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UNITED STATES ATOMIC ENERGY COMMISSION

PRELIMINARY DESIGN AND COST ESTIMATE
FOR THE PRODUCTION OF CENTRAL STATION
POWER FROM AN AQUEOUS HOMOGENEOUS
REACTOR UTILIZING THORIUM-URANIUM-233

Edited by
H. G. Carson
L. H. Landrum

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Nuclear Power Group
Chicago, Illinois

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AEC RESEARCH AND DEVELOPMENT REPORT

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THE PRODUCTION OF CENTRAL STATION
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FOREWORD

This report covers the results of a study made by Nuclear Power Group of the Aqueous Homogeneous Reactor when applied to the generation of central station electric power.

The work covered here completes the investigation planned. The scope of this activity consisted of examining the technical and economical aspects of the reactor system using the thorium U-233 fuel cycle.

In conducting the study Nuclear Power Group was given assistance by the Oak Ridge National Laboratory and the 18 equipment manufacturers listed in the Appendix. This assistance was a significant contribution to the work of the study group.

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ABSTRACT

The design and economics of the Aqueous Homogeneous Reactor as basically under development at the Oak Ridge National Laboratory are presented. The reactor system utilizes thorium-U-233 fuel. Conditions accompanying reactor systems generating up to 1080 mw of net electrical energy are covered.

The study indicates that a generating station, with a net thermal efficiency of 28.1%, might be constructed for approximately \$240/kw and \$200/kw at the 180 mw and 1080 mw electrical levels, respectively. These values result in capital expenses of approximately 4.72 and 3.86 mills/kwh.

A major part of fuel cost is the expense of chemical processing. It is therefore advantageous to schedule fuel through a relatively large processing system since fixed charges are insensitive to chemical plant size. By handling fuel through a plant large enough for processing 200 kg of thorium per day, total fuel costs of about 1 mill/kwh result. This cost for fuel processing appears applicable to generating stations up to about 540 mw in size, decreasing to about 0.6 mills/kwh at the 1080 mw level.

Operating and maintenance expense, including heavy water cost on a lease basis, varies between 1.34 and 0.89 mills/kwh for 180 and 1080 megawatts respectively. If the purchase of heavy water is required, 0.3 to 0.4 mills/kwh must be added.

It is concluded that the Aqueous Homogeneous Reactor may produce electrical power competitive with conventional generating systems when the remaining technical problems are solved. It is felt that the research and development now programmed by the Oak Ridge National Laboratory will solve these problems and affect costs favorably.

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INTRODUCTION

Contract and NPG. This report covers a portion of the work performed under a contract between Nuclear Power Group, consisting of American Gas and Electric Service Corporation, Bechtel Corporation, Commonwealth Edison Company, Pacific Gas and Electric Company, and Union Electric Company of Missouri, and the United States Atomic Energy Commission. The contract specifies a survey and study of the Commission's reactor development activities to determine the technical and economic feasibility of the Group's building and operating a nuclear reactor for the production of central station electric power. The contract initially was executed on October 27, 1953 and was extended for one year on October 27, 1954.

Objective of Study. The objective of this study is to evaluate the application of an aqueous homogeneous reactor system to the generation of central station electric power. This objective involves the selection of a suitable reactor, the preparation of necessary plant designs, and the development of cost estimates which will permit comparing the present and future prospects for the aqueous homogeneous reactor system with other reactors and with conventional power generating facilities.

Liaison with ORNL. The nuclear calculations and designs used in this study are based on work in progress at Oak Ridge National Laboratory. This organization has most generously made available the time of its staff members to advise on various aspects of the nuclear design. Without such assistance this study could not have been undertaken.

Assistance from Manufacturers. A number of equipment manufacturers were approached with requests for design information and price estimates for major components. All responded in a spirit of helpful cooperation and made available such pertinent information as they possessed or could develop within the time available. A list of principal manufacturers who supplied information for this study is included in the Appendix.

Organization of Report. The Summary presents, in abbreviated form, the results of the NPG study of the application of aqueous homogeneous reactor systems to the generation of electric power. The major section in the report contains a description of the plant and its operation, estimates of capital and operating costs, and a discussion of the results and recommendations based on the study. Details relative to the selection of the process and power cycles, as well as nuclear constants and calculations, are contained in the Appendix.

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SUMMARY

SCOPE OF STUDY

The scope of the study covered by this report consists of an examination of the technological and economical aspects of the aqueous homogeneous reactor, basically as being developed at the Oak Ridge National Laboratory. Of prime interest are the economics under conditions of central station application and the influence of power plant size on economics. Conditions applicable to generating station sizes ranging from 94 to 1080 megawatts of net electrical generating capacity are presented in the report.

TECHNICAL ASPECTS

Reactor System

The reactor system covered in the report is the two region type operating on the thorium-uranium-233 cycle. This system appears to offer comparatively good possibilities of breeding. It requires only thorium as a fuel supply. The homogeneous reactor also provides ease in fuel handling, an important factor in utilizing uranium-233.

Fuel Processing Plant

The scope of study includes the additional consideration of the fuel processing plant because uranium-233 must be processed from irradiated fuels either locally or at some point apart from the power plant.

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Operating Conditions

While the report deals with several plants of different electrical capacities, emphasis was directed, in final analysis, toward a plant having a net electrical capacity of 180 mw. This tendency may be accounted for by the fact that greater economic promise exists in larger plants which are of most interest to electric utility companies.

Pertinent operating conditions of the 180 mw net electric generating station used in the report are listed in Table I, next page.

Technical Problems

There are a number of problems that remain to be solved or on which development needs to be advanced prior to or concurrent with the design of a full scale unit. The major problems are included in the following categories:

Corrosion Under Irradiation. Studies need to be continued toward the end of assuring equipment life such that amortization rates that are planned can be fulfilled.

Component Development. Development, particularly on steam generators, catalytic recombiners and reactor vessel design requires continuation to assure minimum operation interruptions.

Materials Handling. The preparation and handling of fuel over a long period of time, especially thorium oxide suspensions, should undergo further research and development in order that its behavior in the reactor loop and in the processing and storage system will be better understood and in order that satisfactory handling procedures can be evolved.

Power Density. Although indications are that power densities of 100 kw per liter can be achieved in the core, experimentation should be carried on to confirm this. Likewise, densities anticipated in the blanket require experimental confirmation.

Instrumentation. Instrumentation suitable for operation of an aqueous homogeneous reactor with dependability necessary to electric power generation should be further explored and developed.

Fuel Processing. Further development of procedures and techniques for decontaminating fuel should be carried out to provide more adequate data for designing a full scale unit.

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SUMMARY

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TABLE I -- Operating Conditions - 180 mw Electrical, Plant

<u>General</u>	<u>Core System</u>	<u>Blanket System</u>
Thermal Power, mw	360	280
Fluid	UO ₂ SO ₄ - D ₂ O sol	ThO ₂ - D ₂ O disp
Concentration, gm/liter		
U-233	1.80	8.00
Th-232	-	1000.00
Primary System - Pr., psia	1800	1800
Reactor inlet T	497 F (258 C)	497 F (258 C)
Reactor outlet T	572 F (300 C)	572 F (300 C)
System volume, liters	28,760	41,785
Maximum fluid velocity, ft/sec	33.6	28.7
Loop head loss, psig	58	80
<u>Pumping Requirements</u>		
Total capacity, gpm	33,775	28,783
Estimated efficiency, %	60	60
Estimated total hp	1900	2250
<u>Steam Generators</u>		
Number of units	3	3
Surface sq ft/unit	14,800	12,650
Feedwater inlet T, F	405	405
Steam temp., F	480	480
Steam pressure, psia	566	566
Thermal capacity/unit, mw	119	93
<u>Gas Condenser</u>		
Type: Horizontal, straight tube, single pass, shell-and-tube exchangers with internal eliminator		
Number of units	1	1
Surface area, ft ²	800	800
Feedwater inlet T, F	405	405
Steam temp., F	480	480
Steam pressure, psia	566	566
Thermal capacity/unit, mw	3.5	3.5
<u>Fuel</u>		
Total fuel in system, kg		
U-233	51.8	334.3
Th-232	-	41,785
Fuel burnup, gm/day		
U-233	447	332
Th-232 (consumption)	-	826
Fuel removed		
grams U-233/day	210	496
kilograms thorium/day	-	62.0
liters/day	117	62.0

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Operating and Analytical Procedures. Development of operating and analytical procedures should be advanced so that a minimum of attention by operating personnel is needed and operational interruptions can be held to a minimum.

ECONOMIC ASPECTS

Costs and economics are considered for generating station sizes ranging from 94 to 1080 megawatts of net electrical output. Conditions which appear to be applicable to station sizes of immediate interest, i.e., the 94 and 180 mw installations, are shown in Table II.

TABLE II.-- Power Costs (80% Capacity Factor)

	Reactor Size	
	Net electrical	
	94 mw	180 mw
Capital expense	6.3 mills/kwh	4.72 mills/kwh
Fuel - includes shipping cost to central location	0.98	0.98
Operating cost		
Operating and maintenance	1.60	1.00
Heavy water expense	0.42	0.34
Total Power Cost	9.30	7.04

Capital expense is based upon a 25 year life. Heavy water expense is based upon a lease charge. In the event that purchasing of heavy water is required, this cost will be approximately doubled.

Fuel cost is based upon processing in a centralized plant and upon paying inventory charges and receiving credit for breeding gain. With no inventory charges, fuel cost is decreased from 0.98 to about 0.80 mills/kwh. If an inventory charge is made and no credit is received for breeding gain, the fuel cost increases from 0.98 to about 1.2 mills/kwh.

Table III, below, indicates the capital expense under varying lifetimes of the reactor plant equipment. This appears of interest because with existing data and information, accurate reactor lifetime prediction is impossible.

TABLE III. -- Capital Expense Vs Reactor Lifetime
(80% Capacity Factor)

Total Generating Station Cost	\$43,495,000
Reactor Equipment	10,730,000
Remainder	\$32,765,000

<u>Reactor Equipment Life</u>	<u>Plant Life Exclusive of Reactor Plant</u>	<u>Capital Expense Mills/kwh</u>
25	25	4.72
20	25	4.77
15	25	4.89
10	25	5.14
5	25	5.98

The influence of generating station size on costs is illustrated in Fig. 1, page 7. This illustration is based on centralized fuel processing, paying 4% on inventory and receiving credit for breeding gain. A 25 year life was used for the plant. Costs vary between 9.3 and 5.5 mills/kwh at the 94 and 1080 mw levels. This condition is of interest in exploring the potential of the homogeneous system

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FIG. 1. -- Generating Station Size Vs Total Costs

CONCLUSIONS

The aqueous homogeneous reactor system:

- a. Operates with a net thermal efficiency of 28.1%.
- b. Provides a way of utilizing thorium in a power breeder. The breeding gain frees the system from dependence upon uranium supplies.
- c. Has no requirement for fuel fabrication. Therefore, fuel handling problems are minimized.
- d. Is controlled by fuel concentration and a negative temperature coefficient. No control, shim, or safety rods are required.
- e. Becomes more promising, economically, in large electrical generating installations. Power costs approximate 7 mills/kwh at 180 mw.
- f. Is developed to a point where problems in its design, construction, and operation have been isolated. Significant progress is being made toward the solution of such problems.

PLANT DESIGN AND OPERATIONGENERAL LAYOUT

The over-all concept for the 180 mw plant, illustrated in Fig. 2 on page 10, consists of an integrated reactor plant, turbine plant and fuel processing plant.

The general layout of this installation is such that additional reactor units, under an expansion program, can be added in an easterly direction. It is assumed that the site will occupy approximately 1000 acres on a river front where adequate cooling water will be available.

Reactor Building and Sphere

The reactor system with its steam generators is inclosed within a concrete structure that provides radiation shielding and, in addition, acts as a blast shield in the event of a catastrophe. The structure, along with the reactor system auxiliaries, is further inclosed in an insulated steel sphere, 175 feet in diameter. The sphere is designed to withstand an internal pressure of 40 psia, the estimated equilibrium pressure which would result from a failure of both the reactor coolant and steam generating systems.

Turbine Building

The turbine building is of steel frame construction with insulated metal siding. It houses a turbine-generator unit and its auxiliary equipment, the control room for the reactor and turbine, shops, locker room, and other facilities. Sufficient laydown and turnover space is provided in the turbine building for all shell, bolt, rotor, governor, stator, bearing, exciter, and valve work. A railroad spur enters the building providing a construction and maintenance facility. Fig. 3, 4 and 5, pages 11, 12, and 13 show the operating floor plan, the basement plan, and an elevation of the turbine plant, respectively.

FIG. 2. -- Property Plat

FIG. 3. -- Operating Floor Plan Turbine Plant

FIG. 4. -- Longitudinal Section of Turbine Plant

Control Room and Controls. All plant operation is controlled from a single centralized control room as shown in Fig. 3 , page 11. Instrumentation is provided to enable the operators stationed in the control room to monitor each major piece of equipment and to supervise the over-all plant operation with convenience and safety. In addition to the conventional steam plant instruments, wired television and numerous remotely located microphones give the control room operator a means of observing equipment normally checked by a roving operator. An extensive annunciator and alarm system automatically gives a signal when a piece of equipment is operating abnormally. In addition, the control system is designed to automatically trip out-of-service important items of equipment that may operate in a faulty manner.

Figures and diagrams showing the primary instruments and controls for the reactor and turbine-generator power plant are as follows:

Instrument and Control Symbols	Fig. 15	page 38
Reactor Nuclear Instrumentation	Fig. 13	page 34
Reactor Process Instruments and Controls	Fig. 14	page 37
Piping and Instrument Steam Flow Diagram	Fig. 19	page 44
Piping and Instrument Condensate Flow Diagram	Fig. 18	page 43
Off-Gas System	Fig. 16	page 39

Auxiliary Equipment. Auxiliary equipment is located in the basement of the turbine building. In addition to the condenser and condensate pumps already mentioned, the boiler feed pumps, control and service air compressors, miscellaneous pumps and other equipment are located as shown in Fig. 4 , page 12. The feedwater heaters are situated parallel to the turbine. Working space surrounding the heaters, required for servicing, is provided at intermediate levels by grating. Auxiliary hoists are also provided for withdrawing the heater shells for maintenance. The lowest pressure heater is located in the neck of the condenser in order to avoid the large multiple extraction lines that would otherwise present a difficult problem in layout.

Power Transformers and Electrical Switchyard

Two 100 mva - 14.4/138 kv 3 ϕ power transformers and two 10 mva - 14.4/4.16 kv 3 ϕ auxiliary power transformers are located outside the turbine building. Underground cables carry the 138 kv electric power to the switchyard.

The switchyard, shown in Fig. 2, page 10, contains terminal facilities and equipment for four 138 kv transmission lines and one 10/12.5 mva emergency auxiliary transformer.

Crib House

Four pumps of 60,000 gpm capacity and travelling screens are located in a crib house at the circulating water inlet on the bank of a river.

Service Building

A service building, 80 ft. wide by 200 ft. long, contains offices and shop facilities for the plant. It consists of a partial basement and two floors (except for the portion occupied by the machine shop). The basement contains the feed-water treating plant as well as service facilities for the building. Shops and stores are located on the first floor, along with the instrument laboratory, maintenance offices, and locker rooms. The second floor includes plant offices, the health physics department, cafeteria, chemistry laboratory, and a conference room. The machine shop is entered by a spur track for delivery of equipment and is served by a 25 ton traveling crane.

Fuel Processing and Off-Gas Buildings

The fuel processing plant is included to permit estimating fuel decontaminating costs for use in the unit cost analysis.

The fuel processing and off-gas buildings are of concrete in order to meet structural and shielding requirements of the proposed fuel decontamination process. In addition, the fuel processing building is designed roughly in the shape of a "U" so that installed equipment will provide the shortest route for the fuel being decontaminated.

Radioactive xenon, krypton, and iodine are routed through equipment in the off-gas building. Here they are adsorbed and retained until radioactivity reaches a point where they may be safely discharged into the atmosphere. Radioactive waste streams are directed into permanent waste storage facilities immediately north of the off-gas building, where they are permanently confined.

REACTOR PLANT

Operating and Design Criteria

The reactor power plant consists of a single two-region reactor system operating at a power level of 640 mw. A system flow sheet is shown in Fig. 6 page 17. The reactor consists of a 6 ft. diameter spherical core which is operated at a power level of 360 mw. This, in turn, is surrounded by a 2 ft. thick blanket which is operated at a power level of 280 mw.

Saturated steam is produced at 566 psia and 480 F. A throttle flow of 2.6×10^6 pounds of steam per hour is utilized to generate a gross electrical output of 190 mw. This corresponds to gross over-all plant efficiency of 29.7 per cent.

Core System

Core Solution. The core solution, containing 1.80 gm per liter of U-233 as uranyl sulfate dissolved in heavy water, circulates at a rate of 33,800 gpm under a pressure of 1800 psia. The solution enters the core at a temperature of 497 F and leaves at a temperature of 572 F.

Heavy Water Decomposition. A copper sulfate catalyst is in the core solution to limit the decomposition of heavy water. Sufficient copper sulfate is provided to limit the net decomposition to 20% of the amount that would result without the catalyst. The decomposition gas aids in stripping poison gases, such as xenon.

Liquid and Gas Recirculation. After leaving the core, the solution enters a centrifugal gas separator where the liquid and the explosive mixture of deuterium and oxygen are separated.

The liquid, after circulating through six heat exchangers in parallel which comprise three steam generator units, is returned to the core by three canned rotor circulating pumps. In each steam generator unit 119 mw of heat is removed from an 11,300 gpm stream of liquid. Steam is produced in the generators at 566 psia and 480 F.

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FIG. 6. -- PROCESS FLOW SHEET

The gas mixture, diluted below the explosive limit with a recirculated gas stream of 300 cfm that contains helium, oxygen, and D_2O , flows into a recombiner accompanied by 100 gpm of entrained liquid. The entrainment is removed in mist separators for return to the liquid system. The D_2 and O_2 are recombined in passing through a catalyst bed containing platinized alumina pellets. Heat liberated in the recombiner increases the temperature of the gas from 572 F to 667 F. The hot gases are cooled to 572 F in a gas condenser, where 3.43 mw of heat is liberated and D_2O is condensed at a rate of 17 gpm. A portion of the D_2O is used to wash the mist separators and to purge the pump bearings and rotor cavities. The remainder of the D_2O is returned to the system through the storage tank or is held in the condensate storage tank.

Removal of Insoluble Fission Products. A side stream of approximately 20 liters/min. (equivalent in one day to the total content of the core system) circulates through a mechanical cyclone separator for removal of insoluble fission products from the core solution. A stream containing the separated fission products of approximately 117 liters/day flows from the cyclone to the fuel processing building.

Blanket System

Blanket Slurry. The blanket slurry consists of 1000 gm of thorium per liter as thorium oxide in D_2O . The slurry also contains 8 gm of U-233 per liter.

Liquid and Gas Recirculation. It is circulated by 3 canned-rotor pumps, at a rate of 28,800 gpm under a pressure of 1800 psia, through a gas separator and through six heat exchangers in parallel comprising three steam generator units each rated at 93 mw. The slurry enters the blanket of the reactor at 497 F and leaves at 572 F.

The gas and liquid recirculating systems for the blanket are similar to those for the core. The gas condenser for the blanket system has a heat output of 2.1 mw.

Fuel Storage and Off-Gas Systems

Both the core and blanket systems are equipped with high pressure fuel storage tanks. All gases bled from the high pressure systems and from the high pressure storage tanks pass through a pressure reducing station to the off-gas system. This system consists of a low pressure recombiner and recombiner condenser, cold traps where the last traces of D_2O are removed, and fission product adsorbers. The gases leaving this system pass to the stack or to a holdup tank when required. In the holdup tanks the activity that may be present decays to a safe level before the gases are discharged to the stack.

007029.000

Fuel Concentration Control

Initial charging of the systems with D_2O is accomplished by the use of 75 gpm capacity canned rotor centrifugal pumps. Tube and shell evaporators are provided in both systems for concentrating the fuel and blanket material. Pulsafeeder type pumps of 3 gpm capacity are used to inject concentrated materials or D_2O into the systems.

Reactor Pressurizer System

A mixture of helium and oxygen in the ratio of 5 to 1 is used to pressurize the system. Excess pressure on the system is relieved by a flow of gas to the off-gas system. The core and blanket pressurizer systems are intimately connected so that the differential pressure across the core vessel can be maintained well within 100 psi. This insures against possible damage to the vessel resulting from momentary power surges.

Components

Reactor Vessel

Fig. 7, page 20, illustrates a two-region spherical reactor vessel which satisfies the established requirements for a power reactor. It consists of a 6 ft. diameter zirconium core-tank with a shell thickness of $\frac{1}{2}$ in. inclosed in an 11 ft. 4 in. internal diameter pressure vessel with a wall thickness of approximately 6 in. of SA 212 Grade B carbon steel clad with 347 stainless steel. The annulus formed by the space between the core-tank and the pressure vessel is the reactor blanket.

Reentry Type Core. A core tank utilizing concentric inlet and outlet connections was chosen in this design study. This type core tank in a bolted-head pressure vessel permits convenient replacement should its useful life prove to be less than the life of the pressure vessel. The core tank is also provided with a 2 in. internal dump line.

Preliminary calculations indicate that the $\frac{1}{2}$ in. thick zirconium core tank can be subjected to a differential pressure of nearly 400 psi before reaching its yield strength.

FIG. 7. -- REACTOR VESSEL

NO.	DATE	REVISION	BY	REACTOR VESSEL	
SCALE: 1/4" = 1'-0"				NUCLEAR POWER GROUP	
DRAWN: H. J. GILBERT				CHICAGO, ILLINOIS	
DATE: 11-20-58				DRAWING NO.	
CHECKED: _____				HR-P-5	
APPROVED: _____					

Core Inlet and Outlet Connections. The fuel solution enters the core through the 24 in. inner pipe and exits through the annulus between the two concentric pipes forming the inlet and outlet connections for the core tank. See Fig. 7, page 20, The area of the annulus is equivalent to the area of a 24 in. pipe. One mechanical joint is required to attach the zirconium core tank to the stainless steel outlet pipe.

Blanket Inlet and Outlet Connections. The slurry enters the blanket through a 24 in. connection in the bottom of the pressure vessel and exits through three 14 in. connections located near the top of the vessel.

Thermal Shield. Calculations (see page 93) and advice from Oak Ridge personnel indicated the need for a thermal shield of 4 in. thickness. This shield is provided in the form of two 2 in. thick SS formed plates. A 2 ft. spherical annulus, unobstructed except for the core outlet pipe, is provided between the core tank and the inside face of the thermal shield to form the blanket. A 2 in. space is maintained between the two thicknesses of thermal shield and between the outside face of the outer shield and the inside face of the pressure vessel. Sufficient flow of the slurry is maintained between and around the shield segments to insure proper cooling.

Head Filler. A laminated filler is provided in the head to reduce the blanket inventory. Sufficient flow passages in the filler insure proper cooling.

Vessel Closure. The bolted head closure utilizes two Flexitallic gaskets, asbestos encased in stainless steel, having a low pressure leakoff between gaskets. The internal diameter of the closure is slightly greater than 6 ft. to allow for core tank removal. The bolts for this head design are stressed to approximately 40,000 psi, much higher than presently allowed by code. Similar bolted joints are provided in the inlet and outlet piping connections to the core as well as in the core tank dump line.

Vessel Support. The vessel is supported by a vertical cylindrical type skirt attached by welding to the shell slightly below the equator. The reactor vessel has an operating weight of approximately 130 tons.

Gas Separator

The gas separators used in this reactor system are basically the same as proposed by Oak Ridge National Laboratory in report ORNL-1642 (see page 130) for a two-region reactor system.

Location and Piping. The gas separators, one in each of the core and blanket systems, are external to the reactor vessel. The two streams enter their respective gas separators through a 30 in. O.D. guided vane 90° miter joint connected to the 24 in. O.D. circulating system pipe. Each separator provides a 16 ft. horizontal run, measured from the center-line of the inlet miter joint to the center-line of the outlet miter joint.

Centrifuging and Dilution Runs. Rotation vanes at the entrance of the horizontal run direct the liquid to the perimeter of the 30 in. pipe causing a vortex to form into which the gas bubbles are centrifuged. A mixture of oxygen, helium, and D₂O is circulated through the central void to dilute the emerging gas below the explosive limit. The effective separation length from the diluting gas inlet nozzle to the outlet nozzle is 7½ ft. Before leaving the horizontal run of 30 in. pipe the liquid passes through recovery vanes and then exits through a 30 in. O.D. guided vane 90° miter joint into the 24 in. O.D. circulating system pipe.

Material of Construction. The material of construction for all components of the gas separators is 347 stainless steel.

Steam Generators

Number and Location. There are six steam generator units in the reactor system -- three are in the core loop and three are in the blanket loop. Each steam generator unit consists of two heat exchangers connected to a common steam drum. Fig. 8, page 23, illustrates the unique compartmentalized feature of the Griscom-Russell Co. heat exchanger design incorporated in this report. Important steam generator data are presented in Table IV on page 24.

Controlling Steam Quality. A minimum 4 in. steam dome is maintained in the heat exchanger shells to provide disengaging sections for the steam. By limiting the velocity of the steam in the risers to 1500 fpm, the moisture carry-over to the steam drum is held to five per cent. The 42 in. I.D. carbon steel drum contains a moisture eliminator which in turn assures a minimum steam quality of 99 per cent.

Reliability and Fabrication of Heat Exchangers. The need for reliable heat exchangers in the homogeneous reactor system cannot be over-emphasized. Much credit must be given to the representatives of the Babcock and Wilcox Company and the Griscom-Russell Company who cooperated to achieve conceptual designs which are acceptable for use in a nuclear power plant.

Rev.	Date	By	Check	Remarks
1				
2				
3				
4				
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9				
10

FIG. 8. -- Steam Generator Assembly

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TABLE IV. -- Steam Generator Data

	<u>Core System</u>	<u>Blanket System</u>
Number of Units	3	3
Number of Heat Exchangers/Unit	2	2
Number of Eliminators/Unit	1	1
Compartment Tube Sheet, D in.	5 3/8	5 3/8
Heat Exchanger Shell, D in.	60	67
Compartment Pipe, D in.	2(Sch.80)	2(Sch.80)
No. of Compartments	38	48
No. of Tubes/Compartment	19	19
Total No. of Tubes/H.E.	722	722
Nominal Length of Tubes, ft.	62.5	42.25
Heat Exchange Surface/Unit, sq.ft.	14800	12650
Preheating Zone Surface/Unit, "	1684	2200
Tube O.D., in.	5/8	5/8
Tube Thickness, in.	0.065	0.065
Tube Material	347 SS	347 SS
Shell Material	SA 212 Gr.B,C.S.	SA 212 Gr.B,C.S.
Steam Drum, D in.	42	42
Steam Drum Material	SA 212 Gr.B,C.S.	SA 212 Gr.B,C.S.
Feedwater Inlet Temp, F	405	405
Steam Temp, F	480	480
Shell Operating Pressure, psia	566	566
Shell Design Temp, F	600	600
Shell Design Pressure, psi	1250	1250
Tube Side Inlet Temp, F	572	572
Tube Side Outlet Temp, F	497	497
Tube Side Operating Pressure, psia	1800	1800
Tube Side Design Temp, F	600	600
Tube Side Design Pressure, psi	2000	2000
Number of Passes	1	1
Velocity - Tube Side, fps	13	9
Pressure Drop - Tube Side, psi	20	17
Holdup Volume/Heat Exchanger, ft ³	60	52

As stated before, the heat exchanger concept included in this report is a Griscom-Russell development. However, a Babcock and Wilcox concept of a horseshoe bend exchanger using 1 in. O.D. cored tubes appears to be equally promising and should be given further consideration. Unfortunately, time precludes the inclusion of both concepts in plant layouts. It may be noted, however, that neither of the conceptual designs enjoys a price advantage based on the cost data supplied by the two manufacturers.

Some positive steps have been taken in the development of this conceptual design in regard to reliability and ease of fabrication. By using U-bend tubes in the exchangers, the need for an expansion joint in the shell or in the connection to a floating tube sheet is eliminated. This adds reliability to the unit since any expansion joint subject to even infrequent work is a potential and likely source of trouble.

Utilizing the compartmentalized concept in the heat exchangers offers added reliability, ease of fabrication, and a means by which maintenance of the units becomes practical. The individual "bottles" consisting of nineteen U-tubes attached to their tube sheets in the eccentric pipe reducers by rolling and welding, can be fabricated and tested as units before installation in the exchanger. The drilling of the "bottle" tube sheets presents practically no difficulty because they are relatively thin, being only 5 3/8 inches in diameter. Similarly, the drilling of the exchanger head for insertion of the 2 in. inlet and outlet pipes to the "bottles" presents no unusual fabrication problems.

Inspection is the key to successful tube making. This has been taken into consideration in the selection of tube size. Ultra-sonic testing of the finished tube appears to be the most practical method of eliminating all tubes with detrimental defects. Until recently it has been possible to inspect only tubes 1 in. O.D. and larger by this method. For smaller diameter tubes, the tube rounds are ultra-sonic tested before processing into final tube form. However, recent information indicates that it is now practical to inspect finished tubes in diameters as small as 1/2 in. Therefore, in this concept additional tube inventory holdup has been sacrificed for the expected gain in tube reliability by going from the customary 3/8 in. tubes to 5/8 in.

Maintenance of Heat Exchangers. Although the goal is theoretically "leak-proof" heat exchangers, it seems prudent to incorporate provisions in the design for maintenance. This has been done in the compartmentalized concept. Should a leak occur, it is practical to seal off the "bottle" in which the leak occurred by plugging the 2 in. inlet and outlet pipe connections, rather than remove an entire heat exchanger.

Primary Circulating Pumps

Six hermetically sealed motor centrifugal pumps are required to recirculate the core and blanket fluids.

Core Pumps. The three pumps in the core system are rated at 11300 gpm each, with a 58 psi differential pressure, handling solution at 1800 psi system pressure. The pump motors are rated at 650 hp. These pumps are vertical with the main impeller mounted on the lower end of a shaft on which also is mounted the motor rotor. The motor rotor and bearing chamber are separated from the impeller and volute by means of a labyrinth seal. D₂O from the high pressure condensate tank is injected into the bearing chamber and continuously flushed through the labyrinth thus minimizing the corrosion on the rotor and bearing parts. Both the radial and thrust bearings are of the fluid piston type. The drive motors are induction type suitable for 3 phase 60 cycle 4160 volt power supply. They have impervious liners in the stator bore for hermetic sealing, and an outer housing totally enclosing the stator as a second safeguard against loss of system fluid. The motor stator windings are cooled by a liquid passing through tubular conductors installed in the stator. All material in contact with the core solution and D₂O is stainless steel, except for the impeller, labyrinth inserts, impeller nut, and wear rings which are titanium.

Blanket Pumps. The three pumps in the blanket system are rated at 9600 gpm each, with an 80 psi differential pressure, handling slurry at 1800 psi system pressure. The pump motors are rated at 750 hp. In other respects the blanket pumps are identical to the core pumps.

Installation and Maintenance. The pumps are installed vertical, floating in the pipe system. A 14 in. suction is located at the bottom of each pump with the discharge horizontal. By unbolting the top flange, the entire pump mechanism can be removed, leaving only the high-pressure pump casing in the pipe line thus facilitating maintenance.

Reactor System Piping

A visual concept of the primary pipe layout can be gained from reference to the reactor plant plan, elevation, and isometric drawings shown in Fig. 9, 10, 11, and 12 on pages 27, 28, 29, and 30 respectively. The major plant components are arranged to achieve a feasible piping system, proper consideration being given to pipe flexibility, fluid velocities, and pressure drops.

NO	DATE	REVISION	BY	OPERATING FLOOR PLAN HOMOGENEOUS REACTOR	SCALE $\frac{1}{8}'' = 1'-0''$ DRAWN: <u>MARTON</u> DATE: <u>1-5-99</u> CHECKED: APPROVED:	NUCLEAR POWER GROUP CHICAGO ILLINOIS DRAWING NO. <u>HR-B-9</u>

FIG. 9. — OPERATING FLOOR PLAN HOMOGENEOUS REACTOR

FIG. 10. -- CROSS SECTION "A-A" HOMOGENEOUS REACTOR.

NO.	DATE	REVISION	BY	<p>CROSS SECTION 'A-A' HOMOGENEOUS REACTOR</p>
<p>SCALE $V_R = 1.0$ DRAWN <u>HOMER L.</u> DATE <u>1-10-68</u></p>				<p>NUCLEAR POWER GROUP CHICAGO, ILLINOIS</p>
<p>CHECKED _____ APPROVED _____</p>				
<p>DRAWING NO. WR-B-10</p>

FIG. 11. -- LONGITUDINAL SECT. "B-B" HOMOGENEOUS REACTOR

NO.	DATE	REVISION	BY	LONGITUDINAL SECT. B-B HOMOGENEOUS REACTOR
SCALE 3/8" = 1'-0" DRAWN: MERTZ DATE: 1-12-55 CHECKED: APPROVED:				NUCLEAR POWER GROUP CHICAGO, ILLINOIS DRAWING NO. HR-B-11

FIG. 12. -- HOMOGENEOUS REACTOR PLANT CONCEPT

Primary Fluid Piping. All piping used to convey D_2O and core and blanket fluids is type 347 stainless steel. No² valving is included in the core and blanket primary fluid loops. Large size valves were considered for this system, but were rejected because their effectiveness is questionable and their expense unjustifiable.

The system consists mainly of 24 in., 20 in., and 14 in. O.D. pipe made by the hollow forged and bored process to specified wall thickness, approximately schedule 140, suitable for 1800 psi working pressure at 600 F with an appropriate corrosion allowance. The entire system is welded except for the two bolted flange joints at the inlet and outlet connections to the core.

The maximum velocity of the core solution is 33.6 fps with an estimated pressure drop of 22 psi in the piping. The maximum velocity of the blanket slurry is 28.7 fps with an estimated piping pressure drop of 47 psi.

Side Stream Piping. All reactor side stream piping is type 347 stainless steel. Approximately 75 bellows type stem sealed small size 347 SS valves, equipped with suitable valve operating mechanisms for remote operation are included in the various side streams.

Reactor Building

Sphere

The reactor plant is housed in a 175 ft. diameter steel sphere similar to the West Milton Sphere. Design pressure is 40 psia which requires a nominal plate thickness of 3/4 in. The sphere is buried to a depth of 50 ft. allowing the reactor vessel to be located below grade for natural ground shielding. The Chicago Bridge and Iron Company, builders of the West Milton Sphere, gave assurance that this depth of burial presents no difficulties in the design, erection, or proper functioning of the sphere.

The steel shell plates are A 201 Gr. B carbon steel. All seam welds are fully x-ray inspected. The design stress for the steel plate is approximately 17000 psi and tested to a stress of 13,500 psi.

The various electrical, condensate, water, compressed air, and steam lines which penetrate the sphere shell pass through concentric nipples welded to the sphere with flexible seal joints between the conduit and its nipple. The major penetrations, other than the two 5 ft. x 9 ft. pressure lock openings for access to the sphere, are the two 20 in. steam headers.

Insulation is applied to the outside surface of the sphere to reduce the possible stresses in the shell caused by temperature changes and also to aid in the control of temperature inside the sphere.

Ventilating equipment is provided, capable of one-half air change per hour. During normal operation of the reactor plant, little or no air is vented. The air within the sphere is circulated by fans and cooled or heated as required to maintain the desired ambient temperature.

Crane

The reactor room crane is a circular type with a 150 ft. span and a 75 ton capacity. An auxiliary 10 ton hook also is provided. This eight wheel crane is capable of handling any reactor plant component plus portable shielding which might be required when moving radioactive objects.

Shielding

The radioactive reactor plant components are inclosed by a barytes concrete structure which serves both as a biological and as a blast shield. The top of this housing is 35 ft. above grade elevation. The side walls are 5 ft. thick and the top shield is 6 ft. thick except for the 8 ft. thick section directly over the reactor vessel. Compartment walls are provided within the housing to facilitate flooding of individual component sections. The floor and side walls of each of the compartments are lined with 1/8 in. stainless steel plate to prevent radioactive contamination of the concrete. Stepped plugs are provided in the top shield to permit access to the components. Suitable strongbacks are provided for the plugs to prevent the possibility of their becoming a missile if an explosion occurs within the housing. The portion of the shielding around the reactor vessel which is below grade, is 4 ft. thick.

A slight negative pressure is maintained within the housing by continuously withdrawing a small quantity of air and sending it to the stack for dispersal. Although personnel are not within the sphere during normal operation, it is advantageous to keep the radioactive argon produced within the housing from leaking into the sphere proper. Therefore, the negative pressure within the housing insures against outward leakage. The quantity of air removed is regulated to control the ambient air temperature in the component compartments.

Maintenance in Reactor Building

The basic skills required to maintain a reactor plant are identical to those required in a conventional plant. However, the job procedures are complicated by the hazards of exposure to radiation. The radioactive reactor plant components are located in individual compartments which are flooded with water to protect workers performing maintenance. When applicable, portable lead shields are used in lieu of the water shield. The following is a concept of a job schedule for maintenance work on reactor components;

- a. Drain core and blanket systems.
- b. Dry primary systems with hot air.
- c. Rinse systems with light water. Acid washing is also employed when required.
- d. Drain rinse solutions to waste holdup tanks.
- e. Flood compartment with light water where portable protective shielding is not adaptable.
- f. Perform work with extension tools. (Remote control pipe cutting machines are now available and remote welding tools should soon be available.)
- g. If light water shielding is used, it is pumped from compartment to decay tanks at completion of job.
- h. Circulate hot air to dry out primary systems.
- i. Follow normal charging and startup procedures.

Reactor Plant ControlNuclear Instrumentation

The nuclear instrumentation for the reactor plant is shown in Fig. 13, page 34. It is simple, utilizing standard components.

Neutron Flux Monitor at Startup. Two independent fission chamber circuits are positioned at different distances from the reactor core to provide a wider instrument range for following the neutron flux level during startup. Both fission chamber circuits connect to log count rate meters, and a switching system allows the operator to select either range for recording and indicating. For operating, a scaler is included on one of the circuits.

FIG. 13. -- Nuclear Instrumentation

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Period Range Monitor. A compensated ion chamber circuit consisting of a log "N" amplifier, indicator and recorder, period meter and indicator, and an alarm, follows the period range of reactor operation. This circuit is connected to a trip bus which automatically shuts down the reactor if the period increases too rapidly.

Power Range Monitor. Two additional compensated ion chambers monitor the power range of the reactor. One power level indicator and recorder can be switched to pick up the signal from either chamber circuit. An alarm from each ion chamber also is connected to the trip bus and automatically shuts down the reactor if power levels become too high.

Health Hazard Monitors. Ten radiation survey detectors and their appropriate circuitry are located strategically throughout the plant area for health hazard control. All waste or air disposal points are also monitored to detect contamination so that it may be held to a minimum.

Nuclear Control

A well designed aqueous homogeneous power reactor is self-regulating, stable and inherently safe. For this reason, a high degree of nuclear instrumentation coupled to automatic systems designed to scram the reactor during excursions is not required. The nuclear instrumentation is used primarily to monitor and record the reactor power level and to give the operator a "feel" for the neutron flux level. Automatic safety circuits which initiate a slow shutdown are provided only to override an operating error or some control equipment failure.

Neither safety rods nor a rapid dump system is considered necessary for safe operation. During one year of operation, it was found that the control rods on the HRE were unnecessary, and also that a scram dump system for safety was not required. The present design of the HRT has no control, shim, or safety rods. The control of the reactor considered in this report is accomplished primarily by (1) negative temperature coefficient of reactivity and (2) fuel concentration.

Stability and Power Regulation. The high negative temperature coefficient of reactivity provides a self-contained automatic method of power regulation. Other factors remaining constant, as the temperature of the core fuel solution rises, the nuclear reactivity drops. If, for any reason, the core becomes slightly supercritical, the neutron flux level rises slowly, additional heat is generated and the core temperature tends to rise. This rising core temperature causes a reduction of the reactivity and the reactor stabilizes itself at a power level which keeps the core effective temperature at equilibrium.

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Not only does the negative temperature coefficient of reactivity provide a strong stabilizing action, but also automatic power regulation. The reactor power level rapidly follows the steam demand of the turbine-generator unit. If the system electrical load rises, the speed sensitive turbine governor adjusts the steam admission valves to increase the steam flow into the turbine. A greater quantity of heat is removed from the primary coolant and the effective core temperature tends to drop. The negative temperature coefficient of reactivity characteristic automatically raises the reactor power level to compensate for this increase in load and stabilizes at the previous equilibrium temperature.

Fuel Concentration and Equilibrium Temperature. The concentration of fuel in the core is a second method of reactor control. Whereas the temperature effect on the reactivity stabilizes and regulates the power level, the fuel concentration determines the effective equilibrium temperature of the coolant at which point the reactor is just critical. As the fuel concentration is increased, the equilibrium temperature rises. So pronounced is this effect that the fuel concentration of the reactor can be calibrated as a function of the equilibrium temperature. Fresh fuel is added to the primary system to compensate for fuel burnup or to increase operating temperature. The fuel is diluted, when required, to reduce the concentration and the effective operating temperature.

Primary System Control and Instrumentation

The fuel process systems for the core and blanket are essentially the same. Therefore, instrumentation and control for the core and blanket are substantially identical. In the discussion that follows, explanations are confined to a consideration of instrumentation and control for the core system. This system is illustrated in Fig. 14, page 37. The instrument and control symbols are in Fig. 15, page 38. The off-gas system, common to both the core and blanket, is shown instrumented in Fig. 16, page 39. For the control designs and estimates of this report, commercially available and conventional components are used whenever possible.

Operation.

The core and blanket systems may be operated independently, but the pressure differential in the reactor vessel must be maintained within design limits.

Operation may be divided into four distinct categories, i.e. startup, steady state, normal shutdown, and automatic shutdown.

PIPING SYMBOLS

	VALVE NORMALLY OPEN
	VALVE NORMALLY CLOSED
	CONTROL VALVE ELECTRICALLY OPERATED
	CONTROL VALVE PNEUMATIC OPERATED
	CHECK VALVE
	RELIEF VALVE
	BUTTERFLY VALVE
	ORIFICE
	FLOW NOZZLE
	REDUCER
	MAIN FLOW
	SECONDARY FLOW
	INSTRUMENT AIR LINE
	INSTRUMENT ELECTRICAL LEAD

INSTRUMENTATION SYMBOLS

	INSTRUMENT SYMBOL INSTRUMENT NUMBER
LOCAL	
	INSTRUMENT SYMBOL INSTRUMENT NUMBER
REMOTE BOARD MOUNTED	
	LOCAL NON-INDICATING TRANSMITTER
	LOCAL INDICATING TRANSMITTER
	MANUAL-AUTOMATIC CONTROL SUB-PANEL IN CONTROL ROOM
	LOCAL TEST POINT

INSTRUMENT			FUNCTION			ELECTRICAL DEVICES		
MEASURED VARIABLE	MEASURING DEVICES	DEVICES	CONTROL DEVICES	DEVICES	RECORD	ALARM	TRIP	SWITCH
PRESSURE	PI	PR	PC	PIC	PRC	PA	PT	PS
TEMPERATURE	TI	TR	TC	TIC	TRC	TA	TT	TS
FLOW	FI	FR	FC	FIC	FRC	FA	FT	FS
LEVEL	LI	LR	LC	LIC	LRC	LA	LT	LS

PREFIXES

A - AUTOMATIC
C - CONDUCTIVITY
D - DIFFERENTIAL
H - HIGH VARIABLE
L - LOW VARIABLE
P - POSITION
RH - H² CONCENTRATION
R - RADIATION
V - VALVE
M - MANOMETER
W - DENSITY

SUFFIXES

A - ALARM
AN - ANALYZER
C - CONTROL
I - INDICATOR
L - LIGHT
R - RECORDER
S - SWITCH
T - TRIP
V - VALVE
X - TEST POINT

FIG. 15. -- Symbols For Use On Flow Diagrams

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FIG. 16. -- OFF-GAS INSTRUMENT FLOW SHEET

Startup. Normal startup of the core or blanket is manual. The pertinent elements in the procedure are as follows:

- a. Using the fuel loading pumps, fill the primary core and blanket loops of the reactor including gas separator, recombiner tank, heat exchangers and primary pumps, with dilute fuel solution.
- b. Pressurize the primary loops with gas from helium-oxygen makeup system.
- c. Start the fuel circulating pumps, gas blower, and the fuel condensate pump.
- d. Allow the primary system to heat up from the resultant friction heat due to pumping.
- e. Adjust and balance all liquid levels, flows, temperatures, and pressures.
- f. Start fuel feed pump to supply concentrated fuel, until the reactor reaches the desired operating temperature.

Steady State. The steady state power operation of the reactor depends on maintaining the proper balance of system pressure, liquid level, fuel temperature, and fuel flow. (Since fuel is continuously withdrawn from the system for processing, the operator must add fuel to maintain the temperature within operating limits.) The operator may maintain balance of the core and blanket systems manually or automatically.

Normal Shutdown. The condensed procedure for normal shutdown is as follows:

- a. Remove turbine-generator unit from service. Close turbine stop valves and steam generator stop valves. Vent a small amount of steam to quench tanks maintaining the reactor at a low power level and compensating for residual heating effects.
- b. Reduce fuel concentration by continuously adding D_2O . Maintain liquid levels by draining excess fuel to storage tanks.
- c. Reduce the circulation rate of primary solution.
- d. Depending on the purpose of the shutdown, the fuel solution is either held in the primary system or drained to the storage tanks. If the fuel solution is held in the primary loop, circulation is continued at a reduced rate until the concentration is below that required for reactor criticality at ambient temperature.

Automatic Shutdown. Automatic dumping of the fuel solution to the storage tanks in order to shut down the reactor provides protection against operational errors as well as system component or control and instrumentation equipment failures.

TURBINE-GENERATOR AND AUXILIARIES

Heat Balance and Steam-Condensate System

The thermal cycle for the over-all plant is shown in Fig. 17 page 42. Throttle steam at 550 psia, dry and saturated, flows through two 20 in. headers to a tandem compound, double flow turbine. Five extractions are provided for feedwater heating, resulting in a full load feedwater temperature of 404.7 F. At a back pressure of $1\frac{1}{2}$ in. Hg, the gross thermal efficiency is 29.7%.

The steam-condensate cycle with its piping and controls is relatively conventional in arrangement. Fig. 18 page 43 is the condensate diagram including instrumentation, and Fig. 19 page 44 is the steam and drain diagram.

Turbine-Generator Unit

Turbine. The turbine has a high pressure single flow section in tandem with an opposed double flow low pressure section. The last stage of the 1800 rpm turbine uses 38 in. buckets. The turbine has a standard downward exhaust. The heat balance diagram shows extraction at five points -- three from the low pressure double flow section, one from the crossover point, and one extraction from the high pressure single flow section. Grooved type buckets are used to provide for moisture removal at each extraction point. The use of such buckets obviates the necessity for external moisture separators. Extracted moisture along with extracted steam is taken to the feedwater heaters.

Generator. The generator is rated 246,000 kva at 30 lb. hydrogen pressure. It is a double winding machine, 14,400 v, 0.64 short circuit ratio, separately excited by means of a motor-generator set, as described in the electrical portion of the report.

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FIG. 17. -- HEAT BALANCE DIAGRAM

FIG. 18. -- Flow Diagram -- Condensate Piping

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FIG. 19. -- Flow Diagram -- Steam and Drains

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Condenser

The condenser is of conventional horizontal design enclosed in a single shell. It has single pass circulating water flow and contains 160,000 sq. ft. of surface. Tubing is phosphorized Admiralty, 1.0 in. by 18 BWG, 30 ft. long. Deaeration is effected in the hot well. The condenser is rated for 70 F circulating water and a velocity in the tubes of 7 fps. The condenser selection is based on representative site conditions.

Condensate Pumps

Three, half-size vertical pit type condensate pumps take suction from the hot well, backed up by an external makeup water tank. Condensate is pumped through the air ejector condenser, through a separate drain cooler, and through the three low pressure feedwater heaters to the suction header for the boiler feed pumps.

Boiler Feed Pumps

Three motor driven boiler feed pumps are provided, each capable of handling one-half of the normal full load flow. They pump the feedwater through the two high pressure heaters, and by means of individual control valves, the feedwater is distributed to the steam generator units.

Feedwater Heaters

The feedwater heaters are all of horizontal, U-tube design, with one heater provided for each extraction stage. All but the lowest pressure heater (No. 5, Fig. 18 page 43) have integral drain coolers. The drains cascade to the condenser.

Steam and Feedwater Piping

The steam and feedwater piping is carbon steel. The steam piping from the steam generators to the turbine stop valves and the feed piping from the feedwater stop valves to the heat exchangers are schedule 120 to handle a maximum pressure of 1250 psi. The remainder of this piping is schedule 40, designed for a maximum pressure of 600 psi.

Turbine-Generator Plant Control

As previously indicated, all plant operation is controlled from a centralized control room. The primary instruments and controls for the turbine-generator plant are shown in the Steam Flow Diagram, Fig.19, page 44, and in the Condensate Flow Diagram, Fig.18, page 43.

Steam Radioactive Contamination Control. Steam generator stop valves are provided in each steam line connecting the individual steam drums to the steam headers. Should a heat exchanger tube leak and the steam become highly contaminated, these valves will close rapidly and isolate the faulty steam generator unit from the system. The size of the steam drums is adequate to allow time for the detection of excess radioactivity and to close the stop valves before an appreciable amount of contaminated steam flows to the turbine. A quick closing valve in the feedwater line to each steam generator unit shuts off all condensate flow to the unit. At the same time the primary fluids are dumped to the high pressure storage tanks. As the pressure is reduced on the primary side, a corresponding pressure is maintained on the secondary side by means of a power relief valve which vents the steam through a closed line to a steam-water eductor submerged under water in the quench tank.

This protection will minimize the amount of radioactive contamination carried over into the steam and condensate system of the turbine-generator unit. In addition, it confines any leakage of D_2O and fuel to the steam generator shell from which D_2O and fuel will be recovered. The turbine-generator unit, feedwater heaters, condenser and all associated equipment can be operated in a conventional manner. Although additional provisions are made for remote operation, a standard system utilizing roving operators and equipment attendants is employed.

Steam Generator Feedwater Control. The water level in each steam generating unit is maintained by a three element type of controller. A signal proportional to the steam flow from a steam generator unit is balanced against a similar signal which is proportional to the condensate flow into the steam generator unit. The difference between these two signals is combined with a third signal from the water level detector of the controller and used to actuate the control valve in the feedwater line to the steam generator unit.

Feedwater Heaters and Condenser Control. Five steam extraction points are provided for feedwater heating and the removal of moisture from the turbine. The wet steam is bled from the turbine to five feedwater heaters. Normally, the drains of each heater cascade directly to the next lower pressure heater; drains from the last heater pass through the drain cooler directly to the condenser. A control valve is installed in each

feedwater heater drain line to maintain the desired water level in the heater. In case any of the heaters is out of service during plant operation or if the amount of drain water becomes excessive due to heater tube leaks, bypass lines drain the excess water directly to the condenser. The water level in each feedwater heater is controlled by a locally mounted, level indicating, controller which transmits a signal through a manual automatic sub-panel in the control room to the control valves in the drain lines. A similar control system on the condenser adjusts the flow of makeup water to maintain a constant condenser water level.

Turbine-Generator - General. The homogeneous reactor lends itself nicely to conventionally designed instrumentation and control of the turbine-generator. The supervisory instrumentation, control devices and accessories include the standard line of accessories normally provided by the turbine-generator manufacturer and, in addition, most of the commercially available optional accessories. Although the turbine-generator unit probably will be started from the floor and checked occasionally by a roving operator, the unit is designed to be operated remotely from the control room.

Turbine-Generator Supervisory Instrumentation. The turbine-generator supervisory instruments and their functions are as follows:

- a. The turbine shaft eccentricity meter records the trueness of the turbine shaft at a given position on the shaft. It is a measure of the amount of bend or bow in the shaft.
- b. The vibration recorder measures and records the amplitude of transverse vibration of each bearing pedestal.
- c. The turbine shell expansion recorder measures and records movements of the shell away from the foundation anchor point.
- d. The turbine differential expansion recorder measures and records the relative expansion between the turbine shell and the rotor.
- e. The speed and camshaft position recorder measures and records the shaft rotational speed of the turbine during the startup and shutdown periods and the amount of opening of the steam admission valves during the time the unit is tied to the electric system.
- f. The shaft position recorder measures and records the axial position of the shaft to detect a thrust bearing failure.

Turbine-Generator Protection. Numerous devices are provided to protect the turbine-generator from various types of failures. Detection of any of the following troubles will automatically initiate a complete shutdown of the unit by tripping closed the turbine stop valves and tripping open the two generator circuit breakers and the two auxiliary power transformer circuit breakers:

- a. Turbine shaft overspeed.
- b. Loss of exhaust hood vacuum.
- c. Excessive vibration.
- d. Excessive generator or transformer differential currents.
- e. Excessive ground current.
- f. Excessive negative sequence current.
- g. Loss of generator excitation.
- h. Excessive generator field ground current.
- i. Thrust bearing failure.
- j. Low main oil pressure.
- k. Low bearing oil pressure.
- l. High exhaust hood temperature.
- m. Excessive bearing oil temperature.
- n. Generator motoring (with time delay).

Remote manual trips are located in the control room and on the front standard of the turbine-generator. An annunciator alarm system warns the operator of troubles with the turbine-generator unit and, if the unit is automatically shut down, indicates which device initiated the trip signal.

Electrical Plant

The single line diagram for the electrical plant is given in Fig. 20 page 50. This diagram includes the generator electrical connections, the 138 kv transmission substation, and the auxiliary power bus arrangement.

Generator Electrical Connection. The generator is a double winding machine rated at 14.4 kv. Although a double winding stator adds slightly to the turbine-generator unit cost, the advantages gained are desirable. Generator fault currents are greatly reduced by use of the double winding. In addition, greater operating flexibility provides additional safety and plant reliability.

An isolated phase bus connects the generator windings to the low voltage side of two 100/133 mva 3 phase power transformers. The high voltage transformer windings connect to the 138 kv bus through two sets of oil circuit breakers.

FIG. 20. -- Electrical Single Line Diagram

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A number of relays are used to detect electrical fault conditions in the generator, the isolated phase bus, and the connecting transformers. Several sets of differential current relays provide the major protection; in addition, ground relays, loss of field relays, and negative sequence current relays are utilized.

Transmission Substation. The substation switching arrangement selected for this design consists of a main bus, split into two sections, and a transfer bus. Four 138 kv transmission lines carry the electric power to the load center. These four lines also insure a reliable power source into the station. They are protected by directional comparison carrier current relays and three element directional impedance relays. Instantaneous automatic oil circuit breaker reclosing may be incorporated depending on the transmission system characteristic. The 138 kv oil circuit breakers are rated at 1200 amperes and interrupting capacity of 3.5 million kva.

Auxiliary Power. The auxiliary power requirements for the entire plant are supplied by a 4160 volt bus, a 480 volt bus, and a 250 volt DC bus. All motor loads above 100 hp rating are connected to the 4160 volt bus. The small motor load is on the 480 volt bus; the emergency and control power is supplied by the DC bus.

The 4160 volt auxiliary bus is divided into three sections for reliability. Each section is fed by a separate transformer. Two 10 mva 14.4/4.16 kv auxiliary power transformers supply power directly from the two windings of the generator to sections 1 and 3. Disconnects in the isolated phase bus between the generator and the auxiliary power transformer tap permit backfeed from the 138 kv bus. The third auxiliary power transformer, rated at 10/12.5 mva 138/4.16 kv, supplies power directly from the 138 kv bus to section 2. This transformer has ample capacity to carry the total station auxiliary load during shutdown and startup conditions.

In addition, an emergency source of power is connected to the 4 kv bus. Depending on the plant location, the interconnected system, and terrain, the emergency power source may be either an underground 4 kv feeder or a 1000 kw diesel generator unit.

As indicated on the single line diagram for the electrical plant, the 4 kv motor load for each function is divided evenly between the three sections. This distribution of motor load permits the loss of any one section without danger to the plant, but requires reduction in load.

Motor loads smaller than 100 hp rating are supplied in a similar manner by the 480 volt bus. Loss of either section is not considered hazardous to plant operation.

Control power for relaying, oil circuit breakers, auxiliary switchgear, and electrically operated valves is supplied from the 250 volt DC bus. Emergency lighting and small emergency backup motors such as the unit emergency seal oil pump and the emergency bearing oil pump are connected to the DC bus.

Two 40 hp 25 kw regulated motor generator sets supply power to the DC bus. Normally, only one unit is operating; the second is held in reserve. Two batteries with 250 volt 70 ampere 8 hour ratings float on the DC bus at all times.

The nuclear instrument AC power is obtained from two DC powered motor generator sets. This method provides continuous AC instrument power even during periods of complete loss of auxiliary power. In addition, the numerous lightning and system switching transients will not ride through the power supplies to give false instrument readings. The motor generator sets have automatic frequency and voltage regulators and harmonic frequency current filters. In case both motor generator sets fail simultaneously, an automatic throwover to the 4 kv auxiliary bus is provided.

Exciter. Two 475 kw 250 volt 1900 ampere motor driven exciters are provided to supply the field excitation for the generator. The induction type driving motor for each exciter is rated at 700 hp. Normally, only one exciter is operating; the second exciter is held as a reserve machine available for use as the occasion arises.

Flywheels are provided to increase the rotational inertia so that momentary reduction in driving motor torque will not substantially affect the excitation supplied to the generator field. Each exciter is equipped with its own field rheostat, switchgear, manual control equipment, and automatic regulating equipment. The automatic regulating equipment includes a rotating type pilot amplifier which is directly connected to the exciter. Both exciters are operated remotely, from the control room.

FUEL PROCESSING

Function and Plant

The fuel processing system, to a large degree, is patterned after developments conducted by Oak Ridge National Laboratory dealing with handling fuel from power breeders. Processing has the two-fold function of removing fission products from the core solution and thereby minimizing their poisoning effect, and of isolating and decontaminating the U-233 that is formed in the blanket. The process entails the following two phases:

- a. Removal of solid fission products from the core solution by centrifugation.
- b. Isolation and decontamination of U-233 and thorium from the blanket slurry, along with the centrifuge discharge, using the Thorex process.

The processing system, other than the centrifugal equipment for continuous handling of core solution, is illustrated in flow sheet Fig. 21 page 54 and plant drawing Fig. 22 page 55. The fuel processing plant and its integration with the over-all power generating station is illustrated in Fig. 2 page 10. The processing unit is of adequate size to handle the output of three reactors and the layout of the plant is such that expansion to a three reactor and turbine plant installation is possible without appreciable change to the fuel processing system.

Removal of Fission Products. The removal of the fission products from the core solution is effected continuously in centrifugal or cyclonic type separators integrally associated with the reactor. This is possible because a high percentage of the fission products are insoluble in the core solution. The separating equipment is of adequate size to handle about 28,000 liters of core solution per day, equivalent to the volume of the entire core system. A small flow of 117 liters per day, containing the concentrated fission products, is sent to the Thorex solvent extraction plant for decontamination. The remainder, 27,880 liters, is returned directly to the reactor.

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FUEL PROCESSING FLOW SHEET

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Isolation and Decontamination. The Thorex process involves solvent extraction to isolate the thorium and uranium and to effect decontamination of these materials. The problem of decomposition of solvent under irradiation dictates limiting the activity level to about 2.0 beta wathours per liter and requires cooling the blanket and core materials for approximately 55 and 95 days, respectively. These materials are kept separate until they reach the extraction column feed step. Even though the blanket and core streams are kept separated until the feed preparation step, their routing is essentially parallel. Provisions are available in the process for mixing the two streams earlier in the sequence of operation in the event that this is indicated by process modifications; for example, stream dilution to lower specific activity.

The Fuel Process

Dissolution and Decay Storage. Fuel to be decontaminated is received from the reactor plant into the evaporator-dissolvers indicated in cell A, Fig. 22 page 55.

The blanket slurry processing rate is 62.0 kg/day of thorium. This discharge from the reactor contains 1000 gm/liter of thorium as oxide in heavy water, about 8 gm/liter U-233 and one gm/liter of protactinium.

The core fuel processing rate is 210 gm/day of U-233 in solution. The solution consists of an equilibrium mixture of 1.8 gm/liter of U-233 and higher uranium isotopes as sulfates in heavy water.

The heavy water accompanying both the blanket and core fuel is evaporated, collected in cell B and immediately reused. The dried fuel is dissolved or slurried in a light water system and the resultant solution or slurry is stored in critically safe and cooled decay storage tanks indicated in cell C. After the appropriate decay period, material in cell C to be decontaminated is discharged to feed preparation tanks in cell D where reagents are added to render the solution amenable to extraction.

In doing this, the core solution and the blanket slurry are mixed and dissolution is made in concentrated (about 13 molar) nitric acid with a fluoride ion catalyst. After dissolution, the excess nitric acid is removed by evaporation, concentrated in a fractionating column and reused in subsequent dissolutions.

First Stage Extraction. The first stage extraction, in cell E, provides a system for separating protactinium, thorium, uranium, and fission products. About 40% tributylphosphate in a hydrocarbon diluent is used as an extractant in pulse columns. Protactinium and fission products accompany the aqueous stream from the first column; thorium accompanies the aqueous stream from the second column; and uranium is stripped from the solvent in the third column.

The greater portion of the fission products in the protactinium stream of the first column is sent to the waste evaporator in cell F where suitable concentration is effected, and where protactinium decay storage tanks are located. Protactinium remaining at this point is about 25% of that in the original blanket material discharge. After the concentration of the protactinium stream, it is sent to decay storage where it is maintained for 250 to 300 days, allowing decay to uranium-233 before moving to the second stage of extraction.

The thorium stream, as nitrate, coming from the second column in cell E is routed to thorium oxalate decomposition tanks in cell H where the nitrate is converted to thorium oxalate. After this conversion, water is removed by evaporation and the oxalate is dried and decomposed at about 900 F to thorium oxide. Heavy water is added and the thorium oxide-heavy water slurry is rerouted to the blanket storage system.

The uranium stream, as nitrate, overflowing the stripping column in cell E is sent through the U-233 isolation columns located in cell H. These columns are the ion exchange type. U-233 is eluted with an appropriate reagent from the ion exchange columns as uranyl acetate, converted to uranyl sulfate, and evaporated to dryness in critically safe vessels located in cell H. Then heavy water is added to produce a concentration of about 1 molar uranium. This heavy water-uranyl sulfate solution is routed to the core solution storage system in the reactor plant.

Second Stage Extraction. Isolation and decontamination of the U-233 resulting from the decay of protactinium is done in the second stage extraction system located in cell G. The amount of U-233 handled here is about 2.7% of the total U-233 processed from the reactor blanket. The uranium overflows the stripping column and follows the same routing as the uranium stream from the stripping column in cell E. Extraction is effected in the pulse columns utilizing about 7% tributylphosphate in a hydrocarbon diluent as an extractant.

Solvent Recovery. Solvent recovery equipment located in both cell E and cell G is identical, consisting of four cascading-type mixer-settler stages. A series of four washes is made; that is, water, caustic, acid, and water. It is necessary to maintain the solvent separate in cell E and in cell G inasmuch as different concentrations of tributylphosphate are used in first and second stage extractions.

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Auxiliary Equipment. The chemical makeup and feed system is located generally in a central area of the processing plant at an elevation such that the gravity flow into the process cells can be utilized to a large extent. The operating corridor and associated instrument panels are located immediately below the chemical makeup system in the central area.

Liquid Waste System. Throughout the fuel processing plant suitable waste holdup tanks are provided in order that samples may be taken for analysis. In the event that the performance of equipment is below standard, solutions may be recycled through the process.

Drainage tanks for the collection of waste are located in cells J and K. Such wastes are concentrated in the evaporator shown in cell L and the condensate collected in cell H prior to being discharged through a monitored waste system.

The tanks located in cells J and K are utilized, also, to receive wash solutions from the reactor plant during periods of cleaning plant equipment to perform maintenance. These wash solutions may contain adequate fuel to warrant their processing. Provisions are included, therefore, for routing materials from cells J and K to cell A for normal routing through the fuel recovery plant.

The final storage of concentrated radioactive wastes is made in tanks fitted with reflux condensers for heat removal. These tanks are located outside the processing building, as shown in Fig. 2 page 10.

Off-Gas System. In instances where venting of vessels is required, it is done through the vent tunnel, filters, and then to the stack.

Dissolver off-gas resulting from the dissolution of fuel is diluted with air to render a non-explosive mixture of hydrogen, then routed through a caustic scrubber located in cell D and compressed to approximately 100 psi for surge storage. Such surge storage is advisable to minimize the size of equipment downstream from the off-gas scrubber. This equipment is operated continuously whereas dissolver operation is batchwise. From the surge storage tank the gas is filtered and sent through charcoal-type adsorbers to remove gaseous fission products before it is exhausted to the stack. The off-gas building is shown in Fig. 2 page 10.

ECONOMIC ANALYSIS

CAPITAL COST ESTIMATE

Capital cost estimates are shown in Table V, page 60. These costs include expenditures necessary for the over-all generating station with the exception of fuel processing facilities which are considered as a part of fuel costs and are treated in a later section.

Although the main effort was directed toward investigating the costs of the 180 mw unit, consideration was also given to a 94 mw size. Multiple units of the 180 mw size were used in analyzing generating station costs up to 1080 mw.

In addition to the breakdown of costs shown in Table V the over-all cost data are shown in graphical form in Fig. 23 page 61, where capital cost in dollars per kilowatt of generating capacity is shown as a function of generating station size.

Fig. 24, page 62, shows fixed charges/kwh at 80% capacity factor operation and a 25 year life as a function of generating station size. The influence of shorter equipment life over the reactor portion of the plant is shown in Table VI, page 63.

FUEL COST ESTIMATE

Since in an aqueous homogeneous breeder type reactor the main fuel cost is for decontaminating the generated and partially spent fuel, it is necessary to consider the cost of constructing and operating fuel processing plants. Such costs are shown graphically in Figs. 25 and 26, pages 64 and 65, in which capital costs and total operating expenses for plants with different capacities are illustrated. Here an appreciation is gained of the advantage in unit cost when processing through large installations. This advantage can be interpreted in terms of a large generating station or in terms of shipping fuel to a special location for decontamination.

TABLE V. -- Capital Cost Estimate

Account	Description	In Thousands - \$			
		94 MW	180 MW	540 MW	1080 MW
310	Land	350	350	350	350
311	Structure and Improvements	8,290	10,320	26,760	51,500
312.1	Boiler Plant Equipment	1,840	2,794	8,080	15,100
312.2	Reactor Plant Equipment	7,040	10,735	30,960	58,000
312.4	Fuel Handling Equipment and Structure	160	243	700	1,320
314	Turbine-Generator Equipment	6,260	10,523	29,950	56,800
315	Accessory Electrical	1,570	2,163	5,400	8,800
316	Miscellaneous Plant	900	1,035	1,630	2,380
343	Switchyard Equipment and Structure	1,420	1,952	4,720	8,950
	Preliminary Operation and Test	370	580	1,440	2,700
	Engineering and Design	<u>1,800</u>	<u>2,800</u>	<u>7,010</u>	<u>13,100</u>
	TOTAL in \$	30,000	43,495	117,000	219,000

A contingency allowance is included in each account per the following schedule:

Conventional portion (less turbine) -	5%
Reactor portion	- 10%
Turbine-generator unit	- \$100,000

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FIG. 23. -- Generating Station Size Vs Capital Cost

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TABLE VI . -- Capital Expense

Total Generating Station Cost	\$43,495,000
Reactor Equipment	10,730,000
Remainder	<u>\$32,765,000</u>

<u>Reactor Equipment Life</u>	<u>Plant Life Exclusive of Reactor Plant</u>	<u>Capital Expense Mills/kwh (80% C.F.)</u>	<u>Fuel</u>	<u>Operating, Maintenance, and D₂O</u>	<u>Total Power Cost Mills/kwh (80% C.F.)</u>
25	25	4.72	0.98	1.34	7.04
20	25	4.77	0.98	1.34	7.09
15	25	4.89	0.98	1.34	7.21
10	25	5.14	0.98	1.34	7.46
5	25	5.98	0.98	1.34	8.30

FIG. 25. -- Relation of Capital Cost
to Fuel Processing Rate

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FIG. 26. -- Relation of Annual Costs to Fuel Processing Rate

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In addition to the processing costs, there are charges for inventory and consumption. These charges are made in accordance with the following schedule:

a. Fuel grade U-233	\$16/gm
b. Isotopically pure U-233	36/gm
c. Thorium	22/kg
d. Inventory	4%/year
e. Processing loss	0.5%

The combination of processing cost as interrelated with different cost structures shown in Figs. 25 and 26, together with fuel inventory and consumption costs are shown graphically in Fig. 27, page 67. Fuel cost is plotted as a function of generating station size. The influence of different methods of considering inventory charges and breeding gain credits is illustrated.

Of particular significance is the thought of scheduling material through a relatively large installation rather than building an individual processing plant for a small reactor. For example, as indicated by the dotted line in Fig. 27, page 67, a 540 mw plant which would handle about 170 kg of thorium and about 2 kg of U-233 per day would result in fuel costs ranging from about 0.8 to 1.2 mills/kwh, depending on how inventory and breeding gain is handled. If an individual plant of, for example, 180 mw supported its own fuel plant, costs would range from about 2.1 to 2.6 mills/kwh.

OPERATING COST ESTIMATE

Personnel Costs

A personnel chart for a 180 mw net electrical output integrated homogeneous reactor power plant is shown in Fig. 28 on page 68. The chart, comprising a complete plant component of 226 personnel, includes those required for fuel processing. To give a truer fuel cost estimate, the payroll and overhead charges for the 83 individuals directly employed in fuel processing are included in the fuel cost and do not appear in the Table VII. Except for a few additional technicians, the caliber of personnel required is comparable to that required for conventional power plants. The plant operators are organized on a 4-crew basis, each employee working an average of 42 hours per week.

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BASIS - 1. INVENTORY CHARGE AT 4% PER YEAR
 2. INVENTORY INCLUDES HOLD-UP OF FUEL IN SHIPPING AND PROCESSING OF 6 MONTHS
 3. COSTS AS SHOWN EXCLUDE SHIPPING TO CENTRALIZED PLANT. UNDER CONDITION OF DOING THIS AS INDICATED BY DOTTED LINE (---) 0.05 MILLS/KWH MUST BE ADDED TO COVER THIS. THUS TOTAL COST WHEN HANDLING FUEL FOR ② THROUGH 540 MW - FUEL PLANT SIZE IS $0.93 + 0.05$ OR 0.98 M/KWH

FIG. 27. -- GENERATING STATION SIZE VS FUEL COST

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FIG. 28. -- Personnel Organization Chart

A centralized organization, provided with the required shop and portable tool facilities, performs all the maintenance required for the entire integrated plant.

TABLE VII. -- Operating Cost (Exclusive of Fuel)

<u>Operation</u>	<u>Yearly Cost</u>	<u>Mills/kwh</u>
Payroll	\$ 321,000.00	0.255
Payroll Tax, Accident Insurance, Pension, and Employee Welfare @ 16%	51,000.00	0.040
Transportation and Stores Handling	6,000.00	0.005
	<u>\$ 378,000.00</u>	<u>0.300</u>
<u>Maintenance</u>		
Payroll	\$ 480,000.00	0.381
Payroll Tax, etc. @ 16%	77,000.00	0.061
Transportation and Stores Handling	9,000.00	0.008
	<u>\$ 566,000.00</u>	<u>0.450</u>
<u>Material and Supplies</u>	\$ 315,000.00	0.250
<u>Heavy Water</u>		
Leased	\$ 428,000.00	0.340
Purchased @ \$28/lb	856,000.00	0.680
<u>Total</u>		
With leased D ₂ O	\$1,687,000.00	1.34
With purchased D ₂ O	2,115,000.00	1.68

Note: Unit costs in mills/kwh are based on an 80 percent plant capacity factor.

Heavy water costs are considered both on the basis that it may be purchased and on the basis that it might be leased. The latter method appears to be reasonable inasmuch as the water would undergo little if any damage during use. Also, it could be returned to the Commission when no longer needed in the reactor, its only use as far as a utility is concerned.

Variation of Heavy Water Costs with Plant Size

Table VIII summarizes the heavy water costs for four sizes of plant. The two larger size plants are three and six multiples of the 180 mw size respectively. The costs provide for a loss of 5 percent of the heavy water inventory per year.

TABLE VIII -- Heavy Water Costs

<u>Plant Size</u>	<u>System Volume Liters</u>	<u>Mills/kwh</u>	
		<u>Leased</u>	<u>Purchased</u>
94 mw	45,100	0.42	0.83
180 mw	70,500	0.34	0.68
540 mw	211,600	0.34	0.68
1080 mw	423,300	0.34	0.68

Cost Summary

Unit costs for the four size plants including the costs of operation, maintenance, material and supplies, and heavy water (both leased and purchased) are summarized as follows:

TABLE IX . -- Operating Unit - Cost Summary

<u>Plant Size</u>	<u>(Total excluding fuel) Mills/Kwh</u>	
	<u>D₂O Leased</u>	<u>D₂O Purchased</u>
94 mw	2.02	2.43
180 mw	1.34	1.68
540 mw	0.99	1.33
1080 mw	0.89	1.23

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DISCUSSION OF RESULTS

ECONOMICS

Capital

The homogeneous reactor system indicated in Table V, page 60, costs about \$240/kw and \$200/kw at 180 and 1080 net mw respectively.

Basis of Estimates. These costs are based upon estimates which reflect both present day price indices and engineering judgment related to known technology. With the accumulation of experience, improvement associated with the development of a new industry should certainly be realized. The costs presented are considered conservative particularly from the standpoint of projecting present circumstances and their economic potentials.

Depreciation. The effect of rapidly depreciating the reactor portion of the plant is interesting from the standpoint of equipment lifetime uncertainty. This is particularly true in considering operation of a first reactor unit. The expense which may be expected with varying periods of reactor equipment amortization is shown in Table VI, page 63.

Fuel Cost Estimates

In a power breeder, the principal cost of fuel is the decontaminating costs for partially spent fuel and the fuel formed. In processing for decontamination in a plant such as Thorex, the cost per unit of fuel handled decreases materially when quantities increase. This makes it possible to assign appreciably lower values to fuel costs as the generating station capacity is increased from 180 to 1080 mw. This amounts to a reduction from approximately 2.5 to 0.8 mills per kwh, respectively.

The breakdown of such fuel costs is shown in Table X, following.

TABLE X. -- Breakdown of Fuel Costs

<u>Processing</u>	<u>180 mw Mills/kwh</u>	<u>1080 mw Mills/kwh</u>
Amortization	1.16	0.30
Operation	0.75	0.15
Maintenance and Supplies	0.35	0.08
	<u>2.26</u>	<u>0.53</u>
<u>Fuel Inventory*</u>		
Uranium - 4% @ \$16/gram + 0.5% loss	0.26	0.26
Thorium - 4% @ \$22/kg + consumption	0.01	0.01
	<u>2.53</u>	<u>0.80</u>
Credit for breeding gain	0.22	0.22
	<u>2.31</u>	<u>0.58</u>

*Inventory includes 6 months out-of-reactor material to handle "hold up" created by shipping and routing through decontaminating plant.

Fuel Decontaminating Plants. By processing fuel through a centralized plant, the advantages gained by larger rate of processing can be realized. Additional charges of 0.05 mills/kwh must be assigned for shipping costs, but by scheduling fuel discharged from several small systems for processing through a larger plant a saving of 1.5 mills/kwh may be realized. This results in a fuel cost of approximately 1.0 mills/kwh. These conditions are illustrated graphically in Fig. 27, page 67. This centralized processing appears reasonable because of the existence of a pilot plant at ORNL which, it is understood, is designed to handle a quantity discharged from three to four reactors of the 180 mw size.

Operating Cost Estimate

Unit Cost of Operating Personnel. Operating personnel requirements, excluding those for fuel processing, are estimated to be 0.3 mills/kwh based on the organization chart shown in Fig. 28, page 68.

Unit Cost of Maintenance. Maintenance of the turbine unit is considered to be the same as in a conventional plant which entails approximately 1% per year of capital investment. Maintenance of the reactor is estimated to be of the same magnitude as that of an industrial chemical processing plant in the order of 3% per year of capital investment.

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Operation and maintenance cost of fuel handling are assigned primarily to the fuel processing plant and are included in the fuel cost. Operating and maintenance costs based on an 80% capacity factor are summarized as follows:

TABLE XI. -- Operating Costs (180 MW Plant)

	<u>Mills/Kwh</u>
D ₂ O (lease basis)	0.34
Operation	0.30
Maintenance	
Turbine-generator plant - 1%	0.13
Reactor plant - 3%	<u>0.57</u>
TOTAL	1.34

PROBLEMS

There are several major problems now under study at ORNL which require resolution to insure the success of the thorium thermal power breeder. Predominant among these are the following, which have been covered and in all probability will be covered in greater detail in ORNL documents.

Corrosion under Irradiation

Although a considerable amount of corrosion data has been collected in simulated fuel solutions in the absence of irradiation, the amount of information on the conditions that are applicable when irradiation exists is limited. Some of the information on the effects of irradiation indicates that corrosion is accelerated. Whereas it appears that inadequate data exists presently for firm selection of materials of construction and for intelligent planning of equipment life, test programs at ORNL aimed at resolving the problems are gaining momentum.

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Component Development

There are several problems in equipment design and construction which are being investigated actively under the direction of ORNL. Included among these are the catalytic gas recombiner, the simulated reactor core tank, and gas separators. Probably, the item which represents the biggest problem is the steam generator. Because of the large amount of transfer surface required (approximately 0.45 sq. ft./kw generating capacity) there is an economic incentive to build very large generator units to provide a large amount of transfer surface in one steam generator. Because of the high value of fuel there is a need to make the surface to volume ratio high. This is being accomplished with small, say 3/8 in. diameter, tubes or by providing solid cores for somewhat larger tubes. The fabrication and inspecting of exchangers with several thousand tubes such that integrity at high temperature and pressure can be assured represents a major problem. Maintaining such equipment under the condition of reactor operation is difficult.

As is discussed under Plant Design and Operation, the heat exchanger design entails isolation of sections of the heat exchanger during maintenance rather than handling the entire exchanger.

This phase of the reactor plant design warrants substantial effort in optimizing maintenance programs so that an attractive capacity factor for the turbine plant can be assured.

Handling Core Solution

The core solution of 1 to 2 grams per liter of U-233 as uranyl sulfate in heavy water presents relatively minor problems in handling so long as extreme temperatures (less than 150C and no more than 335 C) are avoided. The material is a stable solution, its physical properties approaching those of heavy water.

Handling Blanket Slurry

The blanket material as currently under study consists of thorium oxide slurry in heavy water. Variations in viscosity influence handling and heat transfer. This change develops under variations in working of the slurry or in temperature changes as well as in the presence or absence of irradiation. The apparent or effective particle size appears to change under these conditions.

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The handling of a slurry presents problems of particles settling in low velocity spots and of excessive erosion at high velocity. The buildup of solids on the thermal shield, as an example, may aggravate maintenance in the reactor as well as effecting heat release from the shield.

The influence of methods and specific conditions of thorium oxide preparation on the slurry handling characteristics is under study at ORNL. The investigation of additives which will assure a low viscosity and slowly settling material is likewise under study.

Fuel Processing

The section of fuel processing entailing continuous removal of fission products from the core by centrifugation requires more development than does the remainder of the plant. Should the centrifugation plan be unsuccessful, processing could be carried out completely in the Thorex unit. However, indications are that continuous removal of fission products introduces considerable savings and will justify a sizable development budget. A pilot plant program is in operation on the Thorex system at ORNL. Because of the high amount of background experience in the successful development and design of similar nuclear fuels handling systems, it is considered that fuel can be processed successfully and economically.

Power Density

Satisfactory power density of the HRE of ORNL is understood to have reached as much as 50 kw/liter. There are no apparent limitations to the power density that can be achieved in such reactors. However, some limitation either directly or indirectly probably does exist. This limitation is unknown. The design of the reactor in this report is based on approximately 100 kw/liter maximum power density. It may be possible to exceed this value, whereas the probability of not realizing this density appears remote.

REF: 4537ED

SELECTION OF PRIMARY VARIABLES AND EQUIPMENT

Reasons for the selection of the particular operating conditions and equipment for the homogeneous power breeder are set forth below. In those cases where a physical property of the system limits a variable, the choice may be obvious. Other choices must be made by considering broad objectives, or by involved optimization studies, which were to a large degree beyond the scope of this work. A high amount of dependence has been placed on the experience and information developed at the Oak Ridge National Laboratory.

Power

After consideration of all relevant factors, it was decided that about 200 mw was as large a block of power that NPG interests would want to produce with one reactor. It appeared also that this size would give good insight into the potential of the homogeneous power breeder. In addition, this was the size selected for the boiling reactor study being conducted concurrently.

Two Region Vs One Region

While the choice of the reactor type was made on a somewhat arbitrary basis, the two-region reactor was chosen for the following reasons:

- a. For the sizes considered, two region seems to have somewhat better neutron economy.
- b. Concentrations of uranium can be kept to a low value in the core, thus lessening the corrosion problem.
- c. Fuel solution environment may permit the use of SS-347 as the material of construction, limiting the introduction of exotic materials such as zirconium to the core tank. Fabricating this tank appears to present no problems that are insolvable.
- d. The two-region system is given chemical processing flexibility. For example, it permits different processing rates for the two regions and use of mechanical separation for the core.

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- e. The two-region system permits better evaluation of the core region since it remains more like the HRE No. 1 core. Knowledge gained in HRE No. 1, such as the catalytic effect of CuSO_4 , can be more readily applied.
- f. The two-region system restricts the ThO_2 slurry unknowns to the blanket part of the system. It may lessen the presumed problems of controllability of reactors containing slurries. At this time it is doubtful whether thorium could be introduced into the reactor at the required concentration in a form or compound other than the oxide.

Power Density

ORNL has stated that there are no apparent prohibitions on the power density. For the core, tests conducted in HRE indicated that CuSO_4 will eliminate the water decomposition problem which previously limited the power density. There may be a limit, as yet unknown, such as hydrodynamic stability, which might depend on the particular design. At any rate, there is no known reason for being unduly pessimistic at this time. In ORNL-1642, the Laboratory considered 100 kw/liter a reasonable extrapolation from the peak power of 50 kw/liter achieved in HRE. There was no recombination in the core of the ORNL-1642 design. It is considered that in this design, the core of the reactor will operate at 112 kw/liter which appears reasonable in view of the facts stated above.

It is contemplated that 280 mw will be extracted from the blanket of this reactor which compares to 130 mw removed from a blanket of the same volume reported in ORNL-1642. The over-all power density for the blanket system has not changed appreciably because the volume of the external heat removal system has increased proportionately. Oak Ridge has made optimization studies which indicate that the optimum blanket power may be below the value selected here. However, such studies appear to be influenced to a large degree on fuel processing and inventory economics rather than on nuclear limitations. From the studies made as a basis for this report, which are on a different basis from the ORNL work, indications are that it is advantageous to go to higher blanket powers. There remains a question, nevertheless, as to the properties and behavior of the slurry under these conditions. In addition, the recombination problem remains unsolved, even though there is some preliminary information that the ThO_2 particles catalyze the reaction. One of the consequences of poor properties would be a reduction in the power which could be removed practically from the blanket.

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For the reactor as a whole, it is considered confidently that the total power can be extracted by increasing core power if the stated blanket power is not achievable. This confidence stems from (1) the flexibility of the homogeneous system and (2) the experience to date which has used this flexibility to great advantage in solving difficulties. The consequences of higher core powers, while trying to maintain a total reactor power at 640 mw heat, have not been evaluated. There would be some penalty, although probably not prohibitive, in increased chemical processing rates for the blanket.

Temperature

The top temperature of 572 F was chosen because at higher temperatures the solution separates into a uranium rich phase and a uranium deficient phase. Lower temperatures may be ruled out on the following grounds:

- a. The CuSO_4 catalyst at a given concentration is less effective at lower temperatures.
- b. Corrosion rates appear to be less at temperatures above 480 F. This may be attributed to the character of the film on the metal.
- c. Highest temperatures attainable are preferable for best thermal efficiency.

A ΔT or temperature increase of the fluid flowing through the reactor of 75 F was chosen. The general range of ΔT was based on the limitations cited above and some practical considerations of flows and heat exchanger design. The final selection of 75 F was arbitrary. However, limited studies tend to support this and to indicate that there is some latitude in ΔT before the power cost is affected significantly.

Pressure in Primary and Secondary Systems

The pressure in the primary system must be high enough to prevent boiling. The vapor pressure of water at 572 F is about 1250 psia. A pressure of 1800 psia was selected for the system. This degree of much overpressure is influenced by the required partial pressure of D_2 and Q_2 for satisfactory operation. The selection made is considered conservative.

A steam pressure of 565 psia was selected. This pressure has been used by others in studying the aqueous homogeneous reactor and was supported by a limited cost study involving heat exchangers, inventories, fuel, and turbine. Within the limits of the study for a reactor ΔT of 75 F and a top temperature of 572 F, results of the study indicated that 565 psia was in the right range.

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Primary Core Piping Velocity

The economic pipe diameter or velocity for the piping which connects the reactor and the external heat removal system is calculated by optimizing the pipe, pumps, inventory, and pumping costs.

Pipe. The cost of SS-347 pipe varies approximately as the weight or amount of material. The material in this pipe for schedule 140 in sizes ranging from 8" - 20" varies approximately as the internal diameter to the 1.9 power. Further, using cost figures supplied by the Babcock & Wilcox Company for their hollow forged and bored SS-347 pipe and using a basis for fabrication and erection of 120% of the material cost, the following equation can be written:

$$C_p = \text{Cost pipe, } \$/\text{year-ft.} = (a+m)(\$5.00)(D_i)^{1.9} \quad (1)$$

where: $(a+m)$ = amortization and maintenance, assumed to be 20%

D_i = internal diameter of pipe, in.

Pumps. The pump costs were estimated from an equation in CF-54-5-26 (See page 131). The amortization and maintenance for this report was assumed to be 30%. (10 year life)

$$C = \text{Cost pump, } \$/\text{year} = 0.30[6.14 \text{ gpm} + 83.3 (\text{whp}) + 2.56 \Delta p] \quad (2)$$

where: gpm = gallons/min. flow
whp = water horsepower
 Δp = pressure drop, lbs/ft²

Inventory. The inventory charges are given by:

$$C_I = \text{Inventory charge, } \$/\text{year-ft.} = rA\rho S \quad (3)$$

where: r = annual charge for non-depreciating item
 A = cross sectional area of pipe in ft²
 ρ = density of solution lb/ft³
 S = cost of solution in \$/lb

Substituting in the above equation:

$$C_I = \frac{0.12(D_i)^2 \pi (50)(46)}{(4)(144)}$$

Basis:

The density of D₂O at 300 C. is ~50 lb/ft³
Each liter contains 2 grams of U-233
U-233 is worth \$16/gm
D₂O is worth \$28/lb
Annual charges are 12%

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Pumping. Cost of pumping is calculated in the following manner:

$$\frac{dF}{dL} = \frac{4fV^2}{2g_c D} \quad (\text{Fanning Equation-circular pipe})$$

$$\text{but } V = \frac{4w}{\pi D^2 \rho}$$

Therefore,

$$\frac{dF}{dL} = \frac{4f}{2g_c \pi^2} \frac{16w^2}{D^5 \rho^2} = \frac{32fw^2}{g_c \pi^2 \rho^2 D^5}$$

where f = friction factor, dimensionless

w = mass flow in 1000 lb/hr

g_c = conversion factor,
lb_m-ft

lb_f-sec²

ρ = density, lb/ft³

D = internal pipe diameter, in.

Then the equation becomes:

$$\frac{dF}{dL} = \frac{32fw^2}{32.2 \pi^2 \rho^2 D^5} \times \frac{2.49 \times 10^5}{12.95 \times 10^6} = 0.01938 \times \frac{10^5 \times fw^2}{\rho^2 D^5} \frac{\text{lb}_f}{\text{lb}_m}$$

From McAdams, p. 119 (see page 131), in N_{Re} range 5000 to 200,000,

$$f = \frac{0.046}{(N_{Re})^{0.2}}$$

$$f = \frac{0.046}{\left[\frac{DV\rho}{\mu} \right]^{0.2}} = \frac{0.046}{\left[\frac{D4w}{\mu \pi D^2} \right]^{0.2}}$$

Upon substituting the proper units, which are also consistent with the dimensions already used, the following results:

$$f = \frac{0.046}{\left[\frac{4w \times 1000 \times 12}{2.42 \mu \pi} \right]^{0.2}} = \frac{0.046}{\left[\frac{6300 w}{\mu D} \right]^{0.2}}$$

where μ = centipoise

D = inches

ρ = lb/ft³

w = 1000 lb/hr

SELECTION OF PRIMARY VARIABLES AND EQUIPMENT

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$$f = \frac{0.0084 w^{0.2} D^{0.2}}{w^{0.2}}$$

$$\frac{dF}{dL} = 0.154 \times 10^2 \frac{w^{1.8} \mu^{0.2}}{\rho^2 D^{4.8}} \frac{lb_f}{lb_m}$$

The cost of pumping, C_{pp} , may then be expressed as:

$$C_{pp} = \frac{0.006 w^{2.8} \mu^{0.2} Y K}{D^{4.8} \rho^2 E} \text{ \$/year} \quad (4)$$

where Y = hours of operation/year = 7000
 K = cost of power in \\$/kwh = 0.007
 E = efficiency of pump = 0.60

Summing equations (1), (2), (3), and (4), the total annual cost per foot of pipe becomes:

$$C_T = (a+m)(5)(D)^{1.9} + 0.30 [0.14 \text{ gpm} + 83.3 (\text{whp}) + 2.56 \Delta P] + \frac{0.03 \pi (D)^2 (2300)}{144} + \frac{0.006 w^{2.8} \mu^{0.2} Y K}{D^{4.8} \rho^2 E} \quad (5)$$

Inserting in equation (5), the values that remain constant:

$$C_T = D^{1.9} + 1.50 D^2 + 4.6 w + \Delta P (25.25 \times 10^{-5} w + 0.77) + 1.9 \times 10^{-4} \frac{w^{2.8}}{D^{4.8}}$$

where:

$$\begin{aligned} \Delta P &= 0.154 \times 10^2 \frac{w^{1.8} \mu^{0.2}}{\rho D^{4.8}} \\ &= 0.194 \frac{w^{1.8}}{D^{4.8}} \end{aligned}$$

Substituting:

$$C_T = D^{1.9} + 1.50 D^2 + 4.6 w + \frac{w^{1.8}}{D^{4.8}} [4.95 \times 10^{-5} w + 0.149] + 1.9 \times 10^{-4} \frac{w^{2.8}}{D^{4.8}}$$

Differentiating total annual cost/ft. of pipe with respect to diameter:

$$\frac{dC_T}{dD} = 1.9 D^{0.9} + 3.0 D - 2.38 \times 10^{-4} \frac{w^{2.8}}{D^{5.8}} - 0.71 \frac{w^{1.8}}{D^{5.8}} - 9.13 \times 10^{-4} \frac{w^{2.8}}{D^{5.8}} = 0$$

$$1.9 D^{6.7} + 3.0 D^{6.8} = 11.51 \times 10^{-4} w^{2.8} + 0.715 w^{1.8} \quad (6)$$

Equation (6) above is used to calculate the economic diameter and optimum velocity for the reactor core fluid at a power of 360 mw. The results are presented in the following table:

TABLE XII-- Economic Velocities -- Core Piping

$\frac{\Delta T}{F}$	$\frac{W}{lb/hr \times 10^{-3}}$	$\frac{D}{in.}$	$\frac{V}{ft/sec}$
25	41,000	23.6	75
50	20,500	18.0	64.5
75	13,667	15.0	62.0
100	10,250	13.4	58.0

Primary Blanket Piping Velocity

The same basic equations as illustrated above are used to calculate the optimum blanket piping diameter. The following values for the properties were employed:

$$\begin{aligned}\rho &= 113 \text{ lb/ft}^3 \\ \mu &= 0.5 \text{ centipoise} \\ S &= 42.50/\text{lb slurry}\end{aligned}$$

The following equation, which relates the optimum pipe diameter and weight of fluid flowing, results:

$$1.9D^{6.7} + 6.28D^{6.8} = 2.33 \times 10^{-4} w^{2.8} + 0.452w^{1.8}$$

The optimum diameter and velocity for a $\Delta T = 75$ was calculated. The results are tabulated below:

$\frac{\Delta T}{F}$	$\frac{W}{lb/hr \times 10^{-3}}$	$\frac{D}{in.}$	$\frac{V}{ft/sec}$
75	26,900	14.5	57

The high optimum velocities for both core and blanket fluids are a result of the expensive fluids, the high cost of pumps and piping, and rapid amortization. Allowing the inventory charges to approach zero decreases the optimum velocity to 43 fps. It is included, corrosion permitting, that it would be advantageous to operate at higher velocities than those used in the design.

Chemical Processing Rates and Costs

Core Processing. In this study an examination was made of the economic chemical processing rates for the two-region homogeneous power breeder. It was impossible to make a conclusive quantitative analysis of the problem because of the unavailability of complete and accurate information. The qualitative results obtained indicate that power cost is rather insensitive to variations in processing rates from approximately 3% to 8% poison.

In ORNL-1642 and ORNL-CF-54-7-63 (See references No. 5 and 13), for slightly different conditions, results indicate that the economic rate is around 7% poison. Most of the basic information used in the following calculation has been taken from Oak Ridge reports. However, different cost figures have been used and individual interpretations have been made on information available.

The chemical processing cycle is optimized by considering the variation of processing costs versus loss of neutrons to the fission products as the rate changes. The value of neutrons lost to the fission products is given by:

$$\$/\text{day loss to F.P.} = 1.64 \times 10^{-15} \phi V \left[\frac{0.8 T_1 T_2}{T_1 + T_2} + 0.2 T_2 \right]^*$$

* Basic equation from ORNL-1642.

Where: ϕ = average core flux, neutrons/cm²sec

V = value of U-233, \$/gm

T₁ = cycle time for cyclones, days

T₂ = cycle time for Thorex, days

The high cross-section gas and high cross-section stable fission products are not included since they are unaffected by processing rates in the economic range. A constant to account for these poisons can be added to the poison fraction calculated from the optimization study. Other conditions and assumptions are as follows:

- a. Yield of fission products important in chemical processing rates - 1.8 atoms/fission at an average cross-section of 30 barns.
- b. 80% of these fission products are insoluble and removed by the cyclones.

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- c. The cyclone cycle is one core system volume/day.
- d. The average core flux is 2.3×10^{15} neutrons/cm²sec.
- e. The value of U-233 is \$20/gm.
- f. Loss of a neutron is equivalent to a loss of a U-233 atom.

The total cost of chemical processing is given by the following factors:

$$\begin{aligned} \text{Total cost processing} &= \text{losses of U-233 during processing} \\ &+ \text{cost of Thorex processing} + \text{cost of decay storage} + \text{cost} \\ &\text{of cyclone processing, or} \\ \text{Total Cost Processing, \$/day} &= \frac{NFV}{T_2} + \frac{K}{T_2 \cdot 0.6} + \frac{XN}{T_2} + \frac{N}{T_2} + \frac{DVA}{365} + C \end{aligned}$$

Where: N = Total grams of uranium in core system
 F = Fractional loss of uranium during processing
 V = Value of U-233, \$/gm
 K = A constant related to the fixed costs of the Thorex plant
 X = Operating costs for processing core solution, \$/gmU
 D = Decay time for core solution, days
 A = Leasing charge for uranium, %/year
 C = Total cost of cyclone processing, \$/day

The heavy water is recovered in a day or two and returned to the reactor. The inventory charge for this is found to be negligible in relation to the other costs. The cost of recovery and losses of D₂O likewise can be neglected.

The following values* were used to evaluate the processing costs:

N = 45,000
 F = 0.005
 V = 20
 K = 18,300
 X = 1 and 3
 D = 95 (ORNL)
 A = 6.0
 C = 150

*Unadjusted to later information.
 Will not affect conclusions.

The core and blanket streams are processed in plant and equipment common to both. It is difficult to select that part of the chemical processing plant which is assignable to core processing. Without a comprehensive study, the variation of costs with processing rates is open to question. The Thorex processing costs are given by the second and third factors in the equation. It is arbitrarily assumed that 1/3 of the plant

fixed charges are due to uranium processing. It is estimated that this charge is \$1000/day at a rate of 450 gm/day of uranium and that it varies inversely as the 0.6 power of the cycle time. The operating costs are taken as either \$1 or \$3 per gram processed.

The results are shown in Fig. 29, page 86. The total cost curve is shallow and the cycle time can vary from 100 to 200 days, changing the power cost only about 0.15 mills/kwh. For this study a cycle time of 200 days was selected, corresponding to a poison fraction of 5.2% or approximately 7 - 8%, when the gases and high cross-section fission products are added.

If the cost figures are varied, such as value of U-233, the shape of the total cost curve will change appreciably. By increasing values of uranium, sharper minimums are obtained and less economic latitude exists in the selection of processing rate.

Fig. 30, page 87, shows the results if the cyclones are not in the system. The minimum, as could be expected, has moved to shorter cycle times, and at a higher poison fraction. Comparing points at the same poison fraction (5.2%) from the two figures, indications are that a differential of about \$2300/day exists, which is equivalent to about 0.6 - 0.7 mills/kwh. Capitalizing the annual saving, one could invest $\$5 \times 10^6$ to develop and install the cyclones. In ORNL-CF-54-7-63, Oak Ridge estimated that removing the cyclones would make a difference of 0.1 - 0.15 mills/kwh in power cost. Even this differential means that one could afford to spend about $\$1 \times 10^6$ for the cyclones. The difference between estimated savings here and that of Oak Ridge is due to the higher costs used for the Thorex processing in this report.

Blanket Processing. The blanket is processed at the rate necessary to maintain the equilibrium concentration of uranium. An excess of 452 grams of uranium is formed in the blanket per day over what is burned therein. At a concentration of 8 grams/liter, 56.5 liters per day are removed from the blanket for processing in the Thorex plant.

Heat Exchanger Design

Basic Requirements. The ideal heat exchanger for the homogeneous reactor must fulfill rather stringent requirements. Among the more important are low inventory, small pressure drop, reliability, and ease of fabrication. Preliminary thoughts were directed toward minimizing the heavy water and uranium inventory at the expense of decreased reliability and present feasibility of fabrication. It became apparent after discussions with heat exchanger manufacturers, that relaxing the inventory requirement would lead to a more practical design.

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FIG. 29. -- Optimization of Core Solution
Cycle Time Through Thorex Plant

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FIG. 30. -- Optimization of Core Solution
Cycle Time Through Thorex.
No Hydroclones

Minimizing the inventory necessitates use of small diameter tubes since the surface to volume ratio of a tube varies inversely as the diameter, or use of larger cored tubes. The smallest tube considered was 3/8 in. O.D. which resulted in the use of approximately 30,000 tubes in order to obtain sufficient flow cross section for the total coolant at a velocity of about 10 fps. Calculations show that little can be gained in over-all heat transfer coefficient or reduction in heat transfer area by increasing the primary fluid velocity above 10 - 12 fps. Available information indicates that inside film coefficient is not controlling; consequently, increasing the velocity would result in an increase in tube length, and a very rapid increase in pressure drop with no compensating decrease in inventory. A maximum allowable pressure drop of 30 psi across the exchangers was used.

Design Concept. Tentatively two heat exchanger designs were selected as offering promise in meeting design and operating requirements. One is a Griscom-Russell concept of a "bottle" U-tube heat exchanger. The other is a cored-tube-U shell exchanger by Babcock and Wilcox. A discussion of these designs follows:

"Bottle" Exchanger. This design has two principal advantages: (1) It minimizes the risk during fabrication; (2) it may give operating flexibility by making it feasible to isolate a defective tube or joint without the necessity for removing an entire heat exchanger.

One of the more difficult components to fabricate in a large high pressure heat exchanger containing a large number of tubes is the tube sheet. Drilling several thousand small diameter (3/8 in. - 1/2 in.) holes in a thick tube sheet is difficult and costly. Grouping the tubes into bundles, providing small tube sheets for each bundle, and carrying just one 2 in. pipe through the tube sheet virtually eliminates these difficulties. In addition, the consequences of rejection at any stage in fabrication are lessened by compartmentalizing.

As to operating aspects, it is considered that a method can be worked out for locating the bottle containing a defective tube or joint and isolating this bottle from the system. This design could obviate the necessity for removing an entire shell for any tube failure. It would reduce outage time for maintenance and permit operation until a number of bottles have been eliminated since one bottle represents less than 0.25% of the total heat removal system.

Calculations show that the total heat transfer area for the core exchangers is 44,500 sq. ft. (based on outside tube surface). The greatest uncertainty in the calculation of the over-all heat transfer coefficient is in the selection of scale coefficients. A value of 2000 was chosen for each scale, inside and outside the tubes. There are additional uncertainties in the design of the blanket heat exchangers because the physical properties of the slurry are unknown. The estimated properties presented in ORNL-CF-54-5-59 (see reference No.18) were used.

The physical constants used in the calculations and the results obtained are summarized in Table XIII, page 90.

U-Shell-Exchanger-Cored-Tube. An alternate design using 1 in. O.D. cored-tubes appears to be another practical present-day design. The tubes can be inspected ultrasonically and expanded into the tube sheet with known techniques. The number of joints at the tube sheet is reduced which not only decreases the probability of leakage, but facilitates fabrication of the tube sheet. The tube sheet would be larger in diameter than a sheet for a comparable exchanger employing smaller tubes, but would be easier to drill because of the fewer large diameter holes. A heat exchanger of high integrity would result from the fabricating viewpoint. The penalty for this advantage is an increase in heat exchanger inventory of about 20% over a heat exchanger employing 3/8 in. O.D. tubes. Under cost assumptions made in this study, this inventory increase means an added fuel cost of perhaps 0.05 mills/kwh. The order of magnitude costs obtained from the Babcock and Wilcox Company indicate that nearly \$600,000 might be saved in fabricating these exchangers over those with 3/8 in. O.D. tubes. The reduced capital expenditure more than offsets the increased inventory charge.

Heating and Thermal Stress in Pressure Vessel

The calculations of the thermal stresses to be encountered in the pressure vessel design made in this report are adequate for conceptual design and economic studies. In analyzing this problem, the assumptions employed are:

- a. The spherical shell is an infinite slab, 6 in. thick.
- b. Incident thermal neutron current is 1.6×10^{12} n/cm² sec.
- c. Incident primary 5 mev gamma current is equal to neutron current.
- d. All incident energy is absorbed.

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TABLE XIII. -- Physical Constants and Heat Transfer Data

	Core	Blanket
Physical Properties, Fluids		
ρ , lb/ft ³	50	112
μ , centipoise	0.1	0.5
k , Btu/hr-ft. F	0.33	0.47
c_p , Btu/lb-F	1.2	0.49
Heat transferred, Btu/hr		
Boiling section	1100 x 10 ⁶	833 x 10 ⁶
Preheat section	130 x 10 ⁶	123 x 10 ⁶
Temperatures, F		
Primary fluid		
Inlet	572	572
Outlet	497	497
Secondary		
Inlet	405	405
Outlet	480	480
Primary Fluid Velocity, fps	11.6	7.8
Heat Transfer Coefficients, Btu/hr ft ² F (Based on outside tube surface)		
Boiling Section, over-all	540	500
Inside tube	4750	3000
Outside tube	3300	3300
Scale (In and Out)	2000	2000
Preheat Section, over-all	500	370
Log Mean ΔT , F		
Boiling Section	51.5	53.0
Preheat Section	51.5	53.0
Heat Transfer Area, Total, ft ²		
Boiling Section	44,500	38,000
Preheat Section	39,450	31,400
	5050	6600
Pressure Drop, psi, Header to Header	26	22.8

- e. There is uniform volumetric heat generation in the metal.
- f. There is steady-state temperature distribution.
- g. Heat transfer coefficient inside the vessel is 200 Btu/hr ft²F.
- h. Heat transfer coefficient outside the vessel wall is 0.1 Btu/hr ft²F.
- i. Maximum allowable ΔT in pressure vessel wall is 25 F.

Case I - No Thermal Shield. By using the neutron and gamma currents stated above and the graphs shown in NAA-SR-942 (see reference No.17), the uniform volumetric heat generation was calculated to be 15,450 Btu/hr ft³ in the shell. The following additional operating conditions and properties were specified:

k = thermal conductivity of vessel shell = 25 Btu/hr ft F

t_{in} = temperature of fluid inside vessel = 572 F

t_{out} = ambient temperature outside vessel = 100 F

The neutral plane or the plane from which heat flows in either direction can then be calculated. The temperature distribution was calculated from the following equation:

$$t_x = -t_a + \frac{q'''}{2k} (a^2 - x^2)$$

where

t_x = temperature at any point "x" measured from neutral plane.

q''' = volumetric heat generation in Btu/hr ft³.

x = distance from neutral plane to a point "x", ft.

t_a = temperature of surface, F.

k = thermal conductivity of metal, Btu/hr ft F.

a = distance in feet from neutral plane to surface.

The resulting temperatures are as shown in Fig.31

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FIG. 31 -- TEMPERATURE DISTRIBUTION
IN VESSEL WALL
(WITHOUT THERMAL SHIELD)

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The heat transfer coefficient on the outside corresponds to an insulated vessel and results in virtually all the generated heat flowing to the inside. By improving the cooling on the outside of the vessel (at least up to the point where the inner and outer surface temperatures are equal), the maximum difference between the average temperature and the temperature at any point in the metal would decrease. The maximum ΔT (71F) in this case occurs at the inside edge which corresponds to an approximate thermal tensile stress of 15,400 psi. This stress alone exceeds the allowable working stress, and is almost a factor of three higher than the stated maximum ΔT of 25F. It was anticipated that a thermal shield would be required.

Case II - 4 in. Thermal Shield - 6 in. Vessel Wall.

Under the same flux conditions stated in Case I, the uniform volumetric heat generation in a 4 in. shield would be 23,630 Btu/hr ft³. The thermal shield is made of SS-347 and is assumed to be cooled equally well on both sides. The thermal conductivity of SS-347 is about 11. The resulting temperatures in the thermal shield are as shown in Fig. 32, page 94.

The maximum ΔT is 19 F, and the corresponding maximum thermal stress will be approximately 5500 psi. This is a reasonable stress, in particular since there is virtually no pressure stress across the shield. The stress could be decreased by separating the 4 in. shield into two 2 in. shields without impairing the shielding effect. Engineering flow tests will be required to establish the proper shield design and flow characteristics.

From graphs in NAA-SR-942 estimates can be made of the primary and secondary gamma-energy which emerges from the thermal shield and generates heat in the pressure vessel. The total emergent energy under the assumptions of the incident neutron and gamma currents is 1320 Btu/hr ft². Converting this to a uniform volumetric heat generation rate in the 6 in. thick slab representing the vessel wall, 2640 Btu/hr ft³ results.

All the heat generated in the pressure vessel wall is assumed now to flow to the inside. This assumption closely approximates the conditions of Case I. The difference in temperature between the inside wall surface and the outside is 13.2 F. The maximum difference between the average temperature and the temperature at any point in the wall occurs at the inside surface and is 7.3 F. This is well within the specified limit of 25 F.

For purposes of this study, it was concluded that a 4 in. thermal shield would reduce the thermal stress in the pressure vessel wall to a safe value.

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FIG. 32 -- TEMPERATURE
DISTRIBUTION IN
THERMAL SHIELD

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Heating of Biological Shield

The heating in the biological shield has been estimated to be 50 Btu/hr ft², or approximately 20,000 to 25,000 Btu/hr. Provision has been made for cooling the biological shield because only a small thermal gradient can be tolerated.

Pumping Power

Core. The calculated pressure drop for the core system is 58.4 psi, divided as follows:

26.1 psi	- heat exchangers
22.3 psi	- primary piping
10.0 psi	- reactor core (estimated)
58.4 psi	- total

This pressure drop is equivalent to 168 feet of core solution and appears to be reasonable. The major portion of the pressure loss is due to elbows in the system. They are undesirable from another standpoint in that accelerated corrosion might take place at these points because of increased turbulence. In subsequent phases of design, further effort would be made, within inventory and design limitations, to decrease the drop.

The total power required for the core circulating system is 1900 hp or 1420 kw at 60% pump efficiency.

Blanket. The calculated pressure drop for the blanket system is 79.4 psi, as follows:

22.8 psi	- heat exchangers
46.6 psi	- piping
10.0 psi	- reactor blanket (estimated)
79.4 psi	- total

This pressure drop is equivalent to 102 ft. of blanket slurry. The slurry density and viscosity used in the calculations were 112 lb/ft³ and 0.5 centipoise respectively. Some difficulty has been experienced with viscosity changes in the slurry after pumping, heating, and cooling, but it has been assumed that this can be eliminated.

The total power required for the blanket circulating system is 2230 hp or 1670 kw at 60% pump efficiency.

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Reactor Building Sphere

Assuming that a central station power reactor would be located near a center of population, a secondary safeguard should be provided to prevent radioactive contamination of the area should a catastrophe occur. The potential for such an accident exists, but the probability of occurring is low. The assumption is made that all the fission products and other materials such as plutonium or uranium will be released to the atmosphere unless contained. In the homogeneous reactor, where the fuel and fission products are in intimate contact with the water at high pressure, a mechanical failure, such as a ruptured pipe, could cause a release of a dangerous amount of radioactivity. The accepted method of containment at the present time is to provide a pressure-tight building for the reactor.

In this design a pressure-tight spherical vessel is proposed as a satisfactory safeguard. No effort has been made to compare this approach with other methods or even to other types of pressure-tight containers. The problem is recognized and the provision is sufficient for this preliminary design. The precedent established by the construction of the large sphere at West Milton influenced this choice.

Calculations of the equilibrium pressure and temperature were made for a 175 ft. diameter sphere, which is the smallest sphere that will contain the equipment in the reactor system. The main assumptions and conditions follow:

- a. Primary and secondary fluid systems are ruptured.
(total lb-mols water = 25,850)
- b. All the heavy and light water is available for flashing.
- c. All the energy of the fluids is used in flashing.
- d. All the available energy is represented by the water at temperature and pressure, i.e. any additional energy as a result of a runaway has not been considered.
- e. The sphere is originally filled with air at one atmosphere and 100 F.

The calculated equilibrium conditions were as follows:

TABLE XIV. -- Equilibrium Conditions for Reactor Sphere

<u>Temperature</u>	<u>Total Pressure</u>	<u>Partial Pressure, Water</u>	<u>lb-mols water vaporized</u>
230 F	39.5 psia	21.0 psia	7950

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On the basis of this calculation a 175 ft. sphere designed for 40 psia appears to be adequate to contain all materials that may be released should a catastrophe occur.

NUCLEAR PHYSICS

SUMMARY

The reactor design chosen consists of a 6 foot diameter spherical core of uranyl sulfate dissolved in heavy water. The core is contained in a 1/2 inch thick tank of Zircaloy II and is surrounded by a 2 foot thick spherical blanket of thorium oxide and uranium in heavy water. The reactor is operated at 300 C at 640 mw heat power, 360 mw from the core and 280 mw from the blanket. The computed concentration of U-233 required is 1.80 gm/liter for the core and 8.00 gm/liter for the blanket. The breeding gain of this reactor is 0.18 without fission products in the core. The processing rate selected results in a fission product poison level of 7% in the core. This, in effect, reduces the breeding gain to 0.11. The core concentration is approximately proportional to the poison level maintained. The doubling time of this reactor is 21.6 years at an excess U-233 production rate of 49 gm/day.

The calculations made by Oak Ridge for a reactor closest to the selected reactor design in physical dimensions and operating conditions of this study were checked in detail by means of the two-group, two-region method. The concentration and breeding gain were then determined by extrapolating the Oak Ridge data.

An important advantage of the aqueous homogeneous reactor is its flexibility of operation. The reactor can be started with U-235 and produce U-233. The concentration of U-235 needed for criticality is 2.0 gm/liter in the core and 9.0 gm/liter for the blanket. Initially, the conversion ratio is 0.95. This increases to 1.00 after 130 days. From this time on, no more U-235 feed is required.

Another significant advantage of the homogeneous reactor is that this reactor can always be made critical by slight changes in concentration in the event that calculations are in error. No physical changes are required.

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INTRODUCTION

This design produces power and fuel in an excess of that which is burned. Fission products and a chain of higher order isotopes of uranium are produced during operation of the reactor. Fission products are poisonous to the reactor and must be economically minimized by an appropriate rate of chemical processing. Both the breeding gain and the uranium isotope buildup at steady state operating conditions are dependent on the processing rate. Equations are developed showing how these factors affect the reactor calculations.

Many interrelated factors and their effect on operating conditions have been analyzed. Among these are the effect of physical size, temperature, chemical processing rate, and power levels on breeding gain and fuel concentration. Tables of the physical characteristics of the reactor as well as sample calculations, neutron balance, and nuclear constants are presented in later sections.

TWO-REGION REACTOR CHARACTERISTICS

The characteristics of a two-region thermal breeder reactor are exhibited in Table XV, page 100. Such factors as its physical dimensions and operating conditions are included.

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TABLE XV.- Characteristics of Two-Region Thermal Breeder Reactor

Total Power - 640 mw
Exit Temp. - 300 C (572 F)
Breeding Gain - 0.11 (7% poisons)

Core

Power, mw	360
Diameter, ft	6
Volume (core), liters	3,200
(core + external system)	28,760
Power density, kw/liter	112.5
Average flux (core), neutrons/cm ² sec	2.3×10^{15}
(core + external system)	2.5×10^{14}
U-233 concentration, gm/liter	1.80
Total U-233 in system, kg	51.8
Burnup U-233, gm/day	447

Blanket

Power, mw	280
Thickness, ft	2
Volume (blanket), liters	14,350
(blanket + external system)	41,785
Power density, kw/liter	19.5
Average flux (blanket), neutrons/cm ² sec	1.1×10^{14}
(blanket + external system)	3.8×10^{13}
Th-232 concentration, gm/liter	1000
U-233 concentration, gm/liter	8.00
Total U-233 in system, kg	334.3
Burnup U-233, gm/day	332
Consumption Th-232, gm/[REDACTED]	828

Physical Dimensions, Composition, and Operating Conditions

Reactor Core. The reactor core is spherical in shape and is 6 ft. in diameter. It contains a fuel concentration of 1.80 grams of U-233 per liter of solution, as uranyl sulfate (UO_2SO_4) in heavy water (D_2O). It is estimated that the core contains 3% voids due to gas bubbles, resulting in a reduction in average core density. The properties of the core material were adjusted accordingly. The core power is 360,000 kw heat at a power density of 112.5 kw/liter. The core container, separating the core from the blanket, is made of 1/2 in. thick Zircaloy II which has low neutron absorption properties.

Operation of the reactor will result in splitting of the U-233 atoms producing heat and fission products. The fission products are poisonous to the core and are removed by continuous chemical processing. The processing rate leaves a constant 7% fission product poisoning in the core under steady state operating conditions.

Reactor Blanket. The reactor blanket is a spherical annulus 10 ft. O.D. and 2 ft. thick. It contains 1000 gm/liter of thorium as oxide and uranium in heavy water. No void corrections were made for the blanket. The blanket power is 280,000 kw heat at a power density of 19.5 kw/liter.

Operation of the reactor results in neutron capture by the thorium and the buildup of isotopes of Pa-233, U-233, U-234, U-235, U-236, and fission products. At equilibrium conditions, 8.00 gms U-233/liter of slurry are maintained in the blanket. For every atom of U-233 burned in the reactor core, 1.11 atoms of U-233 are produced in the blanket. Thus the blanket produces a breeding gain of 0.11, with 7% fission product poisoning in the core.

The blanket slurry undergoes a continuous chemical processing which separates the fission products, protactinium, thorium, and uranium isotopes. The protactinium-233 is held in storage to decay to U-233 and is then separated from the fission products. The U-233 is fed to the core to maintain the reactor criticality. The gain in U-233 may be sold. The consumed thorium-232 is replaced to maintain the blanket fuel.

Core Diameter vs Core Uranium Concentration

The variation of core diameter with core uranium concentration at different temperatures⁽¹⁾ is plotted in Fig. 33, page 102. Since the calculations for U-235 or U-233 systems are similar, the results on the performance of converters are directly applicable to breeders. It is seen that decreasing the temperature lowers the required fuel concentration, thereby decreasing the neutron leakage from the core. However, the conversion ratio also decreases since relatively fewer neutrons reach the blanket.

FIG. 33 THE REQUIRED CONCENTRATION OF U-235 IN $\text{UO}_2\text{SO}_4\text{-D}_2\text{O}$ HOMOGENEOUS SPHERICAL REACTORS

(FROM CF-51-10-110, PAGE 11)

FIG. 34 VARIATION OF CORE U-235 CONCENTRATION
WITH CORE AND BLANKET TEMPERATURE

(FROM CF-51-10-110, PAGE 28)

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Variation of Core Concentration with Core and Blanket Temperature

The variation of core concentration and core tank absorption Σ_{at} with core and blanket temperature is shown in Fig. 34, page 103⁽¹⁾. "Maintaining the core temperature constant but lowering the blanket temperature at constant thorium concentration, leads to a decrease in core concentration of fuel since the neutron reflecting property of the blanket is enhanced with the higher density of the D₂O. Furthermore, the flux at the shell is increased, contributing to greater neutron absorption by the core tank"⁽¹⁾.

Variation of Core Concentration with Blanket Thickness and Thorium Concentration

"The effect of the concentration of the thorium in the blanket on the fuel concentration in the core is shown in Fig. 35, page 105, while the effect on the conversion is shown in Fig. 36, page 106. Decreasing the concentration of thorium in the blanket and hence its macroscopic absorption cross-section results in a more effective reflector which, in turn, reduces the critical concentration of the core. However, the conversion ratio also decreases with decreasing thorium concentration and this is attributed to two factors: first, an increase in slow neutron flux at the core shell resulting in increased absorption by the shell and second, increased losses to the D₂O in the blanket.

Decreasing the blanket thickness from 2 ft. to 1 ft. increases the leakage losses to 16% of the neutron production, resulting in a decrease in conversion ratio from 0.98 to 0.73 for a 5 ft. core and 500 gm/liter thorium at $\Sigma_{at} = 0.010$. On the other hand, a 3 ft. thick blanket yields only a small increase in conversion ratio, from 0.96 to 1.00 at $\Sigma_{at} = 0.013$."⁽¹⁾

Breeding Gain vs Physical Dimensions and Operating Conditions

The breeding gain is sensitive to many variables and its prediction requires two group calculations. Factors affecting the breeding gain are reactor dimensions, uranium concentrations, temperature, power level, thorium concentration, reactor fission product poisons, neutron leakage and absorption, and economics.

One of the reasons for selecting the dimensions of this reactor as having a 6 ft. diameter core and a 2 ft. thick blanket is that for this combination a variation in the blanket power has essentially no effect on the breeding gain. This is illustrated in Fig. 37, page 108. A change in blanket U-233 concentration also has little effect on breeding gain since the blanket concentration is directly proportional to power for the reactor selected.

FIG. 35 VARIATION OF CORE U-235 CONCENTRATION WITH
BLANKET THICKNESS AND THORIUM CONCENTRATION
(FROM CF-51-10-110, PAGE 26)

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FIG. 36 VARIATION OF TH-232 CONVERSION RATIO WITH BLANKET THICKNESS AND THORIUM CONCENTRATION

(FROM CF 1975-10-170, PAGE 27)

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For a 10 ft. diameter core it is found that the breeding gain is directly dependent on the ratio of the blanket power to core power (see page 43 of reference 1). As the blanket power is increased the fission product poison and the buildup of U-234, U-235, and U-236 also increases. At first the breeding gain increases due to the greater neutron flux at the tank. This more than compensates for the increased tank absorption and neutron leakage out of the reactor. However, a maximum breeding gain is reached at a 50% blanket power to core power ratio. At higher blanket power levels the fission products become dominant and the breeding gain decreases approximately proportional with the percent fission product poison in the core.

The breeding gain is dependent on the poisons in the core which, in turn, are dependent on the processing rate. The processing rate is determined by balancing the cost of reducing the poison level against the value of U-233 lost when neutrons are absorbed by poisons. The reactor studied here operates at 7% poisons in the core. According to Oak Ridge (page 68 of reference 3), a good approximation of the effect of poisons in the core upon breeding gain is obtained by subtracting the core poison fraction (0.07) from the breeding gain.

The breeding gain is decreased by 0.02 when the reactor operating temperature is increased from 250 C to 300 C (page 71 of reference 3).

Factors Which Vary with U-233 Blanket Concentration

Figs. 37, 38, and 39, pages 108, 109, and 110, exhibit the factors which vary with U-233 blanket concentration. Fig. 37 indicates that the core U-233 concentration decreases slightly with U-233 blanket concentration. Core tank absorptions are essentially constant. Fig. 38 indicates that all the higher order isotopes increase with the U-233 concentration in the blanket. A similar variation is displayed in Fig. 39 for fission product absorptions and neutron leakages.

Flexibility of Operation for Startup and Steady State Conditions

U-233 Breeder Reactor. The reactor can be started by using U-233 only in the core and Th-232 in the blanket. During operation and at a high flux the characteristics of the reactor will change until equilibrium conditions are reached after long time operation. This is due mainly to the rapid buildup of fission products and uranium isotopes. No power will be drawn from the blanket until sufficient U-233 has been produced in the blanket from the conversion of Th-232 by neutrons. The breeding gain will initially be 0.18 and will decrease in proportion to the fission product buildup in the core. At steady state with 7% core poison the breeding gain will remain constant at 0.11.

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Fig.37 REACTOR POWER, BREEDING GAIN, CORE U-233 CONCENTRATION, AND CORE TANK ABSORPTION RATIO VERSUS BLANKET U-233 CONCENTRATION. DATA PLOTTED FROM ORNL-CF-53-12-1, P.21

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Fig. 38 BLANKET ISOTOPE NEUTRON ABSORPTION RATIOS VERSUS BLANKET U-233 CONCENTRATION. DATA PLOTTED FROM ORNL-CP-53-12-1, P. 21

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Fig. 39 BLANKET VARIATIONS OF POWER, LEAKAGE, AND FISSION PRODUCTS VERSUS U-233 CONCENTRATION IN BLANKET. DATA PLOTTED FROM ORNL-CF-53-12-1, P. 21

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If, before starting, 8 gm/liter of U-233 is placed in the blanket, then the reactor can immediately be operated at full power and near steady state operating conditions. Since no fission products exist at startup, the uranium 233 concentration for the core can be reduced 7% and increased as core poisons build up.

During operation at 100% power (640 megawatts) and steady state conditions, the reactor system reproduces all the U-233 it needs by means of thorium conversion. The entire reactor system contains 386.1 kg of U-233. This is burned in 495 days at the rate of 779 gm/day. There is 49 gm/day excess production due to the breeding gain being 0.11. At this rate, the time to produce an excess full reactor system is 21.6 years.

It should be pointed out that fuel grade U-233, obtained from some source other than the blanket, can be substituted for pure U-233. The analysis of this material will dictate the concentration required for reactor operation.

The question of whether to start up the reactor without uranium in the blanket and slowly approach steady state or to start immediately at near steady state with blanket uranium is a matter of economic consideration.

U-235 Converter Reactor. If U-233 is unavailable, it can be produced by the temporary substitution of U-235 for U-233 in the reactor. The fissioning of U-235 will make available the required neutrons for the transmutation of Th-232 to U-233. Similar to the use of U-233, the uranium 235 can be placed either in the core alone or in both the core and blanket. Power extraction, conversion ratio, and economics will vary according to the method chosen and the U-235 concentration used. If uranium 235 is placed only in the core, about 2.45 gm/liter (94% U-235, 6% U-238) will be needed to start the reactor. Under these conditions the conversion ratio of atoms of Th-232 transformed to U-233 per atom of U-235 burned, will be about 0.95.

If U-235 is used both in the core and in the blanket, the required U-235 concentrations will be 2.0 gm/liter in the core and 9.0 gm/liter for the blanket. The initial blanket conversion ratio of the U-235 reactor will be 0.95. As U-233 is produced and blended into the core, the blanket conversion ratio will increase to the ultimate value of 1.11 at which time all the U-235 will have been replaced by U-233. The time required for 95% replacement of U-235 by production of U-233 is approximately 6.8 years. Actually, the conversion ratio will increase to 1.00 or greater after 130 days of U-235 feeding. The total U-235 feed required for the 130 days is about 800 grams. From this time on, further U-235 feeding is unnecessary, and the remaining U-235 in the reactor will be replaced by U-233.

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Plutonium Reactor. The flexibility of the reactor permits the possibility of producing plutonium in addition to power. In general, all that would be necessary is the use of U-235 in the core or both the core and blanket and substitution of U-238 for Th-232 in the blanket. Although the present reactor geometry might lead to production of less than one atom of plutonium for each atom of U-235 burned, nevertheless, the economic value of plutonium may more than compensate for this possible deficiency.

HEAVY ISOTOPE BUILDUP AND FISSION PRODUCT POISONING

Power is produced in a reactor by fission. The fission products produced are poisonous to the reactor and must be removed. Operation of a reactor also results in production of higher isotopes of uranium. The aqueous homogeneous reactor requires the selection of a chemical processing rate to minimize or balance the effects of fission product poisoning and isotope buildup at steady state operating conditions. The factors involved in the buildup of heavy isotopes and fission product poisons are presented in the following sections. Their consideration results in mathematical equations for buildup and poisoning in the core and blanket at steady state conditions.

Buildup of Isotopes in the Blanket

Thorium 232 in the blanket is converted according to the chain of events shown in Fig. 40, page 113.

The decay and buildup of isotopes can be calculated from the usual radioactivity decay laws. Starting with pure thorium 232 (ThO₂-D₂O slurry), and maintaining constant concentration in the blanket, and irradiating it with neutrons, the following is obtained:

$$\frac{dN(02)}{dt} = - N(02) \sigma_c(02)_{\text{thermal}} \phi_t - N(02)_{\text{resonance}} \phi_r \quad (1)$$

for thorium 232, let $\rho = \frac{\text{thermal absorptions} + \text{resonance absorptions}}{\text{thermal absorptions}}$

Therefore

$$\frac{dN(02)}{dt} = - \rho N(02) \sigma_c(02)_{\text{thermal}} \phi_t \quad (2)$$

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FIG. 40. -- Isotope Buildup

- Let N = Number of atoms
 R = Fraction of the blanket system volume removed for processing per second
 λ = Decay constant - different for each isotope
 ϕ = Average thermal flux for the blanket and blanket system combined
 ρ = Ratio of total (thermal+resonance) to thermal absorptions in thorium 232
 σ_c = Microscopic capture cross section of isotope
 σ_f = Microscopic fission cross section of isotope
 σ_a = Microscopic absorption cross section of isotope
02 = Thorium 232 isotope
13 = Protactinium 233 isotope
23 = Uranium 233 isotope
24 = Uranium 234 isotope
25 = Uranium 235 isotope
26 = Uranium 236 isotope

Capture of neutrons by thorium 232 transmutes it to thorium 233. This decays almost immediately (23 minutes) to protactinium 233 by means of β emission. A negligible amount of thorium 234 is formed also by means of neutron capture. This is ignored. The results are represented by the following equation:

$$\frac{dN(13)}{dt} = \rho N(02) \sigma_c(02) - N(13) \sigma_c(13) \phi - \lambda(13) N(13) - RN(13) \quad (3a)$$

$$\sigma_a = \sigma_f + \sigma_c ; \quad \sigma_f(13) = 0$$

$$\frac{dN(13)}{dt} = \rho N(02) \sigma_a(02) - N(13) \sigma_a(13) \phi - \lambda(13) N(13) - RN(13) \quad (3b)$$

where R represents a chemical processing rate.

Pa-233 decays (half life 27.4 days) by β emission to U-233. At the same time Pa-234 is formed by neutron capture. However, comparison of the rates of transmutation of Pa-233 to both Pa-234 and U-233 indicates that only a very small amount of Pa-234 is formed. For simplification of the buildup equations, Pa-234 is excluded. The resonance absorption for elements and isotopes greater than thorium 232 is negligible and the need for a further use of ρ is thereby eliminated.

With respect to the above criteria the following equation represents the situation:

$$\frac{dN(23)}{dt} = \lambda(13) N(13) - N(23) \sigma_f(23) \phi - N_{23} \sigma_c(23) \phi - RN(23) \quad (4a)$$

$$= \lambda(13) N(13) - N(23) \sigma_a(23) \phi - RN(23) \quad (4b)$$

U-234 is formed by neutron capture of U-233. There is also a small contribution from Pa-234 which, due to the fast decay (1.1 minutes and 6.7 hours), can be considered as coming directly from Pa-233. The following equation results:

$$\frac{dN(24)}{dt} = N(13) \sigma_c(13) \phi + N(23) \sigma_c(23) \phi - N(24) \sigma_c(24) \phi - RN(24) \quad (5)$$

Continuing the neutron irradiation as per Fig. 40, page 113, the following isotope buildup equations are obtained:

$$\frac{dN(25)}{dt} = N(24) \sigma_c(24) \phi - N(25) \sigma_f(25) \phi - N(25) \sigma_c(25) \phi - RN(25) \quad (6a)$$

$$= N(24) \sigma_c(24) \phi - N(25) \sigma_a(25) \phi - RN(25) \quad (6b)$$

$$\frac{dN(26)}{dt} = N(25) \sigma_c(25) \phi - N(26) \sigma_c(26) \phi - RN(26) \quad (7)$$

The extra neutrons produced by fission of U-235 are absorbed by U-234 and U-236, so that the net neutron gain from this source is small.

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The half lives of uranium isotopes decaying to neptunium are very large, thus the decay rate is very small. Corrections therefore have not been included in the above equations. A very small number of the atoms will finally become plutonium.

For the steady state condition, equations (1) through (7) are equated to zero: $\frac{dN}{dt} = 0$. Solving, rearranging, and substituting in the equations then leads to the following isotope ratios.

Steady state conditions in the blanket:

$$\frac{N(23)}{N(02)} = \frac{\rho \sigma_a(02) \phi \lambda(13)}{[\lambda + \sigma_a(13) \phi + R] [\sigma_a(23) \phi + R]} \quad (8)$$

$$\frac{N(13)}{N(02)} = \frac{\rho \sigma_a(02) \phi}{\lambda(13) + \sigma_a(13) \phi + R} \quad (9)$$

$$\frac{N(24)}{N(02)} = \frac{N(13)}{N(02)} \frac{\sigma_c(13) \phi}{[\sigma_a(24) \phi + R]} + \frac{N(23)}{N(02)} \frac{\sigma_c(23) \phi}{[\sigma_a(24) \phi + R]} \quad (10)$$

$$\frac{N(25)}{N(02)} = \frac{N(24)}{N(02)} \frac{\sigma_c(24) \phi}{[\sigma_a(25) \phi + R]} \quad (11)$$

$$\frac{N(26)}{N(02)} = \frac{N(25)}{N(02)} \frac{\sigma_c(25) \phi}{[\sigma_a(26) \phi + R]} \quad (12)$$

Equations (8) through (12) were used in the reactor blanket calculations for the determination of the buildup of concentrations of heavy isotopes.

Buildup of Isotopes in the Core

From Fig. 40, page 113, it can be seen that as uranium 233 is burned in the core fission products are produced and, by the absorption of neutrons, the formation of uranium 234, 235, and 236 results. Plutonium, neptunium, and uranium 237, 238, and 239 are produced in negligible quantities.

If pure U-233 is fed into the core the following equations will account for the isotope buildup. Processing rate and the α and β emissions are neglected.

$$\frac{dN(23)}{dt} = - N(23) \sigma_f(23)\phi - N(23) \sigma_c(23)\phi \quad (13)$$

$$\frac{dN(24)}{dt} = N(23) \sigma_c(23)\phi - N(24) \sigma_c(24)\phi ; \sigma_f(24) = 0 \quad (14a)$$

$$= N(23) \sigma_c(23)\phi - N(24) \sigma_a(24)\phi \quad (14b)$$

$$\frac{dN(25)}{dt} = N(24) \sigma_c(24)\phi - N(25) \sigma_f(25)\phi - N(25) \sigma_c(25)\phi \quad (15a)$$

$$= N(24) \sigma_c(24)\phi - N(25) \sigma_a(25)\phi \quad (15b)$$

$$\frac{dN(26)}{dt} = N(25) \sigma_c(25)\phi - N(26) \sigma_c(26)\phi ; \sigma_f(26) = 0 \quad (16a)$$

$$= N(25) \sigma_c(25)\phi - N(26) \sigma_a(26)\phi \quad (16b)$$

ϕ is the thermal neutron flux averaged over the core and external core system.

Setting $\frac{dN}{dt} = 0$ for the steady state, gives

$$\frac{N(24)}{N(23)} = \frac{\sigma_c(23)}{\sigma_a(24)} \quad (17)$$

$$\frac{N(25)}{N(23)} = \frac{\sigma_c(23)}{\sigma_a(24)} \frac{\sigma_a(24)}{\sigma_a(25)} = \frac{\sigma_c(23)}{\sigma_a(25)} \quad (18)$$

$$\frac{N(26)}{N(23)} = \frac{\sigma_c(23)}{\sigma_a(25)} \frac{\sigma_c(25)}{\sigma_a(26)} \quad (19)$$

If the core is fed with fuel processed from the blanket, the fuel will contain U-233 and also small quantities of U-234, U-235, and U-236.

Let $\frac{N_B(23)}{t}$ be the rate at which U-233 from the blanket is fed into the core. If the U-233 concentration in the core is to remain constant, then the rate at which decontaminated blanket-bred uranium is returned to the core must equal the rate of uranium burnup in the core. The differential equations representing these conditions are presented in the following expressions:

$$\frac{dN_c(23)}{dt} = \frac{N_B(23)}{t} - N_c(23) \sigma_a(23) \phi \quad (20a)$$

where $N_c(23)$ = number of core U-233 atoms.

If all the Pa-233 in the blanket is allowed to decay before separation, then $N_{B_{Total}}(23) = N_B(23) + N_B(13)$.

Let $\frac{N_{B_{Total}}(23)}{t} \equiv M$. Then

$$\frac{dN_c(23)}{dt} = M - N_c(23) \sigma_a(23) \phi \quad (20b)$$

$$\frac{dN_c(24)}{dt} = \frac{N_B(24)}{t} + N_c(23) \sigma_c(23) \phi - N_c(24) \sigma_a(24) \phi \quad (21a)$$

where $\sigma_a(24) = \sigma_c(24)$

$$\begin{aligned} \text{Now, } \left(\frac{N_B(24)}{t} \right) \cdot \left(\frac{N_{B_{Total}}(23)}{N_{B_{Total}}(23)} \right) &= \left(\frac{N_{B_{Total}}(23)}{t} \right) \left(\frac{N_B(24)}{N_{B_{Total}}(23)} \right) \\ &= M \frac{N_B(24)}{N_B(23) + N_B(13)} \\ &\equiv M\alpha_{43} \end{aligned} \quad (21b)$$

Therefore

$$\frac{dN_c(24)}{dt} = M\alpha_{43} + N_c(23) \sigma_c(23) \phi - N_c(24) \sigma_a(24) \phi \quad (21c)$$

$$\frac{dN_c(25)}{dt} = \frac{N_B(25)}{t} + N_c(24) \sigma_a(24) \phi - N_c(25) \sigma_a(25) \phi \quad (22a)$$

$$= M\alpha_{53} + N_c(24) \sigma_a(24) \phi - N_c(25) \sigma_a(25) \phi \quad (22b)$$

$$\text{where } \alpha_{53} \equiv \frac{N_B(25)}{N_B(23) + N_B(13)} \quad (22c)$$

$$\frac{dN_c(26)}{dt} = \frac{N_B(26)}{t} + N_c(25) \sigma_c(25) \phi - N_c(26) \sigma_a(26) \phi \quad (23a)$$

$$= M\alpha_{63} + N_c(25) \sigma_c(25) \phi - N_c(26) \sigma_a(26) \phi \quad (23b)$$

$$\text{where } \alpha_{63} \equiv \frac{N_B(26)}{N_B(23) + N_B(13)} \quad (23c)$$

At the steady state $\frac{dN}{dt} = 0$. Solving equations (20b), (21b), (22b), and (23b) gives

$$\frac{N_c(24)}{N_c(23)} = \frac{\sigma_c(23)}{\sigma_a(24)} + \frac{\sigma_a(23)}{\sigma_a(24)} \alpha_{43} \quad (24)$$

$$\frac{N_c(25)}{N_c(23)} = \frac{N_c(24)}{N_c(23)} \frac{\sigma_c(24)}{\sigma_a(25)} - \frac{\sigma_a(23)}{\sigma_a(25)} \alpha_{53} \quad (25)$$

$$\frac{N_c(26)}{N_c(23)} = \frac{N_c(25)}{N_c(23)} \frac{\sigma_c(25)}{\sigma_a(26)} + \frac{\sigma_a(23)}{\sigma_a(26)} \alpha_{63} \quad (26)$$

By setting $\alpha_{43} = \alpha_{53} = \alpha_{63} = 0$, equations (24), (25), (26) reduce exactly to equations (17), (18), (19). The error introduced by the approximation is very small and results in the isotopic ratios in the core being independent of blanket composition. Equations in the form of (17), (18), and (19) were used in our core isotope ratio calculations.

Fission Product Poisons

Fission product poisons in this reactor can be classified into three major groups:

1. Gases. This group gives a relatively small contribution of 0.5% to the poison fraction (page 63 of reference 3).
2. High Cross Section Isotopes. This group transmutes to other nuclides of lower cross section by the absorption of neutrons. At 250 C the microscopic absorption cross section, σ_a , is about 40,000 barns. The yield is 0.008 atoms/fission (page 50 of reference 2).
3. Low Cross Section Isotopes. This group transmutes into other isotopes of approximately the same cross section. At 250 C the microscopic absorption cross section, σ_a , is 40 barns. The yield is 1.6 atoms/fission (page 50 of reference 2).

Core. The fission product poisons in the core are removed at a processing rate which takes into account both the losses in breeding gain and the cost of chemical processing.

Two processes are used for the removal of fission products from the core. Eighty percent (80%) of these products are insoluble and are removed by the hydroclones. Twenty percent (20%) are soluble and must be removed by the Thorex process.

The total poison fraction for all fission product groups combined was determined by the following expression exhibited on page 63 of reference 3.

$$p = 0.005 + \frac{0.0073}{f} + 0.122 \frac{AP_c}{M_1 f} \left(\frac{0.8T_1T_2}{T_1 + T_2} + 0.2T \right) \quad (27)$$

Economic and nuclear consideration resulted in a value of 7% poison level in the core.

Blanket. The fission product poison level depends upon the chemical processing rate established to maintain a fixed U-233 concentration in the blanket.

Only one processing method (Thorex process) is used for the removal of fission products from the blanket in contrast to that used for the core.

The total poison fraction for all fission products was determined by the following expression, as may be evolved from page 50 of reference 2.

$$p = 0.005 \frac{\sum_f \sigma_a(H) \phi y(H)}{\sum_a(23) [\sigma_a(H) \phi + R]} + \frac{\sum_f \sigma_a(L) \phi y(L)}{\sum_a(23) R} \quad (28)$$

where $\sum_f = N(23) \sigma_f(23) + N(25) \sigma_f(25)$

H = the high cross section fission poison group

L = the low cross section fission poison group

TWO-GROUP TWO-REGION ANALYSIS

Two-group two-region calculations are used to determine the critical concentration, the neutron balance, and the breeding gain in a two-region reactor. To determine the critical concentration one must know such factors as the reactor dimensions, the core and blanket compositions, the isotopic cross sections, the core tank composition and thickness, and the temperature.

Theory

The following two-group two-region boundary conditions and equations are based on the methods and nomenclature of references 1 and 4. The textbook "The Elements of Nuclear Reactor Theory" by Glasstone and Edlund was also consulted.

The boundary conditions are:

1. Vanishing of fast and slow fluxes at the outside edge of the reactor.
2. Finite, non-negative fluxes exist everywhere in the reactor.
3. Continuity of fast and slow fluxes at the interface between core and blanket.
4. Continuity of the blanket and core fast currents at the interface.
5. Balance of slow currents at the blanket-core interface to include tank absorption.

The conventional two-group diffusion equations for the fast and slow fluxes in the core and blanket are as follows:

Core

$$D_{fc} \nabla^2 \phi_{fc} - \Sigma_{fc} \phi_{fc} + k_c \Sigma_{sc} \phi_{sc} = 0 \text{ for fast flux} \quad (29)$$

$$D_{sc} \nabla^2 \phi_{sc} - \Sigma_{sc} \phi_{sc} + \Sigma_{fc} \phi_{fc} = 0 \text{ for slow (thermal) flux} \quad (30)$$

Blanket

$$D_{fb} \nabla^2 \phi_{fb} - \Sigma_{fb} \phi_{fb} - \Sigma_{rb} \phi_{fb} + \frac{k}{p} \Sigma_{sb} \phi_{sb} = 0 \text{ for fast flux} \quad (31)$$

$$D_{sb} \nabla^2 \phi_{sb} - \Sigma_{sb} \phi_{sb} + \Sigma_{fb} \phi_{fb} = 0 \text{ for slow flux} \quad (32)$$

Boundary conditions 1 and 2 are applied to the general solutions of the above equations.

$$\begin{array}{l} \text{Boundary} \\ \text{Conditions} \end{array} \left\{ \begin{array}{l} \textcircled{1} \quad \phi_{fb}(a_2) = \phi_{sb}(a_2) = 0 \\ \text{where } (a_2) \text{ is the outer radius of the blanket.} \\ \textcircled{2} \quad 0 < \phi < \infty \end{array} \right. \quad \begin{array}{l} (33) \\ \\ (34) \end{array}$$

This gives the following specific solutions of the differential equations:

$$\phi_{sc} = \frac{A}{r} \sin B_{ic}r + \frac{B}{r} \sinh B_{rc}r \quad (35)$$

$$\phi_{fc} = \frac{A''}{r} \sin B_{ic}r + \frac{B''}{r} \sinh B_{rc}r \quad (36)$$

$$\phi_{sb} = \frac{E}{r} \sinh B_{ib}(a_2 - r) + \frac{G}{r} \sinh B_{rb}(a_2 - r) \quad (37)$$

$$\phi_{fb} = \frac{\alpha_{ib}}{r} E \sinh B_{ib}(a_2 - r) + \frac{\alpha_{rb}}{r} G \sinh B_{rb}(a_2 - r) \quad (38)$$

Boundary conditions 3, 4, and 5 are then applied to the specific solutions:

$$\begin{array}{l} \text{Boundary} \\ \text{Conditions} \end{array} \left\{ \begin{array}{l} \textcircled{3} \quad \phi_{fc}(a_1) = \phi_{fb}(a_1) \\ \quad \phi_{sc}(a_1) = \phi_{sb}(a_1) \\ \quad \text{where } (a_1) \text{ is the core diameter} \\ \textcircled{4} \quad -D_{fc} \nabla \phi_{fc}(a_1) = -D_{fb} \nabla \phi_{fb}(a_1) \\ \textcircled{5} \quad -D_{sc} \nabla \phi_{sc}(a_1) + D_{sb} \nabla \phi_{sb}(a_1) = \sum_a (\text{tank}) t \phi_{sc}(a_1) \end{array} \right. \quad \begin{array}{l} (39) \\ (40) \\ (41) \end{array}$$

This gives a system of four homogeneous equations. To obtain a set of solutions the determinant of the equations must be zero. This is the critical equation for the reactor and the means by which critical fuel concentrations are calculated.

Core Concentration

In practice, the requirements of engineering, chemistry, physics, and economics are taken into account in determining the reactor specifications and operation. If reactor dimensions, temperature, and blanket composition are chosen, then the core concentration remains to be determined. Using the two-group calculations and a trial and error method, a core concentration is assumed and the thickness of core tank which will result in a critical reactor is then obtained. The core concentration calculations can be conveniently subjected to two arithmetic error checks since there must be an equivalence of the fast and slow fluxes at the interface between core and blanket.

Neutron Balance

Further development of the two-group, two-region equations leads to the calculation of neutron balance. This means that one can determine what fraction of neutrons are absorbed or leaked in any fashion per neutron absorbed in an atom of U-233. A check for arithmetic error is at hand in that the total number of neutrons absorbed or leaked in any way per neutron absorbed in U-233 must equal the number of neutrons produced by fission per absorption of a neutron in U-233.

Breeding Gain

The breeding gain is obtained from the neutron balance, where the breeding gain is defined as the net difference between or the excess of U-233 atom production compared with U-233 atom losses. In this reactor U-233 production is brought about by absorptions of neutrons in thorium whereas U-233 losses are due to absorption of neutrons by protactinium, and by U-233. This can be expressed mathematically in the following manner:

$$\begin{aligned} \text{Breeding gain} = & \frac{\text{Abs. in Th-232}}{\text{Abs. in core U-233}} - \frac{\text{Abs. in Pa-233}}{\text{Abs. in core U-233}} \\ & - \frac{\text{Abs. in (core + blanket) U-233}}{\text{Abs. in core U-233}} \end{aligned} \quad (42)$$

Calculated and Extrapolated Results

All reactor calculations are modeled directly after the methods developed at Oak Ridge and presented in references 1, 2, 3, 4, and 5. The calculations made by Oak Ridge for a reactor closest to the selected reactor design in physical dimensions and operating conditions of this study were checked in detail using their constants. A resultant breeding gain of 0.22 was obtained as against a value of 0.20 presented by Oak Ridge. The concentration and breeding gain for the power reactor was then determined by extrapolating the Oak Ridge data. The method and results of the extrapolations are discussed below.

Fuel Concentration. Figs. 37, 38, and 39, pages 108, 109, and 110, are graphs plotted from the Oak Ridge data presented on page 21 of reference 2. The dashed portion of the curves is an extrapolation of the data. A blanket concentration of 8.00 gm U-233/kg Th-232 and a core U-233 concentration of 1.10 gm/liter for zero poison level in the core is indicated by Fig. 37 for 280 mw of blanket power at 250 C. Reactor operation at 300 C instead of at 250 C requires a 50% increase in core U-233 concentration (page 71 of reference 3), presenting a requirement of 1.65 gm/liter. The core concentration, in addition, is approximately proportional to the core poison level, thus giving a resultant requirement of 1.80 gm U-233/liter for the core (page 68 - 72 of reference 3).

Breeding Gain. The breeding gain was obtained from the neutron balance. The neutron balance was obtained from the extrapolated portion of Figs. 37, 38, and 39 by using as abscissa the 8 gm/kg Th-232 in the blanket for 280 mw power at 250 C. The numerical values obtained are presented in Table XVI and result in a breeding gain of 0.204 with no poisons in the core.

TABLE XVI. -- Neutron Balance for Two-Region Reactor at 250 C *
(No Poisons in Core)

<u>Neutron absorptions</u>		
<u>Absorption in U-233 in core</u>		
Core	U-233	1.000
	U-234	.090
	U-235	.090
	U-236	.014
	Sulphur	.002
	Fission Products	.000
	D ₂ O	.034
Core Tank	(Based on Zirconium)	.060
Blanket	Th-232 (resonance)	.782
	Th-232 (slow)	1.371
	Pa-233	.024
	U-233	.925
	U-234	.011
	U-235	.005
	Fission Products **	.161
	D ₂ O	.003
Leakage	Fast	.095
	Slow	.029

$$\text{Breeding Gain} = (0.782 + 1.371) - (1.000 + 0.925 + 0.024) = 0.204$$

* Data extrapolated from Figs. 37, 38, and 39.

** Steady state based upon processing rate required to maintain U-233 concentration in blanket.

Corrections for the 7% core poison level in the reactor of this study result in the breeding gain decreasing to 0.134. Operation of the reactor at 300 C increases the neutron losses so that breeding gain is reduced by 0.02 as against operations at 250 C (page 71 of reference 3). This results in a net breeding gain of 0.114.

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It would be expected that a detailed two-group calculation would yield the results indicated above for the fuel concentration and breeding gain at steady state operating conditions for the reactor under study.

The results obtained by extrapolation of the influences of temperature and poison level on the concentration and breeding gain are summarized in Table XVII.

TABLE XVII. -- Effects of Temperature and Poison Level on Concentration and Breeding Gain

Reactor Operating Temperature	Core U-233 Concentration in Grams/Liter for Poison Levels of		Breeding Gain for Poison Level of	
	0%	7%	0%	7%
250 C	1.10	1.20	0.204	0.134
300 C	1.65	1.80	0.134	0.114

NUCLEAR CONSTANTS

TABLE XVIII. -- Thermal Neutron Cross Sections at 250 C (barns)

Isotope	Absorption, σ_a	Capture, σ_c	Fission, σ_f	Reference
Th-232	5.2	5.2	0	9
Pa-233	100	100	0	2
U-233	443	40	403	2, 9, 11
U-234	60	60	0	9
U-235	478	75	403	9, 11
U-236	6	6	0	9

The above nuclear constants are not Maxwelllized. They were Maxwelllized by means of division by $2/\sqrt{\pi} = 1.128$ before being utilized in the reactor calculations.

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Cross sections were obtained by using the reference values at 2200 meters/sec and assuming a $1/v$ cross section to correct for temperature.

The U-235 cross section was corrected for its non $1/v$ variations.

TABLE XIX. -- Constants for D_2O

	<u>20 C</u>	<u>250 C</u>	<u>Reference</u>
Density, gm/cm ³	1.105	0.876	6
Diffusion length squared, cm ²	1.35×10^4	3.05×10^4	9
Absorption cross section, cm ⁻¹	7.07×10^{-5}	4.20×10^{-5}	2
Diffusion coefficient, cm			
Core (slow)	0.84	1.14	2
(fast)	1.30	1.66	1
Blanket (slow)	0.8846	1.152	1
(fast)	1.180	1.429	1
Fermi Age, cm ²			
Core	120	190	2
Blanket	121.1	186.8	1

The above nuclear constants have not been corrected for void fractions. However, before working calculations were made, the nuclear properties in the core that are sensitive to density variations were corrected for a 3% void fraction.

TABLE XX. -- Miscellaneous Constants

			<u>Reference</u>
U-233	Neutrons/fission, ν	2.53	11
	Neutrons/absorption, η	2.30	11
U-235	Neutrons/fission, ν	2.48	11
	Neutrons/absorption, η	2.09	11
Pa-233	Half life, days $T_{\frac{1}{2}}$	27.4	9
D ₂ O	Macroscopic resonance absorption (blanket), cm^{-1} , Σ_{rb}	.004619	1
Zirconium	Macroscopic absorption cross section, (core tank), cm^{-1} , $\Sigma_a \text{Zr}$	0.009493	1
Sulphur	Microscopic absorption cross section at 250 C, cm^{-1} , $\sigma_a(S)$	0.37	1
Oxygen	Microscopic absorption cross section at 250 C, cm^{-1} , $\sigma_a(O)$	$< 10^{-4}$	1

For more information and additional discussion of nuclear constants see references 1 and 2.

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Babcock and Wilcox Company
Bailey Meter Company
Chicago Bridge and Iron Company
Crane Company
General Electric Company
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