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REACTORS-POWER ✓

UNITED STATES ATOMIC ENERGY COMMISSION

A SODIUM-GRAPHITE REACTOR STEAM-ELECTRIC STATION FOR 75 MEGAWATTS NET GENERATION

By
Edward F. Weisner
Warren M. Sybert ✓

March 22, 1955 ✓

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Downey, California ✓

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REACTORS-RESEARCH AND POWER

AEC RESEARCH AND DEVELOPMENT REPORT

A SODIUM-GRAPHITE REACTOR STEAM-ELECTRIC STATION FOR 75 MEGAWATTS NET GENERATION

CLASSIFICATION CANCELLED

DATE FEB 28 1957

For The Atomic Energy Commission

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ABSTRACT

The major design features, nuclear characteristics and performance data for a nuclear fueled central station power plant of 75, 000 kw net capacity are presented. The heat source is a sodium cooled graphite moderated reactor. The design of the reactor takes full advantage of the experience gained to date on the Sodium Reactor Experiment (SRE); the plant described here is a straightforward extension of the smaller experimental SRE, which is now under construction. The fuel elements are made up of rod clusters and the moderator is in the form of zirconium canned graphite elements. The performance of the reactor has been based on conservative temperatures and coolant flow velocities which result in a plant with "built-in reserve." Thus, as experience is gained and anticipated improvements in reactor fuel elements and construction materials are proven, the performance of the plant can be increased accordingly.

Two reactor designs are described, one for operation with slightly enriched uranium fuel elements and the other for operation with thorium-uranium fuel elements. The associated heat exchangers, pumps, steam, and electrical generating equipment are identical for either reactor design.

An analysis of turbine cycles describes the particular cycle chosen for initial operation and discusses a method by which modern central station performance can be initially obtained.

The purpose of this report is to establish the design and performance data which are required to enable reliable estimates of the plant construction and operating costs to be made.

I. GENERAL DESCRIPTION

This report presents the results of a preliminary design study of a nuclear fueled central station power plant capable of producing 75,000 kw of net electric power. The purpose of the study was to determine performance data and sufficient design information so that realistic construction and operating cost estimates could be made. The basic philosophy which has been followed in this study has been to adhere to the sodium-graphite design technology presently embodied in the Sodium Reactor Experiment, currently under construction. The resulting design is a straightforward extension of the SRE program.

The over-all layout of the plant is shown in Fig. 1, and the reactor building and equipment arrangement is shown in Fig. 2. The complete plant consists of (1) a reactor core cooled by a primary sodium loop, (2) intermediate heat exchangers, (3) secondary coolant loop, (4) steam generators, (5) turbogenerator, condenser, transformers, switchgear, and (6) auxiliary equipment.

The reactor core is contained in a large diameter steel tank located below ground and cooled by liquid sodium. The sodium coolant is separated into four independent loops and flows from the reactor by means of radial discharge pipes to four intermediate sodium-to-sodium heat exchangers. The loops are contained in individually shielded vaults located in four quadrants around the reactor tank and below ground level. The four individually shielded quadrants permit maintenance on any one of the four loops without complete shutdown of the plant.

The secondary non-radioactive sodium from each intermediate heat exchanger is pumped above ground to the steam generators. The steam generators are separated into two buildings located on either side of the main reactor building and are each fed by the secondary sodium from the two intermediate heat exchangers nearest that side of the building. The location of the steam generators is sufficiently removed from the main building and the electrical generating equipment to eliminate damage to these units in case of a tube failure which might result in a sodium-water reaction. The steam from each exchanger unit is piped to the turbogenerator located to the rear of the main building. The location of the electrical transformers, switchgear enclosure, and cooling tower area are only shown schematically; the exact location would, of course, be dictated by specific plant location and needs.

The reactor building is rectilinear in shape, approximately 100 feet in width, 230 feet in length, and has a high bay section the entire length of the building with low bay areas to either side. A large fuel handling crane spans the high bay area and is capable of traveling the entire length of the building. The controls and instrumentation are housed in three rooms one above the other in the high bay area and at the rear of the building.

New and spent fuel storage is provided by means of individual element containers in the form of vertical pipes located below ground level in the high bay area and to the front of the building. Approximately two hundred new and two hundred spent fuel element storage containers are provided. These fuel elements are transferred to and from the plant by truck. A truck door is provided so that this fuel transfer can be made directly with the fuel handling crane in the high bay area.

The reactor building also provides space for fuel dismantling operations, a machine shop, sodium service area, inert gas service area, and office area. It is to be noted that all radioactive materials are confined to the reactor building, and that all materials in this building are compatible and will not react chemically with each other.

Sections II and III of this report give the description of two reactor designs for this plant. The first section is devoted to a reactor design for use with uranium fuel and the second section to a reactor design for use with thorium-base fuel. The two reactors are identical in all design aspects. They differ only in type of fuel element employed and certain physical dimensions. The best choice of reactor design for a particular application depends on such factors as the relative costs of uranium and thorium fuels, fabrication and processing costs in each case, the financial structure of the utility, etc. This subject is beyond the scope of this report; however, sufficient information is given for each design so that a choice can be made by further evaluation of these factors.

In each reactor design the fuel is in the form of rod clusters which are cooled by liquid sodium. The moderator is in the form of graphite contained in thin-walled hexagonal shaped zirconium cans. The core moderator cans have a thin-walled zirconium process tube along the longitudinal axis in which the fuel elements are suspended. The primary coolant flows up through these process channels and cools the surfaces of the fuel clusters. Reflector elements are

identical in dimensions with the moderator elements but do not contain process tubes. The core matrix is made up from a close packed array of these elements and rests on a stainless steel grid placed in the bottom of the stainless steel reactor tank.

The top of the reactor tank is sealed off by means of a 5-foot thick circular shield. The top shield contains a hole and plug for each process channel and it is through these holes that the fuel assemblies are loaded and unloaded. A mild steel thermal shield, thermal insulation, and concrete shield are located outside of the reactor tank and together with the adjacent earth, complete the biological shield.

The maximum sodium and zirconium temperature in each reactor design has been limited to a conservative 950° F, which results in a mean sodium outlet temperature of approximately 925° F. The maximum fuel temperature used for the design calculations was 1200° F for the uranium fueled design and 1600° F for the thorium design. The nuclear characteristics and performance for each reactor design based on these temperatures are given. Various possibilities which offer means for improved performance are discussed in the respective sections.

Section IV of this report describes the basis of selection of the initial turbine cycle as well as methods of attaining improved performance either initially or in the future. A three-heater non-reheat regenerative cycle with throttle conditions of 800 psig/825° F is the choice for initial operation, based upon a primary sodium mean temperature of 925° F. The 925° F limitation is conservative and there exists a reasonable expectation that actual operating temperatures may be raised in the future. This will permit higher steam conditions with a consequent decrease in the plant heat rate. It is estimated that an increase in the sodium temperature of not more than 100° F is sufficient to produce 1250 psig/900° F steam. The prospect of achieving these improved conditions is sufficiently promising that serious consideration be given to the installation of a turbine plant which would be suitable, with minor rework, to utilize 1250° psig/900° F steam at a future date.

The attainment of 1450 psig/1000° F and 1450 psig/1000° F/1000° F steam conditions initially by means of an auxiliary combustion heater in conjunction with the reactor is briefly discussed. Future improvement in reactor performance may also result in increased heat production either with or without higher operating temperature. Provision can easily be made in the plant layout for additional future generating capacity.

The last section discusses the conclusions reached from this design study. It is believed that this report gives a feasible design of a full scale nuclear power plant and sufficient design information is given to obtain realistic construction and operating cost estimates from which the over-all economic evaluation of bus bar power costs can be made.

II. URANIUM FUELED REACTOR

A. GENERAL

The reactor core consists of an array of closely packed, canned, hexagonal graphite cells contained within a large steel tank (see Fig. 3). The tank is surrounded by a thermal shield and a layer of thermal insulation. Since the reactor is located entirely below ground level, the concrete foundation provides adequate biological shielding except, of course, for the top. The top shield is composed of several feet of heavy concrete and could be removed should major repairs to the core be required.

The reactor has been designed to produce a thermal power of 250 megawatts with the maximum zirconium temperature limited to 950° F. The maximum uranium temperature has been limited to 1200° F. The sodium inlet temperature to the reactor has been selected as 500° F, so that the desired steam pressure may be produced. The number of fuel channels and velocity of coolant through the core were chosen so that a mean bulk outlet temperature of at least 925° F would result while satisfying the above requirements. See Table I for reactor design and performance data.

The lattice spacing was selected as 10 inches and the required enrichment as 1.8 per cent on the basis of a preliminary cost analysis. Although these quantities cannot be completely optimized until a specific application and cost structure are defined, these values are sufficiently close to an optimum to permit reasonably accurate estimates to be made of construction and operating costs.

B. FUEL ELEMENTS

The fuel elements consist of clusters of 19 rods. Each rod is made up of ten slightly enriched uranium fuel slugs 0.455 inch in diameter by 12 inches long. The rods are contained in a stainless steel jacket 0.495-inch OD by 0.010-inch

wall (see Fig. 4). The jacket is sealed at the top and bottom with a plug and, except for a helium filled expansion space, is filled with a bonding material (either sodium or NaK). The average thickness of the bond is 0.010 inch. The rods are spaced from each other by helically wrapped steel wires.

The rods are joined to a common hanger rod by which they are suspended in the coolant channels located in the moderator elements. The hanger rod is supported from the top shield by a shielding plug. On each assembly there is an orifice plate to throttle the flow of coolant to each fuel cluster. The size of the orifice plate is determined by the location of the element in the reactor so that the bulk coolant outlet temperature may be maximized.

C. MODERATOR AND REFLECTOR ELEMENTS

The moderator elements are graphite columns contained within sealed zirconium cans having 0.035-inch wall thickness. There are small gaps between the zirconium and graphite to allow for thermal expansion, radiation growth and clearance for assembly. A nominal clearance of 0.100 inch is provided between adjacent can walls to permit sodium coolant to flow between these elements.

A typical canned moderator element is shown in Fig. 5. The canned reflector elements are similar except that they do not contain fuel channels or coolant tubes.

There are 233 moderator elements forming an active core 13.3 feet in diameter by 10 feet high. A 2-foot reflector surrounds the core both in the radial and axial directions resulting in over-all dimensions of 17.3-foot diameter by 14-foot height.

D. CORE TANK

The core tank is 17.5-foot ID with 2-inch walls made from stainless steel plate. Near the bottom of the tank is a core support plate which locates and supports the moderator and reflector elements. Beneath this plate is a second plate, or tube sheet, forming two plenum chambers, one between the two plates, and one between the lower plate and the bottom of the core tank. The coolant channels extend through both plates so that the main sodium flow is from the lower plenum chamber up through the coolant tubes. The flow between the moderator and reflector elements is from the upper plenum chamber. Each chamber is force-fed by separate downcomers located just inside the core tank wall with the division of flow being adjustable.

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E. CORE COOLING SYSTEM

The primary sodium coolant removes heat from the fuel elements, the moderator and reflector elements, and the thermal shield. This useful heat represents more than 98 per cent of the total energy produced in the reactor. The heat generated in the neutron shield and losses through the thermal insulation are removed by a separate low temperature cooling system. The low temperature coolant is an organic liquid such as toluene, which is chemically nonreactive with sodium.

The sodium cooling system is divided into primary and secondary sections so that the radioactive primary coolant is contained in a small volume and is separated physically from the steam generator units. There are four parallel primary loops (see Fig. 6). The coolant in each loop leaves the core tank through pipes welded to the tank at a level just above the top of the moderator and reflector units. The coolant then flows to the inlet end of a vertically-mounted centrifugal pump. Two pumps in parallel are provided in each loop. The frozen seal sodium pumps now being developed for the SRE are suitable for application here.

Because of the radioactivity induced in the sodium, that portion of the primary sodium system not inside the reactor must be surrounded by a gamma shield. The sodium-to-sodium heat exchangers, primary system piping, and pumps associated with each loop are located in a vault adjacent to the reactor and below ground level. There are four such vaults, one for each loop. Each vault is covered by 6-foot deep removable shielding beams.

The secondary non-radioactive sodium system receives its heat from the primary system with approximately a 30° F loss in temperature. There are two secondary loops with four pumps in parallel in each loop. The secondary sodium gives up its heat to steam in the superheaters, evaporators, and economizers.

F. CONTROL AND SAFETY ELEMENTS

A total of 18 control and safety elements are provided. Each element operates within a stainless steel thimble, as in the SRE. The thimbles, however, extend through the core in regular coolant channels rather than in special channels located at the corners of moderator elements as in the Sodium Reactor Experiment. From experience gained to date in the fabrication of the moderator cans for the

Sodium Reactor Experiment it appears that the present design will enable considerable savings in the cost of manufacturing the moderator units.

Each control rod is made up of a stainless steel tube surrounded by boron rings.¹ The rod size is determined by the amount of heat generated within the rod, the maximum temperature limits, and heat transfer considerations. The rod is cooled by conduction and radiation across a 10-mil helium annulus between the boron steel rings and the thimble. The thimble wall is cooled by the primary sodium flowing between the process channel and thimble. The purpose of the thimble is to exclude sodium and sodium vapors from the drive mechanisms. It would be desirable from a nuclear point of view to avoid the use of the steel thimbles, but the physical properties of zirconium at these temperatures are not sufficiently known to permit zirconium thimbles to be specified at this time. From a heat transfer standpoint it would be desirable to eliminate the thimbles and cool the rods directly with sodium, but this would require the development of seals or mechanisms to operate in sodium or sodium vapor. Should these be developed successfully, the thimbles should be discarded.

The safety elements are as large in diameter as the channel dimensions will permit since they normally must be out of the reactor during operation and therefore do not present the heat transfer problem encountered in the control rod design. The safety elements may either be boron steel rods or boron-containing balls. Both types are still under consideration for the Sodium Reactor Experiment. The final selection will depend upon the results of development work now in progress.

G. THERMAL SHIELD AND INSULATION

The core tank itself actually serves as a partial thermal shield but is normally not designated as such. The thermal shield proper is made of mild steel interlocking slabs, approximately 6 inches total thickness, located adjacent to and just outside of the core tank. The upper thermal shield is supported from the top biological concrete shield. The top shield is pierced by vertical holes, aligned with the fuel channels, and normally filled by shield plugs which support the fuel element hanger rods.

Thermal insulation is placed between the thermal shield and the biological shield or foundation walls to reduce the heat losses. Most of the heat generated within the thermal shield is radiated or conducted back into the core tank and

removed by the sodium coolant. The temperature of the thermal shield will vary from around 500° F below the reactor to approximately 1200° F in the central portion of the upper thermal shield.

H. FOUNDATION, SHIELDING, AND SHIELD COOLING

The foundation concrete is lined with a mild steel plate. Cooling tubes welded to this plate contain an organic liquid which is circulated to remove the heat conducted across the insulation and generated in the foundation concrete and steel liner. A design temperature of 150° F has been selected as a maximum for the concrete. Since the reactor is entirely below ground, the earth and foundation concrete provide adequate biological shielding. The foundation material is regular reinforced Portland concrete. The thickness and reinforcing requirements will depend on the subgrade conditions at the site.

The upper neutron shield is designed so that it can be removed should this be necessary. Unlike the Sodium Reactor Experiment, which is an experimental reactor, the upper shield will not be designed to be rotated for ready removal of moderator and reflector units.

I. INSTRUMENTATION

Instruments will be provided to monitor the reactor neutron flux during start-up and operation. Figure 7 is a block diagram of the proposed reactor control system. Three types of chambers will be used to cover the wide range of neutron flux involved in going from shut-down conditions to full operating power. These are fission chambers, gamma-compensated ion chambers, and uncompensated ion chambers. This selection has been made so as to provide continuous information over the range of approximately 10 decades which is attainable with a neutron source.

The fission chamber signals will be amplified and read on a counting rate meter. This arrangement will provide neutron flux indication over a range of 4 decades. The location of these 4 decades of range with respect to the power level may be altered by remotely positioning the fission chambers. The gamma-compensated ion chambers will each cover a range of approximately 6 decades. They will be arranged so as to overlap each other and also overlap the coverage provided by the fission chambers. The signal from this type of chamber will be passed through a logarithmic amplifier and differentiated to obtain an output

proportional to reactor period. The amplifier output will also be displayed on a recording instrument with a 6 decade strip chart. The output from the uncompensated ion chambers, of the parallel circular plate type, will be used primarily for flux monitoring at power levels from about 1 decade below to 1/2 decade above full power level. By the use of scale changing resistors in the amplifier input circuit, this range may be made to fall in any portion of about 7 decades of range from normal full power level downward. This feature is particularly useful in the initial start-up of the reactor.

Instrumentation will also be provided in each of the sodium circuits to measure the flow rate, temperature, and pressure at several different points. The sodium level in the reactor and in the surge tanks will be continuously measured. The outlet temperature of the sodium in each coolant channel will also be continuously monitored.

J. AUXILIARY SYSTEMS

1. Sodium Service Systems - All of the equipment required for melting, filtering, and transferring sodium to the primary and secondary coolant systems comprises the sodium service system. This system will be used intermittently, and all lines and vessels will be empty when not in use. All the vessels and piping that may contain sodium will be electrically heated to approximately 300° F. The system is designed to perform the following functions:

- a. Purge air from the system and maintain an inert atmosphere in the pipes and vessels
- b. Melt and transfer virgin liquid sodium to either the primary or secondary fill tanks
- c. Recirculate and filter used sodium from either fill tank
- d. Circulate sodium to the cold traps in the primary and secondary cooling circuits to flush out precipitated contaminants
- e. Drain sodium from the pipes and vessels

Sodium will be charged to the system through a melting furnace and a filter. The sodium will then be pumped to both the primary and secondary fill tanks. A check valve will prevent radioactive sodium from being accidentally forced from the primary fill tank back to the transfer tank. The sodium service system will also be used to recirculate sodium for filtering and cold trapping. Contaminated sodium will be drawn off to portable containers for disposal. Each cold trap in

the reactor coolant system may be isolated from its loop and flushed with sodium from the sodium service system.

2. Gas Handling System - An inert gas, either helium or nitrogen, is to be used wherever an inert atmosphere is necessary to prevent oxidation of either graphite or sodium. Specifically, helium will generally be used where the atmosphere has direct contact with either sodium or graphite. Nitrogen will generally be used where there is normally no direct contact such as the primary and secondary system piping compartments. The gases will be stored in high pressure storage tanks. Flow from the high pressure storage units will be to low pressure storage tanks through pressure-reducing valves. The helium required for the free sodium surface will be noncirculating. It will first be passed through a NaK bubbler to remove oxygen. The gas handling systems for this reactor are merely enlarged versions of the systems being planned for the SRE and do not differ from the SRE design in any significant detail.

3. Fuel Handling System - The fuel handling system will replace the irradiated fuel elements with new fuel elements while the reactor is shut down. In addition, this system is designed to store, inspect, clean, and transport new and spent fuel elements (see Fig. 2). The fuel handling system is comprised of the following major components:

- a. New fuel storage area; new fuel elements will be stored in individual cells in an inert gas atmosphere
- b. Fuel cleaning cells; fuel elements removed from the reactor will be cleaned of sodium by a water spray in the cleaning cells.
- c. Storage cells; spent fuel elements will normally be stored in cells provided in the storage area. These cells are cooled by an organic liquid.
- d. Fuel transfer cask; while the reactor is shut down, the fuel transfer cask will be used to transfer new fuel elements to the reactor and to remove spent fuel elements from the reactor and then transfer them to a cleaning cell. After reactor operation is resumed, the cask will be used to transfer the irradiated elements from the cleaning cells to the storage cells. The design permits making and breaking gas-tight connections between the cask and the reactor loading face, the new fuel storage cell, or the cleaning cells.
- e. Fuel shipping cask; irradiated fuel elements must be transported from the storage area to chemical processing facilities in a fuel shipping cask.

K. REACTOR BUILDING

The reactor building is shown in Fig. 2. Its location relative to the rest of the equipment is shown in a proposed plant layout (Fig. 1). The reactor building houses the reactor, the fuel handling cask and cranes, the control rooms for the entire plant, the new fuel storage area, the spent fuel cleaning cells, and storage facilities, the entire primary coolant system, the pumps of the secondary coolant system, and auxiliary equipment areas, such as the sodium handling system, the inert gas systems, and shield cooling system.

The high bay area extends over the top of the reactor, the new fuel storage area, the spent fuel cleaning and storage cells, and the fuel dismantling area. The fuel handling coffin traverses this entire high bay area by means of an overhead crane.

The control rooms are located at one end of the reactor building. There are three rooms located one above the other with a service elevator connecting them. Each room overlooks the reactor on one side and the outdoor turbine-generator unit on the opposite side.

The primary sodium system equipment is located in underground rooms adjacent to the reactor. There are four parallel loops in the primary system, one in each quadrant. A 25-ton traveling crane is located on each side of the reactor building and is used for installing, handling, and repairing the equipment located in the underground rooms.

L. DESIGN PERFORMANCE

The reactor has been designed so that when operating within the specified temperature limits it will produce 250 megawatts thermal power. The maximum uranium temperature has been limited to 1200° F, the maximum zirconium temperature to 950° F. The coolant inlet temperature is 500° F. The maximum sodium velocity through the central channels is 9 ft/sec. Other pertinent design and performance data are given in Table I.

The neutron economy is given in Table III for the start up conditions. This will be approximately the same throughout the core life since it is not planned to feed the plutonium back into the reactor once it has been removed. The initial conversion ratio is 0.72. The lattice spacing and fuel enrichment for the initial loading were chosen as 10 inches and 1.8 atomic per cent respectively on the

basis of a preliminary cost analysis. The number of loaded channels is 215. This loading provides an initial excess reactivity of 2 per cent, which should be sufficient to allow for at least 3000 Mwd/t irradiation.

M. ALTERNATE PERFORMANCE AND DESIGN CONCEPTS

A major portion of the five-year Sodium Graphite Reactor engineering development program is devoted to operating tests in the Sodium Reactor Experiment of improved uranium fuel elements. Concurrently, vigorous programs at a number of Atomic Energy Commission sites are being applied to fuel material development, and advances are being made both in heat treating methods and through alloy research. There are good prospects that the maximum permissible uranium temperature will be raised as a result of metallurgical improvements and also through the use of fuel elements of special shapes, for example, rods having small axial voids. An increase in maximum uranium temperature will permit a corresponding increase in the thermal power which can be produced in the reactor described here. In this event, the reactor cooling must be expanded to be capable of removing the increased power. This can be accomplished either by overdesigning the heat transfer systems initially, in anticipation of improved performance, or by allowing certain provisions for adding additional heat transfer equipment and turbogenerator capacity when needed. At this time the latter method appears preferable.

A second major objective of the Sodium Reactor Experiment program is to investigate the feasibility of operating sodium-zirconium systems at temperatures above 1000° F. With the determination of maximum allowable oxygen contents at these higher temperatures and the development of methods for maintaining satisfactory oxygen concentrations, it is reasonable to expect that the 950° F limit adopted in the present design will prove to be overly conservative. Higher maximum sodium temperatures will permit both increased thermal power from the reactor and improved steam conditions.

There are several alternate possibilities for improving the reactor design which merit further study. These alternates, however, differ to a greater or lesser degree from the Sodium Reactor Experiment design and will therefore require closer scrutiny before they can be definitely recommended for this application.

The neutron economy could be improved if several moderator cells could be canned as a unit. The amount of intercell zirconium and sodium used when each cell is individually canned could be reduced to half by using a square lattice and canning four cells as a unit. This would permit the fuel enrichment to be reduced and/or the conversion ratio increased. It has not been possible in this preliminary study to investigate the manufacturing questions, heat transfer problems, graphite temperatures, and thermal stresses which would result.

Also, if the steel thimbles provided for the control and safety elements could be replaced by zirconium thimbles, or omitted entirely, as mentioned in Section II-F, the enrichment and conversion ratio could similarly be improved. The problems here are mainly associated with the development of reliable seals against sodium and assurance of trouble-free operation.

III. THORIUM URANIUM-235 FUELED REACTOR

A. GENERAL

The reactor design discussed in this section differs in only one major aspect from the design discussed in the preceding section, namely, the use of thorium-U²³⁵ fuel elements. The reactor consists of closely packed, canned, hexagonal graphite cells contained within a cylindrical steel tank. The thermal rating of the reactor has been taken as 250 megawatts with the maximum zirconium temperature limited to 950° F. The sodium inlet temperature to the reactor is 500° F. The calculated central thorium temperature is 1600° F, but this is not considered to be limiting. The core size was determined mainly from nuclear considerations. While the required amount of power could be removed from a smaller core, the neutron leakage would be quite high.

The lattice spacing was selected as 8.0 inches and the required atomic fraction U²³⁵ in thorium as 3.70 per cent on the basis of a preliminary cost analysis. See Table II for reactor design and performance data.

B. FUEL ELEMENTS

The fuel elements consist of clusters of 7 rods. Each rod is made up of ten slugs of thorium-U²³⁵ alloy 0.648 inch in diameter by 12 inches long. The slugs are contained in a stainless steel jacket 0.688-inch OD by 0.010-inch wall. The

jacket is sealed at the top and bottom with a plug and, except for a helium filled expansion space, is filled with a bonding material (either sodium or NaK). The average thickness of the bond is 0.010 inch.

The rods are joined to a common hanger rod by which they are suspended in the coolant channels. On each assembly there is an orifice plate to throttle the flow of coolant to each cluster so that the bulk coolant outlet temperature may be maximized.

C. MODERATOR AND REFLECTOR ELEMENTS

The moderator elements are graphite columns contained within sealed zirconium cans similar to the Sodium Reactor Experiment and similar to those used in the uranium fueled reactor (Section II) except for the over-all dimensions. The can walls and coolant tube wall thickness are each 0.035 inch. The coolant tube ID is 2.493 inches. There are 203 moderator elements forming an active core 10 feet in diameter by 10 feet high. A 2-foot reflector surrounds the core both in the radial and axial directions resulting in over-all dimensions of 14-foot diameter by 14-foot height.

D. CORE TANK

The core tank is identical to that described in Section II-D except that the ID is 14.2 feet. The core cooling system, the control and safety elements, the thermal shield and insulation, the foundation, shielding and shield cooling, the instrumentation, the auxiliary systems, and the reactor building are the same as those described in Section II.

E. DESIGN PERFORMANCE

The reactor has been rather conservatively designed. The maximum sodium-zirconium temperature has been limited to 950° F. The size of the reactor, the fuel element dimensions, and the lattice spacing were selected from nuclear and economic considerations. Since the maximum required reactor power is 250 megawatts, the center temperature of the thorium will not exceed 1600° F with a maximum sodium velocity of 16.2 ft/sec through the central channel. Table II gives additional design and performance data.

The nuclear characteristics change with burn-up since the U^{235} originally present is replaced by U^{233} , but not on a one-for-one basis. The fissionable

material required for make-up is highly enriched U^{235} . The neutron economy of the core is given for the first two complete cycles in Table IV. The conversion ratio improves from an initial value of 0.675 to 0.782 at the end of the second cycle. The conversion ratio should ultimately level off at about 0.81. It is assumed that 40 per cent of the fissionable material is burned each cycle (this is approximately 10,000 Mwd/t). The irradiated material is removed from the reactor after 40 per cent burn-up of the fissionable material and sent to chemical processing. The fission products are removed, the Pa^{233} allowed to decay to U^{233} , and sufficient U^{235} and thorium are added so that the initial excess reactivity is restored. The principal isotopes present in the fuel material are listed in Table V for the first few cycles. Also included in Table V is an estimate of the excess reactivity in a fuel element at the beginning and end of each cycle. It will be necessary to change fuel by zones so that the reactor will be critical at all times.

F. IMPROVED PERFORMANCE

As mentioned in the previous section, the size of the reactor, number of elements, etc. were determined from nuclear and economic considerations. The reactor is capable of producing considerably more power than the amount required to generate 75 megawatts of electricity. If the center temperature of the thorium is increased to 2000° F (which appears to be only slightly less conservative than 1600° F) and the sodium velocity through the central channel is increased to 25 ft/sec, the capability of the reactor would be nearly 400 megawatts thermal, or approximately 125 megawatts electrical, with the 950° F limit retained for the maximum sodium temperature. This improvement in performance does not entail the solution of a basic metallurgical problem, as is required to overcome the uranium alpha-beta phase transformation limitation. The absence of any significant phase transformation in thorium and the generally superior metallurgical properties of this material lend confidence to the belief that the improved performance indicated above will be readily attained.

It was indicated in Section II-M that the Sodium Reactor Experiment program is expected to raise the 950° F limitation on maximum sodium temperature. Advances in this direction will permit a further increase in thermal power from the reactor described here and also will permit improved steam conditions. In fact, the thorium-base fuel element is uniquely capable of taking advantage of higher

sodium temperatures, since very high center temperatures in the metal can be sustained.

G. ALTERNATE DESIGN CONCEPTS - BREEDER BLANKET

The alternate design possibilities mentioned in Section II-M are equally applicable to the thorium fueled reactor. A more important design alternative in the case of the thorium- U^{235} fueled reactor is the addition of a breeder blanket of thorium metal rods. This blanket would be located between the core and the reflector and would serve to absorb a large fraction of the leakage neutron flux, thereby increasing the over-all nuclear conversion ratio to a value approaching unity. The various problems associated with the blanket design have not been investigated in the present study.

IV. POWER PLANT CYCLES

A. BASIC CONSIDERATIONS

1. General - Of fundamental importance in the selection of a power plant cycle is the bus bar cost of the electric energy generated. The ultimate aim of any economic study is the optimization of the variables in order to obtain the lowest possible energy cost. It may be worth while to restate the following points:

- a. Thermal efficiency of a plant is only one factor of many that determines the final energy cost.
- b. There is no single power plant cycle which is optimum for every situation. Many optimization studies have been made for particular reactor configurations and their power plant cycles; of necessity, these studies have been based upon certain assumptions. Thus, they are valid in the general sense that they establish certain boundaries. But the cost of borrowing money, the capital structure of the utility, plant size, plant factor, conventional fuel costs, etc., will differ for each specific case. Thus, each situation is unique and requires its own analysis.
- c. The cycle optimized for the lowest bus bar cost may not necessarily be the cycle chosen. There are many considerations, some intangible, which may make a non-optimum cycle more attractive.

2. Reactor - The particular reactors under consideration have been designed with a considerable degree of conservatism. Thus, it is reasonable to assume that the performance of the reactor during actual operation may well exceed the design values. In this sense the reactor portion of the plant has a "built-in" reserve capability. Greater heat generation from the reactor requires more turbine plant capacity to utilize the additional heat. But aside from this, reserve reactor performance without a change in sodium temperature does not influence the selection of the thermal cycle.

3. Turbine Cycle - Previous optimization studies² have shown that the non-reheat regenerative cycle is the optimum choice at this time. While the super-pressure cycle appears promising for nuclear steam-electric stations, this technology is as yet insufficiently developed. It seems desirable to avoid complications in the conventional portion of a nuclear station, at least until there

exists a more widespread experience in the operation of power reactors. Whether or not the thermal gain available from a reheat regenerative cycle for the proposed nuclear station warrants the extra complexity and costs has not been assessed at this time. A tentative opinion in this respect is that it does not. The present choice is therefore the non-reheat regenerative cycle.

In conventional fuel fired stations, fuel economy can be considerably increased by the use of five or more extraction feedwater heaters, which raise the temperature of the feedwater to e.g., 408° F (66,000 kw preferred-standard, 1250/950° F, five heaters). In a nuclear plant, however, a high feedwater temperature has a detrimental effect. It either lowers the maximum possible steam pressure with a consequent loss in cycle efficiency, or it decreases reactor power output. The decrease in reactor power output does not result in any significant reduction in either capital expenditures or operating costs, whereas the higher feedwater temperature is only obtained by the installation of five or more heaters at a cost of something like \$50,000 per heater. Previous studies have shown that the best regenerative feedwater temperature for nuclear plants is in the range from 250° to 300° F. The attached "Temperature-Difference (or Pinch-point)" diagrams show graphically how higher steam conditions push up the cold end of the reactor coolant temperature with a consequent decrease in reactor power (for a constant hot end temperature).

B. CHOICE OF INITIAL TURBINE CYCLE AND THROTTLE CONDITIONS

A non-reheat regenerative cycle with 800 psig/825° F throttle conditions and a feedwater temperature of 300° F has been selected. At this time, the mean temperature of the primary sodium leaving the reactor has been limited to 925° F. The choice of 800 psig/825° F represents what appears to be the best compromise between a low cycle heat rate, adequate temperature drives in the sodium to water heat exchangers, and permissible moisture in the turbine exhaust.

A three-stage feedwater-heating arrangement was chosen to obtain a final feedwater temperature of 300° F. This will result in a cycle arrangement which can be expanded, should this ever be desired, to the standard five heater arrangement by the addition of 2 heaters without materially changing the duty of the existing three heaters. Preliminary heat balance calculations show a turbine cycle heat rate of 10,050 Btu/kwh for a gross electrical output of 80,800 kw.

The net plant heat rate for the station is estimated at 10,800 Btu/kwh, which compares quite well to a net plant heat rate of 11,300 Btu/kwh for a 66,000 kw preferred standard machine with a 1250 psig/950° F throttle, gas-fired boiler, five heater extraction and gross electrical load of 75,000 kw. The poorer heat rate of the conventional station is due to the inefficiency of the boiler, whereas reactor heat losses are estimated to be less than 2 per cent.

The three heater 800 psig/825° F cycle is not necessarily a final selection. It may be possible, for example, to go up to 850 FTT by increasing the superheater surface. Other changes may appear to be desirable upon further study. But, in the absence of a combustion superheater, the particular cycle chosen is easily attainable with the presently planned conservative reactor coolant temperatures. Thus, it is a good starting point for further investigations and refinements. It can be considered the base cycle. A simplified heat balance for this cycle is shown in Fig. 13.

C. ALTERNATE CONDITIONS

1. General - As stated above, a throttle condition of 800 psig/825° F was selected for initial operation. No factors other than those pertaining to the initial reactor performance have been considered, except that, in anticipation of the alternates set forth below, the initial steam conditions represent the least departure, within economical limits, from standard conventional practice. Therefore, the highest throttle conditions attainable within the present reactor limitations have been chosen. A station solely designed for these conditions is considered to be the basic plant. However, the following two important aspects pertaining to the uranium-fueled reactor could influence the design of the conventional part of the station:

- a. Future increase in reactor performance
- b. Use of an auxiliary combustion heater

2. Present Reactor Operating Conditions - In order to gain a full appreciation of the potentialities of the sodium graphite reactor, a brief and simplified discussion of the basic relationships between reactor operating temperatures and turbine cycle efficiency is necessary. The selection of an initial turbine cycle for a nuclear steam electric plant involves a compromise between two competing factors. These are maximum heat generation from a given reactor and maximum conversion of this heat into electric energy, i. e., maximum thermal efficiency,

where plant electrical output is the product of heat generation and thermal efficiency.

Whereas the amount of heat generation from a combustion fired boiler is inherently limited, a given operating reactor can theoretically produce any desired heat output, or thermal power, from essentially zero up to infinity, provided this heat can be safely removed. Heat removal, i. e. heat transfer from the center of each fuel element, depends on the temperature drive from the center to the surface. Due to the excellent heat transfer characteristics of sodium, the temperature gradient between the fuel element surface and the sodium coolant is quite small. Therefore, heat removal from the reactor depends primarily upon the difference between the center temperature of each fuel element and the average temperature of the liquid sodium flowing up and through each fuel channel.

The availability of the thermal power for conversion to electrical power decreases with a decrease in the average sodium temperature. For a given heat sink, the efficiency of any heat cycle is determined by the average temperature of heat addition to the system. Thus, in order to obtain a high thermal efficiency, a fairly high average sodium temperature is required.

As discussed in the previous sections, a reactor using metallic uranium fuel elements and zirconium-clad moderator presently has two temperature limitations. The first applies to the uranium metal which limits the center temperature of the fuel element. The second limitation pertains to the zirconium metal which limits the sodium outlet temperature.

It is one of the nuclear characteristics of the reactors under consideration that heat generation from each fuel element varies radially from a maximum for the center elements to a minimum for the peripheral elements. If the sodium flow were the same through each fuel channel, then the sodium flowing through the center channels would experience the largest temperature rise and the sodium flowing through the outer channels would experience the least rise. This situation would result in a low mixed mean sodium outlet temperature due to the greater number of outer fuel elements. In this situation only the inner fuel elements are at the maximum permissible center temperature.

Thermal efficiency demands the highest attainable mixed mean sodium temperature. Therefore, the sodium flow through each fuel channel is adjusted by throttling such that the outer channels have the least flow. This causes the

sodium outlet temperature to be a maximum consistent with the aforementioned temperature limitation. As a result, each fuel element operates at a temperature limit, either in maximum center temperature or surface (i. e. sodium exit) temperature. Power output from each fuel element remains unaffected by sodium flow as it is dependent only upon neutron flux distribution.

3. Future Increase in Reactor Operating Performance - An improvement in reactor performance will be the result of either a higher permissible sodium exit temperature or a higher permissible center temperature in the fuel element.

When it becomes permissible to do so, the sodium exit temperature can be raised by reducing the sodium flow through the outer fuel channels until the new temperature limit is reached. This flow reduction causes an increase in the mixed mean sodium temperature without exceeding the present limit for the fuel element center temperature. Also, thermal power output is unaffected by this change. A higher sodium temperature in turn permits higher steam conditions with a consequent gain in cycle efficiency.

If it becomes practical to raise the limit on the central metal temperature, then the thermal power output from the reactor may be increased. In this case cycle conditions are unchanged since they are solely dependent upon the sodium temperatures.

If both the fuel element center temperature and the sodium exit temperature can be raised, then not only will reactor thermal power increase, but the availability of the total reactor thermal power for conversion into electrical energy also increases.

The benefits to be gained from the improvements in performance of a given reactor heat source are then two-fold:

- a. Increase in electric power production from the same heat source, which results in lower bus bar cost
- b. Decrease in bus bar cost as a result of increased thermal efficiency

These gains are associated with the unique capability of the sodium cooled reactor to produce additional thermal power and permit higher cycle efficiencies as soon as operating experience permits an increase in the present temperature limitations.

4. Future Steam Plant Performance

(a) Increased Generating Capacity - If the increase in thermal power output can be achieved with the original reactor then no additional capital investment is required for the reactor portion of the plant. Additional turbogenerator facilities and heat transfer equipment external to the reactor will be necessary, but the unit cost of electric energy will decrease because of the incremental gain due to increased thermal power output from the original reactor. This is true because reactor investment costs are fairly substantial and larger blocks of power will significantly decrease the total investment cost per unit of energy generated. Provision can easily be made in the plant layout for the anticipated additional facilities.

(b) Increased Thermal Efficiency - The effect of an increase in thermal efficiency can be either an increase in electric energy generation or a decrease in fuel feed for the same amount of electric energy. The bus bar cost of electric power is decreased in either case. The increase in over-all plant efficiency is solely due to the higher reactor sodium temperature.

In the present reactor design the maximum primary sodium and zirconium temperature has been limited to 950° F which brings about a mixed mean sodium outlet temperature of about 925° F. A 30° F temperature difference in the primary-to-secondary sodium heat exchangers results in a secondary sodium temperature of 895° F which (for a 70° approach in the hot end of the superheater) permits a steam temperature of 825° F. These relationships are shown in the temperature-difference diagram for this case in Fig. 8. There exists a reasonable expectation that operating experience with the SRE and future metallurgical developments will raise the present temperature limit. The sodium graphite reactor, with its flexible design, is well adapted to utilize higher temperatures and other future technological improvements.

If the steam plant is to be capable of operating at the higher anticipated throttle conditons, certain features must be initially included in the plant design, as well as provisions for certain changes at the time the station goes to higher conditions. Whether or not the additional cost required for this is warranted can only be determined by means of an economic study for each specific situation and is beyond the scope of this general discussion. Even if, in a given case, the additional expenditure can be justified, it seems prudent to make the initial extra

investment as small as possible. This precaution seems advisable since there is only a reasonable expectation, not a definite assurance, that higher sodium temperatures can be attained.

In order to estimate the gains that may result from the expected improvements in reactor performance, two successively higher steam conditions have been examined. The first is the standard steam condition 850 psig/900° F, which represents the least departure from the initial design conditions of 800 psig/825° F. Thus, the additional investment required is probably quite small; yet the net plant heat rate drops from an estimated 10,800 Btu/kwh to 10,500 Btu/kwh, a 2.8 per cent reduction.

The attainment of 850 psig/900° F steam requires a relatively modest extension from the basic design. It was pointed out in Section IV-B that it may be possible to go to 850° F steam in the basic design by increasing the superheater surface and decreasing the temperature approach from 70° F to 45° F. If this change should be economically justified, then the attainment of 850 psig/900° F conditions requires a primary sodium outlet temperature of 975° F, an increase of only 50° F. The pinchpoint diagram for the 850 psig/900° F condition (Fig. 9) shows that this reduction of drive in the superheater does not cause a significant change in the pinchpoint. (Pinchpoint is the temperature difference at the entrance to the evaporator.)

In the same manner, the attainment of 1250 psig/950° F throttle condition requires a mean sodium outlet temperature of 1025° F, an increase of 100° F over the outlet temperature of the present basic design (see Fig. 10). The estimated net plant heat rate drops to 10,160 Btu/kwh, a decrease of 3.1 per cent from the first step (850 psig/900° F) and a decrease of 5.9 per cent from the initial heat rate of 10,800 Btu/kwh.

It is not too important at this time to determine necessary drives in the superheater, particularly since it would require a specific superheater design and an investigation of its behavior during part-load conditions. The significant point here is the fact that an increase in reactor outlet temperature of 50° to 100° F can be expected with a reasonable degree of confidence. The history of steam temperatures in conventional stations, over even so short a period as the last 10 years, provides a striking illustration of metallurgical advances permitting ever higher temperatures. It would seem reasonable to design the steam

plant initially to take advantage of the higher steam conditions anticipated for the future. Whether the choice is made for ultimate operation at 850 psig/900° F or at 1250 psig/950° F will be largely governed by an economic analysis for each specific installation.

There appear to be no undue difficulties involved in designing the steam plant for initial operation at 800 psig/825° F and eventual operation at 850 psig/900° F or 1250 psig/950° F. Also, on the basis of preliminary information from turbine manufacturers regarding feasibility and costs of "convertible" turbines, there appear to be no significant problems in obtaining a turbine designed for initial operation at 800 psig/825° F but suitable for change-over to the higher conditions. There will be no penalty in machine efficiency; i. e., the machine, when operating at 800 psig/825° F or upon rework, at either of the higher conditions, will have an efficiency in each case equal to that of a machine initially designed for the particular throttle conditions. The first cost of the convertible machine will be the same as a standard handbook machine for the higher conditions.

5. Use of an Auxiliary Combustion Heater - The best steam condition attainable at this time with an all-nuclear station and based upon the proposed reactor design is approximately 800 psig/825° F. Nevertheless, the over-all plant heat rate of this all-nuclear station is very good. This arises from the fact that the reactor and its sodium system have an "efficiency" of something like 98 per cent (i. e., heat losses are quite small), whereas the efficiency of a modern boiler, mainly due to the stack loss, is about 85 per cent. Of course, higher throttle conditions will bring about significant reductions in the plant heat rate for all cycles, since the reactor heat losses are nearly constant regardless of temperatures.

Modern throttle conditions of 1450 psig/1000° F can be attained by the use of an auxiliary combustion superheater. The summary table (Table VI) and temperature difference diagrams (Fig. 11 and 12) show the percentage of heat supplied by a combustion superheater for various throttle conditions and the corresponding net station heat rates. In every case the thermal power of the reactor is reduced by the amount of heat supplied by the combustion unit. Also, as shown in the diagrams, the reactor coolant temperature at the cold end has to be raised for successively higher throttle conditions.

The use of a combustion superheater might bring the reheat-regenerative cycle back into the picture. Heat rates and other pertinent data for the 1450 psig/1000°/1000° reheat condition have been estimated. However, in the reheat cycle 20 per cent of the heat is supplied in the fired furnace as against 10 per cent for the 1450 psig/1000° non-reheat case. The lower efficiency of the furnace (say 75 per cent) tends to cancel out the thermal gain of the reheat cycle. Whether or not the installation of a combustion superheater (or superheater-reheater) is warranted may depend upon effective utilization of the hot furnace gases leaving the heater. The use of an exhaust gas turbine is one of the possibilities being investigated. A final determination will require a detailed economic analysis.

V. CONCLUSIONS

A feasible preliminary design has been presented for a sodium cooled, graphite moderated nuclear power plant of 75 megawatt nameplate rating, which is a straightforward extension of the Sodium Reactor Experiment, now under construction. Two reactor designs have been presented, one for use with slightly enriched uranium fuel and one for use with thorium-uranium fuel. A turbine cycle for initial operation has been selected and a number of alternates presented.

The thorium-uranium fueled reactor can be operated at powers considerably above the design value without the need for technological improvements. Provision should be made in the initial design so that additional power plant capacity can be added when this extra power is needed.

Both the uranium fueled and the thorium-uranium fueled reactors can take advantage of anticipated developments which will allow the sodium outlet temperature to be increased. The power plant can be designed to utilize a higher sodium temperature in the future; the use of a convertible turbine will permit initial operation at the conservative conditions presently specified. Consideration has also been given to the feasibility of operating the station initially at higher throttle conditions by the use of a combustion-fired superheater. This warrants further investigation.

From the design and performance data given, it should be possible to make a reliable estimate of the unit plant construction and operating costs. From these the cost of power may be determined for any specified accounting procedure and fuel costs.

TABLE I

PERFORMANCE AND DESIGN DATA - URANIUM FUEL

A. REACTOR

1. Reactor Type	Thermal heterogeneous
2. Reactor Operating Data	
Nominal rated output, Mw (thermal)	250
Average specific power, fuel, Mw/ton of metal (1 ton = 2,000 lb) kw/kg-25	9.2 565
Average specific power, moderator, Mw/ton graphite	3.5 3.5
Average thermal flux in fuel, neutrons/sec-cm ²	2.5×10^{13}
Peak thermal flux in fuel, neutrons/sec-cm ²	4.1×10^{13}
Initial at. % U ²³⁵ in the fuel for 2% k excess	1.80
3. Materials	
Fuel	Uranium metal, slightly enriched in U ²³⁵
Moderator and Reflector	Graphite, each cell canned in 0.035 in. zirconium
Fuel element cladding	0.010 in. type 304 stainless steel
Coolant tube	3.250 in. OD by 0.035 in. wall zirconium
Primary coolant	Sodium
Secondary coolant	Sodium
Thermal shield	Mild steel
Biological shield	Barytes concrete

TABLE I (CONTINUED)

4. Core Nuclear Data

Lattice type	Triangular
Lattice spacing, in.	10.0
Number of cells, total	233
for fuel	215
for control & safety rods	18
Nominal reflector thickness, ft	2
Core diameters (including reflector), ft	17.3
Core height (including reflector), ft	14.0
Core composition, per cent by volume	
Graphite	87.05
Sodium	6.72
Stainless Steel	0.43
Uranium	3.57
Zirconium	1.81
NaK	0.32
Void (gas)	0.10
U^{235} content (initial loading), kg	443
Lattice constants (initial loading with equilibrium poisons) 2% k excess at 600° C. Uranium enrichment 1.8%	
ϵ	1.026
p	0.7564
f (for uranium)	0.8551
η (for uranium)	1.6878
k_{∞}	1.120
L^2 , cm ²	134
τ , cm ²	405
B^2 , cm ⁻²	205×10^{-6}

TABLE I (CONTINUED)

5. Fuel Element Data

Type of fuel element	19-rod cluster
Number of elements in reactor	215
Fuel slug material	Uranium (β -heat treated or 2 weight per cent zirconium alloy)
Diameter of uranium slugs, in.	0.455
Length of fuel slug, in.	12.0
Initial fraction U^{235} in fuel, at. %	1.80
Weight of uranium in reactor, kg	24,600
Fuel element length, ft.	10.0
Jacket wall thickness, in. of SS	0.010
Bond, in. of NaK	0.010

6. Control and Safety Rod Data

Number of control rods	10
Type of rods	Boron steel rod operating inside of steel thimble
Diameter of rods, in.	1.92
Total Δk in control rods, %	~ 10
Number of safety rods	8
Total Δk in safety rods, %	~ 9

B. COOLING SYSTEM DATA

1. Reactor

Coolant	Sodium
Temperature in, °F	500
Mixed mean outlet temperature, °F	925
Highest temperature out, °F	950
Maximum coolant velocity, ft/sec	9
Pressure drop across core, psi	6
Flow rate, lb/hr	6.5×10^6
Maximum heat flux, Btu/hr-ft ²	390,000
Maximum fuel temperature, °F	1200
Total Power, Mw	250

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TABLE I (CONTINUED)

2. Primary Loops

Number of parallel circuits	4
Inlet temperature to primary heat exchanger, °F	925
Outlet temperature from primary heat exchanger, °F	500
Flow rate in each circuit, lb/hr	1.625×10^6
Number of pumps in each circuit	2

3. Secondary Loops

Number of parallel circuits	2
Inlet temperature to superheater, °F	895
Outlet temperature from economizer, °F	470
Flow rate in each circuit, lb/hr	3.25×10^6
Number of pumps in each circuit	4

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TABLE II

PERFORMANCE AND DESIGN DATA - THORIUM URANIUM-235 FUEL

A. REACTOR

1. <u>Reactor Type</u>	Thermal heterogeneous
2. <u>Reactor Operating Data</u>	
Nominal rated output, Mw (thermal)	250
Average specific power, fuel, Mw/ton of metal (1 ton = 2,000 lb) kw/kg-25	23.3 694
Average specific power, moderator Mw/ton graphite	6.2
Average thermal flux in fuel, neutrons/sec-cm ²	3.1×10^{13}
Peak thermal flux in fuel, neutrons/sec-cm ²	5.1×10^{13}
Initial at. % U ²³⁵ in the fuel for 2% k excess	3.70
3. <u>Materials</u>	
Fuel	Thorium-uranium alloy (highly enriched uranium)
Moderator and reflector	Graphite, each cell canned in 0.035 in. zirconium
Fuel element cladding	0.010 in. type 304 stainless steel
Coolant tube	2.563 in. OD by 0.035 wall zirconium
Primary coolant	Sodium
Secondary coolant	Sodium
Thermal shield	Mild steel
Biological shield	Barytes concrete

TABLE II (CONTINUED)

4. Core Nuclear Data

Lattice type	Triangular
Lattice spacing, in.	8.0
Number of cells, total	203
for fuel	185
for control and safety rods	18
Nominal reflector thickness, ft	2
Core diameters (including reflector), ft	14.0
Core height (including reflector), ft	14.0
Core composition, per cent by volume	
Graphite	86.24
Sodium	6.50
Stainless steel	0.43
Thorium-Uranium	4.16
Zirconium	2.25
NaK	0.26
Void (gas)	0.10
U^{235} content (initial loading), kg	360.4
Lattice constants (initial loading with equilibrium poisons) 2% k excess at 600° C	
ϵ	1.000
ρ	0.7936
f (for uranium)	0.6892
η (for uranium)	2.0769
k_{∞}	1.136
L^2 , cm ²	95
τ , cm ²	435
B^2 , cm ⁻²	240×10^{-6}

TABLE II (CONTINUED)

5. Fuel Element Data

Type of fuel element	7-rod cluster
Number of elements in reactor	185
Fuel slug material	Thorium-Uranium alloy
Diameter of thorium slugs, in.	0.648
Length of fuel slug, in.	12.0
Initial fraction U^{235} in fuel, at. %	3.70
Weight of thorium in reactor, kg	9,740
Fuel element length, ft.	10.0
Jacket wall thickness, in. of SS	0.010
Bond, in. of NaK	0.010

6. Control and Safety Rod Data

Number of control rods	10
Type of rods	Boron steel rod operating inside of steel thimble
Diameter of rods, in.	1.8
Total Δk in control rods, %	~9
Number of safety rods	8
Total Δk in safety rods, %	~8

B. COOLING SYSTEM DATA1. Reactor

Coolant	Sodium
Temperature in, °F	500
Mixed mean outlet temperature, °F	925
Highest temperature out, °F	950
Maximum coolant velocity, ft/sec	16.2
Pressure drop across core, psi	16
Flow rate, lb/hr	6.5×10^6
Maximum heat flux, Btu/hr-ft ²	850,000
Maximum fuel temperature, °F	1600
Total power, Mw	250

TABLE II (CONTINUED)2. Primary Loops

Number of parallel circuits	4
Inlet temperature to primary heat exchanger, °F	925
Outlet temperature from primary heat exchanger, °F	500
Flow rate in each circuit, lb/hr	1.625×10^6
Number of pumps in each circuit	2

3. Secondary Loops

Number of parallel circuits	2
Inlet temperature to superheater, °F	895° F
Outlet temperature from economizer, °F	470
Flow rate in each circuit, lb/hr	3.25×10^6
Number of pumps in each circuit	4

TABLE III

NEUTRON ECONOMY OF REACTOR CORE - URANIUM FUEL

Virgin Fast Neutrons	100.00
Fast leakage	5.946
U ²³⁸ resonance captures	22.912
Thermal leakage	1.412
Moderator captures	1.887
Outer coolant tube captures	0.135
Moderator can captures	0.878
Moderator coolant captures	1.370
Control rod captures	1.857
Fuel element captures	63.603
	<u>100.000</u>
Fuel Element Captures	63.603
U ²³⁸ captures	10.866
Xe and Sm captures	2.401
Na coolant captures	1.346
SS cladding captures	1.976
NaK bond captures	0.132
U ²³⁵ fission captures	39.620
U ²³⁵ non-fission captures	7.262
	<u>63.603</u>
Initial conversion ratio = $\frac{22.912 + 10.866}{46.882} = 0.720$	
Second generation neutrons = $39.620 \times 1.026 \times 2.46 = 100.000$	

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TABLE IV
NEUTRON ECONOMY OF A THORIUM URANIUM-235 FUEL CELL

	Beginning of 1st cycle*	End of 1st cycle	Beginning of 2nd cycle	End of 2nd Cycle
Virgin Fast Neutrons	100.000	100.000	100.000	100.000
Fast leakage	8.303	5.955	8.303	6.040
Th ²³² resonance captures	18.930	19.202	18.930	19.185
Thermal leakage	1.354	1.084	1.406	1.106
Moderator captures	1.457	1.646	1.513	1.644
Outer coolant tube captures	0.105	0.118	0.108	0.121
Moderator can captures	0.607	0.632	0.614	0.645
Moderator coolant captures	1.183	1.230	1.196	1.257
Control rod captures	1.535	0	1.787	0.017
Fuel element captures	66.526	70.133	66.143	69.985
	100.000	100.000	100.000	100.000
Fuel Element Captures	66.526	70.133	66.143	69.985
Th ²³²	13.540	16.037	14.268	16.587
Xe and Sm	2.380	2.439	2.382	2.440
Na coolant	0.816	0.958	0.851	0.985
SS cladding	1.611	1.891	1.679	1.945
NaK bond	0.080	0.093	0.083	0.096
U ²³⁵ fission	40.650	29.328	29.692	20.991
U ²³⁵ non-fission	7.449	5.376	5.443	3.848
U ²³³ fission	0	10.966	10.614	19.041
U ²³³ non-fission	0	1.069	1.034	1.857
U ²³⁴	0	0.076	0.067	0.211

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TABLE IV (CONTINUED)

	Beginning of 1st cycle*	End of 1st cycle	Beginning of 2nd cycle	End of 2nd Cycle
Fuel Element Captures (Cont)	66.526	70.133	66.143	69.985
U ²³⁶	0	0.033	0.030	0.056
Pa ²³³	0	0.296	0	0.303
Fission products	0	1.571	0	1.625
	66.526	70.133	66.143	69.985
Conversion ratio	0.675	0.754	0.710	0.782

*A cycle represents 10,000 Mwd/t irradiation

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TABLE V
PRINCIPAL ISOTOPES PRESENT IN A
THORIUM URANIUM-235 FUEL ELEMENT

Cycle*	Isotope - grams					Excess Reactivity - per cent
	U ²³⁵	U ²³³	U ²³⁴	U ²³⁶	Th ²³²	
Beginning of 1st	1946	0	0	0	52600	+2.0
End of 1st	1167	449	18	123	52100	-3.5
Beginning of 2nd	1340	495	18	123	~52600	+2.0
End of 2nd	805	753	49	207	~52000	-3.0
Beginning of 3rd	977	799	49	207	~52600	+2.0

*a cycle represents 10,000 Mwd/t irradiation.

TABLE VI
SUMMARY OF CONDITIONS FOR
VARIOUS TURBINE CYCLES

Basic Data

Plant gross generation - - - - -	80,800 kw (approx)
Plant aux power consumption - - - - -	4,000 kw (assumed)
Plant net generation - - - - -	76,800 kw (approx)
Condenser pressure - - - - -	2 in. Hg abs.
Number of feedwater heaters - - - - -	3
Boiler feed temperature - - - - -	300° F

Heat Balance Assumptions

The following simplifying assumptions have been made:

Auxiliary power consumption - - - - - 5%

Assumed constant for all cases, no allowance has been made for the extra power requirements of the auxiliary combustion heater.

Heat losses in reactor and coolant systems - - 2%

Efficiency of auxiliary combustion heater - - - 75%

No provision has been made for utilization of the hot flue gases except for air preheating.

Boiler feed pump work has been neglected

PERFORMANCE

	Nuclear Station			Nuclear Station With Aux. Combustion Heater	
	800/825	850/900	1250/950	1450/1000	1450/1000/1000
Steam conditions, psig/°F					
Cycle heat rate, Btu/kwh	10,050	9,785	9,465	9,145	8,860
Net plant heat rate, Btu/kwh	10,800	10,500	10,160	10,130	10,100
Net plant thermal efficiency, %	31.6	32.5	33.6	33.7	33.8
Improvement in net heat rate, % (800/825 base)	-	+2.8	+5.9	+6.2	+6.5
Heat supplied by reactor, %	100.0	100.0	100.0	90.0	80.0
Heat supplied by reactor, thermal megawatts	243.0	236.3	228.6	205.2	181.8

Fig. 1. Plant Arrangement

Fig. 2. Reactor Building and Equipment

Fig. 3. Reactor Layout

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Fig. 4. Fuel Cluster

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Fig. 5. Moderator Element

Fig. 6. Schematic Flow Diagram

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Fig. 7. Instrumentation Block Diagram

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Fig. 8. Temperature Difference (Pinchpoint) Diagram
(Steam Conditions: 800 psig/825° F)

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Fig. 9. Temperature Difference (Pinchpoint) Diagram
(Steam Conditions: 850 psig/900° F)

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Fig. 10. Temperature Difference (Pinchpoint) Diagram
(Steam Conditions: 1250 psig/950° F)

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Fig. 11. Temperature Difference (Pinchpoint) Diagram
(Steam Conditions: 1450 psig/1000° F)

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Fig. 12. Temperature Difference (Pinchpoint) Diagram
(Steam Conditions: 1450 psig/100° F/100°F)

PERFORMANCE

GROSS POWER: 80,800 KW
 AUXILIARY POWER (5%): 4,000 KW
 NET POWER: 76,800 KW
 REACTOR HEAT LOSSES: 2 PER CENT
 CYCLE HEAT RATE: 10,050 BTU/KW HR
 NET PLANT HEAT RATE: 10,800 BTU/KW HR
 HEAT SUPPLIED BY REACTOR: 243.0 THERMAL MW

LEGEND

— CONDENSATE & BOILER FEED LINES
 — OTHER WATER LINES
 - - - MAIN STEAM LINES
 - - - OTHER STEAM LINES
 # POUNDS PER HOUR
 h ENTHALPY BTU PER POUND
 °F TEMPERATURE, DEGREES FAHRENHEIT
 TD °F TERMINAL TEMPERATURE DIFFERENCE
 PSIG POUNDS PER SQUARE INCH GAGE
 P POUNDS PER SQUARE INCH ABSOLUTE
 DC DRAIN COOLING SECTION
 "Hg PRESSURE, INCHES MERCURY, ABSOLUTE
 KW KILOWATTS

Fig. 13. Preliminary Heat Balance

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