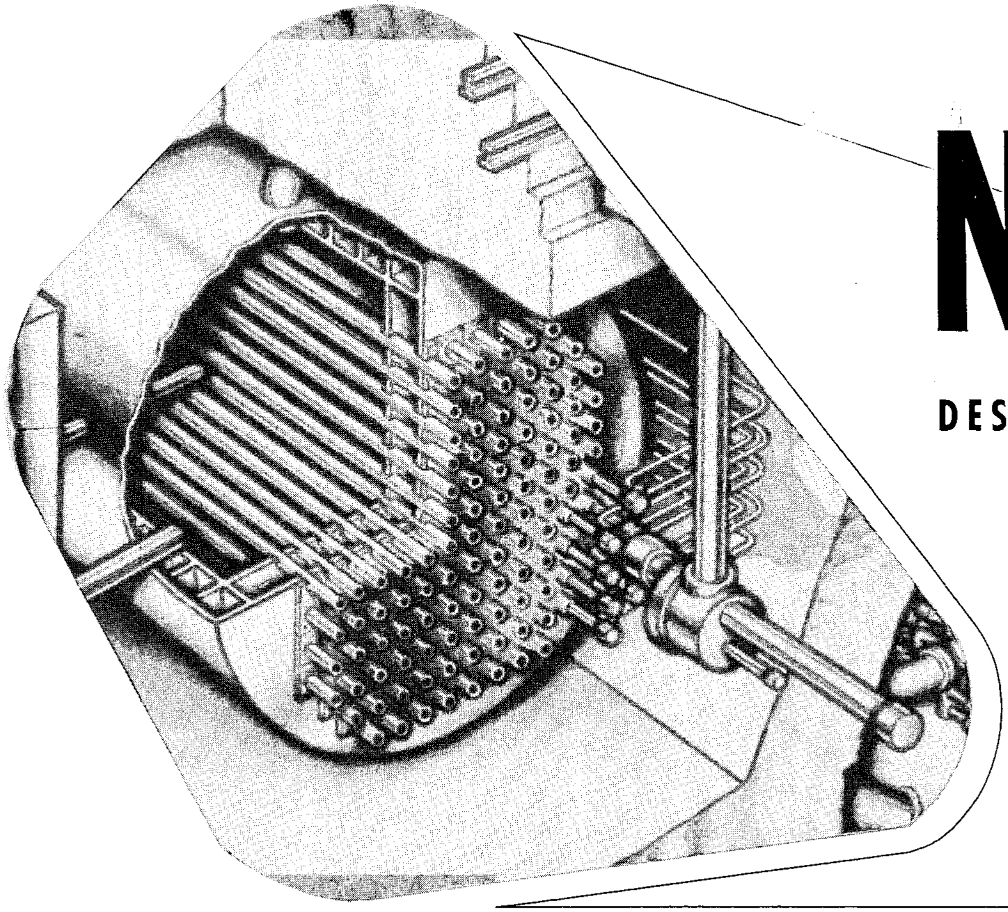
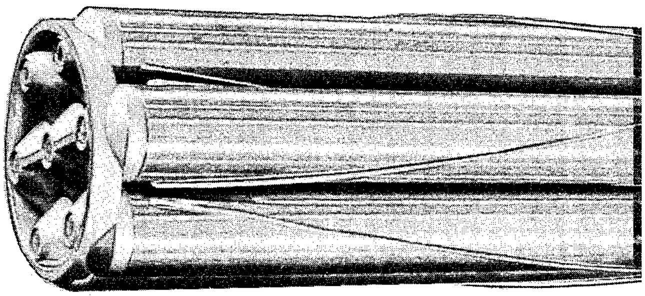


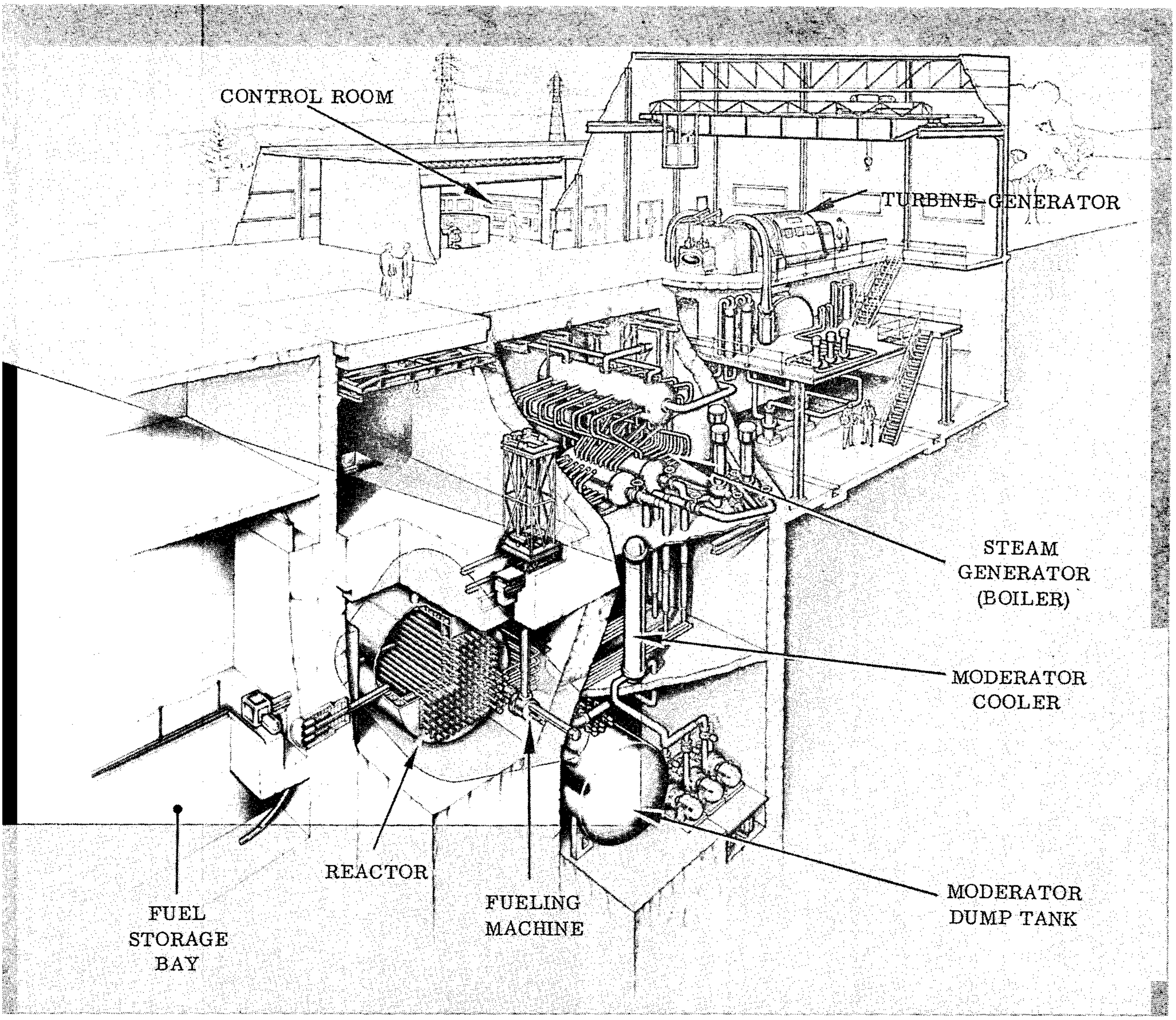
NPD-2

DESIGN DESCRIPTION

CANADA'S FIRST NUCLEAR POWER STATION



A joint project of:
Atomic Energy of Canada Limited
Hydro Electric Power Commission of Ontario
Canadian General Electric Company Limited



NUCLEAR POWER DEMONSTRATION

SUMMARY

Canada's program for the development of economic nuclear power will be very materially aided by the 20,000 KW power station now under construction. The station is a joint project of Atomic Energy of Canada Limited (the Crown-owned company which operates Canada's nuclear energy research and development centre at Chalk River), the Hydro-Electric Power Commission of Ontario, and Canadian General Electric Company Limited.

This nuclear power station, utilizing technology pioneered at Chalk River, has been given the name "Nuclear Power Demonstration, or "NPD." The original design (now referred to as NPD-1) was based on a pressure-vessel type of reactor. The current design, known as NPD-2, features a pressure-tube reactor designed for on-load refueling. Design work on NPD-2 begain in late 1957, and site construction was resumed in mid-1958; the station will be in operation in 1961.

This booklet presents a detailed design description of the NPD-2 station. The station employs several advanced design concepts, and is characterized by a reactor which uses natural uranium oxide as fuel, heavy water near atmospheric pressure as the moderator, and pressurized heavy water as the coolant.

A series of pressure tubes passing through the reactor is used to contain the coolant, as opposed to the use of a large pressure vessel to contain both moderator and coolant. Studies have indicated that the pressure-tube type of reactor is particularly applicable to large heavy-water moderated and cooled power reactors. NPD-2 will, in fact, serve as a prototype for a 200,000-kilowatt station.

The fuel is in the form of short slugs or bundles, each containing seven cylindrical fuel elements composed of uranium oxide sheathed in zirconium alloy tubing. The use of short fuel slugs and an on-load refueling scheme, combined with the simple but important innovation of bi-directional fueling (from opposite ends of adjacent fuel channels), permits the achievement of several significant advantages. For example, a uniformly high burn-up can be obtained from each fuel slug prior to its ejection from the reactor, and the necessary reactivity reserve is relatively small; also, reactor availability will be increased appreciably.

Another distinctive feature of the NPD-2 reactor is the method of control. Power-output control and emergency shut-down control of the reactor are both accomplished entirely through control of the moderator level and to a limited extent of its temperature. No moving control or shut-off rods are required inside the reactor. Moderator level is controlled by a fail-safe helium gas pressure-differential system, and by the force of gravity. Thus, the system possesses a high degree of safety and reliability, particularly in view of the advanced instrumentation scheme using triplicated instruments in dual-coincidence circuits.

The excellent progress of design and development work to date indicates that the NPD-2 station will represent a significant contribution towards the development of safe, reliable nuclear plants for the production of economic nuclear power.

CONTENTS

N P D-2 A DESIGN DESCRIPTION

HISTORY OF THE NPD PROJECT

Canada's interest in the peaceful uses of atomic energy is an outcome of her effort in the last World War. It began when the allied governments selected Chalk River as the site for a pilot reactor which would prove the feasibility of building large heavy-water-moderated, plutonium-production reactors in the event that other methods proved unsuccessful. A zero-energy heavy-water reactor, ZEEP, was erected on the site in 1945, and became the first nuclear reactor to operate outside the U.S.A. This was followed by NRX, which was first operated in 1947 and for many years provided the world's highest thermal neutron flux. In 1957, a larger reactor, NRU, was brought into operation at the Chalk River Project.

All of these research reactors have been of the heavy-water-moderated, natural-uranium type. With many years of experience in construction, operation and maintenance of this type of reactor there came the growing conviction that the heavy-water reactor promises to be the most economical means of deriving energy from natural uranium. The discovery in Canada of some of the largest deposits of uranium in the world encouraged the desire to develop the Canadian type of reactor using natural uranium for power generation.

Early in 1954 a team of engineers and scientists, designated the Nuclear Power Group, was assembled at Chalk River to investigate the feasibility of building natural-uranium, heavy-water power reactors. This team carried out preliminary design studies for a small power reactor which would demonstrate the reliable operation of large power reactors and which would be useful in estimating the economics of similar full-scale power plants. Such a reactor would also provide Canadian experience and training in the design, construction and operation of nuclear power plants. It would also hasten the time when Canada could benefit from this new form of energy and from the industries which would supply its needs.

The results of these studies indicated that a small reactor of this type contained in a pressure vessel was feasible. A pressure-tube design was also considered but rejected on the basis of economics and reliance on new materials.

Atomic Energy of Canada Limited, the Crown company which took over operation of the Chalk River Project in 1952, was formed to do research and development on the peaceful uses of atomic energy. In accordance with this policy A.E.C.L., in 1955, invited leading Canadian industrial engineering companies to submit proposals for the design and construction of a nuclear power station. This station, to be known as the Nuclear Power Demonstration Station (NPD), was to be based on the design study by the Nuclear Power Group.

As a result of these proposals, the Canadian General Electric Co. Ltd. was selected to design and to provide the development for the reactor plant. It was also to contribute $2,000,000 toward the engineering cost of the project, and to serve as prime contractor for the supply and construction of the station.

Proposals for operating the plant were also received from interested public and private utilities. The Hydro-Electric Power Commission of Ontario was selected by Atomic Energy of Canada Ltd. to become the third partner in the project. Ontario Hydro will provide the site, finance and design the conventional plant and station buildings and operate the station as part of its system. The financing of the reactor plant and all reactor development in excess of that contributed by the Canadian General Electric Co. Ltd. will be the responsibility of A.E.C.L., which will own the reactor plant equipment.

Design of the 20,000 kilowatt (electrical) power station using a pressure-vessel reactor concept began in mid-1955. This design is now known as NPD-1.

In the original studies for NPD-1, a pressure-vessel system was selected because of its lower cost for a small reactor. It was also selected because, at that time, there was greater confidence regarding the availability and properties of a steel pressure-vessel than there was regarding the zirconium alloys necessary for the pressure-tube system. By the spring of 1957, detailed design work was well under way and site construction had begun.

In the meantime, during 1956 and 1957, the Nuclear Power Group at Chalk River was working on the preliminary design of a full-scale nuclear station to produce 200 megawatts of electrical power. This design is described in an A.E.C.L. publication entitled "The Canadian Study for a Full-Scale Nuclear Power Plant". The design proposed for this 200 Mw station embodied several advanced concepts. One of the major features was the use of a pressure-tube type of reactor, instead of a pressure-vessel type. Experience in the years since the completion of the preliminary NPD studies had indicated the suitability and availability of zirconium alloys for the pressure-tube design, and this design appeared to be more applicable for large power reactors.

Work on NPD-1 was therefore halted in April 1957 at the request of A.E.C.L. and Ontario Hydro to permit a study of the feasibility of incorporating some of the new concepts of the proposed large power reactor. The study, carried out by the Canadian General Electric Co. Ltd., confirmed that the new design concept is particularly applicable to large units and that it is probable that capacities of several hundreds of megawatts-electrical can be obtained in single units of this type. C.G.E. recommended that the new concept be applied to the Nuclear Power Demonstration Station. The recommendation was accepted.

Contractual arrangements between the three partners remained unchanged and design work based on the new reactor concept was resumed in late 1957. This reactor plant, now known as NPD-2, is scheduled to be in service by the middle of 1961.

GENERAL STATION DESCRIPTION

The NPD-2 Nuclear Power Demonstration Station is a 20,000 kilowatt (electrical) nuclear power station.

It will be erected on a site located on the Ontario side of the Ottawa River, two miles east and downstream from the Hydro Electric Power Commission of Ontario's generating station at Des Joachims and about 16 miles upstream from the Chalk River Project. It is located in sparsely settled country (see Map in Fig. 3).

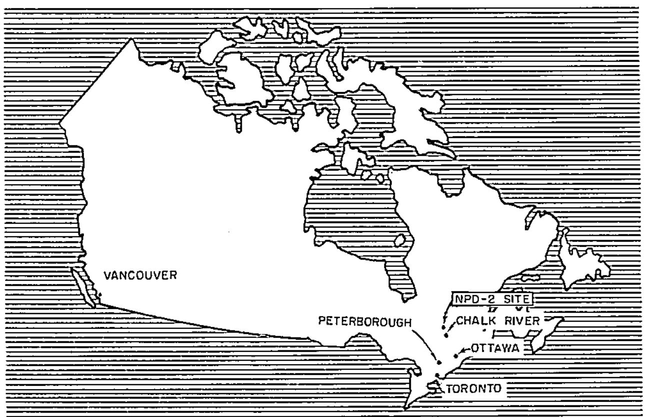


Fig. 3 Map of N P D-2 Location

An artist's conception of the finished plant is shown on the inside back cover. The plant is made up of four sections: the Administration Wing shown at the left, the Control Wing, the Main Section, and the Service Wing. The Main Section contains the reactor-boiler plant and the turbine-generator plant.

The basic station thermal cycle is illustrated by the block diagram in Fig. 4. The reactor-boiler plant consists of the reactor, the steam generator, and the pumps to circulate the heavy-water coolant through the reactor and the boiler. The hot coolant is used to generate non-radioactive steam from ordinary water in the boiler. The turbine-generator plant consists of a single turbine-generator set, a condenser, and the pumps to return the condensate to the boiler in the reactor-boiler plant. The reject condenser, which by-passes the turbine and its condenser, provides an alternate means of loading the reactor for periods when the turbine is not demanding steam.

The station is provided with all the auxiliaries, controls and services necessary for proper operation of the main components.

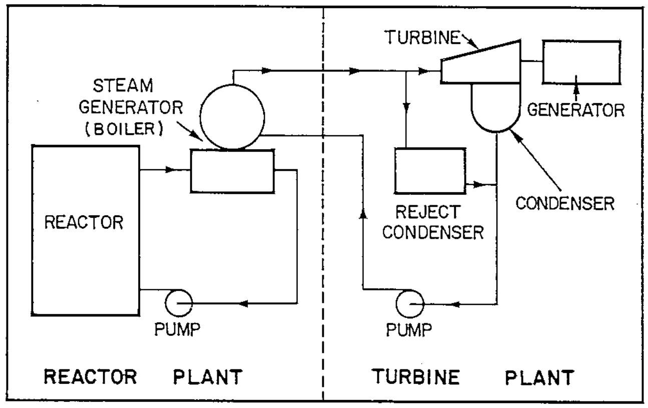


Fig. 4 Basic Station Thermal Cycle

An artist's cutaway of the Main Section of the building is shown on the inside front cover. The main components mentioned above are indicated.

REACTOR

The design of the NPD-2 reactor is as similar in concept as practicable to that of the large nuclear power unit proposed by A.E.C.L. In general, the design embodies the following features:

1. Natural uranium fuel in the form of short lengths of uranium oxide contained in zirconium-alloy sheaths.
2. A pressurized heavy-water coolant.
3. Reactor coolant pressure-tubes of zirconium alloy.
4. An unpressurized heavy-water moderator.
5. A horizontal cylindrical construction.
6. On-power refueling from opposite ends of adjacent fuel channels.

These features will be described in the sections that follow.

FUEL

The fuel rods are the primary heat source within the reactor. The fuel proposed for NPD-2 is natural uranium in the form of sintered oxide (UO_2) pellets contained in zirconium-alloy sheaths.

Two of the main requirements for fuel in a water-cooled reactor are adequate corrosion resistance and dimensional stability. A considerable amount of irradiation experience with UO_2 has been accumulated by Canada, and more work is planned. These tests have shown that UO_2 has excellent resistance to water corrosion, and in addition has adequate dimensional stability under irradiations of up to at least 5000 MWD/tonne (megawatt-days thermal per metric ton of 2205 pounds). The general conclusion is well supported that UO_2 is the most stable form of fuel for natural-uranium reactors. Moreover, as further experience is gained, it seems likely that fuel designs of progressively improved performance and lower cost will be available, so that average operating costs with a UO_2 reactor should be continually reduced.

The function of the sheath material for UO_2 fuel is to provide mechanical protection and to contain fission product gases which are released by the fuel. To perform these functions, good mechanical strength and corrosion resistance at high temperatures are required. At the same time, the sheath should have a low neutron capture cross-section to minimize loss of neutrons. Zircaloy 2 is a proven material which possesses these properties. It is expensive, but the price is dropping with greater demand and supply. The cost of fabrication, although high at present, is also expected to drop as industry masters the required techniques.

In order to achieve a reasonable maximum fuel temperature and surface heat rate, some sub-division of the fuel is necessary. However, the amount of subdivision must be limited in order to minimize cost and resonance capture; the minimizing of resonance capture results in greater safety in the event of coolant loss. Several geometries were considered, including bundles of cylindrical elements with seven, nineteen and thirty-seven elements respectively, and nested annular elements. The choice of cylindrical elements was based on the availability of a large amount of performance data and fabrication experience on such fuel. A seven-element fuel rod was chosen, following optimization studies using a digital computer.

A feature of the NPD-2 design is bi-directional on-power fueling. This innovation uses relatively short slugs of fuel in order to obtain a symmetrical neutron flux distribution across the reactor and a uniformly high burnup from each fuel slug. The fuel slugs are pushed through the reactor a step at a time, and are fed in opposite directions in adjacent fuel channels. The slug length was chosen to suit the computed reactor core size and a practical size of fueling machine. A short slug length also permits a partial cooling-off period in the low-power reflector region at the outlet end of each channel. There are nine slugs in each fuel channel, each slug being 19 1/2" (50 cm) in length.

Figure 5 is a photo and Figure 6 is an outline of such a fuel rod.

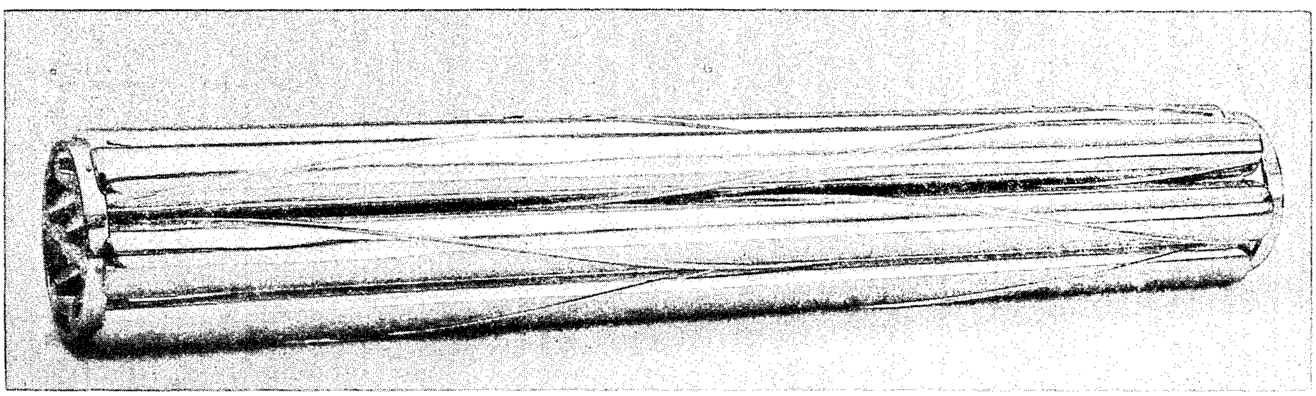


Fig. 5 N P D-2 Fuel Slug

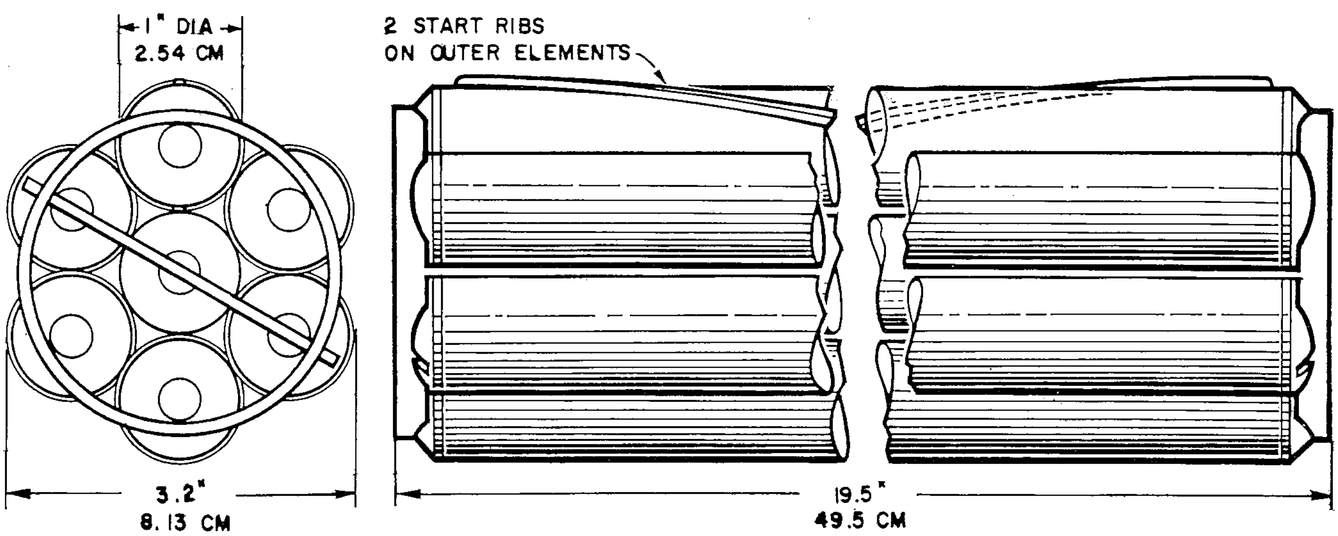


Fig. 6 7 Element Fuel Slug

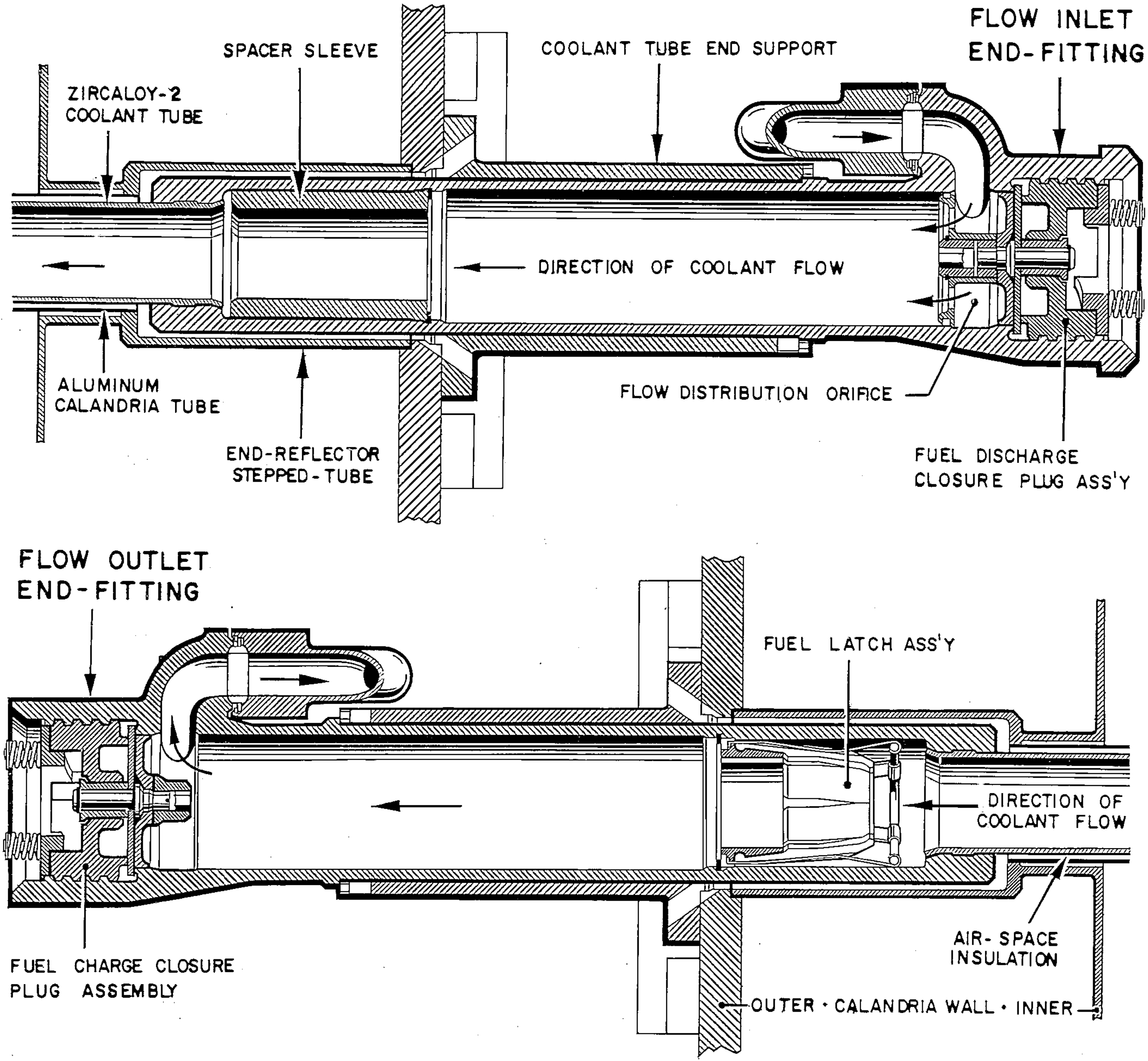


Fig. 7 Coolant Tube Assembly

Tests to confirm these design choices are well under way. Experiments at C.G.E.'s Peterborough Works include measurements of hydraulic and heat transfer properties. In-pile trials in the NRX and NRU reactors at Chalk River will test the stability of the fuel at the desired power output.

COOLANT TUBE ASSEMBLIES

The fuel rods are located inside a pressure-tube assembly which contains the heavy-water coolant. The coolant is circulated through these tubes carrying the useful heat out of the reactor to the steam generator.

Figure 7 shows a coolant tube assembly as it is installed in the reactor. There are one hundred and thirty-two such assemblies, each comprised of a Zircaloy 2 coolant tube attached at each end to a stainless steel end-fitting by an expanded joint.

The zirconium alloy Zircaloy 2 was selected for the coolant tube material, and for the fuel element cladding on the basis of its strength, low neutron absorption cross-section, and its excellent corrosion resistance in high temperature water. The only other materials which could be considered from a reactivity standpoint are alloys of aluminum or magnesium. At the NPD-2 operating temperature the mechanical properties and corrosion resistance of both these materials are considerably lower than those of Zircaloy 2.

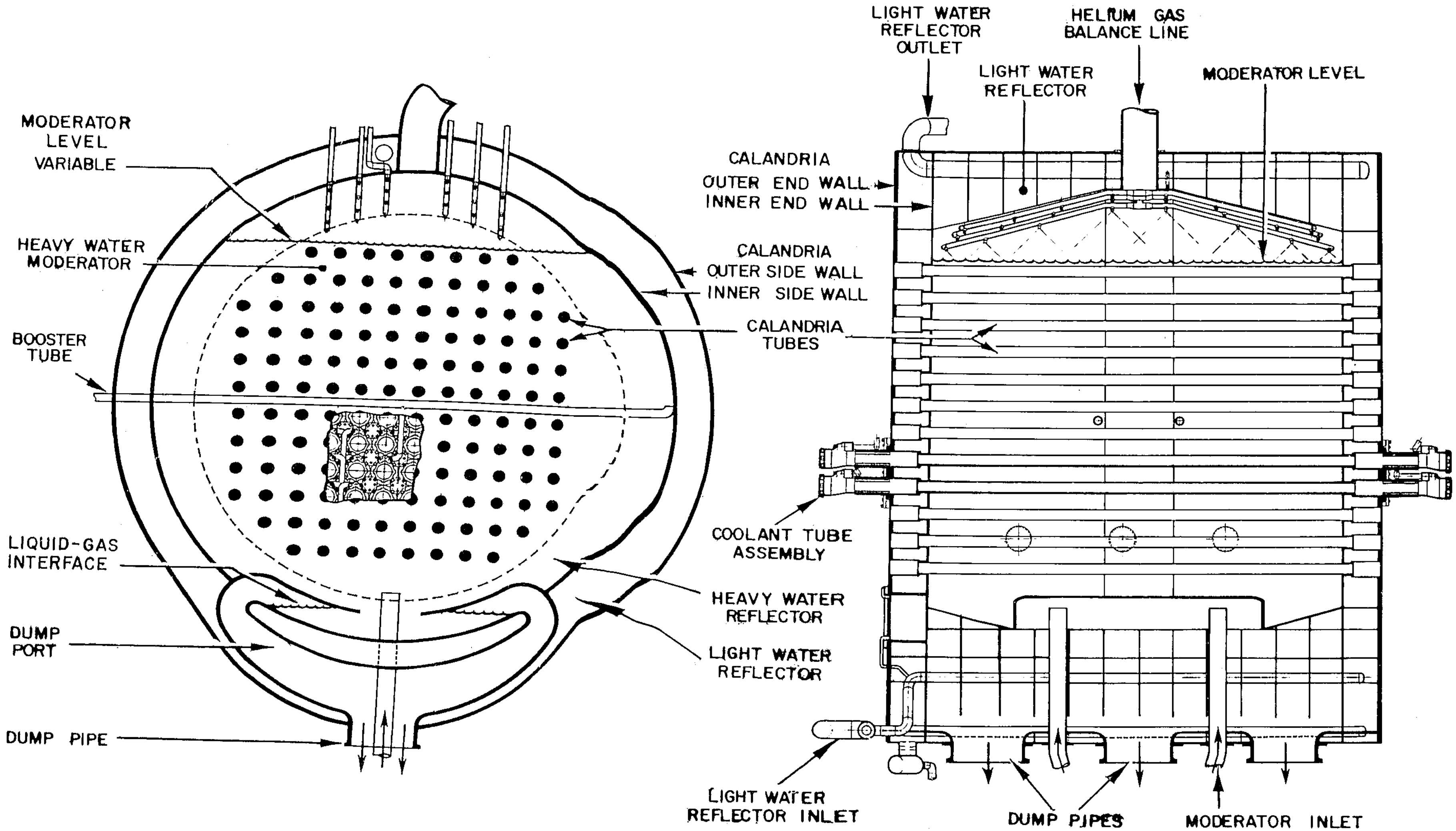


Fig. 8 N P D-2 Reactor Arrangement

Although the end-fittings are almost identical at each end of the coolant tube, the internal fittings of the flow inlet end differ from the flow outlet end. Both end-fittings are equipped with high-pressure closures of the flexible diaphragm type (see Fig. 7) which can be removed for insertion or removal of fuel. Both have sealing faces and attachment lugs for the fueling machines. The flow inlet end-fitting, which also forms the fuel discharge end of the coolant tube assembly, is equipped with a flow distribution orifice. This orifice, which controls the coolant flow into the tube, is attached to the closure plug and is removed with it. Removal of the orifice with the plug provides extra pressure in the coolant channel to overcome the extra flow resistance introduced by the fueling operation; thus, the coolant flow is maintained at an adequate rate at all times.

The difference between the flow end-fittings occurs in the inner space at each end. The flow inlet end-fitting contains a spacer sleeve, while the flow outlet (fuel inlet) end-fitting contains a fuel latch. The fuel latch consists of six spring-loaded fingers which permit fuel slugs to pass into the coolant tube zone of the assembly but act as stops to fuel movement in the reverse direction.

Each coolant tube assembly is held in place in the reactor structure by two end supports which are mounted in the outer end walls of the calandria vessel.

Replaceable coolant tubes is a design feature of the reactor, and development of the special tools required for this operation is part of the NPD project.

CALANDRIA

The calandria is a horizontal cylindrical aluminum alloy vessel with double side and end walls. It contains the heavy-water moderator and reflector within the inner walls, and the light-water reflector and shield between the inner and outer walls. (See Fig. 8) The metal thicknesses and approximate overall dimensions of the calandria are given with the Design Data Summary.

At each end of the calandria, one hundred and thirty-two tubes 1/4-inch (6 mm) thick are welded to both the inner and outer walls, and span the one-foot space between them. These tubes are step-reduced with the smaller diameter at the inner ends. (See Fig. 7)

Spanning the length of the calandria between the inner end walls are one hundred and thirty-two 4-inch (10 cm) inside diameter, 0.052-inch-thick (1.3 mm) aluminum calandria tubes. The ends are expanded into rings which are integral with the stepped tubes.

The net result of the calandria tube and end reflector stepped-tube installation is the provision of one hundred and thirty-two horizontal holes completely through the calandria. The coolant tubes are located within these holes and are centralized within the calandria tubes by internal annular spacers at the centre of each tube. An air space is provided between the coolant tube and the calandria tube to insulate the moderator from the hot coolant.

The calandria is provided with dump ports which provide a minimum area of 20 square feet (1.9 $m.^2$) for rapid removal of heavy water from the calandria via three 24-inch (61 cm) pipes to a dump tank. The port arrangement also provides a horizontal gas-liquid interface which normally supports the heavy water within the calandria by means of a differential gas pressure system, which will be described in detail with the moderator and helium system.

An internal piping manifold at the top of the reactor will supply heavy water to a system of spray nozzles whose function is to cool any calandria tubes which are not submerged in the heavy water moderator, either during operation of the reactor or during reactor shut-down periods, when the main heat transport system may still be hot.

The fully-loaded reactor weighs approximately 200 tons (181 tonnes), and is supported by hanging the calandria structure from four 3 1/2 inch (8.9 cm) diameter, 13 foot (4 m.) long spherically ended steel rods, which in turn are carried from four steel columns anchored to the walls and floor of the reactor vault. The weight of the coolant tube assemblies and their contained load of fuel and heavy-water coolant is also carried by these rod supports via the heavy end plates of the calandria structure.

CORE

The fuel rods, moderator, coolant tubes and the heavy water within them are called the reactor core. The theoretical core is a cylinder 11 feet (335 cm) in diameter and 12 feet 7 inches (384 cm) long. The actual core consists of 132 cells formed by the fuel, calandria tubes and coolant tube assemblies. These cells have a total area of 105 $in.^2$ (678 cm^2) and are spaced on a square lattice having a pitch of 10 1/4 inches (26 cm). They are arranged in an irregular octagon having alternately eight and three cells to a side, the eight-cell sides being horizontal and vertical.

REFLECTOR

The theoretical cylinder containing the one hundred and thirty-two calandria tubes is encircled by an annular heavy-water reflector which has a radial thickness of 21.6 inches (55 cm) at the mid-point and which tapers off to a 5.6 inch (14 cm) radial thickness at the ends. (See Fig. 8) The heavy-water side reflector is not separated from the heavy-water moderator. Surrounding the heavy-water side reflector is a minimum of one foot of light water which serves as both neutron reflector and shield.

FUEL HANDLING

Since the fuel rods for NPD-2 do not require the attachment of shielding plugs, no on-site operations are required on new fuel, other than a simple inspection (for transit damage) and storage in racks until required for use in the reactor.

FUELING MACHINES

Refueling is carried out, with the reactor operating at full power, by means of remotely-operated fueling machines. (See Fig. 9) One machine is operated at each end of the reactor, and the fuel movement is in opposite directions in adjacent fuel channels. The machines are located in a room directly above the reactor vault and separated from it by full-power shielding. Through this shielding floor two slots are provided above the end faces of the reactor through which the two fueling machine heads may be lowered on telescopic tubes to reach the coolant tube ends. When not engaged in refueling, the machines are retracted and shielding gates close the slots. In this position, all parts of the machine are directly accessible for servicing while the reactor is in operation.

For a typical refueling cycle the magazine of machine "A" will be loaded in the fueling machine room with the requirement of new fuel for that day. For normal operation two slugs will be replaced each day. The room will be locked, the ventilation shut off, and the shielding gates opened. Both machines will then be lowered through the slots and homed onto opposite ends of the same fuel channel. The machines will lock on the reactor coolant tube end fittings forming a high pressure seal between the machines and the tube end fittings. The coolant tube end plugs will be removed and stored in the magazines of the machines. Machine "A" will then charge a fuel slug from its magazine into the reactor. This slug will push the spent slug at the far end of the channel into a positive grappling mechanism on machine "B". Machine "B" will then retract this spent slug into its magazine where it will be held until ready for discharge. The machines will replace the end plugs and disengage from the coolant tube ends. This cycle will then be repeated until the required number of channels have been refueled.

On completion of refueling, machine "B" will home onto a fuel discharge chute. This chute runs from the reactor vault through the shielding wall, and ends as a trough on the bottom of the spent fuel storage bay, under 14'-6" (4.4 m.) of water. The chute inlet is above the bay water level. The spent fuel will discharge from the magazine of machine "B" into the chute, slide down it and come to rest in the trough section on the bay floor. The fueling machines will then be returned to their positions in the fueling machine room and the shielding gates closed.

At some convenient time an operator will go to the bay room and, with hand tools, transfer the spent fuel slugs from the troughs to partitioned wire baskets. Each basket will hold 36 slugs. The fuel will remain in the bay for 180 days to permit decay of some of its radioactivity. After this storage period, a basket of fuel will be loaded into a cast iron shipping flask and shipped to the processing or final storage point.

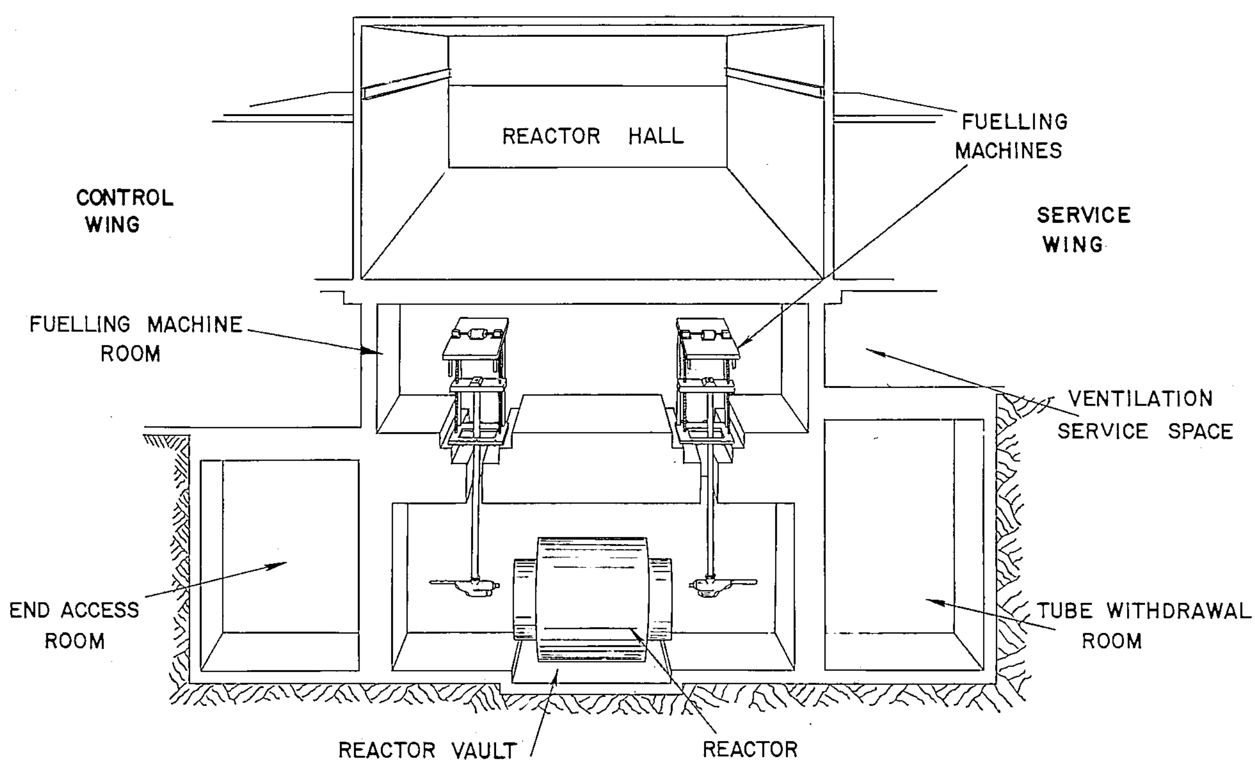


Fig. 9 Section of Building showing Fuelling Machines

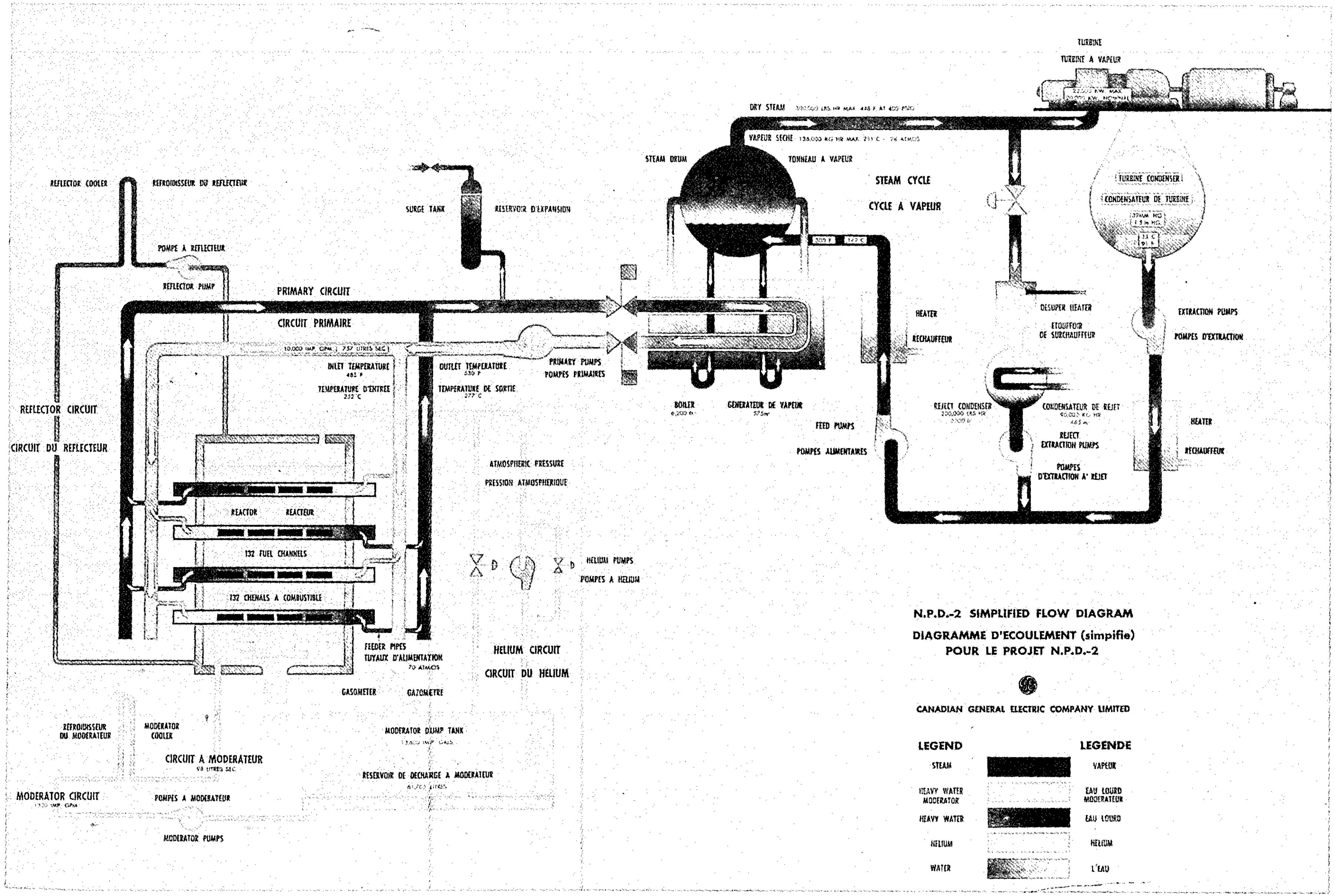


Fig. 10 Overall Fluid Circulation Diagram

The reactor fueling machines will ordinarily be controlled remotely from a central control board. The control will be arranged so that only complete operations need be initiated by the operator, and sequences of actions within an operation will be controlled automatically. The operator, for instance, will call for a machine to engage a particular coolant tube end fitting, to carry out a single fueling operation, to engage the fuel discharge chute, to return to the fueling machine room or to engage the fuel-loading port. All motions of the machine and its components are supervised, and complete information regarding the status of the machine is presented at the control board. An auxiliary control station will be provided, local to the fueling machine room, to facilitate service checks and adjustments of the machine and its sequence controls.

REACTOR PROCESS SYSTEMS

The overall fluid circulation diagram is shown in Figure 10. There are three main reactor process systems: the Primary Heat Transport System, the Moderator and Helium System, and the Reflector System.

PRIMARY HEAT TRANSPORT SYSTEM

The primary heat transport system consists of a boiler (made up of a steam drum and heat exchanger), pumps, valves, heavy-water coolant and the necessary auxiliaries. The coolant is circulated through the reactor coolant tubes by the pumps, carrying the heat produced in the fuel to the boiler, where ordinary water is converted to steam to drive the turbine-generator set.

The primary system delivers 2.8×10^8 BTU/hr. (7.05×10^7 Kcal/hr) to the boiler to produce the maximum of 22 MW (electrical) in the turbine-generator with a thermal-electrical conversion efficiency of 27%. The fuel surface temperature sets an upper limit on the coolant temperatures and thus on the steam conditions in the turbine-generator plant. The maximum sheath temperature of the fuel is 548° F (287° C). Heavy-water coolant circulating through the reactor enters at 485° F (252° C) and emerges at 530° F (277° C), bulk temperature. In the boiler, the coolant transfers its heat to ordinary water forming steam at 410 psig (28 atmospheres). The steam will not be superheated but will go directly to the turbine-generator.

Boiler. A "U-shell-and-tube" heat exchanger and a steam drum constitute the boiler. (See Figs. 11 and 12). The two are connected by risers and downcomers. Feedwater enters the steam drum and mixes with the circulating boiler water. Steam generated in the heat exchanger section promotes natural circulation up the risers and down the downcomers. The steam drum contains cyclone separators which remove moisture from the steam before it leaves the boiler.

The boiler is located at an elevation higher than the reactor to maintain natural convection circulation of the reactor coolant by thermal syphoning. With the main pumps stopped this is capable of removing about 2 MW of heat from the reactor.

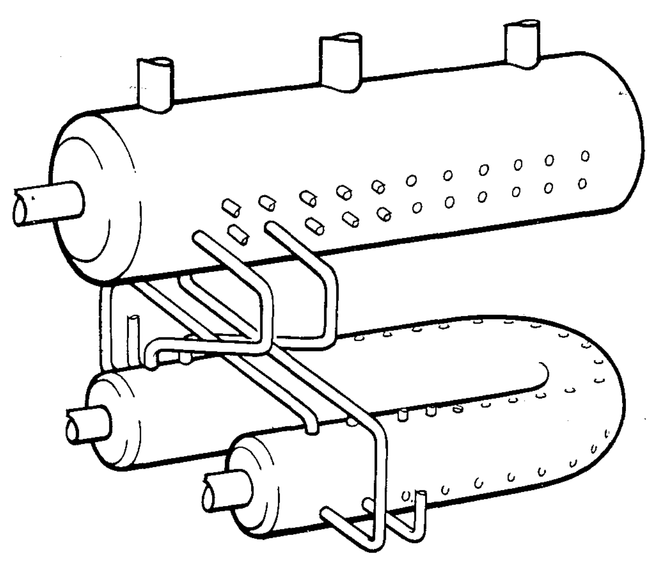


Fig. 11 "V-Shell-and-Tube" Boiler

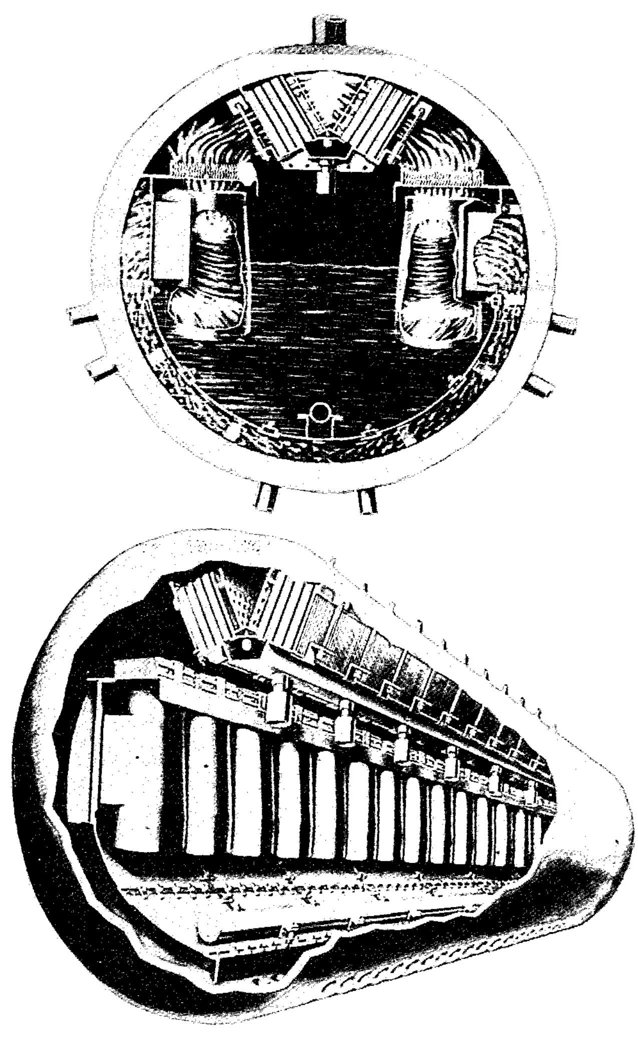


Fig. 12 Boiler Steam Drum

(Courtesy of Babcock-Wilcox and Goldie McCulloch Ltd.)

Primary Pumps. The vertical, single stage, centrifugal pumps will employ a single mechanical shaft seal backed up by a labyrinth gland and will have provision for collecting leakage entering the space between the seal and gland. The canned rotor or hermetically sealed type of pump is considerably costlier both initially and in terms of power consumption. Experimental operation of a shaft-sealed pump under conditions similar to those of NPD-2 indicates that satisfactory operation can be expected. The required total flow of 10,000 Igpm (755 litres/sec.) will be provided by two operating pumps with a third identical pump installed as a standby unit. Emergency short-term capacity for coolant circulation will be provided by means of flywheels. If the power supply to the main pumps fails, the flywheels provide enough energy to turn the pumps until adequate cooling can be obtained by natural convection circulation.

Piping. Each coolant tube within the reactor is coupled to an inlet feeder pipe at one end and an outlet feeder pipe at the other. Feeder pipes from both ends of the reactor are brought to the wall of the reactor vault, penetrate the wall through a labyrinth shield, and emerge in the header room, where the inlet and outlet feeders are routed to connect to the corresponding vertical inlet and outlet headers. The inlet headers are fed by the primary pumps. Coolant from the outlet headers is carried to the boiler. All joints between the steel piping and steel fittings are welded, except feeder pipe connections to the end fittings of the coolant tubes.

Valves. Three main isolating valves are provided in the system. Two are located on the reactor inlet and outlet connections and are provided with motor operators. The third is a manually-operated valve located between the pumps and the boiler. The manually-operated valve will be used in conjunction with one of the other two isolating valves to isolate and drain the pumps or the boiler when repairs are required. In addition to these, check valves are provided on the discharge line from each of the main pumps to prevent excessive backflow through the idle standby pump.

Surge Tank. To accommodate small changes in the liquid volume in the primary system a surge tank is provided. A liquid level control governs the admission or discharge of heavy water to the tank. The space above the liquid in the surge tank is filled with vapour generated by an electric heater. Control of the heater current regulates the pressure in the primary system.

Standby Cooling. To permit maintenance of the main pumps, the steam generator, or the main piping during a reactor shut-down, a separate standby cooling circuit is provided having a capacity of 1% of full reactor power.

Demineralizer. A small flow will be diverted to the primary demineralizer circuit where it will pass through filters and resin beds for purification. The heavy water will first be reduced to a temperature of 120° F (49° C) and atmospheric pressure, in order to minimize capital costs and operating costs.

Fuel Channel Monitoring. To safeguard the reactor and to identify faulty fuel, the following conditions will be monitored for each fuel channel: coolant flow entering the channel, coolant temperature leaving the channel and coolant activity leaving the channel. Any changes beyond the normally-allowed limits will initiate an alarm.

MODERATOR AND HELIUM SYSTEM

The major components associated with the moderator and helium system (see Fig. 10) are listed below:

Moderator Circuit	Helium Circuit
Calandria	Calandria
Dump tank	Control valves
Cooler	Dump valves
Purification system	Dump tank

The calandria has been described previously. The other components are all located externally to the reactor vault. The system will be operated at approximately atmospheric pressure.

The moderator and helium system controls reactor operation through variation of the moderator level in the calandria. The method of control is explained in more detail with the Control System. Briefly, level control is achieved by pumping heavy water from the dump tank into the calandria, and by holding it at the required level by means of a differential gas pressure (helium) system. The heavy water circulates continuously, draining back to the dump tank through the dump ports at the bottom of the calandria.

The helium system provides the atmosphere above the heavy water in the tanks in this system, and the helium piping provides the vent connections between the tanks. The helium is maintained at a pressure slightly above atmospheric within the dump tank. Two helium blowers with bypass control valves maintain the gas pressure differential required to support the necessary height of heavy water in the calandria.

Six large quick-opening dump valves are provided to equalize the helium pressures above and below the heavy water in the calandria, when a signal from the protective system calls for a reactor trip (shut-down). The heavy water then flows by gravity through the large dump ports into the dump tank, quickly shutting down the reactor. The dump tank is located below the calandria in the dump tank room. The capacity of the tank is 13,600 Imp. gallons (62,000 litres), so that it can receive and store the full calandria heavy-water content.

A 1500 cu. ft. (42.5 cu. m.) gasholder will accommodate gas volume changes with temperature and barometric pressure and provide a reasonable reserve to allow for sampling and minor leaks.

The moderator circuit has provision for cooling and purification of the heavy-water moderator. Heat is generated in the moderator by radiation absorption and by neutron collision, and some heat is also transferred into it from the primary heat-transport system. This heat is removed by the moderator cooler, which discharges the heat to river cooling water. The total cooling load imposed on the moderator cooler is equivalent to 7.6 per cent of the useful reactor thermal output of 81 MW. With the reactor in full power operation, the moderator temperature can be controlled at levels ranging between 120° F (49° C) and 180° F (82° C). Circulation of the heavy water is provided by the moderator pumps, which are of the shaft-sealed type. During normal operation, two pumps supply the required flow of 1300 Igpm (98 litres/sec.) with a third pump on standby.

At times when the reactor is operating with the calandria partly full or when the reactor is shut down, heat in those calandria tubes not immersed in the heavy-water moderator would cause distorting stresses in the calandria structure. To dissipate this heat, part of the moderator returning to the calandria is discharged to spray piping in the upper area of the calandria to spray-cool the calandria tubes in the free-board space.

Piping material in the moderator system is aluminum. Joints and closures will be welded insofar as is practicable.

To maintain the chemical purity of the heavy-water moderator, a purification system employing filters and mixed-bed resins is employed.

REFLECTOR SYSTEM

The light-water reflector located in the space between the inner and outer calandria walls is provided with a circuit, external to the reactor vault, for filling, emptying and purifying the water, and for circulating the water for cooling and temperature control. The equipment for this purpose is located in the active auxiliaries room.

CONTROL SYSTEM

The control system for the station includes the reactor-boiler control system, the reactor fueling control, turbine-generator control, electrical control, radiation monitoring, and miscellaneous instrumentation and local controls such as those for the water treatment plant. This section will describe the reactor-boiler control and its associated instrumentation and operating procedures. The other control systems are described with the appropriate sections.

The reactor-boiler control system can be divided into two major sections according to function, the Regulating System and the Safety System. The former provides an automatic means for starting the reactor, for controlling reactor-boiler conditions at the desired operating power level, and for shutting down the unit on a scheduled program. The safety system independently supervises various reactor conditions and, should any of the critical conditions leave the range of acceptable values, this system will automatically and rapidly shut down the reactor.

THE REGULATING SYSTEM

The regulating system (see Fig. 13) is designed to maintain constant steam pressure at the turbine throttle by controlling reactor output, through variation of the moderator level. Moderator level, as described with the Moderator and Helium System, is varied by controlling the differential gas pressure that supports the moderator; this is accomplished through the use of control valves which bypass the helium blowers.

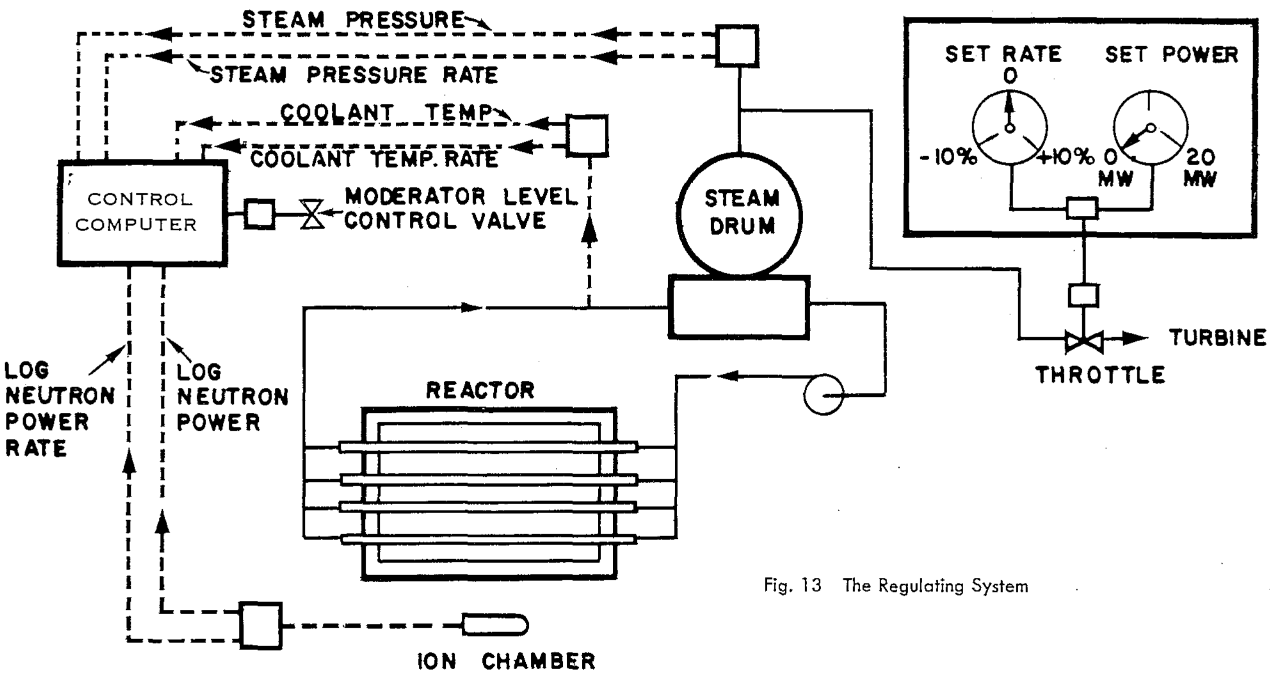


Fig. 13 The Regulating System

Although the regulating control loop responds primarily to deviations in steam main pressure, other signals exercise limits on the response of the control loop. These signals represent neutron flux, reactor period, coolant temperature and rate of change of coolant temperature. (Neutron flux varies over a wide range of values and can best be measured on a logarithmic or log N scale. The reactor period is the rate of change of log N power.) If, for instance, the steam pressure signal, as the result of a break in the steam main, were to call for increased power to restore the pressure, the neutron flux signal would impose a ceiling that would prevent the control system from coercing the reactor to produce more than normal full power.

Moderator level control provides rapid regulation of reactor output. In addition, there are two other methods for more gradual adjustment of reactivity. The first of these is variation of moderator temperature between normal limits of 120° F (49° C) and 180° F (82° C), to provide a reactivity adjustment of approximately 7 mk.* This may be used for poison override and, if necessary, for reactivity adjustment to suit a changing fuel composition.

The second method for gradual adjustment of reactivity is the use of the booster. This is an enriched fuel rod which can be inserted on the horizontal diameter at a rate of one foot per minute, to provide a designed reactivity boost, when fully inserted, of 2.5 mk. The booster can be used for short-term poison override and will extend the override time for NPD-2 by about fifteen minutes. Although only one booster is presently intended, sites for two boosters are being provided.

The rates of reactivity variation for the different control methods are as follows:

1. Moderator level:

   The net flow of moderator into the calandria is limited by the capacity of the helium blowers. The maximum rate of reactivity increase, pertaining to the most reactive core conditions, is 0.29 mk/sec. With the core at equilibrium (after many months of operation), the rate of reactivity change available for control is approximately ± 0.04 mk/sec.; this rate applies for average operating conditions, with the calandria almost filled with heavy water. Fig. 14 shows the relationship between reactivity change and percent moderator volume.

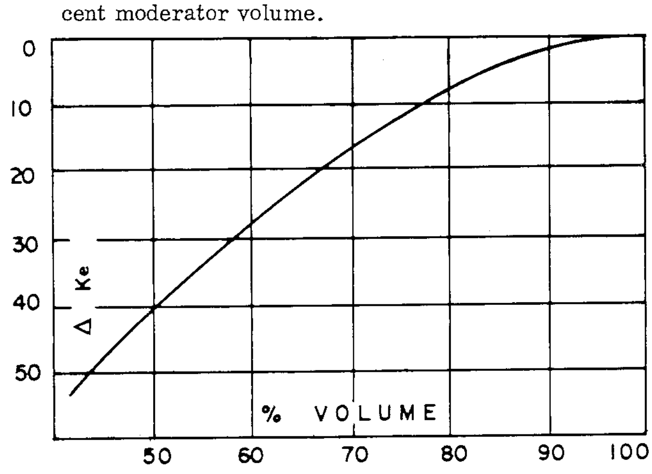


Fig. 14 Reactivity Change and Volume Relationship

2. Moderator temperature:

   The maximum positive reactivity rate that can be effected by moderator cooling with a full calandria is .015 mk/sec. This corresponds to full cooling with the moderator at its maximum temperature of 180° F (82° C) and river water at 40° F (4.5° C).

3. Booster:

   The booster has a total effect of 2.5 mk and can be inserted at a rate of one foot per minute, corresponding to a maximum reactivity rate of approximately 0.01 mk/sec.

SAFETY SYSTEM

The safety system supervises certain of the critical system variables, among which are the neutron flux, reactor period, coolant temperature, and rate of change of coolant temperature. When any of these variables exceeds a set limit, the supervisory system reacts automatically to shut down the reactor as quickly as possible. This is accomplished by opening the six large helium dump valves, thus removing the pressure differential that supports the moderator in the calandria; the heavy water then dumps rapidly by gravity into the dump tank.

The rate of reactor shut-down by dumping is at least 3 mk in the first second. Drainage of the heavy water is completed in about 15 seconds. Figure 15 shows the relationship between reactivity and time during dump. Figure 16 shows the way that power is reduced following a dump.

INSTRUMENTATION

Instruments and components are applied, insofar as possible, in a fail-safe manner. For example, failure of the heavy-water pumps, helium blowers, or power to the valves, results in the reactor shutting down.

Critical instruments are triplicated and connected in a coincidence circuit. In this circuit, a fault signal from a single instrument annunciates an alarm only. Fault signals from a pair of instruments are required to trip the reactor. Triplication of instruments and the coincidence requirement for tripping has the following advantages:

1. It has a much higher probability of correctly indicating a fault condition than does a single instrument.

2. It reduces considerably the probability of spurious trips due to instrument failure.

3. It makes it possible to test a control channel frequently and to service components during station operation while maintaining full protection.

Both the safety system and the power control loop in the regulating system are triplicated. For example, signals from the three power control loops are compared and averaged. Any loop in serious disagreement with the average is disconnected from its control function and the

* The term "k" is used to describe the reactivity (or neutron multiplication factor) of a nuclear reactor. The term "mk" (milli-k) refers to a change of 0.1% in "k".

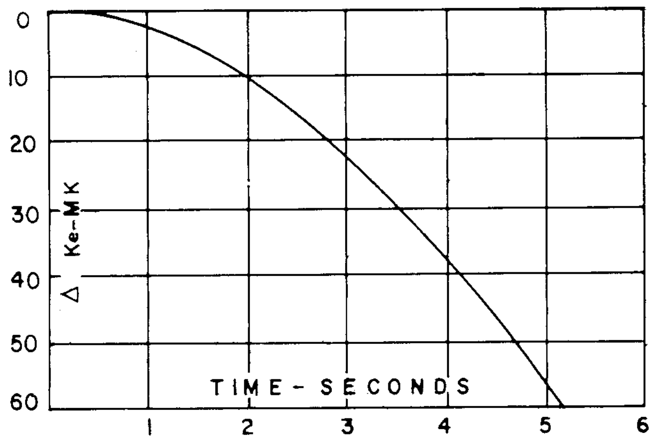


Fig. 15 Reduction of Reactivity following Moderator Dump

fault is annunciated. Disagreement between the three results in an automatic shut-down.

All major operating decisions are based on information derived from indicators, recorders and annunciators which are located in the Control Room. Floor-mounted control panels and a benchboard operating console contain all the necessary instrumentation. Both the reactor plant and turbine-generator plant are controlled from the one location.

START UP PROGRAM

Moderator level control is used for start-up as well as shut-down of the reactor. Start-up can be initiated only by deliberate action of the operator.

Before start-up can take place, certain conditions must exist. These conditions are as follows:

1. The calandria is empty.
2. The helium dump valves are open.
3. Flow is being diverted from the calandria to the dump tank by the moderator diversion valve.
4. The moderator is being cooled at the maximum rate.
5. The level control valves are open.
6. The regulating circuit recognizes low log N demand instead of steam pressure demand.
7. The tripping functions that caused the previous dump have been cleared.

The reactor is first brought to a low power level. To accomplish this, the following sequence of events is followed:

1. The dump valves are closed by deliberate action of the operator.
2. Tests of the operation of certain specified protective instruments are carried out by simulating the fault conditions and checking for operation of the dump valves.
3. As an additional safety feature, these tests must be carried out within a preset time.

If these tests are performed satisfactorily within the specified time, the operator may start the reactor with a single action. This results in closing the moderator diversion valve and connecting the regulator control valves into the regulating circuits. During this start-up procedure, the increase of neutron power is governed by the preset limit period and by the demanded log N value.

It is intended that the reactor be held at the low-power operation, two or three decades down from full power, at all times when power output from the turbine-generator is not required. This has the advantage of maintaining the reactor in a known controlled state at all times.

It is quite possible that high-power operation may not be required until many hours or days after the reactor was raised to low-power operation. Therefore, in order to guarantee that the rise in power will be safely carried out, tests of critical protective equipment, such as the period instruments, are performed. When proof of the satisfactory completion of these tests is obtained, and if they have been completed within a specified time, the operator can transfer the control from log N to steam pressure. During the increase in power, the rate of increase is under the control of several variables which are not permitted to exceed preset limits. The increase in power proceeds automatically until the control system establishes demanded pressure in the steam main.

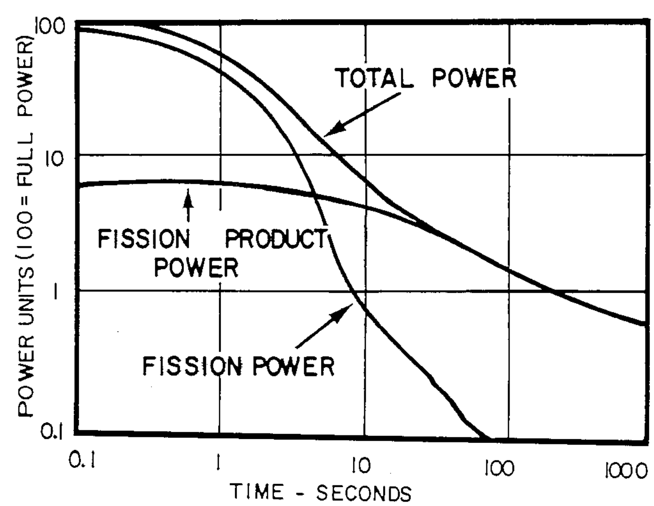


Fig. 16 Reduction of Reactor Power following Moderator Dump

EMERGENCY SHUT-DOWN

When any of the critical variables exceeds a set limit, the protective system rapidly shuts down the reactor. The following sequence of events occurs:

1. All six helium dump valves are opened, permitting the moderator to fall into the dump tank.

2. The moderator diversion valve cuts off the inflow of moderator to the calandria.

3. The level control (regulating) valves are opened fully.

4. Steady-state control is transferred from constant steam pressure to hold a low log N power.

5. The moderator cooling is increased to a maximum to reduce moderator temperature as quickly as possible.

6. The origin and results of the fast trip are annunciated.

7. The times at which all devices in the protective circuit operated are recorded.

8. The heat-transport system pressures and temperatures are automatically controlled at operating levels to facilitate rapid start-up.

SCHEDULED SHUT-DOWN

Periodic shut-down will take place to allow maintenance and inspection of inaccessible areas. The following sequence is followed:

1. Power output from the generator and the reactor is reduced gradually by closing the turbine throttle valve.

2. When the power output has been reduced to a sufficiently low value, the operator switches the reactor control to low log N power. This reduces the primary coolant activity to a low level.

3. At the same time, the generator breaker is tripped and steam flow to the turbine is cut off.

4. The reject condenser is automatically loaded to remove fission-product heat from the reactor and to maintain operating steam pressure. If lower temperatures are desired, the plant can be cooled by manually controlling the reject condenser. At 1% full power or less, use of the standby cooling circuit permits isolation of the large primary circuit components.

AUXILIARY SYSTEMS

WASTE DISPOSAL SYSTEM

The wastes from NPD-2 have been grouped according to whether they are gaseous, solid or liquid. Disposal of gaseous wastes is described in the following section (Ventilation System).

Solid wastes from NPD-2 may consist of spent fuel, damaged equipment parts, or other material. All material from the active section of the station will be suitably wrapped or boxed to confine real or potential contamination, and will be sent to the A.E.C.L. disposal area. Inactive items may be disposed of by normal methods.

Liquid wastes from NPD-2 will be handled in a manner similar to that used for other reactor installations. Liquid wastes known to be inactive will be discharged to the river. Domestic sewage will be treated and ultimately discharged to the river. Wastes which may be active will be monitored. If the level of activity is sufficiently low, they will be mixed with the condenser outfall and the discharge to the river will be maintained at better than drinking tolerance. A recording radiation meter will provide a record of the activity released with the liquid effluent from NPD-2 to the Ottawa River. Liquid wastes too active for this method of disposal will be collected in storage tanks, which will be periodically emptied into a tank truck for transportation to the A.E.C.L. disposal area. The flow diagram of the drainage system for NPD-2 is shown in Fig. 17.

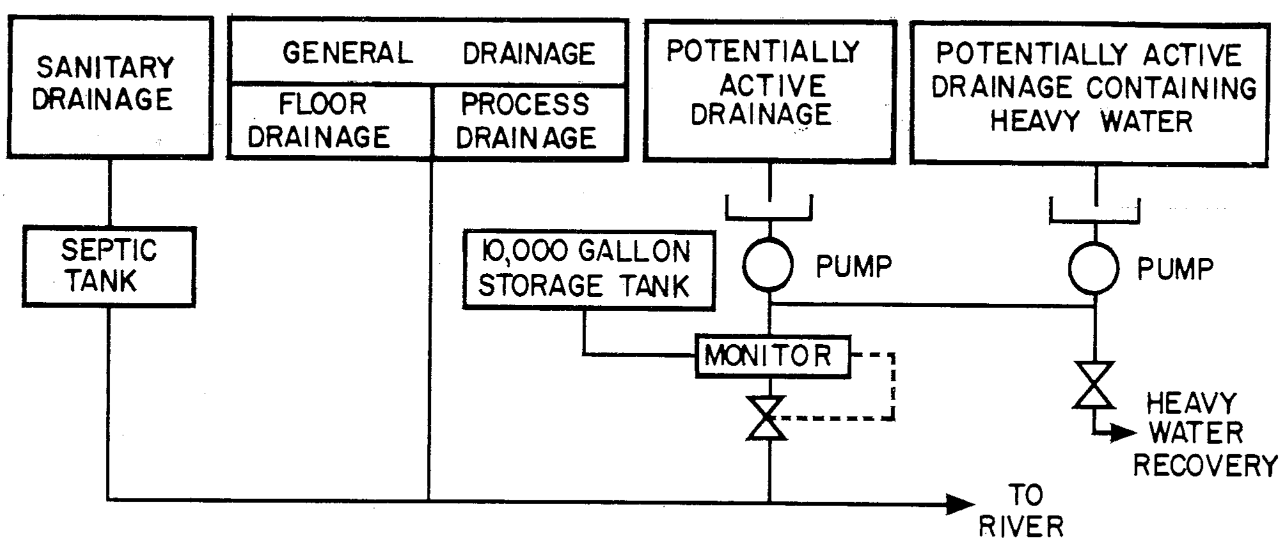


Fig. 17 Drainage Flow Diagram

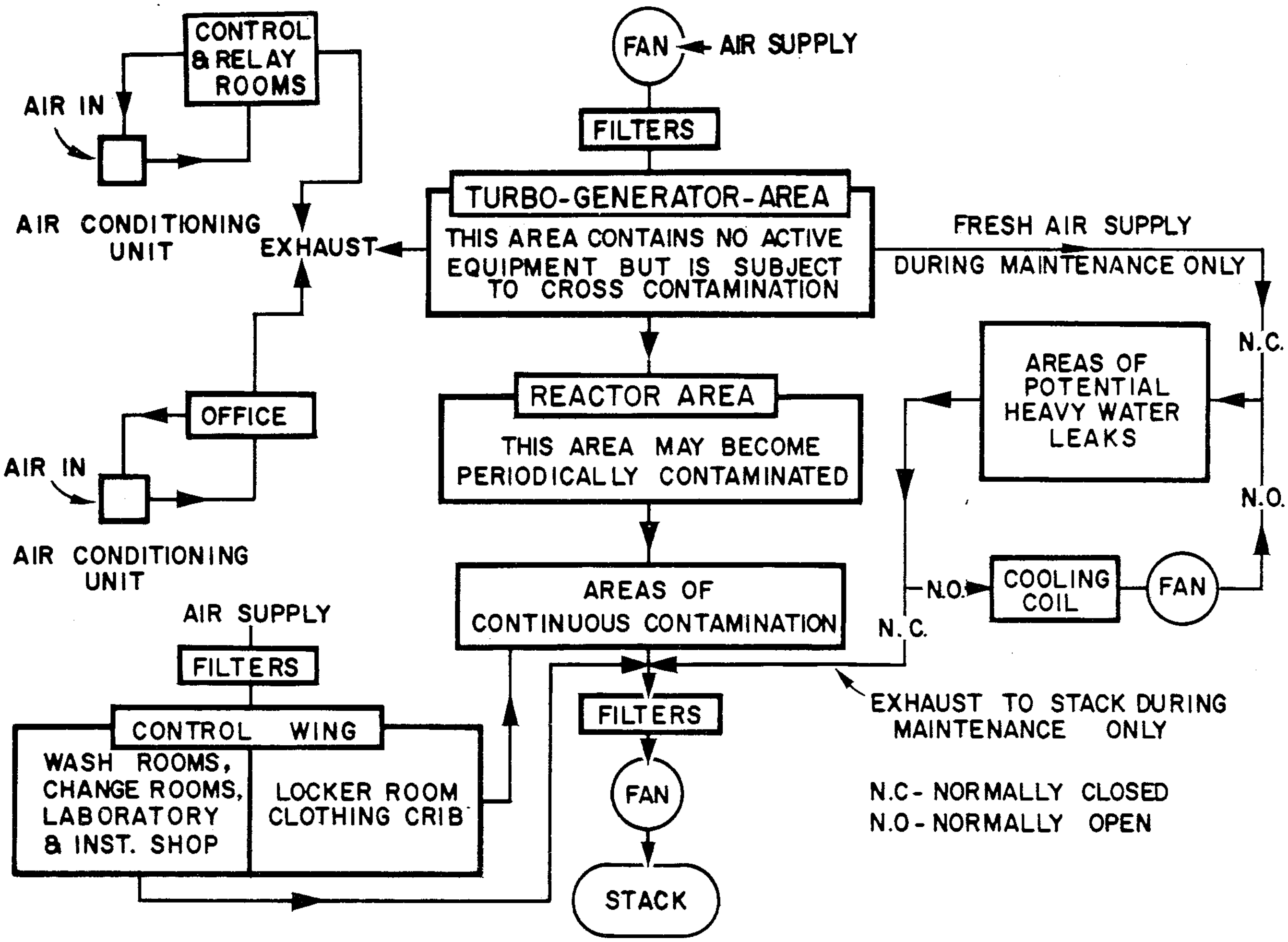


Fig. 18 Ventilation System

VENTILATION SYSTEM

The ventilation system is designed to handle both the normal requirements of a conventional station and the special requirements of the nuclear components in a single integrated system. The conventional requirements are those involving air changes in working areas, and control of the ambient temperature and humidity. The special requirements involve control of the movement of air borne radioactive materials, the recovery of heavy water escaping from the process, and heat removal from some of the process components and their structures. The main exhaust system provides the means by which filtered outside air is introduced into the working areas and then moved through zones of progressively increasing potential contamination before being discharged through filters to the stack. These filters are capable of removing particles down to 0.3 microns in size with 99.5% efficiency. When the reactor is in operation, the normal exhaust rate will be 22,000 cfm (10,000 l/sec). When the reactor is shut down for maintenance, purging of compartments will take place before personnel may enter, at which time the building exhaust will discharge up to 32,000 cfm (15,000 l/sec). A block diagram of the system is shown in Fig. 18.

HEAVY WATER CONSERVATION

Great care is given to the integrity of the heavy-water systems by using welded connections wherever possible. In addition, since small but acceptable leaks may occur in many components, the whole of the space surrounding the reactor, steam generator, and heavy-water piping is sealed, effective vapour barriers being used in all walls and floors. An air cooling and washing system is provided to recover any heavy-water vapour.

ELECTRICAL SYSTEM

The electrical system has been kept as similar as possible to that found in the average thermal power station. It has been designed to have a high degree of reliability to assure reactor safety during all phases of operation, including periods when the reactor is shut down.

Power is generated by the station at 13.8 KV. A 13.8/115 KV step-up transformer will connect the station with the Ontario Hydro system. Two breakers separate the NPD-2 generator from the transformer, and the station service takeoff is between the two breakers; thus, the station auxiliaries may be powered from either the turbine-generator or from the Hydro system. There will be two three-winding station service transformers to provide 2400 and 600 volts. Power at 2400 volts will be supplied to the main heavy-water circulating pumps and other large loads. Diesels and batteries supply emergency power for essential loads under shut-down conditions if Hydro power is unavailable.

There are four station service buses, numbered in order of decreasing reliability. The class four buses are 600 volt and 2400 volt AC, fed from the station service transformers. The class three bus will normally be fed from the class four 600 volt bus, but on failure of the class four bus will be supplied from a diesel generator set. The class two bus is a 120 volt AC bus fed from the class one bus through MG sets; it will supply some AC loads, such as certain instrumentation, which require the degree of reliability afforded only by the station battery. The class one bus is a DC bus normally fed from the class three bus through rectifiers, but on failure of the normal source it will be supplied from batteries; it will supply all the station DC loads.

WATER SUPPLY SYSTEM

The water supply for the plant is obtained from a pump-house located on the bank of the Ottawa River.

Water is supplied to the plant in three separate systems: circulating water for the condensers, process water, and a combined system for boiler make-up and domestic water.

COMPRESSED AIR

All requirements for compressed air are supplied from one reliable source of adequate capacity to maintain the design pressure of 80 psig (5 atmos.). To further ensure continuous instrument operation, air is automatically cut off from all non-essential services if an abnormal drop in air pressure should occur. In the event of air failure, all instruments will fail safe.

TURBINE-GENERATOR PLANT

The turbine is a single-cylinder, 3600 rpm unit with fifteen pressure-compounded impulse stages. The normal continuous rating of the turbine-generator set is 20,000 KW, and the maximum capability is 22,000 KW. Steam is supplied to the throttle at 400 psig (27 atmos.) dry and saturated, and the exhaust pressure is 1.5 in. (3.8 mm.) Hg. absolute. Three extraction points are

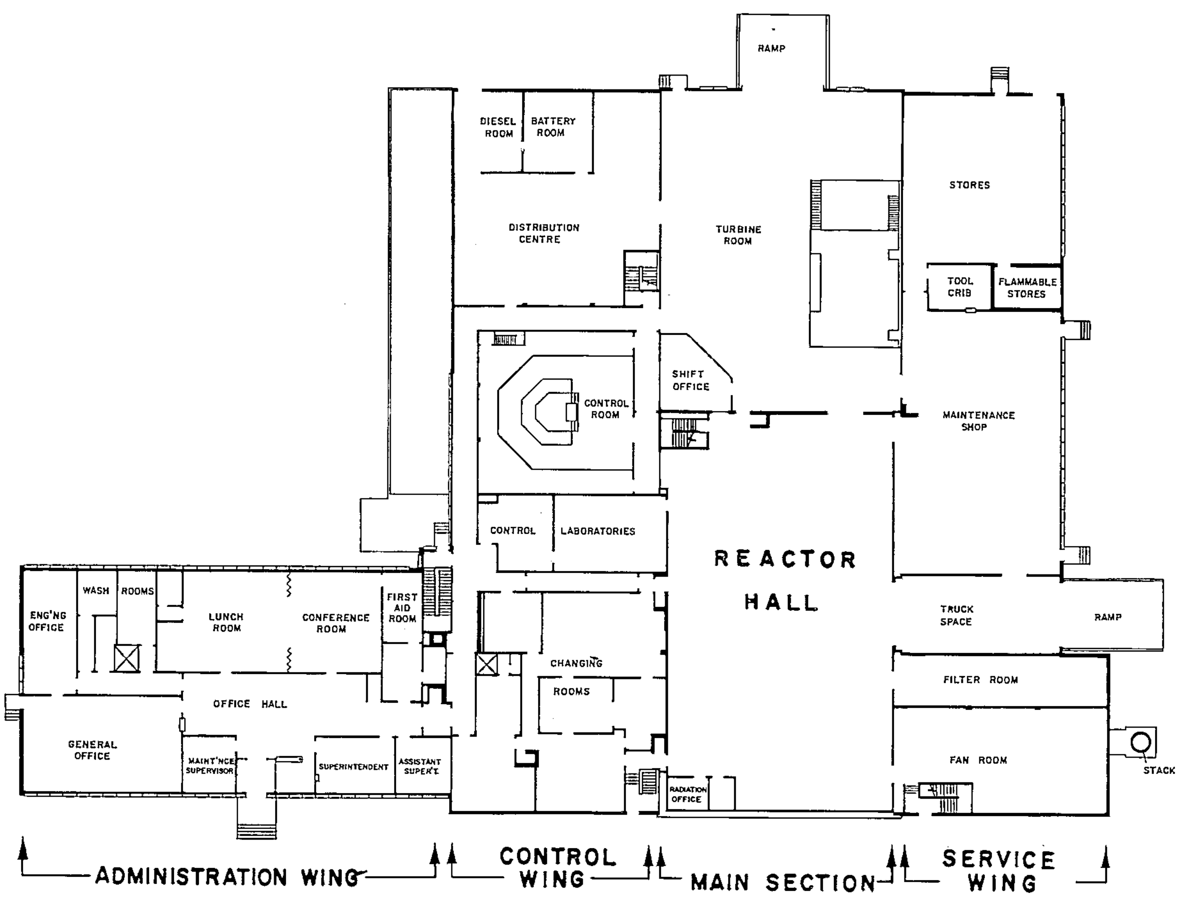


Fig. 19 Plan of Building

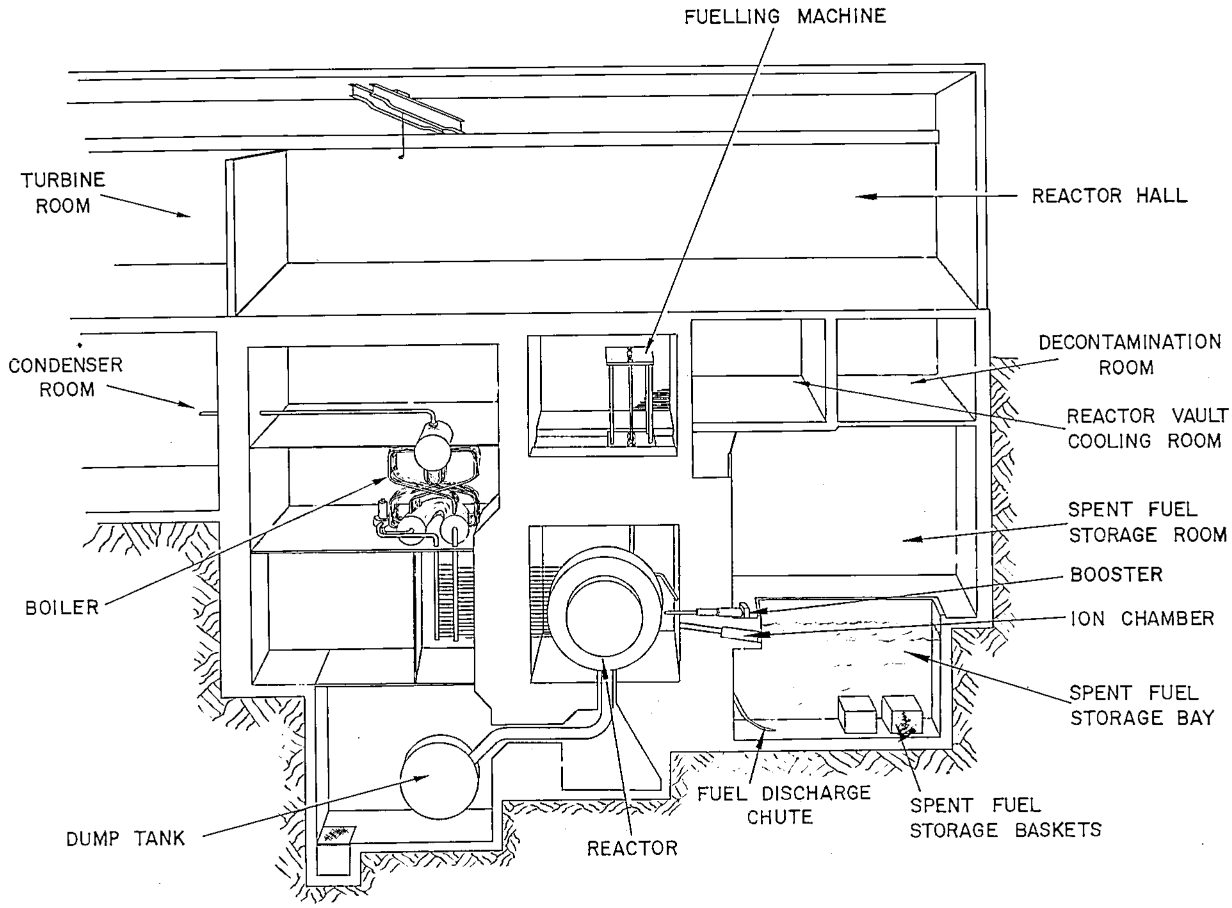


Fig. 20 Section of Building

provided for feedwater heating to 300° F (149° C). At maximum capability (22,000 KW) the turbine cycle heat rate is 12,550 BTU per kwh (3,160 Kcal/kwh) and the steam flow to the throttle is 296,000 pounds per hour (134,000 kg/hr).

As the steam is not superheated, special drainage belts are provided in the wet area of the turbine, and the collected water is led either to a feedwater heater or to the exhaust chamber. In addition, two water separators take all the steam from the high-pressure section of the cylinder after stage 8 and return practically dry steam to the low-pressure section of the cylinder. The exhaust steam to the condenser will be 11.5% wet at 20,000 KW load.

The generator is rated 20,000 KW at 0.85 power factor, 13,800 volts, three phase, 60 cycles, 3600 rpm with main and pilot exciters. It is totally enclosed and provided with an air cooling system. The condenser is a 22,000 ft.^{2}, two-pass, central flow deaerating unit designed for a cooling water inlet temperature of 60° F (15.5° C). Two full-duty motor-driven air pumps will evacuate the condenser and two full-duty motor-driven extraction pumps will be installed. The two circulating water pumps are vertical motor-drive units each rated at 12,600 Igpm (950 litres/sec.).

The feedwater heating plant consists of one low-pressure heater with drain cooler, a glands condenser, and two high-pressure heaters which raise the condensate temperature from 91° F (33° C) to a final feed temperature of 300° F (149° C) with 5 per cent make-up. Treated river water will be used for boiler make-up. Three 50% capacity boiler feed pumps are provided.

A reject condenser having 5000 ft.^{2} (465 m.^{2}) of surface area is provided in parallel with the turbine and main condenser to permit reactor operation independently of the turbine-generator set. This is a convenience during the early operation of NPD and provides an alternative load for the reactor when the turbine-generator is off the line for periods which could poison the reactor if it remained shut down. The reject condenser operates at atmospheric pressure and has a capacity to condense 200,000 lb./hr. (90,000 kg/hr.) of steam.

BUILDING

An outline floor plan of the building on the main operating floor is shown in Fig. 19.

The plant is made up of four attached structures: the Administrative Wing, the Control Wing, the Main Section, and the Service Wing.

The Main Section is approximately 60 ft. (13 m.) wide and 180 ft. (55 m.) long with a partial wall separating the 100 ft. (31 m.) long reactor plant from the 80 ft. (24 m.) long turbine plant. Both the reactor plant and turbine plant will be served by a single crane travelling on rails extending the length of the Main Section. The partial dividing wall is just low enough to permit the crane to pass over it.

The reactor plant is the only portion of the station that has any unusual construction features or materials. By placing the nuclear portion of the plant below ground, shielding is in great part supplied by the surrounding rock. This portion of the plant, in which the reactor and all components associated with radioactive materials will be located, is known as the Wells Area. It is illustrated in Fig. 20.

The wells area is divided into the Accessible and Inaccessible Wells Areas. The accessible area includes the south end of the wells area (the right hand side in Fig. 20.) It is accessible during reactor operation and is used for spent fuel storage and component decontamination. It also includes the fueling machine room (upper middle in Fig. 20.) The inaccessible wells area includes the north end of the wells area (the left hand side of Fig. 20) which is inaccessible during operation of the reactor. This area houses all equipment requiring shielding during reactor operation. The inaccessible area also includes the reactor vault (lower middle in Fig. 20) which is inaccessible at all times.

Two biological shields are provided: a shut-down shield, equivalent to 4 1/2 ft. (137 cm.) of heavy concrete, and an operating shield equivalent to 2 1/2 ft. (76 cm.) of heavy concrete. The shut-down shield surrounds the reactor vault; the areas immediately outside it are accessible only during periods of reactor shutdown. These areas house the boiler, primary pumps, and other major components, and are in turn enclosed by the operating shield. With the reactor in operation, radiation originating in the reactor or process equipment is reduced to a safe level by the combined effect of the shutdown and operating shields. The Administrative Wing on the west side houses the offices, lunchroom and first aid rooms. Normal personnel entry and exit to the building are through this wing. The Control Wing, between the Administrative Wing and the Main Section, contains the control room, control laboratory, heating plant, change rooms, and other service rooms. The Service Wing on the east side houses facilities for equipment maintenance, storage of spare parts, and the ventilation ducts, fans and filters.

OPERATING STAFF

It is estimated that NPD-2 will be managed and operated initially by a total staff of sixty-three, as shown by the organization chart in Fig. 21. This number is dictated partially by the diversity of skills required to operate and service the components, and partially by the experimental or pilot-plant nature of the station, particularly in the area of fuel behaviour. The same total num-

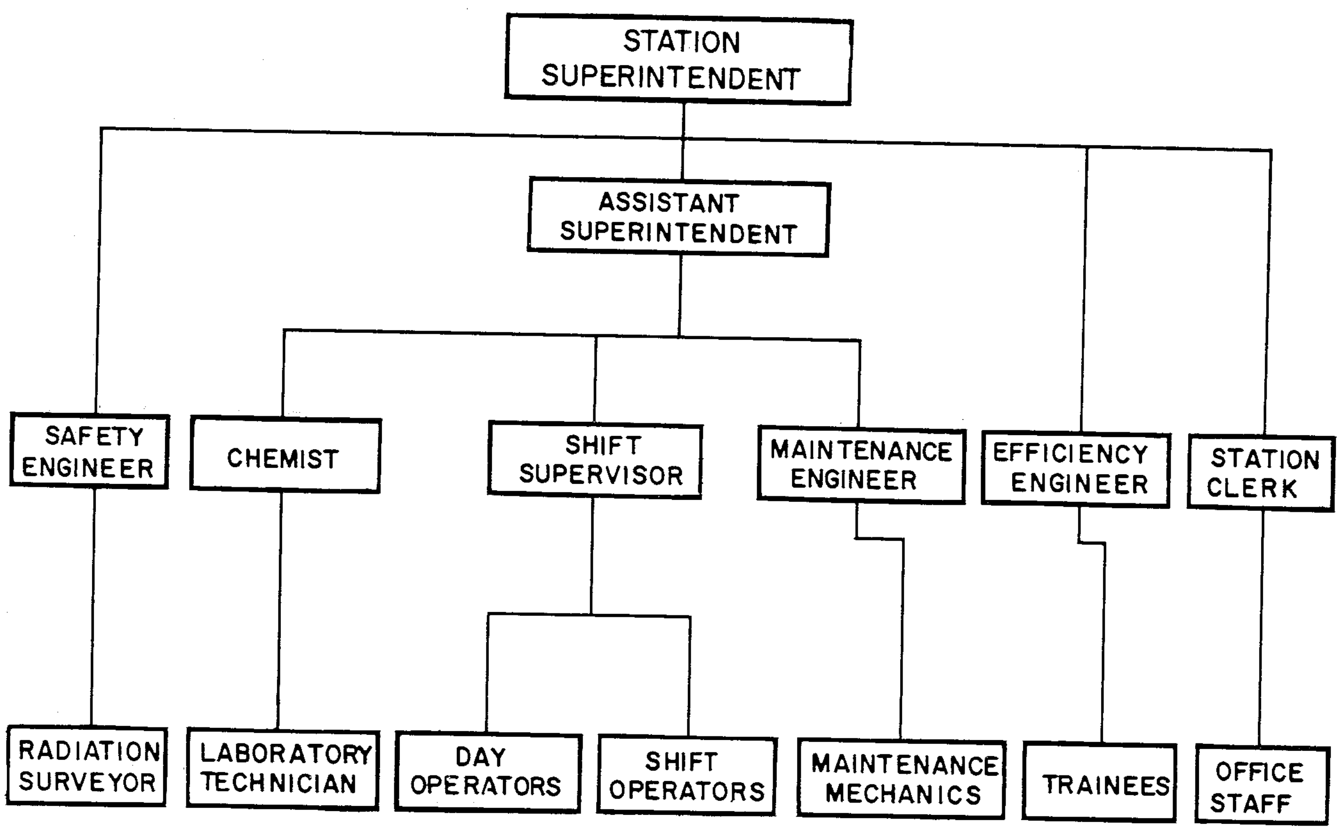


Fig. 21 Operating Staff Organization Chart

ber should be capable of operating a much larger unit of similar form. It is expected that this initial total may be satisfactorily reduced after the station operation has settled down to a steady routine two or three years after operation begins. The total staff of sixty-three is made up as follows: five shifts of six men, ten supervisory and technical personnel, a maintenance and technician staff of nineteen and a clerical staff of four. The professional staff of fifteen consists of the administration and technical staff and the shift engineer from each of the five shifts.

Acquiring the staff for a new station involves both the selection and the training of the personnel prior to station operation. Since the nuclear power industry has not yet advanced to the point where experienced personnel are readily available when required, more preliminary training must be given than would be the case for a conventional station.

Seven key personnel will receive basic training to complement their previous experience. The basic training of this group involves at least one year's training or previous experience in nuclear reactor operation and in thermal power plant operation. The thermal power plant training is being given at the R. L. Hearn generating station of the Hydro-Electric Power Commission of Ontario in Toronto. The nuclear reactor training is being given by Atomic Energy of Canada Limited at Chalk River.

Having completed this basic training, a further detailed training program directed toward the NPD-2 project itself will be provided by the Canadian General Electric Co. Ltd. for the key personnel. This program will consist of:

1. An intensive course of study, lectures and discussions at C. G. E.'s Peterborough Works as an introduction to details of the project design.
2. Preparation of the station descriptive, operating, and testing manuals in cooperation with personnel of C G. E., Ontario Hydro, and A. E. C. L.
3. Training the balance of the staff.
4. Participation in the test program during and after plant completion, with particular emphasis on equipment performance, characteristics, and operation.

   Other members of the operating staff will be trained by the seven key personnel with supplemental training where required by A. E. C. L. or others.

SAFEGUARD APPRAISAL

A large amount of radioactivity is produced and stored within a nuclear reactor (mostly as fission products in the fuel). A measure of the safety of the reactor is the degree to which the system prevents or contains the release of this activity.

NPD-2 embodies several features, either inherent in the system or incorporated in the design, which prevent the release of fission products to the atmosphere. The main features are described below:

1. The advanced control system prevents conditions arising that might lead to the release of activity.

   Two systems, almost completely independent are provided: the Regulating System, to safely control the reactor as required by turbine steam demand or neutron power, and the Protective System, to trip the reactor whenever any one of seven parameters is exceeded. Each system employs triplicated channels operating on a two out of three principle to trip the reactor; operation of only one instrument will annunciate an alarm. This design permits testing of a complete channel while the reactor is at power.

   This triplication of instruments plus the ability to test on power means that the chance of the control system being in an unsafe condition can be reduced to a negligible value by testing at frequent intervals.

   The extensive use of automatic control helps prevent accidents arising from operating error.

   Since reactivity control is primarily by moderator level it is impossible for the reactor to go critical without automatically supplying shut-down potential in the moderator height. Incorporated in the design is the feature that permits the operator, after a reactor trip, to bring the reactor back to critical at a very low power after checking the neutron power and period instruments. This keeps the reactor in a controlled state at all times.

   The positive rate of addition of reactivity is limited. The reactivity is controlled by moderator level, and the positive rate of change of the level is limited by the capacity of the helium blowers. The moderator cannot be raised, i. e., reactivity added, faster than this equipment permits. There is one "booster" rod for use in poison override, but the rate of addition of the limited reactivity in the rod (2.5 mk) is very low.

   The fuel possesses a strong negative reactivity coefficient with temperature, which acts as a self-regulating feature of the reactor.

   The very reliable power supply for critical equipment and instruments results in a negligible probability of an accident due to loss of power.

   Refueling is done by changing individual slugs containing only 30 pounds of UO_2 and will normally be carried out with the reactor at power. Therefore, the reactivity effect of new fuel will be readily detected and automatically compensated.

2. The several barriers between the fission products (in the fuel) and the atmosphere serve as effective containment.

The fuel is sintered UO_2 pellets, a form reasonably retentive of gaseous fission products for hundreds of hours at the NPD-2 operating temperatures, and completely inactive with high temperature water. It has a high melting point of 5100° F (2816° C) and an extremely high boiling point. In the reactor, the UO_2 is contained in 1 inch (2.54 cm) diameter Zircaloy 2 tubes, 19 1/2" (49.5 cm) long, seven of which are bound together to form a fuel slug.

Zircaloy 2, the sheathing and coolant tube material, has good strength and very low corrosion in water at the operating temperatures of NPD-2.

The primary heat transport system is a closed loop designed to give the greatest integrity possible. The entire system is designed according to the ASME-ASA pressure vessel and piping codes. Systems following these codes have a remarkable record of failure-free performance.

The reactor and boiler system is placed in what is essentially a sunken strong room. This area will contain within its bounds the release from any likely accident. A dousing system in the area condenses the release at the source.

3. Since NPD-2 is a pressure-tube type of reactor, the high-temperature and high-pressure fluid in the reactor is confined to relatively small channels. The moderator and the surrounding reflector, at atmospheric pressure and a relatively low temperature, are a large potential heat sink, and minimize the stored thermal energy available for dispersion of activity.

The NPD-2 concept also has some undesirable characteristics but these have been recognized and accounted for by the design:

1. NPD-2 has a positive coolant void coefficient, meaning that a loss or boiling of the coolant adds reactivity and tends to increase the power. The protective system guards against this by providing shut-down at a rate greater than that necessary to overcome the positive effect. Any serious change in coolant conditions will initiate a shut-down.

2. Zirconium reacts exothermically with water or steam at high pressures. However in NPD-2 it is almost impossible for the Zircaloy 2 to reach the temperature necessary for the reaction while still in contact with water or steam.

In summary, because of the various features incorporated in the reactor-boiler control system it is considered that NPD-2 is particularly safe from a control standpoint. Similarly, the several provisions to ensure cooling of the fuel render a loss-of-cooling accident a very improbable event. However, even a loss-of-cooling accident is looked after. These features, in addition to the various structural barriers to the escape of system fluids, mean that the probability of the emission of dangerous quantities of radioactive materials from the station is extremely low.

CONCLUSION

The NPD-2 Nuclear Power Demonstration reactor is unique among reactors now under design or construction. It uses natural uranium in the form of uranium oxide as the fuel; it employs bi-directional on-load fueling using short slugs of fuel, and the control and protective system is based on moderator level variation exclusively. Solution of the problems connected with the design and construction of NPD-2 will add significantly to reactor knowledge.

The many years of applicable experience at Chalk River in research, development, and reactor operation, and the results of design and development work to date give every reason to be confident that NPD-2 will achieve its objectives.

While NPD-2 is designed particularly with Canadian needs in mind, other nations with access to the large amounts of natural uranium that are now being discovered throughout the world should find the NPD-2 type of reactor of considerable interest.

SUMMARY OF NPD-2 STATION DATA

GENERAL

Reactor Type:

Fuel:	Natural Uranium Oxide (UO_2)
Moderator:	Heavy Water
Coolant:	Heavy Water
Purpose:	Demonstrator
Owner:	(Joint) Atomic Energy of Canada Ltd. and Hydro-Electric Power Commission of Ontario
Designers:	
Reactor Plant:	Canadian General Electric Co. Ltd.
Conventional Plant:	Hydro-Electric Power Commission of Ontario
Prime Contractor:	Canadian General Electric Co. Ltd.
Location:	Near Des Joachims, Ontario
Commissioned:	June, 1961
Thermal Power Output:	82,500 KW
Electrical Power Output:	20,000 KW (Gross) 17,000 KW (Net)

REACTOR

Fuel

Form:	Sintered Natural Uranium Oxide (UO_2) Pellets
Cladding:	Zircaloy 2
Length of Fuel Slug:	19½ in. (50 cm.)
Type of Slugs:	Bundle of 7 cylindrical elements
Number of Slugs:	9 in each of 132 sites (8 within the active zone)
Number of Pellets:	22 in each element
Length of Pellets:	0.832 in. (2.11 cm.)
Diameter of Pellets:	0.937 in. (2.38 cm.)
Thickness of cladding:	0.025 in. (0.0635 cm.)
Maximum Surface temp:	547° F. (287°C.)
Maximum internal temp:	4000° F. (2200°C.)
Maximum heat flux:	225,000 BTU/hr. ft.^{2} (7×10^{-2} KW per cm.^{2})

Coolant Pressure Tube Assemblies

Coolant flow:	Bidirectional in adjacent tubes
Coolant velocity in tubes:	15.7 ft./sec. max.
End Fitting Material:	Steel
Tube Material:	Zircaloy 2
Number of Tubes:	132
Tube wall thickness:	0.163 in. $^{+.008}_{-.012}$ (0.414 cm.)
Tube inside diameter:	3.25 in. (8.25 cm.)
Tube length:	13′ -3″ (404 cm.)
Assembly overall length:	19′ -2″ (585 cm.)

Calandria

Form:	Horizontal Cylinder
Material:	Alcan 57S Aluminum alloy
Nominal Diameter:	17′ -0″ (518 cm.)
Overall length:	15′ -0″ (457 cm.)
Inner Wall Thickness	¼ in. (0.64 cm.)
Outer Wall Thickness:	
Sides:	½ in. (1.27 cm.)
Ends:	1½ in.(3.81 cm.)
Calandria Tube Inner Diameter:	4 in. (10 cm.)
Calandria Tube Wall Thickness:	0.054 in. (1.42 cm.)

Reflector

	Primary	Secondary
Material:	Heavy Water	Ordinary Water
Form:	Liquid Annulus	Liquid Annulus
Quantity:	34,600 lb. (15,660 kg.)	100.000 lb. (45,300 kg.)
Radial Thickness:		
At centre	21.6 in. (55 cm.)	Min. 11.8 in. (30 cm.) (part Shield)
At Ends	5.6 in. (14 cm.)	

Core

Number of cells:	132
Lattice Type:	Square
Lattice Pitch:	10¼ in. (26 cm.)
Fuel mean cross-section:	4.64 in.^{2} (29.6 cm.^{2})
Sheath mean cross-section:	0.78 in.^{2} (5.0 cm.^{2})
Coolant mean cross-section:	2.87 in.^{2} (18.5 cm.^{2})
Total cell area:	105 in.^{2} (678 cm.^{2})
Theoretical core diameter:	11′ -0″ (335 cm.)
Moderator Volume:	1200 ft.^{3} (33.83 m.^{3})
Fuel burn-up (expected average):	4300 MWD per tonne
Active zone fuel capacity:	34,848 lb. of UO_2 (15.800 kg.)

Nuclear Characteristics

Thermal Flux Average:	2.6×10^{13} n./cm.^{2}/sec.
Thermal Flux maximum:	1.1×10^{14} n./cm.^{2}/sec.
Flux distribution—radial:	0.65
—axial:	0.68

REACTOR PROCESS SYSTEMS

PRIMARY HEAT TRANSPORT SYSTEM

Coolant Material:	Heavy Water
Quantity (Total):	28,060 lb. (13,000 kg.)
Total Flow:	5.14×10^6 lb/hr (650 kg./sec.) 10,000 Igpm (755 litres/sec.)
Outlet Temp:	530° F. (277° C.)
Outlet Header Pressure:	1021psig (73 kg./cm.^{2})
Inlet Temp:	485° F. (252° C.)
Inlet Header Pressure:	1113 psig (79 kg./cm.^{2})

Boiler

Number and Type:	1 horizontal "U-shell-and-tube" heat exchanger with steam drum
Total Surface Area:	6200 ft.^{2} (577 m.^{2})
Tube Material:	Inconel
Total Evaporation Rate:	300,000 lb./hr. (136,000 kg./hr.)
Heavy Water Velocity:	15.4 ft./sec. (4.7 m./sec.)
Heavy Water Pressure Drop	21.1 psi (1.5 kg./cm^2)
Feed Water Inlet:	300° F. (149° C.)

Primary Coolant Pumps

Number and Type:	3 shaft-sealed centrifugal at 50% capacity.
Capacity (each):	5000 Igpm (379 litres/sec.)
Operating Head:	140 psi (9.8 kg./cm.^{2})
Electrical rating:	800 HP each
Material:	11-13% chrome steel

Primary Coolant Piping

Material:	A.S.T.M. Type A.106, Schedule 80 Carbon Steel
Main Piping Diameter: ..	16 in. (40.7 cm.)
Pump Branch Diameter:	10 in. (25.4 cm.)
Feeder Pipe Diameter: ..	1½ in. (3.8 cm.)

Standby Cooling Circuit

Heavy Water Flow Rate:..	80 Igpm (6.1 litres/sec.)
Pumps:	3-50% capacity at 1.5 HP each
Cooler:	
Number and Type: ..	1 shell and tube heat exchanger
Maximum Heat Duty:	1.0 MW

MODERATOR AND HELIUM SYSTEM

Moderator Cooling

Flow Rate:	1300 Igpm (98 litres/sec.)
Normal Temp Range:	120° F. (49° C.) to 180° F. (82° C.)

Heat Exchanger

Number and Type:	1 shell and tube
Heat Duty:	6.5 MW
D_2O Pumps:	3-50% capacity at 50 HP each

Helium Circuit

Gasholder capacity:......	1500 ft.^{3} (42.4 m.^{3})
Helium Blowers:	2-100%capacity at 10 HP each
Helium flow rate:	180 cfm (84 litres/sec.)

REFLECTOR SYSTEM

Light Water Flow Rate: ..	1,000 Igpm (75.8 litres/sec.)
Pump:	1 at 30 HP
Cooler:	1 shell and tube heat exchanger

CONTROL SYSTEM

REGULATING SYSTEM

Fine Control:	Moderator volume, varied by gas balance system

Maximum positive reactivity rate while under normal operation: 0.04 mk./sec.

Coarse Control:

Booster rods:	Motor driven
1 rod worth:	2.5 mk.
Max. reactivity rate (approx.):	0.01 mk./sec.
Moderator Temp:	
Temp. range:	120° F. (49° C.) - 180° F. (82° C.) (Temperature can be lowered to 90° F. (32° C.) after dumping)
Temp. coefficient of reactivity:	−.038 mk./°F. for equilibrium fuel

Control Sensitivity:

Power output held within 2%

SAFETY (SHUT-DOWN) SYSTEM

Moderator completely evacuated from reactor (Total Worth: 1000 mk.)
In one second, reactivity reduced 3 mk.
In four seconds, reactivity rate: 20 mk./sec.

ELECTRICAL SYSTEM

A.C. Supply Voltage: ..	2400 V, 600V and 120V
D.C. Supply:	125V

TURBINE-GENERATOR

Turbine

Number and Type:	1 impulse-condensing single cylinder with separators
Maximum Capacity:......	22,000 KW
Inlet Temp:..............	450° F. (232° C.)
Inlet Pressure:............	400 psig (27 kg./cm.^{2})
Exhaust Pressure:	1.5 in. (3.8 m.m) Hg. abs.

Generator

Number and Type:	1 totally enclosed
Maximum Capacity:......	25,882 KVA at 85% PF.
Voltage:	13.8 KV
Phase:..................	3
Frequency:	60 cycles/sec.

Condenser

Type:	Two pass, central flow, deaerating unit
Total surface area:	22,000 ft.^{2} (2,050 m.^{2})
Extraction Pumps:	Vertical Two-stage 550 Igpm (42 litres/sec.) against a head of 140 ft. (43 m.)
Circulating water pumps:	2 vertical at 70%capacity each rated 12,600 Igpm (950 litres/sec.) with a head of 150 ft. (46 m.)

Feedwater Heating

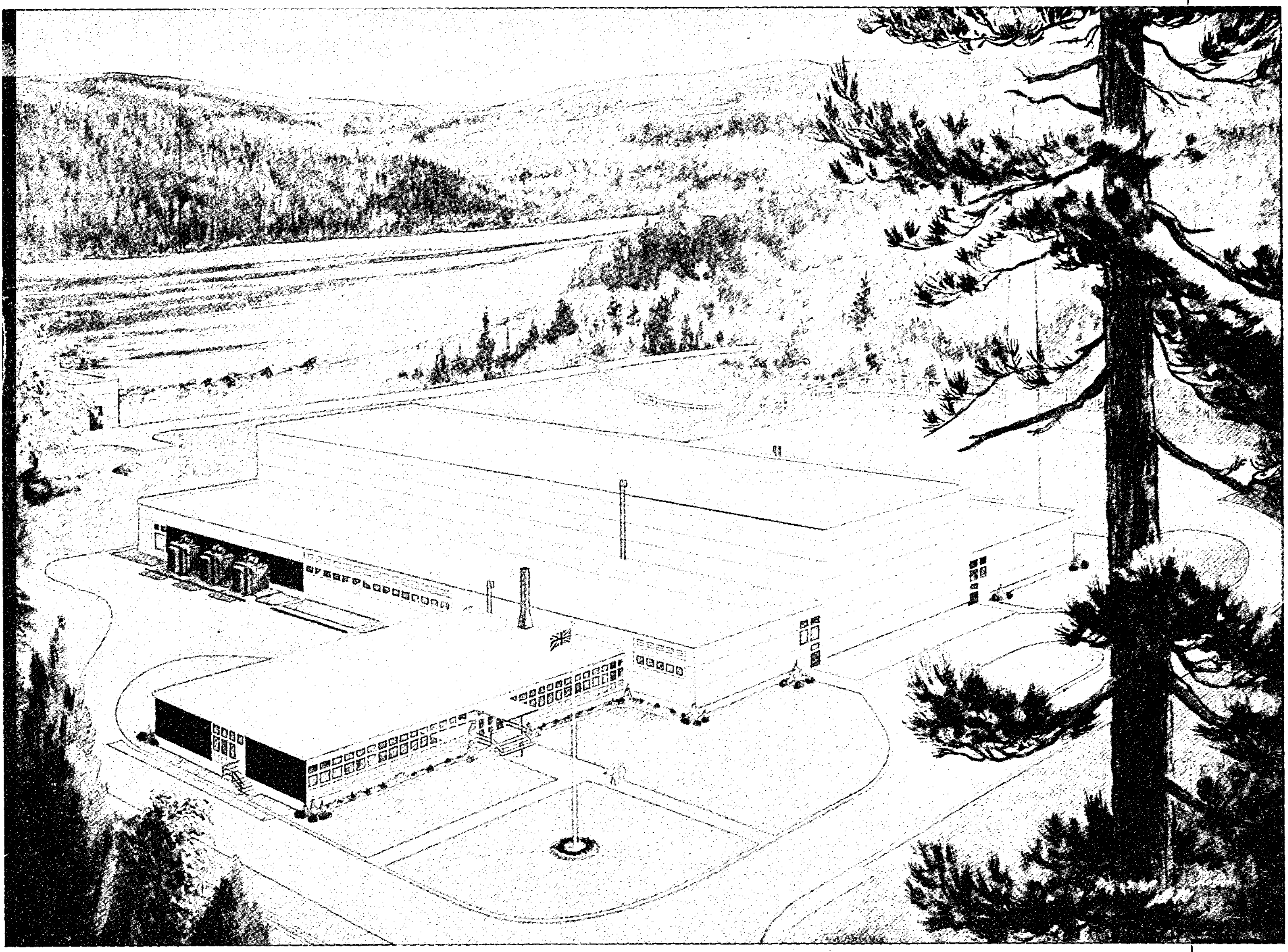
Heaters:	One low-pressure, with a drain cooler and glands condenser Two high-pressure, raise temp. from 91° F. (33° C.) to 300° F. (149° C.)

Boiler Feed Pumps

Number:................	3 at 50% capacity
Rating:	150,000 lb./hr. (68,000 kg./hr.)
Head:..................	1100 ft. (335 m.)

Reject Condenser

Total Surface Area:	5000 ft.^{2} (465 m.^{2})
Capacity:	200,000 lb. of steam/hr. (90,000 kg./hr.)



ARTIST'S CONCEPTION OF N P D-2 BUILDING

Booklet prepared by Civilian Atomic Power Department, Canadian General Electric Company Limited, Peterborough, Ontario, Canada, for distribution at the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, September, 1958.