutes one of the principal parameters in this task.

(2) Metal-Water Reactions. In the event of a core meltdown in a water cooled reactor, a very high chemical energy release could result from one or both of the following events: (a) reaction of the cladding with the coolant releasing hydrogen, and (b) reaction of liberated hydrogen with oxygen to form an explosive mixture.

It can be shown that the chemical energy which can be released from these reactors is greater than the nuclear energy released during a reactor excursion. A number of research tasks were initiated to study the nature and mechanism of metal-water reactions. The status of these tasks

follows:

Studies with aluminum have shown that the reaction between aluminum and water below 1200°C is negligible. Experiments with zirconium and Zircalloy 2 showed essentially no reaction at their melting points.

Another task concerned the investigation of damage caused by extremely rapid steam formation produced by a thermal transient. Steam explosions were initiated by the rapid electrical heating in water of copper, zirconium, silver, iron, and aluminum. Only in the case of zirconium was the steam explosion followed by evidence of a metal-water reaction. These tasks provided insufficient information for thoroughly understanding and predicting metal-water reactions and additional work was started.

Present work will emphasize both laboratory and in-pile experiments. In the laboratory tests, the metal-water reactions have been approached as nonisothermal reactions. Since there is no single experimental technique which can accurately provide the data needed, the closest approximations will be used. The methods chosen are melting of wires by condenser discharge in steam or water and using solenoid valves to allow rapid steam contact with a molten metal sample heated in vacuo.

In-pile tests using nuclear heat to melt down a fuel element in an autoclave to obtain correlations of the laboratory studies will be performed in the TREAT reactor.

The only experiments run to date have been the condenser discharge studies of .060" wires for zirconium, aluminum, uranium and stainless steel. At temperatures up to the melting point of zirconium oxide, the pressure and temperature rise was extremely rapid.

One task which is nearing completion concerns both the study of chemical kinetics of metal-water reactions and the analysis of nuclear

excursions to determine conditions which favor a metal-water reaction. Materials considered in this study were sodium, aluminum, and zirconium. Accomplishments have included theoretical and experimental determination of the maximum rate of reaction possible between liquid metal and water vapor.

(3) Fission Product Release Program. The amount of activity in the core of a reactor after it has reached high burnup levels is large enough to have serious consequences to a widespread area if totally released and uniformly distributed. Furthermore, very little is known about the percentage of gases and particulate matter released and the conditions affecting release. As a consequence, engineers must design containment shells and core vessels which would be safe in the event of this total release, and safeguards in general are predicated on what may be an overly conservative estimate.

Prior to the initiation in 1957 of experimental programs for the determination of fission product release at Oak Ridge National Laboratory, it had been necessary for reactor engineers to assume 100% release of the fission products formed in the core. The immediate objective in the early stages of this work was to obtain data on the release phenomena so as to determine what assumptions of activity release were realistic. The program, until now, has therefore concentrated on measuring the activity release of relatively low burnup material by slow meltdowns (ca. two minutes).

These exploratory experiments have been performed with aluminum Geneva type (dispersion) fuels, APPR type (SS-UO_2) fuels, GE-HTRE fuels, MTR type (uranium-aluminum) fuel, STR type (uranium-zirconium) fuel, and PWR type (UO_2-zirconium) fuels.

The results of these experiments have shown that there is some retention by the melt of fission product gases, which is very strongly dependent on the melting temperature, and on the time of heating to melt. The variation in total fission product release has been from approximately 10% to 95%.

This first stage of the program is now essentially complete, and the second stage, systematic studies of the important parameters, will begin on or about September 1, 1959. The future plans are outlined in another section of this report.

In addition to the Oak Ridge National Laboratory studies, the following unclassified work has been done and may be of interest. It may be noted that in all these experiments the accident postulated is the loss of coolant accident; the nuclear excursion accident has not been considered.

One task concerned fission product release from a uranium-zirconium PWR type fuel assembly. Of the major constituents, 42 - 58% of the xenon, and as much as 25% of the iodine were released. This qualitatively agreed with the Oak Ridge National Laboratory studies.

Field release tests of fission products are currently under study in a separate task.

Oxidation tests together with simultaneous release of fission products using irradiated natural uranium fuel material constitute still another task.

Lastly, pyrometallurgical processing studies are underway at Argonne National Laboratory. Because of the lack of experimental data on fuel meltdowns, the pyro-processing work at Argonne National Laboratory has been followed closely, since the variables which affect fission

product release in this process will probably be important in catastrophic fuel meltdowns.

c. Containment. Containment costs are estimated to be about ten percent of the total cost of a nuclear power plant so that any increase of knowledge, which would justify a significant reduction in safety factors, could reduce capital costs.

Vapor containment shells are employed to assure retention of radioactive fission products even if they are released from a power reactor in the unlikely event of a maximum credible accident.

In computing the maximum energy that must be restrained, practice varies somewhat, depending upon the type of reactor. However, it is usual to combine the following sources: (1) the initial nuclear energy release, (2) a possible chemical energy release (such as a metal-water or metal-gas reaction in the core), and (3) the thermal and mechanical energy stored in the reactor coolant. Under the worst conditions, the reactor vessel itself is considered to have been ruptured in some fashion so that the coolant spills into the building and the released energy generates a pressure on the outer containment shell. An adiabatic transfer of energy is frequently assumed but a significant reduction in pressure is possible in some cases by considering the transfer of heat to components within the building.

The vapor container itself is designed for the calculated pressure according to Section VIII, Unfired Pressure Vessels, of the Boiler Code issued by the American Society of Mechanical Engineers. This code limits the design stress to roughly one-fourth of the ultimate tensile strength of the material. The large safety factor of containment systems arises

from the pessimistic assumptions in the postulation of the accident and the low design stresses allowed by the Boiler Code.

As a conservative measure, the design is often made to withstand an equivalent amount of energy released from an explosive since the rapid loading from explosives usually does a greater amount of damage. Scale models have been built and tested with explosives in some instances in order to verify the adequacy of structures. Special blast and missile shields are employed on occasion to absorb the explosive energy and prevent damage to vulnerable items either from blast or from missiles generated in the accident. Concrete lining of vapor shells can be used both as radiation shielding and protection against missiles.

The present technology is deficient, not from the standpoint of safety, but in the overdesign that is now necessary due to lack of more accurate information on: (1) possible accidents, (2) the loads generated, and (3) the response of structures to the loads.

The total amount of energy release and the variation of the rate of release with time represent two major unknowns in the loading phase of the containment problem. It is, of course, possible to obtain bracketing values as is the present practice; however, more accurate knowledge may make it possible to decrease the strength of containment structures with no sacrifice in safety.

Even if the timewise loading on the reactor pressure vessel were known, there is still not a great deal of information on the dynamic rupture strength of pressure vessels under rapid loading. Greater knowledge in this field would make it possible to predict the rupture loads more accurately and also to compute the amount of energy dissipated in the rupture.

For the majority of incidents, it is doubtful whether shock waves will be generated. However, present data are not sufficient to rule out the possibility of this type of loading and blast phenomena are studied as an upper limit. Shock wave theory has made tremendous strides within the last fifteen years, due largely to the advent of supersonic flight and atomic bombs but it still remains a young science. Improved knowledge is needed in shock propagation in various materials and the properties of these materials under shock loading. The behavior of some simple structures to shock loading can be calculated reasonably well but these structures seldom occur in problems of practical interest and it is therefore necessary to resort to experiments, either on models or full scale structures.

Only very small leak rates can be tolerated from containment structures due to the necessity of preventing the release of excessive amounts of radioactivity. These leak rates are so low that their measurement becomes a significant problem both in instrumentation and technique. Special precautions are needed in measuring average temperatures since they affect leak test results. Improved and less time-consuming techniques are needed. Raw data on the permeabilities of different building materials are also scarce.

Statistical information on occurrences of failures, ruptures, and other types of accidents is very meager due largely to the suppression of such data by embarrassed firms. A full reporting of such incidents could be extremely helpful in advancing the state of knowledge but acquisition of such information shall probably continue to be difficult.

It is likely that model testing will continue to be the prime source of good data on the probable behavior of a prototype structure. Some energy sources are available for generating various pressures as a function of time but their development needs to be continued to obtain a greater variety.

Schemes for absorption and dissipation of energy appear to be one of the most advantageous means of reducing the cost of containment. Heat sinks, crushable materials, and absorptive linings need to be studied in more detail.

The following programs have been undertaken to establish a more realistic basis for containment design. The work can be roughly separated into the following categories: (1) dynamic and static rupture of pressure vessels, (2) blast transmission, (3) blast loading of vapor containers, (4) mechanics of penetration and fragmentation, (5) earthquake loading, (6) leak testing of steel vapor containers, and (7) leak testing of conventional building structures. The following work is currently underway:

One task concerns the performance of fundamental experiments on the static and dynamic rupture of simple cylindrical shells.

Earlier studies under this task have been made on (a) the energy absorbed in crushing of wood and celotex in a blast shield and (b) the behavior of a reactor installation for a postulated release of energy. In the latter study, an upper limit on the motion of a reactor shield plug has been obtained by assuming that all the energy release was converted to air pressure for driving the plug. It is estimated that

the actual motion of a plug would be a great deal less due to energy absorption by the coolant which was neglected in the simplified theory.

Another task involves general investigations into the problems of blast generation, transmission, and attenuation. Theoretical studies have been made of the chemical energy release and the generation of steam in a reactor excursion. General investigations are being made in blast theory which should lead to improved methods of calculation. Theories of internal dynamic loading and yielding of simplified structures are also under study.

Another study concerns internal blast loading on vapor containers. Initial studies were made on a sphere because its simple geometry made it possible to develop a theory for comparison with experimental results. Tests within the elastic range, for which the theory was valid, showed good agreement, considering the complexity of the problem.

There is some discussion on the applicability of model scaling laws for structural deformation in the plastic range. Cantilever beams have been loaded in the plastic range by blast waves and the evidence indicates that the model scaling laws can be used for engineering purposes even with large permanent deformations. Work has recently been done on loading of a sphere in the plastic range. The effect on stresses in the tanks resulting from such supporting media as water and earth is now being studied.

Fragmentation of material and penetration of steel plates by rods is being studied as a separate task. Control rods and similar structures are efficient penetration devices when propelled longitudinally since they present a large mass behind a small frontal area. Penetration tests were initiated for rod velocities up to 400 feet per second which is roughly

the maximum velocity expected from any accident. The metallurgical condition of the steel has been found to have a large influence on the penetration resistance of the plate even though ultimate strength and ductility of the material is otherwise the same. Energy sources are being developed from explosives and pyrotechnics for use in testing small models where various pressure-time loadings are needed.

The difficulties arising from the earthquake loading of reactors have been studied as still another task.

Some work has been done on methods for calculating the vapor pressure inside a containment shell due to the adiabatic release of pressurized water. Tables and charts have been prepared.

Work is underway to investigate the permeability of building walls, panels, and construction materials. This task will also concern the study of methods for leak testing of industrial buildings.

A preliminary study of underground containment has been made by the Division of Construction and Supply of the Atomic Energy Commission. The costs are estimated to be 3% to 7% higher than the total cost for a plant constructed above ground with conventional steel vapor shell containment.

V

Civilian Power Reactors Program Costs

FY 1950–FY 1959

V. CIVILIAN POWER REACTOR PROGRAM COSTS

Charts have been prepared to show the total cost, trends and emphasis of research, development and construction for the Civilian Power Program for the period FY-1950 through FY-1959. These totals show the actual dollars spent in each project activity and the total dollars spent in nuclear technology. However, there is one very important feature which cannot be expressed in such a specific cost presentation, i.e., the benefits derived from one activity (where the costs are charged) are applicable to other activities. For example, the core material zirconium was developed and charged largely in the Naval Reactors Program. To a lesser degree, there were zirconium development costs in the general nuclear technology program and the pressurized water Shippingport Atomic Power Station costs. The Boiling Water Reactor program, sodium-graphite reactor program and any other reactor concept utilizing zirconium have benefited significantly and directly from these successful zirconium development programs without cost. The same is true of other materials and components throughout the Commission's total program such as UO_2, sodium technology, components, instrumentation, etc.

Cost reports on Nuclear Technology programs were reviewed back through FY 1950 to accumulate costs expended solely for civilian power reactor development. It was not always clearly indicated just what activity was the primary benefactor from a general development program. In those areas such as hot cells where they were built for joint use of the military programs and civilian programs, the costs were not included.

Another large cost area that has not been included in this report are the costs of building the MTR and ETR and CPP. Any costs of irradiation services and fuel processing directly chargeable to a civilian power reactor are included.

No cost totals are included in this report showing the investment of private industry in non-operative arrangements or nuclear development programs.

Figure 32 shows annual costs and the relative expenditure growth of the Atomic Energy Commission's research, development and construction program coupled with the Co-operative Arrangements program in which industry costs on specific power reactor projects are included. (See legend for total.) This figure includes the division between research and development expense and capital (construction) costs.

Figure 33 shows the same annual totals as are shown in Figure 32, however, they are broken down by the cost of AEC Nuclear Technology, AEC Experimental Program, AEC Co-operative Arrangements Cost and Industry Co-operative Arrangements Cost. This figure shows the gradual shift in emphasis to the Co-operative Arrangements program. The legend explanation for Figure 33 is as follows:

AEC Experimental Projects Costs include all direct design, research, development and capital costs expenditures for Civilian Power Reactors and support facilities within the following project activities:

1. Pressurized Water
2. Boiling Water
3. Homogeneous
4. Fused Salt

5. Liquid Metal Fuel
6. Fast Breeder
7. Sodium Graphite
8. Organic Cooled and Moderated
9. Plutonium Recycle
10. Heavy Water
11. Gas Cooled

Contributions and/or investments by private organizations also included in these totals are, as follows:

Atomics International - SRE	$ 2.80 million
Atomics International - OMRE	.75 million
Westinghouse Electric - Shippingport	.50 million
Duquesne Power - Shippingport	17.50 million
Total	$21.55 million

AEC Nuclear Technology Costs include all general design, research, development and capital cost expenditures carried out in support of all civilian power reactor projects and those leading to the establishment of a new power reactor project. The various categories include Physics, Coolants, Heat Transfer, Instrumentation and Control, Shield Development, Manuals and Data Books, Reactor Safety, Fuels and other core materials, Separations Systems and Waste Systems.

Any cost associated with the Materials Testing Reactor (MTR), Engineering Test Reactor (ETR) or the Chemical Processing Plant (CPP) at the National Reactor Testing Station have not been included. It is recognized the Civilian Power Reactor program and industry benefit from these facilities; however, the prime justification and users of

these facilities are the military reactor programs. Similarly, any other smaller facility, such as hot cells, that was not built exclusively for civilian use has not been included.

AEC Co-operative Arrangements Costs include the direct research, development and construction costs, as appropriate, for the following Co-operative Arrangements:

Yankee Atomic Electric	R&D only
Enrico Fermi	R&D only
Consumers Public Power	R&D and construction
Rural Co-operative Power Association	R&D and construction
City of Piqua, Ohio	R&D and construction
Chugach Electric Association	R&D only
Northern States	R&D only
Carolinas-Virginia NPG	R&D only
Florida NPG's	R&D only

There were no costs reported through FY 1959 on the Philadelphia Electric arrangement. Not included in the totals shown was any waiver of fuel use or heavy water use charge.

Industry Co-operative Arrangements Costs include all research, development and construction costs through FY 1959 reported by the private organizations under contracts with AEC in the Co-operative Arrangements program:

Yankee Atomic Electric	R&D and construction
Enrico Fermi	R&D and construction
Consumers Public Power	Construction only

Rural Co-operative Power Association	Construction only
Chugach Electric Association (NDA)	R&D only
Northern States	R&D and construction
City of Piqua, Ohio	No costs reported through 6/30/59
Carolinas-Virginia NPG	R&D only
Florida NPG's	R&D only
Philadelphia Electric	No costs reported through 6/30/59

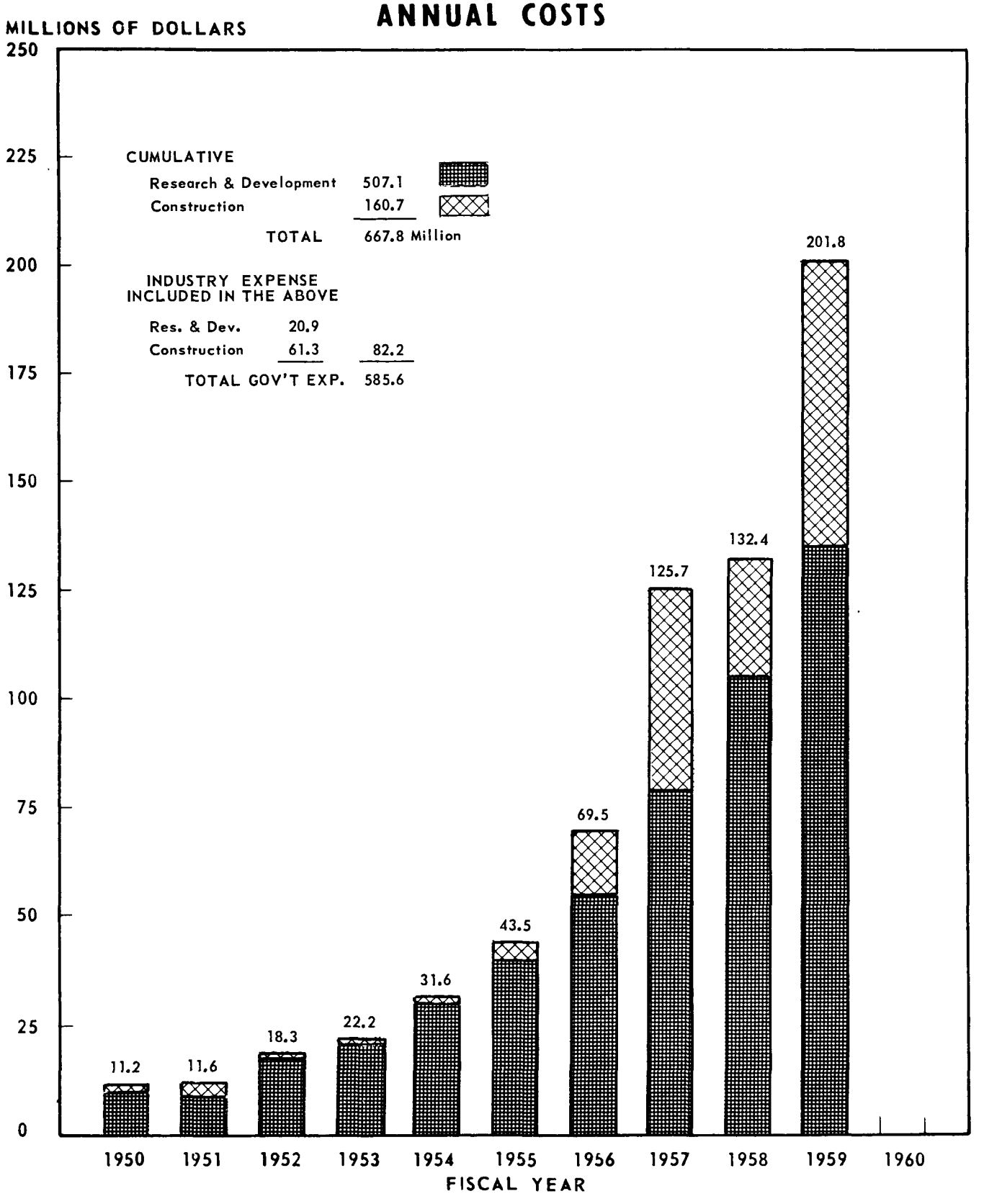
Figure 34 shows the cumulative costs of research, development and construction for each major activity in the Civilian Power Reactor program. This chart shows where the emphasis of the Commission's program has been placed between FY 1950 and FY 1959.

The costs related to each reactor classification is shown in Figure 35.

CIVILIAN POWER REACTOR PROGRAM

RESEARCH - DEVELOPMENT and CONSTRUCTION

ANNUAL COSTS



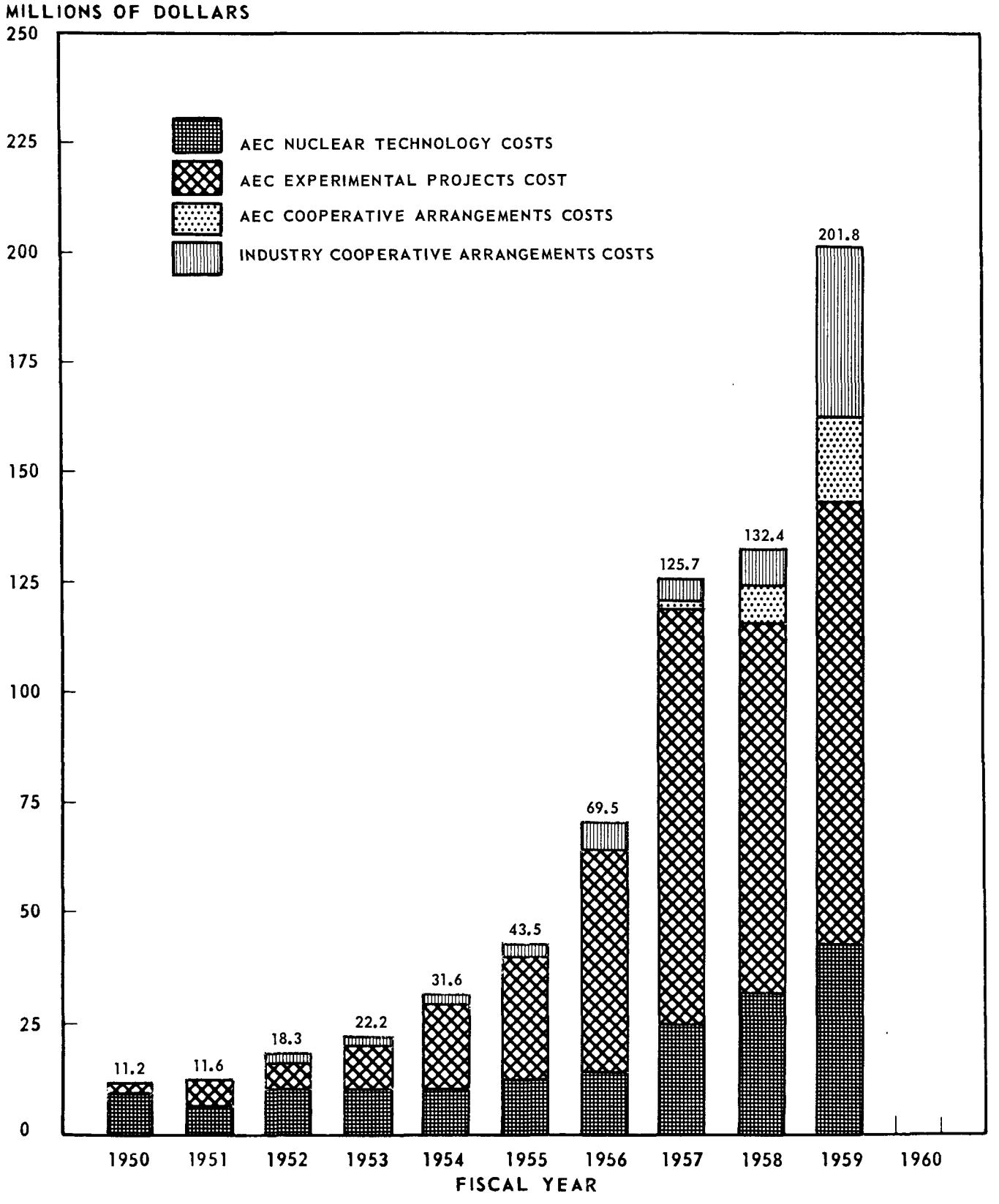
SEPT. 1, 1959

FIG. 32

CIVILIAN POWER REACTOR PROGRAM

RESEARCH - DEVELOPMENT and CONSTRUCTION

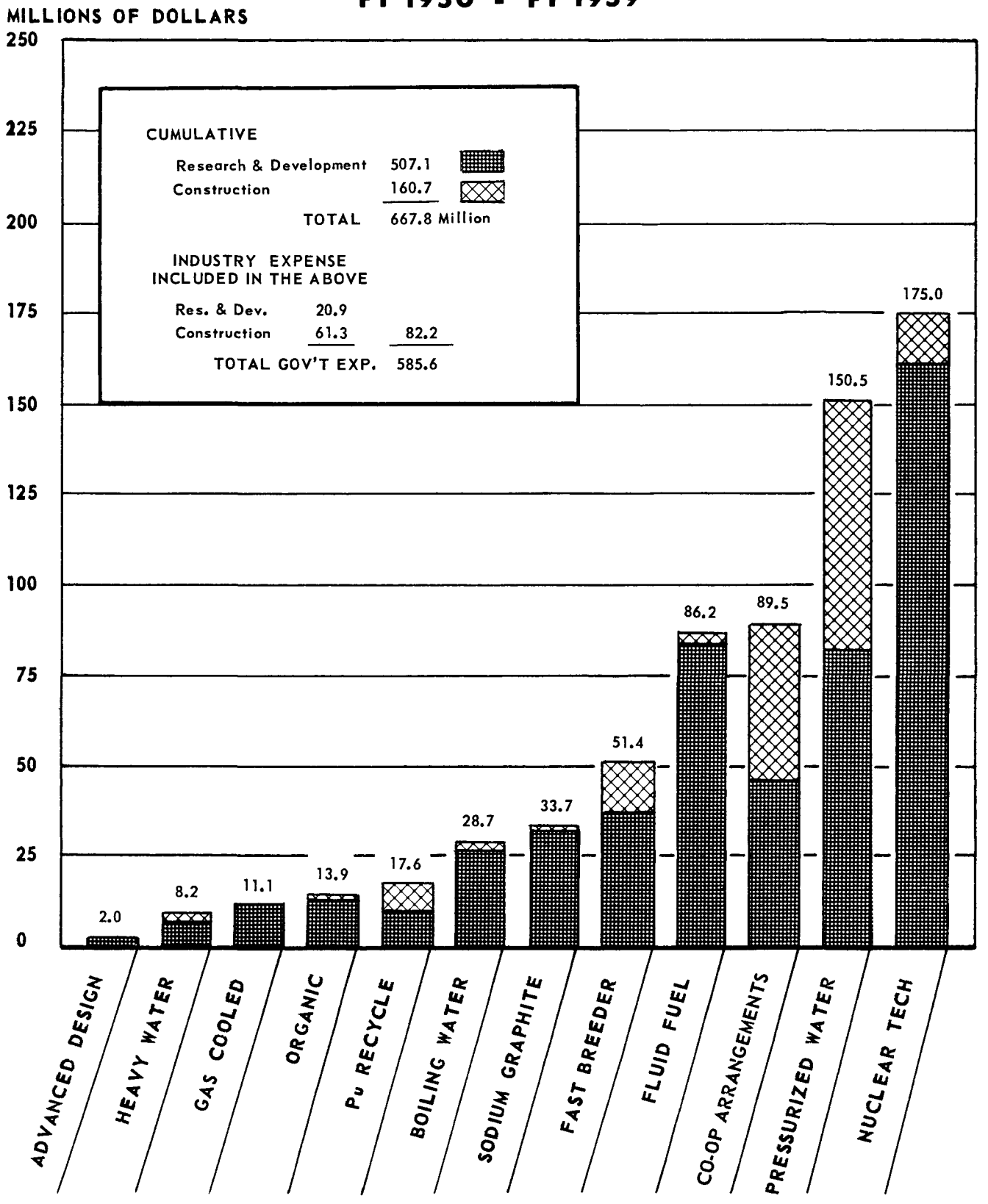
ANNUAL COST - SUB PROGRAM



SEPT. 1, 1959

FIG. 33

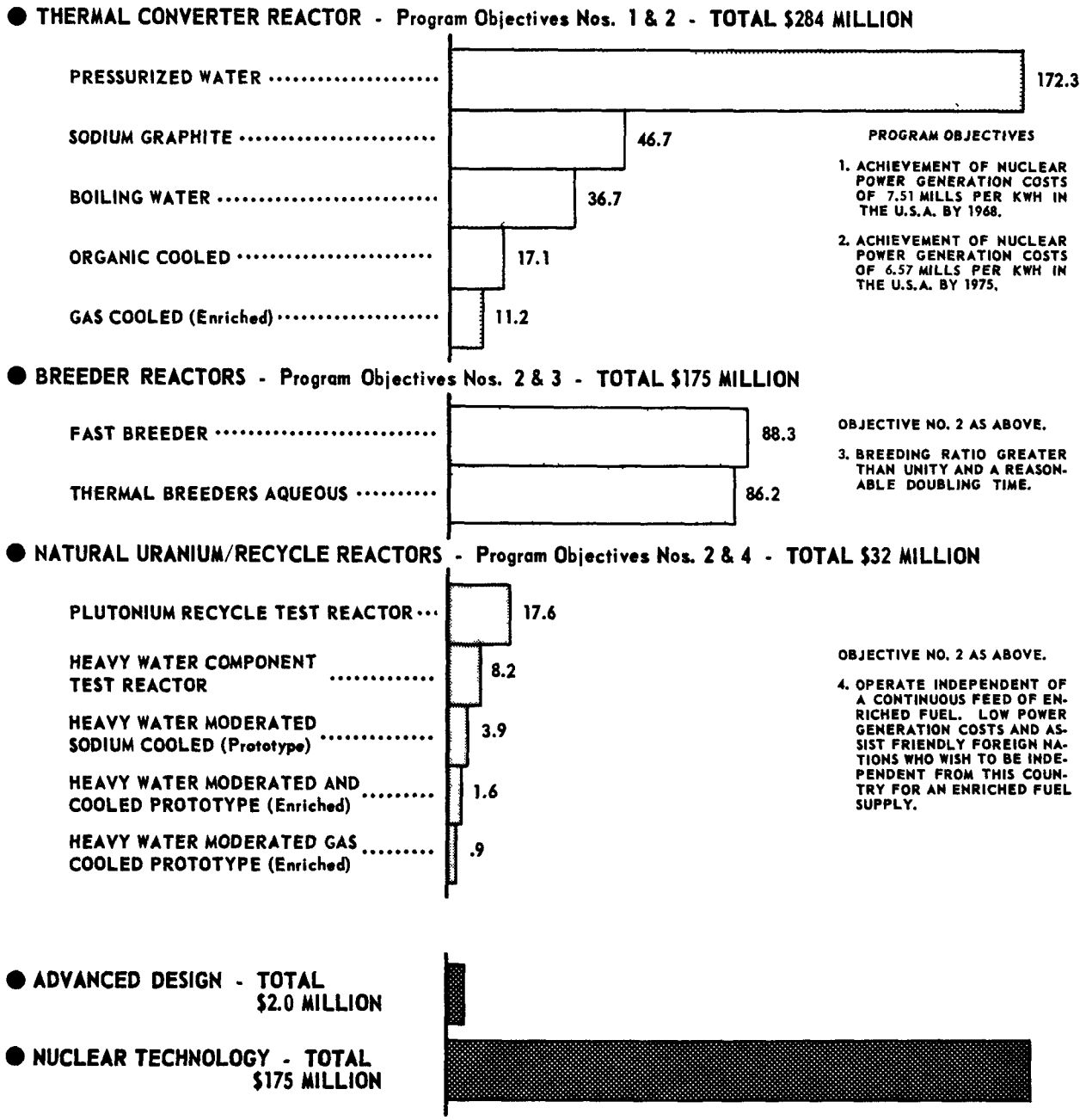
CIVILIAN POWER REACTOR
RESEARCH & DEVELOPMENT & CONSTRUCTION
CUMULATIVE COST BY ACTIVITY
FY 1950 - FY 1959



SEPT. 1, 1959

FIG. 34

CIVILIAN POWER REACTOR
RESEARCH & DEVELOPMENT & CONSTRUCTION
CUMMULATIVE COST FOR EACH REACTOR CLASSIFICATION
FY 1950 through FY 1959



SEPT. 1, 1959

FIG. 35

VI

Appendices

APPENDIX I

The organizations and the key personnel who prepared the detailed technical status reports on which this summary was based are listed below:

Organization	Name	Report Area
GNEC (Combustion Engr)	John M. West Frank Bevilacqua	Boiling Water
APDA	A. Amorosi John Yevick	Fast Breeder
A.I.	Robert Dickinson Ray Beeley	Sodium-Graphite
A.I.	William Parkins Robert Wilson	Organic
Westinghouse	W. E. Shoupp Ted Stern	Pressurized Water
ORNL	Robert Charpie	Gas Reactor
ORNL	Beecher Briggs	A. H. R.
du Pont	Dale Babcock	D_2O Reactor

The above personnel were ably assisted by many capable members of their own organization. Some of the information contained in the reports was supplied by other organizations, such as ANL, Alco, the Martin Company, Babcock and Wilcox, Allis Chalmers, Nuclear Development Corporation of America, Sargent and Lundy, AECL, Los Alamos Scientific Laboratory, General Atomics, UKAEA. In addition to the above personnel, many members of the office of Civilian Power and Nuclear Technology provided valuable data and assistance for this report.

APPENDIX II

Cost Backup Material (Summary)

The nuclear cost data on which the curves in this report are based is divided into three categories. These are capital cost, fuel cycle cost and operation and maintenance cost. The cost breakdown, the system design parameters, and rationale for these costs are included in report SL-1674. The cost breakdown and performance data for the coal-fired fuel plants are included in SL-1564. The information included in the following paragraphs is a brief summary of the groumd rules and the basis for the cost calculations.

I. Capital Cost (Nuclear Plants)

To maintain as much consistency as possible, Sargent and Lundy utilized the following basic assumptions in determining capital costs;

1. The design parameters were based on current technology.

2. Uniform costs were used throughout, with appropriate differences for size, for those components of the power plants that had the same design specifications, such as turbine-generators, condensers, heat exchangers, buildings and structures, etc.

3. Provision for service and auxiliary facilities were made on a common basis between reactor concepts.

4. Uniform construction periods for each reactor size were established for the determination of interest cost during construction, and

5. Cost estimates for all reactor types were prepared with

the same estimating practices.

Basic flow diagrams and general arrangement drawings were prepared for each reactor type, based on data in the status reports. Estimates were made based on this data, using Sargent and Lundy's normal estimating practices and using uniform costs, where applicable, in the different reactor type cost estimates. The assumption and indirects used in determining the cost are as follows:

Indirect cost percentages (based on 325 MWE plant)

General and administrative expenses	12.5% of direct cost
Engineering, design and inspection	14.6% of direct cost plus G & A
Startup cost based on $4\frac{1}{2}$ months of operating cost	
Contingencies	10% of direct and indirect
*Interest during construction	8.1% of total cost

*Based on 6% simple interest rate on money and an estimated expenditure curve over the 36 months' construction period.

The capital cost portion of all power cost estimates were based on 14% annual fixed charges and 80% capacity factor.

II. Fuel Cycle Cost (Nuclear Plants)

Fuel cycle cost estimates were made for three different core sizes for each reactor type with the exception of the gas reactor where sufficient information was not available to justify an attempt to show cost over the size range.

The component parts included in the fuel cycle costs are as follows:

1. Purchase of fuel assemblies and control elements, including spares. The purchase price covers the cost of the following:

a. Conversion of the starting source and fissionable material (enriched UF_6, depleted UF_6, some form of natural uranium, thorium, U-233 or plutonium) into the desired chemical, physical and metallurgical form.

b. Fabrication of the material into the proper size and shape.

c. Cladding, sheathing, or otherwise housing fuel components, providing hardware and fittings, assembling parts into units ready for loading into the reactor.

d. Quality control analyses, inspection, handling and shipping between various points in fuel processing steps.

e. Material losses, cost of scrap recovery and rework.

f. Use charge for fissionable material during fabrication.

g. Purchase of the source material, if natural uranium, depleted uranium or thorium is used.

2. Payment for burnup of leased fissionable material.

3. Payment of the 4% per year use charge for special nuclear material.

4. Payment of shipping charges for transporting irradiated fuel from reactor site to chemical reprocessing plant.

5. Payment for reprocessing and chemical conversion which is required to convert irradiated fuel to decontaminated and pure forms of UF_6, plutonium metal, uranyl nitrate solution of U-233, and thorium

nitrate solution (if thorium is to be recovered).

6. The above charges are reduced by the credit received for the net amount of plutonium or U-233 produced in the nuclear material and delivered to the reprocessing plant.

7. The working capital necessary to keep the fuel cycle operating. The cost of working capital was assumed to be 6% of the amount of money in working capital.

The average fuel burnup and fabrication costs used are as listed in Table 1, following.

Table 1

Type	Fuel	Cladd	Average Exposure (MWD/MT)	Fab. Cost $/kg U
BWR	UO_2	Zr	11,000	140
PWR	UO_2	SS	13,000	110
OCR	U-3½ w/o Mo-0.1 Al	Al	4,500	60
SGR	U-10 w/o Mo	SS	3,000	80 - 110 (function of rod size)
FBR	U-10 w/o Mo	Zr (1/6" Pins)	15,900	480
FBR	Depleted U-2.75 w/o Mo (Blanket)	SS	---	45
D_2O	(Nat) U-Zr	Zr	3,850	50
GCR	(Nat) U	Magnox	3,000	50

The fixed cost data for the fuel cycle is as follows:

1. Reprocessing cost based on $15,300/day (8 + MT of U).
2. AEC schedule for UF_6 cost.
3. Shipping cost (irradiated fuel) = 12.45 $/kg.
4. Conversion of UNO_3 to UF_6 = 5.60 $/kg.

5. Conversion of $PUNO_3$ to Pu metal = 1.50 $/gm.

6. Buyback of Pu (credit) = 12 $/gm.

The inventory cycle was calculated for each reactor type. The use charge was based on fuel inventory not the fuel loading. The detailed fuel cycle costs are included in report SL-1674.

III. Operation, Maintenance and Insurance Cost (Nuclear Plants)

A. Operation and Maintenance. Estimates of operation and maintenance cost were prepared for a 75, 200 and 300 MWE size for each reactor type. The estimates were based on information available from Shippingport, Yankee, Indian Point, Enrico Fermi, and Dresden, and an analysis of the operation requirements for the various reactor types. The ground rules used in establishing operation and maintenance costs were as follows:

1. At this stage of development, operating and maintenance cost for nuclear plants will be higher than conventional plants of corresponding net kilowatt capacity. This is due essentially to the radiological and biological aspects of nuclear plants with their attendant safety problems together with lack of any extended operating experience on which to base judgment to achieve operating economies.

2. In considering the items making up the total cost per KWH of a nuclear plant viz, plant investment charges, fuel costs, and operating and maintenance costs, the latter item of operating and maintenance represents a relatively small proportion of the total cost, being in the order of 10%. This cost relationship merits a somewhat simplified approach in normalizing operating and maintenance costs.

3. Operating and maintenance costs were broken down into:

a. Supervision and engineering labor.

b. Station labor.

c. Fringe benefits, applicable to payroll.

d. Operating supplies, maintenance materials, and services.

e. Extraordinary items. These are items applicable to a particular type of reactor such as the makeup of heavy water in a D_2O moderated reactor, organic liquid in an organic cooled reactor plant, helium in a gas cooled reactor or special maintenance and operating techniques associated with radioactive sodium and other liquids.

B. Insurance Cost (Nuclear Plants). Insurance costs for nuclear plants are substantially higher than for conventional plants. This is due essentially to lack of insurance experience and the liability potential, both on site and off site, to property, personnel and the general public resulting from a nuclear incident which conceivably would release radioactive contaminates. The insurance coverage falls into three categories:

1. An all risk nuclear property insurance for which the rate is in the order of 35¢ per $100 of insurance coverage.

2. Nuclear liability insurance for which an insurance pool of $60,000,000 has been set up by mutually owned and stock holder owned insurance companies. The reactor owner is required to take out $150,000 per megawatt of thermal capacity of this type of insurance

coverage before the plant is eligible for government indemnity insurance. The maximum amount of insurance coverage is $60,000,000 at an annual premium of $260,000. The rates for this nuclear liability or so-called third party insurance are as follows:

Amount	Rate/Million	Annual Premium
First $1,000,000	$40,000	$40,000
Next 4,000,000	20,000	80,000
Next 5,000,000	8,000	40,000
Next 10,000,000	4,000	40,000
Next 20,000,000	2,000	40,000
Next 20,000,000	1,000	20,000
$60,000,000		$260,000

3. Government indemnity insurance at the rate of $30 per MW thermal with a maximum coverage of $500,000,000. The details of the operation and maintenance and insurance cost is included in report SL-1674.

IV. Coal-fired Plant Costs

The coal-fired plant cost data included in this report was based on the following data:

Plant Size-MWE net	25	60	100	200	325
Pressure - psig	850	1250	1450	2000	2000
Temp. Initial/Reheat - F	900	950	1000/1000	1050/1000	1050/1000
Back Pressure - HgA	1.5	1.5	1.5	1.5	1.5
Heat Rate-BTU/KWH	11980	11066	10063	9550	9471
Capital Cost $/KWE(net)	2.96	2.21	1.99	1.78	1.66
Power Generation Cost @ .7 LF.					
Fixed Charges @ 14%	6.76	5.04	4.54	4.08	3.78
Fuel Cost-35¢/10^6BTU	4.19	3.87	3.52	3.34	3.31
O & M Cost	1.19	.86	.76	.58	.42

Total Power Generation Cost M/KWH					
@ 35¢/10^6 BTU	12.14	9.77	8.82	8.00	7.51
@ 25¢/10^6 BTU	10.94	8.67	7.82	7.05	6.57

The details of the coal-fired plant cost are included in report SL-1564.

APPENDIX III

Explanation of Terms Used in Definition of Objectives

The explanation of terms and the reasoning behind the selection of specific numbers and dates used with the objectives are listed below:

Equivalent station design basis is to be interpreted as (1) the same gross electrical output, since a nuclear plant may require a different amount of auxiliary power than a fossil fueled plant; (2) the same site characteristics (not necessarily the same acreage), i.e., soil conditions, cooling water temperature and transportation; (3) fossil fueled plant to have normal, modern turbine conditions, i.e., not to exceed 2000 psia 1050°F/1000°F reheat; (6) turbines to be rated at same exhaust pressure (1½" Hg A).

High energy cost area is a relative statement; however, it is necessary to arbitrarily select a specific figure in order to compare the technological development of nuclear plants against a specific goal. There are areas in this country where the fuel cost exceeds 35 cents/million BTU's; however, this fuel cost was arbitrarily selected as a minimum high fuel cost area.

The specific power generation cost for fossil fueled plants to be used as a target for the 1968 objective is a 325,000 KWE net (348,150 KWE gross) plant rated at 2000 psia 1050°F/1000°F reheat at 1½" Hg Abs. The capital cost (single unit/station) is $166/KWE Net or 3.78 M/KWH, fuel cost (35 cents/million BTU's of 3.31 M/KWH, operation and maintenance cost of .42 M/KWH. Total power cost is 7.51 M/Net KWE based on 1959 cost. No escalation is included.

It is assumed that proper escalation and cost increases will be added to the plants (nuclear and fossil fueled) for the dates on which the comparison is to be made. Cost projections are discussed in Appendix II.

The target date of 1968 is an arbitrary date based on the best apparent compromise between current status of nuclear technology, a reasonable budget estimate and a strong desire to achieve competitive nuclear power at the earliest date.

The budget limit (dollars available) will influence the number of reactor types to be pursued for each program objective. However, the earliest date at which a reactor type may become economically competitive is limited by the status of the technology and the construction time. For example, increasing the budget by a factor of two would not reduce the time required to develop a concept to maturity by a similar amount. This is especially true of those concepts competing for economic power by 1968.

A low energy cost area was arbitrarily selected as 25 cents/million BTU's, based on the same plant conditions as described under high energy cost areas. The power generation cost for a plant installed in this area would be 6.57 M/Net/KWH (based on 1959 cost). A nuclear plant that could provide power at 6.57 M/Net/KWH would be competitive in all but the extreme low cost areas in the U.S. The target date of 1975 for a nuclear plant that could compete in low energy cost was arbitrarily based on two generations of plants ($3\frac{1}{2}$ years construction time each) after the first objective (1968) was realized.

The development of breeder reactors, recycle reactors and natural uranium burning reactors has been set as a fundamental program objective. This is due to the uncertainties in future fuel

reserves, national emergencies and the desire to maintain world leadership in the development of nuclear power for civilian application.

The most promising thermal converter concepts will be picked for future development. More than one concept should be developed so that at least one backup reactor type will be available in case the potential of the most promising concept fails to materialize. It may also be impossible to accurately determine the one concept with the most potential at this status of the technology. There is no reason, however, to continue the development of those concepts competing for the same objective that do not show the same degree of long-range potential.

A lower size limit of 25,000 KWE was set for evaluating thermal converter type reactor plants, so that the comparison could be made between fossil fueled, steam turbine type plants and nuclear plants. Below this range the nuclear plant would also have to compete with the diesel plant, internal combustion plants and other types of prime movers.