CHAPTER 2

Genealogy of CANDU Reactors

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Summary:

This chapter discusses the historical beginnings and evolution of the CANDU reactor. The research and prototype reactors leading to the CANDU design are described with some of their principle technical parameters. Reasons for the choice of key parameters are given. This leads into a review of the evolution of the design of increasingly larger commercial CANDU reactors. The CANDU 6 reactor design has been chosen as a reference and a brief description of its main components with typical design parameters follows with a comparison with the CANDU 9 reactor. The chapter includes a note on the advantages of the CANDU reactor compared with other water cooled reactors and a general review of reactor safety as applicable to most water cooled reactors. It concludes with some technical details of the proposed Advanced CANDU reactor for comparison with existing commercial CANDU reactors.

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1 Canadian Historical Review

1.1 Canadian Nuclear Beginnings

During the first part of the 20th Century, significant discoveries were made in the field of nuclear physics and chemistry, leading to the splitting of the atom and release of energy. Canada became an important player in the development of nuclear technology during and after World War II as a result of the Allied war effort to develop the atomic bomb. Although not directly involved in the development of the bomb, Canada contributed a significant amount of related nuclear research.

Canadian history in nuclear physics essentially started in 1898 when Ernest Rutherford, a New Zealand born British scientist, was appointed Professor of Experimental Physics at McGill University. His early work here was related to radioactivity, and he won a Nobel Prize for this contribution to scientific knowledge in 1908. One of his students, John Cockcroft, who received a Nobel Prize jointly with Ernest Walton in 1951, subsequently came to Canada in 1944 to direct the Montreal Laboratory which had been set up in 1942 as a joint British-Canadian effort. In the intervening period, much fundamental research had been done, and consideration was then being given to the design and building of a reactor which would require more extensive laboratories to be located at Chalk River.

Concurrent work in the United States had led to the building of the first nuclear reactor, in which a self-sustaining nuclear reaction took place in 1942. This reactor was built of graphite blocks in which were distributed lumps of uranium. It was known as a pile due to the method of construction.

During the war years, emphasis in the United States was directed towards the development of the atomic bomb. This required highly enriched uranium or plutonium. Whereas the former required extensive isotope separation facilities, the latter could be created from non-fissile uranium in a reactor and chemically separated from the spent fuel. There was therefore a concerted effort in the United States to build suitable reactors. Therefore, after the war, nuclear research was also directed towards the design of reactors for power production.

After the war, Britain, without enrichment facilities and with no source of heavy water, followed the route of graphite moderated natural uranium reactors to produce plutonium for nuclear weapons, while at the same time developing reactors for power production. As a result, Calder Hall became the first commercial nuclear power plant in the world when the first of four 60 MWe units was connected to the grid on 27 August 1956. It continued to operate until 2003. In the United States, with uranium enrichment facilities in place, the focus was on the development of light water moderated reactors using enriched uranium. Such reactors were highly suitable for submarine propulsion because the reactors were very compact. Furthermore, the vessels could remain submerged for extended periods and could operate extensively between refuellings. The expertise thus developed in steam system components for submarine propulsion greatly influenced and facilitated the design of Canadian power reactors. Canada had supported the war effort by research related to heavy water physics and therefore had the expertise to direct its efforts into the development of heavy water moderated reactors using natural uranium as fuel. In summary, each country focused its efforts in different directions depending upon its relative expertise and resources.

Work in Canada was directed towards use of heavy water as a moderator, and to test the concept, a Zero-Energy Experimental Pile known as ZEEP was built at Chalk River. It was essentially an aluminum cylinder 2 m in diameter and 2.5 m in height surrounded by blocks of graphite. Uranium metal rods clad in aluminum were hung vertically from the top, and heavy water could be pumped in slowly at the bottom. ZEEP was completed and went critical on 5 September 1945, with almost exactly the amount of heavy water that had been predicted by theory. It was the first nuclear reactor to operate outside the United States. This confirmed the design parameters that would be used for the fuel rod lattice for the first functional reactor, designated NRX. ZEEP, however, continued to be used in later years for experimental studies of neutron behaviour.

2 Canadian Research Reactors

2.1 Small Research Reactors

The Zero-Energy Experimental Pile or ZEEP, which had an operating power of just 1 W, enabled tests to be carried out to determine the characteristics of heavy water and uranium lattices. It was also used to determine the neutron energy spectrum in various regions of a fuel bundle.

A larger version of ZEEP, known as ZED-2, was used for studies of fuel and irradiation effects as well as a variety of possible lattices for power reactors. Fuel bundle designs from 7-element to 37-element assemblies were tested to determine their performance.

The Pool Test Reactor, or PTR, was constructed, as its name implies, near the bottom of a deep pool of light water. Because this light water was both moderator and coolant, slightly enriched uranium had to be used as fuel. This arrangement with an easily accessible core made it possible to shuttle samples of fuel and other materials in and out of the reactor core to determine their effect on reactivity as well as to study short term radiation effects on materials.

2.2 The NRX Reactor

The first Canadian nuclear reactor to be designed to operate with a positive power output due to its generation of excess heat was the NRX. This reactor started operation in 1947 with a design thermal power rating of 10 MW, which was increased to 42 MW by 1954. At the time of its construction, it was the most powerful nuclear research reactor in the world and became the most intense source of neutrons available anywhere. It also suffered one of the first major nuclear reactor accidents in the world (power excursion to 80 MW and partial core meltdown), but was rebuilt and restarted within two years. It continued in service until 1993, which was quite a remarkable achievement for an experimental reactor.

The NRX reactor consisted of a vertical cylindrical aluminum calandria with a diameter of 8 m and a height of 3 m. The calandria contained the heavy water moderator. The core consisted of a hexagonal array of approximately 175 tubes with a diameter of 60 mm. These tubes held fuel elements, control rods, and experimental devices. Of these, 12 held control rods made of steel tubes containing boron carbide powder. The fuel elements consisted of uranium metal rods 3.1 m in length and 31 mm in diameter sheathed in aluminum. Each was surrounded by an aluminum tube through which flowed cooling water from the Ottawa River. An air flow in the gap between the calandria tube and the coolant tube was maintained to limit heat flow to the moderator to maintain it at a low temperature.

The key technical parameters for the NRX reactor are given in Table 1. The modern CANDU reactor has all these key elements in a similar but horizontal arrangement.

Table 1 Technical data for NRX

Parameter	Characteristic
Reactor shape	cylindrical
Reactor orientation	vertical
Calandria diameter	8 m
Calandria height	3 m
Fuel channel arrangement	vertical hexagonal array
Number of fuel channels	~175
Fuel channel diameter	60 mm
Fuel channel material	aluminum
Fuel	uranium metal
Fuel rod length	3.1 m
Fuel rod diameter	31 mm
Fuel rod mass	55 kg
Fuel cladding material	aluminum
Moderator	heavy water
Moderator quantity	14 m^3
Coolant	light water
Coolant flow	250 kg/s
Annulus gas	air
Annulus flow	8 kg/s
Control rods	steel tubes containing boron carbide

2.3 The NRU Reactor

The National Research Universal (NRU) Reactor was a significantly advanced version of the NRX. Design was started in 1949 after NRX had been put into operation, and it started self-sustained operation in 1957, some 10 years after NRX. It was designed for a thermal output of 200 MW using natural uranium, but was converted to 60 MW using highly enriched uranium in 1964 and converted again to 135 MW using low enriched uranium in 1991. The NRU supported the Canadian Neutron Beam Centre and served as a test bed for the development of the CANDU reactor. Like NRX, it was when built the most powerful source of neutrons in the world and became the world's leading supplier of radioactive isotopes. It suffered an accident in 1958 when a fuel rod caught fire on removal from the reactor and caused radioactive contamination within the building and to a lesser degree on the surrounding site. As with NRX, the NRU was designed to use natural uranium as fuel and heavy water as moderator, but was cooled by heavy water. With a view to later commercial power applications, it was designed for on-load refuelling.

3 Development of the CANDU Reactor

3.1 The NPD Reactor

The Nuclear Power Demonstration or NPD Reactor was partially modelled on the NRU reactor and was intended to be the first Canadian nuclear power reactor and a prototype for the CANDU design. Design work commenced in 1955, with the important components being designed and fabricated by Canadian General Electric. As a power producer, it had to operate at high temperatures sufficient to generate steam to drive a power producing turbine. These requirements presented some unique technical challenges due to the pressures under which the systems would have to operate. The design electrical output was to be 22 MW. Use of heavy water as a coolant and on-load refuelling had been proven with the NRU reactor, but the higher temperatures and pressures had to be accommodated by some significant changes. Firstly, the fuel was changed from uranium metal to uranium dioxide, which could withstand higher temperatures and was dimensionally stable over long irradiation periods. Secondly, the fuel cladding was changed from aluminum to zirconium alloy, which could also withstand higher temperatures and was corrosion resistant as well as being essentially transparent to neutrons. Thirdly, the fuel channels rather than the calandria were pressurized, leading to a pressure tube design rather than a pressure vessel design, and these pressure tubes would be of zirconium alloy. The pressure tube design led to further considerations to accommodate on-load refuelling. With horizontal pressure tubes rather than vertical tubes as in the NRX and NRU reactors, the fuel could be made into easily handled bundles which could be pushed in from one end and recovered from the other during refuelling while the reactor was at full power. Several unique safety features, such as safety shutdown systems with testability during operation, decay heat removal by natural circulation after shutdown, and a containment to prevent the release of radioactive material in the event of an accident, were incorporated. The NPD reactor started operation in 1962 and was taken out of service in 1987, by which time it had fulfilled its original purpose as a prototype and would have required re-tubing. It had also been an important training centre for future CANDU nuclear plant operators.

3.2 Douglas Point

The Douglas Point plant was the first full-scale CANDU nuclear generating station and was built on the shores of Lake Huron where the Bruce Nuclear Power Complex is now located. The proposed electrical output was 200 MW, and approval was given in 1959. It was basically a scale-up of the NPD reactor with similar design and components. It delivered power to the Ontario Hydro electrical grid system from 1967 until 1984, when replacement of the pressure tubes became necessary and could not be justified on economic grounds. Valuable experience was gained on this plant which was subsequently applied to later CANDU nuclear power plants.

4 Commercial CANDU Reactors

4.1 Commercial Reactor Development

The design of the CANDU reactor began with the Nuclear Power Demonstration (NPD) plant. Certain parameters had to be established within the constraints of existing technology. Basically, the reactor had to be fuelled with natural uranium, with heavy water as a moderator. This required a heterogeneous core arrangement with an array of fuel channels through which coolant flowed. Initial studies suggested a fuel channel cross section of 50 cm^2, which translated into a standard size fuel channel 3.25 in (83 mm) in diameter. To promote good heat transfer from the fuel, a modified hexagonal array of 19 rods or elements was chosen. The minimum spacing between rods based on laboratory testing was determined to be 0.050 in (1.27 mm). Based on irradiation testing, uranium dioxide was shown to have better dimensional stability and corrosion resistance than uranium metal at high burnup levels. This choice did, however, reduce the uranium concentration in the fuel and increased the need for good neutron economy. Zirconium had been selected as fuel cladding due to its corrosion resistance and low neutron absorption, but the quantity of it around the fuel had to be reduced as much as possible. Although a cladding thickness of 0.020 in (0.51 mm) was recommended by the tube supplier, further experimental work led to a thickness of 0.015 in (0.38 mm) and later to 0.013 in (0.33 mm). This resulted in collapsible cladding where the tube wall was compressed onto the fuel once in service.

A major decision was to adopt a pressure tube design rather than a pressure vessel design to avoid the difficulty of constructing a sufficiently large vessel for a heavy water moderated reactor. Zirconium alloy pressure tubes had sufficient neutron transparency to make this possible. A further major decision was to orient the fuel channels horizontally instead of vertically to facilitate on-load refuelling. The fuel could be made in the form of bundles which could be pushed through the channels with no need to link them together. These decisions on reactor configuration made the CANDU unique with regard to other commercial reactor developments. Although some were designed with pressure tubes or separate fuel bundles, only the CANDU has horizontal fuel channels.

With a selected fuel channel length of 4 m, the next question was the optimum number of fuel bundles. A significant advantage of separate fuel bundles was that the entire fuel channel need not be refuelled at one time. Low burnup fuel could be retained while new fuel was added. An analysis of diminishing returns indicated that having more than eight bundles would not yield any advantage, assuming a cosine neutron flux distribution along the fuel channel. This gave a fuel bundle length of 50 cm in a 4 m long fuel channel. This translated into a standard fuel bundle length of 19.5 in (495 mm), which was adopted for future CANDU reactors.

A fuelling machine was required at each end of the fuel channel. One supplied fresh fuel from one end, while the other received spent fuel at the other end. They had to connect to each fuel channel in turn and remove and re-insert closure plugs at the beginning and end of the refuelling process. The fuel configuration in the form of separate bundles facilitated the design of these machines because they did not have to handle a very long fuel assembly. In fact, the design of the fuel string and machines was dictated by the space available in the rock excavation, which had been dimensioned on the assumption that the reactor would have a vertical orientation. Of necessity, therefore, the fuelling machines were designed each with a rotating magazine with slots to hold each bundle separately. A significant advantage of this arrangement was that fuelling could take place from either end of the reactor. Such bi-directional fuelling made it possible to balance the tendency for flux skewing in a partially refuelled channel by fuelling in the opposite direction in an adjacent channel. In the NPD design, coolant flowed in the direction opposite to the fuelling direction, so that the fuel bundles were held in place by the flow of coolant. This established the characteristic bi-directional coolant flow in CANDU reactors.

The calandria vessel and calandria tubes were of aluminum. Initially, there was no spacer between the calandria tubes and the fuel channel pressure tube, but as the design for Douglas Point evolved, a single spacer in the form of an Inconel wire garter spring was introduced. This enabled the pressure tubes to withstand the effect of creep sag. The calandria had a light water neutron reflector surrounding it in a radial direction. To bring about a shutdown, the heavy water moderator could be rapidly drained into a dump tank, which for a small reactor was an effective way of ensuring safe shutdown in the event of an accident.

The heat transport system which removed heat from the reactor core and generated steam in a steam generator required pumps to circulate the coolant. Originally, completely enclosed canned pumps were proposed, but these were later changed to conventional vertical pumps with shaft seals. These had the advantage of being able to incorporate a flywheel to provide increased rotational inertia, which would extend the rundown period after a power failure and enable extra core cooling during the initial stages of high decay heat. The steam generator design was based on those being designed for United States nuclear submarines. It consisted essentially of a U-shaped horizontal heat exchanger in which the heavy water coolant transferred heat to the light water of the steam system. Because the latter operated at a lower pressure, some steam was generated around the tubes. Risers and downcomers circulated the steam and water mixture to a steam drum and returned the water to the heat exchanger. The long circular horizontal steam drum was similar to that of a conventional fossil fuel fired boiler and had internal cyclones for effective steam separation. The overall configuration required inlet and outlet headers at each end of the reactor to accommodate bi-directional coolant flow. The former were supplied by three 50% capacity pumps, and the latter delivered coolant to a single steam generator.

The NPD reactor was followed very closely by the Douglas Point reactor, the latter being essentially an enlarged version capable of operating like a commercial power plant and producing electric power to the grid system on a commercial scale. The site chosen was on Lake Huron, where there was an adequate supply of cooling water to sustain large scale power production. The most significant change was an upgrade in power from 20 MW electrical to 200 MW electrical. This naturally required a larger and more robust calandria, and therefore the structural material was changed from aluminum alloy to stainless steel and the calandria tubes from aluminum alloy to zirconium alloy. Because stainless steel is a stronger neutron absorber than aluminum, the calandria diameter was further enlarged to accommodate an internal heavy water reflector instead of an external light water reflector. The axial shields were moved inwards so that the fuel channel end fittings would be outside the shield and therefore more easily accessible.

The increased size of the Douglas Point reactor required fuel channels with a length of 5 m. This meant more fuel bundles per channel. The one-eighth length fuel bundle was based on a cosine flux distribution, but with flux flattening, such short lengths were not required, and one third length bundles would give virtually optimum fuel burnup. Because the same standard length fuel bundles would be used, the fuelling machines were therefore redesigned to handle two bundles in each slot. This enabled the magazine to be smaller in diameter, but able to accommodate 39 in (991 mm) of fuel with each fuel movement. Other changes in fuelling-machine design provided positive control from both ends of the fuel channel during refuelling and enabled refuelling in the same direction as coolant flow. The advantage of this was that fresh fuel, which would give a higher local power output, would be in contact with cooler water when only part of the fuel channel had been refuelled.

The larger output required increased steam generating capabilities. Furthermore, the longer reactor put the inlet and outlet of each channel further apart. Therefore, it was decided to put steam generators and circulating pumps at each end of the reactor. The same concept of separate heat exchangers and steam drums was retained, but instead of one horizontal heat exchanger, several vertical hairpin tube exchangers were used to feed a single steam drum in a way similar to that in which steam is generated in the water walls of a fossil fuel fired boiler. The multiple smaller heat exchangers could be replaced more easily than a single large one should tube problems arise. With steam generators and circulating pumps at each end of the reactor and with bi-directional flow, the "figure of eight" configuration for the heat transport system was established for this and all future CANDU reactors. A further feature of Douglas Point was the introduction of horizontal inlet and outlet headers above the highest fuel channels, but below the steam generators and circulating pumps. This would enable the system to be partially drained for maintenance on the circulating pumps and steam generators while still maintaining fuel cooling to remove decay heat. In normal operation, pressure was maintained in the heat transport system by a feed-and-bleed system to minimize heavy water inventory.

Pickering was the first large scale multi-unit commercial CANDU plant. This was yet a further scale-up from the 200 MW Douglas Point plant to a 4 x 500 MW plant. The 500 MW electrical unit size was consistent with that of the coal fired units on the Ontario Hydro system. Significant changes to the reactor included longer and wider fuel channels to accommodate the increased fuel inventory. The fuel channels were 6 m in length instead of 5 m. Rather than increasing the number of fuel channels excessively, it was decided to increase the diameter to a nominal 4 in (102 mm) while retaining the fuel element or rod diameter. This resulted in fuel bundles with 28 instead of 19 elements. This required a fuel channel 4.07 in (103 mm) in diameter to maintain the standard minimum element spacing, which was maintained by pads brazed to the element cladding. This then became the standard for all future CANDU reactors. The end shields were redesigned, and the annular space between the pressure tube and calandria tube filled with inert gas to minimize the risk of corrosion.

A further change was the adoption of a double "figure of eight" coolant loop with two hydraulically independent loops. This had the advantage of limiting the consequences of a loss of coolant accident due to a pressure tube or feeder tube break because the fuel channels in half the reactor would remain flooded without the need for emergency coolant injection. The moderator dump system for emergency shutdown was retained, but due to the time needed to drain the large calandria, it was supplemented with gravity operated shutoff rods. The latter system was able to handle most reactor trips, and therefore a dump arrest system was added to prevent dumping, enabling a faster reactor restart for such non-accident related trips.

The larger reactor introduced the possibility of flux oscillation, in which the peak flux wanders from one region of the reactor to another, induced by xenon transients. To overcome this, the reactor was divided into 14 zones, each with a neutron flux detector and a chamber in which an appropriate amount of light water could act as a stronger neutron absorber than the surrounding heavy water moderator and thus achieve local flux control. This regional neutron flux control system became a standard feature of all subsequent CANDU reactors.

An innovative change was the introduction of vertical steam generators which incorporated the vertical U-tube heat exchange surface and steam separation cyclones into a single vessel with provision for water recirculation and feedwater preheating. This design became the standard for CANDU reactors and many pressurized light water reactors, or PWRs.

For a multi-unit station, the concept of a containment surrounding each reactor was maintained and extended by linking the separate containments to a common vacuum building. In the event of a major reactor accident and release of steam inside any containment, the vacuum building would act to suck out and condense the steam, thus establishing a negative pressure in the containment to prevent egress of radioactive material.

Bruce was the next large multi-unit nuclear generating station to be built. This involved a change in the heat transport system. In previous designs, the outlet conditions of the fuel channels varied due to the different power ratings of the channels across the reactor core. To create more uniform conditions at the channel outlets, the central core region required a greater degree of subcooling at the inlet. This was accomplished by having separate preheaters outside some steam generators rather than integral preheaters and passing coolant for the central region only through these preheaters. This complicated the flow system, but obtained better matching of coolant outlet conditions. Another change was made to the steam generators, where the large U-tube heat exchanger as used in Pickering was retained, but without the integral upper steam separating section. Instead, a single long horizontal steam drum was attached to the top of all steam generators at each end of the reactor. The larger water-steam interface reduced level control problems, but introduced thermal stress problems. This design was used on Bruce A, while Bruce B reverted to the separate integral design as used at Pickering.

Studies had indicated that a moderator dump system was too slow for a large reactor, and therefore this was eliminated in favour of faster acting gravity shutoff rods. Because dumped moderator would not be available for channel cooling in the event of loss of coolant, a new system was developed for the next generation of reactors, but applied at Bruce. This was the injection of high pressure water into the heat transport system headers from gas pressurized storage tanks. For Bruce, the initial supply pressure was 800 lbf/in^2 (5.52 MPa), but this was reduced to 600 lbf/in^2 (4.14 MPa) in subsequent reactors.

A further safety issue arose with regard to the shutdown system. Bearing in mind that the CANDU design is a neatly balanced configuration of fuel and moderator, any disruption of this configuration will render the reactor subcritical and inoperable. In the event of failure of the shutdown system to operate in an overpower situation, the reactor would disassemble and shut itself down. Further consideration of the core disassembly philosophy led to addition of a second shutdown system which would remove the need to consider core disassembly for licensing purposes. Because the shutoff rods operated in a vertical plane, a new system operating entirely independently on a horizontal plane was developed. Solid rods were considered, but might suffer mechanical interference, so liquid rods were favoured. However, their effect would be too localized, so liquid dispersion in the moderator was the ultimate choice. By injecting neutron absorbing gadolinium nitrate solution into the moderator through an array of nozzles in the reactor core, a very quick shutdown could be ensured. This system was driven by helium in pressurized storage tanks and became standard on all CANDU reactors. Because removal of the gadolinium nitrate from the moderator was a lengthy process, this became the secondary shutdown system, while the shutoff rods remained the primary shutdown system which operates first in adverse transient conditions requiring a reactor trip.

The next generation of CANDU units included Point Lepreau and Gentilly-II as well as a number of overseas plants. A design for single unit stations was needed for these to be attractive to small utilities. The major change was the requirement for a single large containment for the reactor and heat transport system, including the steam generators, which would condense released steam and contain radioactive products in the event of a large break and loss of coolant accident. Other changes followed a natural evolution in reactor design. The fuel bundles were modified from 28 to 37 element bundles by using smaller diameter elements. The increased heat transfer surface thus created enabled a power increase from 500 MW to 600 MW electrical, while reducing the number of fuel channels from 390 to 380. This reactor therefore became known as the CANDU 600. Furthermore, from 12 steam generators and 16 coolant pumps at Pickering, the design had evolved to 4 steam generators and 4 coolant pumps. Experience in steam generator design had enabled larger components to be built. Comprehensive testing had shown that some boiling could be permitted within the fuel channels, thus increasing substantially the heat removal capability of the coolant. Darlington followed the design of the CANDU 600, but with a degree of scaling up. The heat transport system maintained the same arrangement of two loops in a figure-of-eight configuration, with each loop having a steam generator and a circulating pump at each end. For increased output, the reactor was designed for 480 fuel channels instead of 380. For future reactors of this size, a change in the figure-of-eight arrangement has been proposed. Instead of the two loops being in separate halves of the reactor, they would be interleaved alternately between adjacent channels throughout the reactor. The advantage of this would be to avoid a flux tilt and power pulse in the event of a loss of coolant accident in one loop while still maintaining the advantages of the two loop concept.

4.2 CANDU Reactors in Service

Table 2 lists the CANDU reactors in service in Canada with their commercial service start dates. Of these, Point Lepreau and Gentilly 2 are both CANDU 600 or CANDU 6 reactors, which was the basic reference design for reactors sold abroad. Table 3 lists CANDU 6 reactors in operation in foreign countries. In this text, Point Lepreau has been selected as the reference plant because it is typical of CANDU reactors on a worldwide basis.

Table 2 Canadian CANDU reactors: capacity and service date

Country	Station Name	Gross (MWe)	Net (MWe)	Service Date
Canada	Pickering 1	542	515	1971
Canada	Pickering 2	542	515	1971
Canada	Pickering 3	542	515	1972
Canada	Pickering 4	542	515	1973
Canada	Bruce 1	836	781	1977
Canada	Bruce 2	836	781	1977
Canada	Bruce 3	836	781	1978
Canada	Bruce 4	836	781	1979
Canada	Point Lepreau*	680	635	1983
Canada	Gentilly 2*	675	635	1983
Canada	Pickering 5	540	516	1983
Canada	Pickering 6	540	516	1984
Canada	Pickering 7	540	516	1985
Canada	Pickering 8	540	516	1986
Canada	Bruce 5	877	822	1985
Canada	Bruce 6	877	822	1984
Canada	Bruce 7	877	822	1986
Canada	Bruce 8	877	822	1987
Canada	Darlington 1	935	881	1992
Canada	Darlington 2	935	881	1990
Canada	Darlington 3	935	881	1993
Canada	Darlington 4	935	881	1993

*CANDU-6 Reactor

Table 3 Foreign CANDU 6 reactors: capacity and service date

Country	Station Name	Gross (MWe)	Net (MWe)	Service Date
Argentina	Embalse 1	648	600	1984
China	Quinshan 4	700	640	1984
China	Quinshan 5	700	640	2002
Romania	Cernavoda 1	706	655	1996
Romania	Cernavoda 2	706	655	2007
South Korea	Wolsong 1	679	629	1983
South Korea	Wolsong 2	700	650	1997
South Korea	Wolsong 3	700	650	1998
South Korea	Wolsong 4	700	650	1999

4.3 Direct Steam Generation

Some reactors, such as the boiling water reactor or BWR, generate steam within the reactor core. Due to the cost of heavy water and the formation of tritium, heavy water cannot be used in the steam cycle because some leakage from this cycle is inevitable. Therefore, light water must be used as the coolant. The CANDU with its separate moderator and coolant circuits (unlike the light water PWR and BWR reactors in which the moderator and coolant are the same and in a single loop) can be adapted for direct steam generation by using heavy water as moderator in the calandria and light water as coolant in the fuel channels. Some boiling already occurs in CANDU fuel channels, and this can be increased with vertical channels, where the natural buoyancy effect enhances the forced circulation generated by the circulating pumps. Steam is separated in horizontal steam drums similar to those in the early CANDU reactors. Hence, we have the heavy water-light water reactor, or HWLWR.

The most successful of these was the 100 MWe Steam Generating Heavy Water Reactor, or SGHWR, built in the United Kingdom. This reactor operated as a commercial prototype and produced reliable power to the national grid system from 1968 to 1990, when it reached the end of its design life. It had some novel design aspects, including direct injection of emergency coolant into the centres of the fuel bundles in the event of a loss of coolant accident, and served as a testing ground for various types of fuel. Because light water, which absorbs more neutrons than heavy water, was used as coolant, slightly enriched uranium in the form of uranium dioxide was required. The steam became slightly radioactive due to formation of nitrogen-16 in the reactor, and therefore certain precautions had to be taken by operational and maintenance staff when working around the steam turbine and feedwater system. Gentilly-1, a similar Canadian design, designated the CANDU Boiling Water Reactor or CANDU-BWR and also known as the CANDU Boiling Light Water Reactor or CANDU BLW, was modelled on the SGHWR, but fuelled with natural uranium. Built for an output of 250 MWe, first power was produced in 1971, and full power was achieved in 1972. However, it did not operate successfully. The reactor had a strong positive power coefficient of reactivity when additional voidage due to steam formation in the coolant channels occurred. Although this could be corrected by absorber rods in the core, the reactor remained difficult to control, and its operation was sporadic due to other problems such as condenser corrosion. It was shut down in 1979 and eventually decommissioned in 1984.

5 The Current CANDU Reactor Design

5.1 Plant Arrangement

The general arrangement of the plant is shown in Figure 1. The active part of the reactor is cylindrical in shape and set horizontally. The moderator is contained in the calandria, which is about 6.0 m long and about 7.6 m in diameter for the CANDU 6. Because of the low internal pressure in the calandria, the thickness of its shell and its tubes needs to be sufficient only to be structurally sound and to support the weight of the moderator. Only the pressure tubes located within the calandria tubes are pressurized. These pressure tubes, which contain the fuel and through which the coolant flows, are arranged horizontally in an axial direction. The coolant is fed to and from the pressure tubes by feeder tubes, which in turn are connected to headers situated above the reactor. The headers in turn are connected to steam generators in which heat from the coolant is used to generate steam. Circulating pumps, usually known as heat transport pumps, drive the coolant around the primary circuit. A pressurizer maintains pressure in the primary system to suppress large scale boiling and maintain operating temperatures.

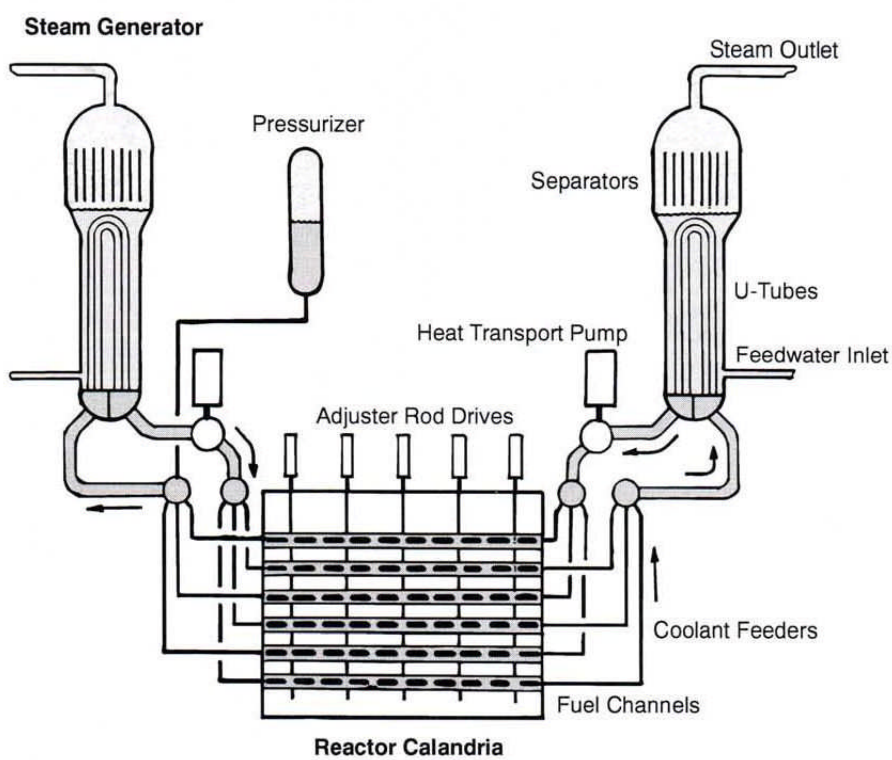


Figure 1 Diagrammatic cross section of a typical CANDU (PHWR)

The moderator absorbs some energy from the neutrons as they are slowed down and also receives some heat by conduction and radiation through the annulus gas between the pressure tubes and calandria tubes as well as conduction through the tubes themselves. The low conductivity of the annulus gas provides the greatest resistance to heat transfer. Cooling of the moderator is therefore required to maintain a temperature of about 70°C, which means that a circulating and cooling system is required. This is shown diagrammatically in Figure 2.

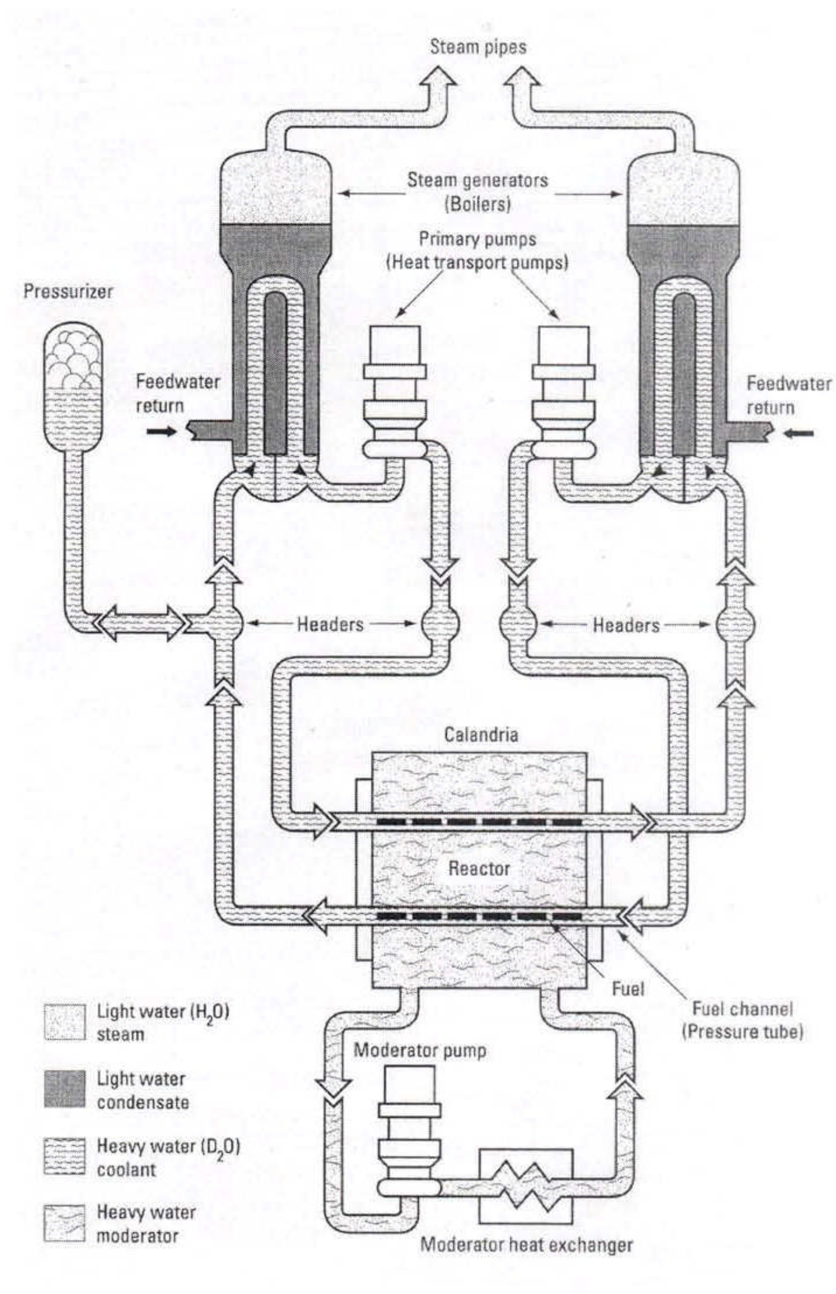


Figure 2 Moderator and coolant circuits

One characteristic when using heavy water as a moderator is that the neutron slowing down distance is relatively long. This determines the spacing between the fuel channels because neutrons need to be able to reach thermal equilibrium with the moderator before entering the next fuel channel. Too short a distance results in under-moderation and too long a distance in over-moderation. CANDU reactors are slightly over-moderated, but while this has some disadvantages, it does provide additional space between the fuel channels for various control devices. Over-moderation enables additional neutrons to be absorbed in the moderator, and when voidage occurs in the fuel channels, fewer neutrons are absorbed, thus increasing the number of neutrons and giving a positive void coefficient of reactivity which must be counteracted in other ways. With a separate calandria, the moderator temperature can be kept relatively low compared with the fuel and coolant. This is advantageous in reducing neutron energy and promoting the fission process due to the increased fission cross section of uranium at lower temperatures. In all CANDU reactors, the moderator in the calandria is maintained at atmospheric pressure and a temperature of about 70°C.

5.2 Fuel Channel Conditions

In the CANDU reactor, coolant enters the fuel channels of the core in a subcooled state, but not excessively subcooled. It leaves the most highly rated fuel channels with a vapour content of a few percent, with boiling having occurred. The lower rated channels have little or no boiling. In this way, nearly constant reactor coolant outlet temperatures can be maintained even though some channels produce more heat than others. Furthermore, the outer channels with the lowest rating can have their flow slightly restricted to enhance their coolant outlet temperatures. Note in this context that for good thermodynamic efficiency, the cycle working fluid should receive heat at the highest possible temperature. Hence, the reactor primary coolant should be at the highest possible temperature so that it can supply heat to the secondary working fluid at a suitably high temperature as well. Limited boiling in some fuel channels of a CANDU reactor enables the average temperature of the coolant entering the steam generators to be very near saturated conditions. An alternative view of the above review is that, by allowing some boiling to occur in the fuel channels, the pressure of the primary circuit can be lower while still maintaining desired steam generator temperatures. This has significant benefits in the design and manufacture of a rather complex primary system with multiple pressure tubes and feeder pipes.

Within individual fuel channels where boiling does occur, conditions are not uniform across the fuel bundles. The circular arrangement of fuel elements in each bundle leads to irregularly shaped and sized coolant channels. More boiling occurs in narrow channels than in wider ones. The horizontal arrangement of the fuel bundles also creates non-uniform mixing between the coolant channels because the lighter vapour tends to migrate upwards, resulting in some vertical segregation of vapour and liquid across the channel. Although turbulence, particularly at the abutting ends of fuel bundles, tends to equalize conditions, some fuel pins are subject to more boiling than others. Boiling does enhance heat transfer, but excessive vapour is detrimental, and therefore fuel sheath temperatures do vary slightly across any particular fuel bundle.

5.3 Comparison of CANDU Reactors

The CANDU reactor has evolved progressively and has been marketed in several other countries. Three standardized versions, CANDU 300, CANDU 600, and CANDU 900, with different capacities were developed. These are now known as CANDU 3, CANDU 6, and CANDU 9 respectively. The original numbers indicate roughly the electrical output in megawatts, but later developments have enabled much greater outputs to be achieved by the respective models. Table 4 gives key technical parameters and shows the range in outputs available from these three models built up with similar basic components.

Although the CANDU 6, for example Point Lepreau, is representative of CANDU reactors in general and most existing CANDU reactors are of this capacity, the CANDU 3 and CANDU 9, for example Bruce and Darlington, were marketed as alternate capacities to suit smaller or larger electrical utility systems respectively. The CANDU 9 is very similar to the CANDU 6, being simply larger in diameter and with increased heat transport capacity. It therefore has the benefit of economies of scale. The CANDU 3, however, was a novel development to minimize the adverse effects of economies of scale, but no examples have been built. It is like a half CANDU 6 with a single loop instead of a "figure of eight" loop with two steam generators at one end and two heat transport pumps at the other, thus maintaining essentially the same size components as the CANDU 6. Coolant flow is in one direction only, and only one refuelling machine is used, which is located at the coolant outlet end so that coolant flow aids the discharge of fuel bundles. A comparison of these three basic designs is given in Table 4. The technical parameters given in Table 4 and those given in Section 6 were extracted from the Atomic Energy of Canada publication "Technical Summary CANDU Nuclear Generating Station" which gives the parameters for the 600 MW and 950 MW units (subsequently the CANDU 6 and CANDU 9). The complete tabulated parameters are given in Appendix C to this chapter. Since publication these parameters have been revised as new developments have allowed the design to evolve particularly with regard to channel power. The parameters given for CANDU 9 (950 MW unit) are therefore not representative of the latest plants for example Darlington which has a lesser number of fuel channels each of which have a higher power rating. However Appendix C is useful for comparison of the two unit sizes and serves well for educational purposes as the data is based on the same design philosophy. A later AECL publication in 1997 "CANDU Technical Summary" gives some technical parameters for the CANDU 6 but not for the CANDU 9.

As the CANDU design evolved with the proposed Bruce B and the later Darlington nuclear plants these were nominally in the 800 MW range resulting in the CANDU 8 design. The AECL publication in 1989 "CANDU 8 Technical Outline" lists the net output of Bruce B as 825 MW and of Darlington as 881 MW. Subsequently the terminology was rationalized as in Appendix B "CANDU: The Evolution". This shows the development of the CANDU up to the four-unit CANDU 600 MW class (Pickering) and the subsequent division into the single-unit CANDU 6 or 700 MW class (Point Lepreau onwards) and the four-unit CANDU 900 MW class (Bruce A & B and Darlington) with the further development of the single-unit CANDU 9 also 900 MW class.

5.4 Power Density

Power density is a measure of how much power is generated per unit volume of core. Generally, the higher the power density, the more compact is the reactor and the lower the capital cost. In all reactors, therefore, there is an incentive firstly to maintain a high neutron flux (to achieve a high heat release rate) right across the reactor core and secondly to maintain a high heat flux (to achieve a high heat removal rate) everywhere in the reactor.

All CANDU reactors contain roughly the same amount of in-core fuel per channel, but the major differences are in the fuel bundles themselves and in the number of channels. The total design power output of a CANDU reactor can be fixed by selecting an appropriate number of fuel channels for the reactor. In a CANDU 3 reactor there are 232 fuel channels, in the CANDU 6 reactor the number of channels is increased to 380, while in the CANDU 9 reactor there are 600 fuel channels.

A new development is the Advanced CANDU Reactor, known as the ACR, which uses slightly enriched uranium as fuel and light water as coolant. This has the benefit of significantly reducing the heavy water inventory. In the proposed ACR, the power density is increased by a smaller calandria diameter, but with an increased number of elements in the fuel bundle and a smaller channel pitch, resulting in a more compact reactor.

Table 4 Comparison of CANDU reactor types

Parameter	Units	CANDU 3	CANDU 6	CANDU 9
Moderator		D_2O	D_2O	D_2O
Coolant		D_2O	D_2O	D_2O
Number of fuel channels		232	380	600
Fuel		UO_2	UO_2	UO_2
Number of elements in bundle		37	37	37
Number of bundles in channel		12	12	12
Number of steam generators		2	4	8
Number of heat transport pumps		2	4	4
Reactor outlet pressure	MPa	10.0	10.3	10.3
Reactor outlet temperature	°C	310	312	312
Reactor coolant flow rate	kg/s	5 300	7 600	13 500
Steam pressure	MPa	4.7	4.7	5.1
Steam temperature	°C	260	260	265
Steam flow rate	kg/s	700	1 050	1 610
Total fission heat	MW	1 441	2 156	3 394
Net heat to steam cycle	MW	1 390	2 060	3 347
Gross turbine generator output	MW	470	676	1 121
Net electrical output	MW	450	626	1 031

6 CANDU Technical Parameters

6.1 Reference Plant

Point Lepreau has been chosen as the reference for this text because it is a typical CANDU 6 reactor. Technical parameters for other CANDU reactors are given here for comparison to show the main differences, but other chapters will generally refer to Point Lepreau characteristics. Figure 3 shows an external view of Point Lepreau. Appendix A shows a cutaway view of a CANDU 6 reactor plant.

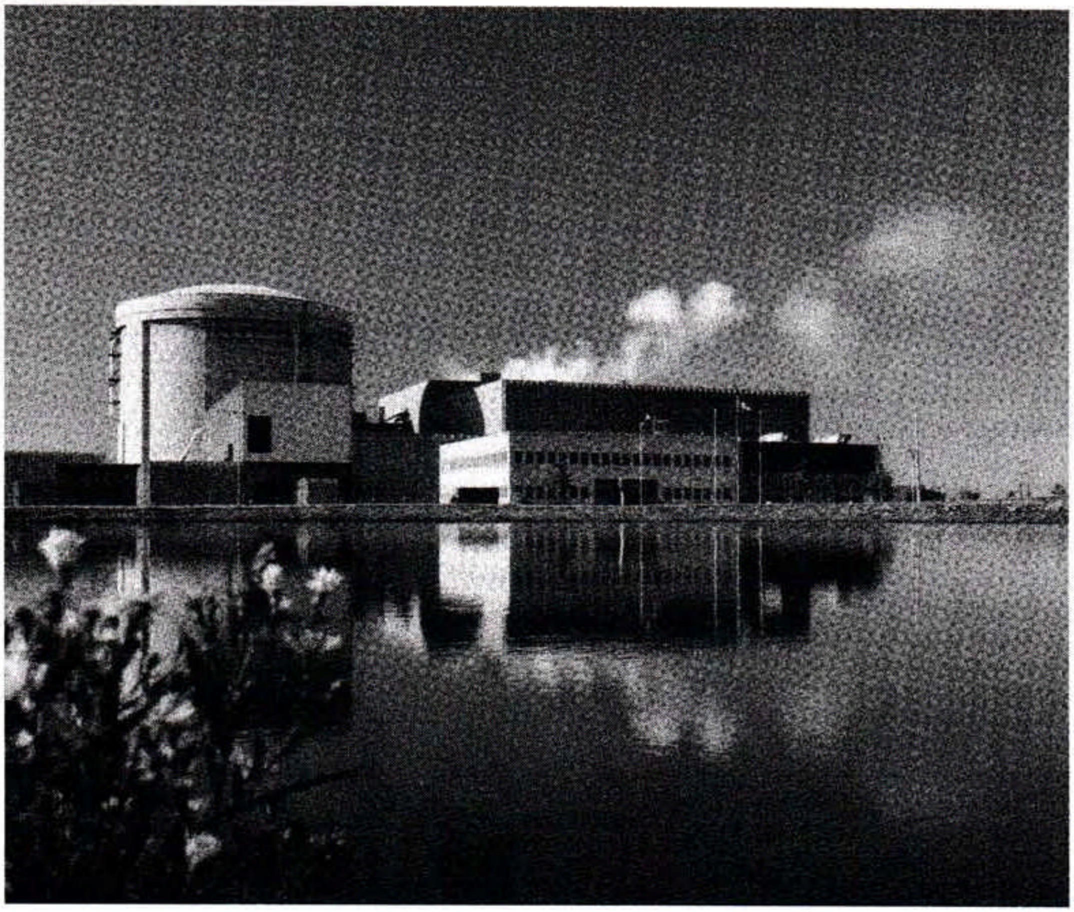


Figure 3 Point Lepreau nuclear generating station

6.2 Reactor Core Arrangement

The reactor core is cylindrical, but is set horizontally, as shown in Figure 4, with horizontal fuel channels. The heavy water moderator is contained in a cylindrical calandria with multiple tubes set in a square array and in an axial direction so that the moderator surrounds all the tubes.

Pressure tubes containing the fuel bundles pass through the calandria tubes. Heavy water coolant under a pressure of about 10 MPa flows through the pressure tubes to remove heat from the fuel bundles. Because the coolant is heavy water, there is minimal absorption of neutrons. Each pressure tube contains twelve fuel bundles, each of which has 37 fuel elements arranged in a circular pattern, as shown in Figure 5. The fuel elements consist of zirconium-alloy tubes filled with natural uranium dioxide pellets. A feature of the CANDU is that individual fuel channels can be refuelled while the reactor is at full power by pushing new fuel elements in at one end of the pressure tube and removing spent fuel elements at the other end, using special refuelling machines which can be attached to the ends of any pressure tube.

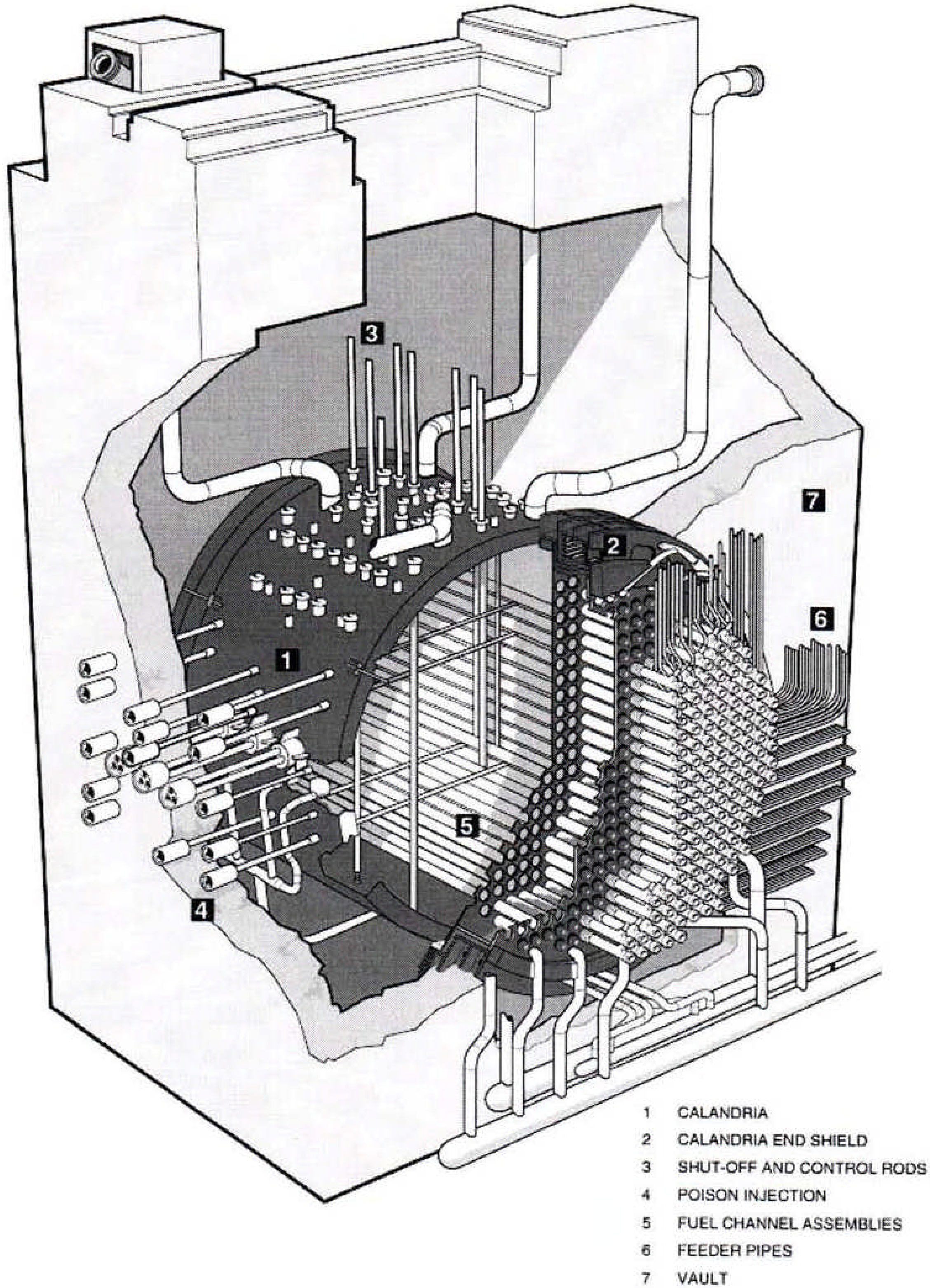


Figure 4 Reactor vault and assembly

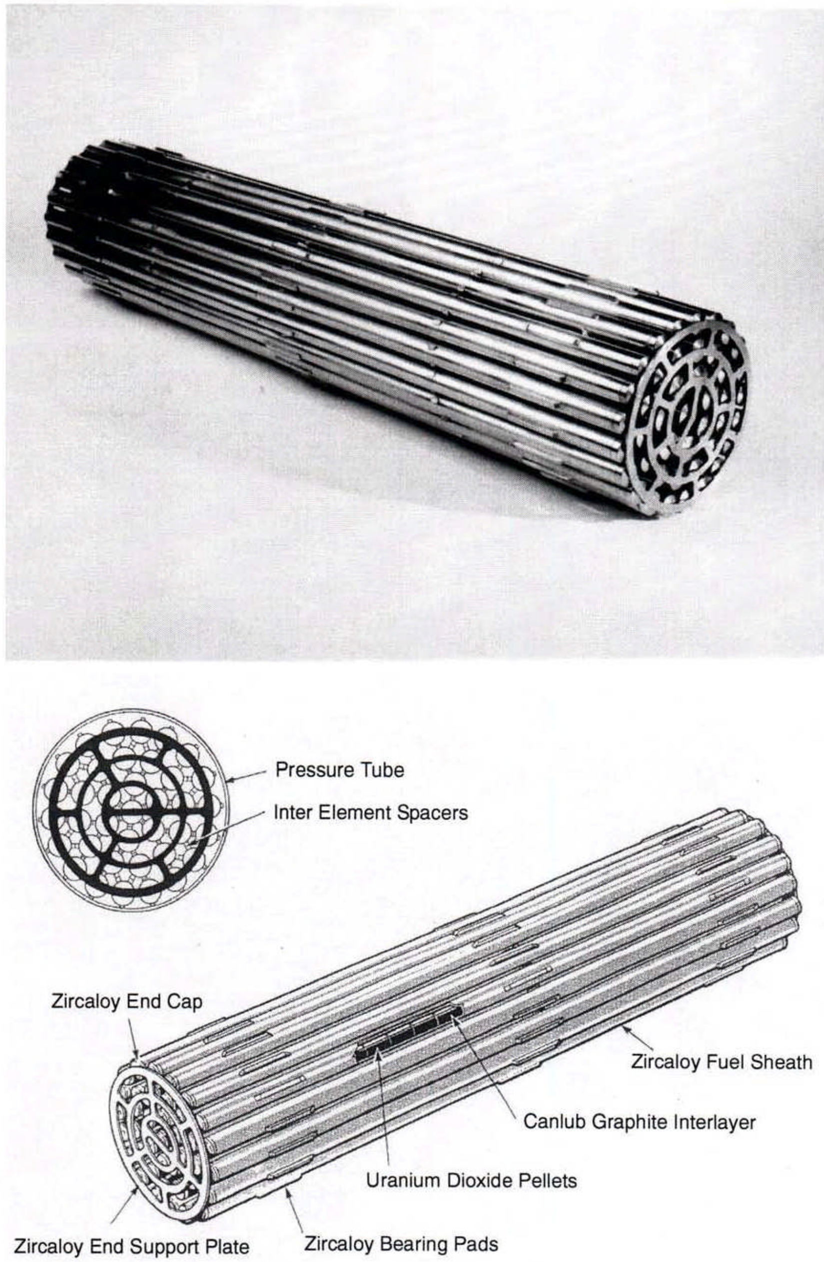


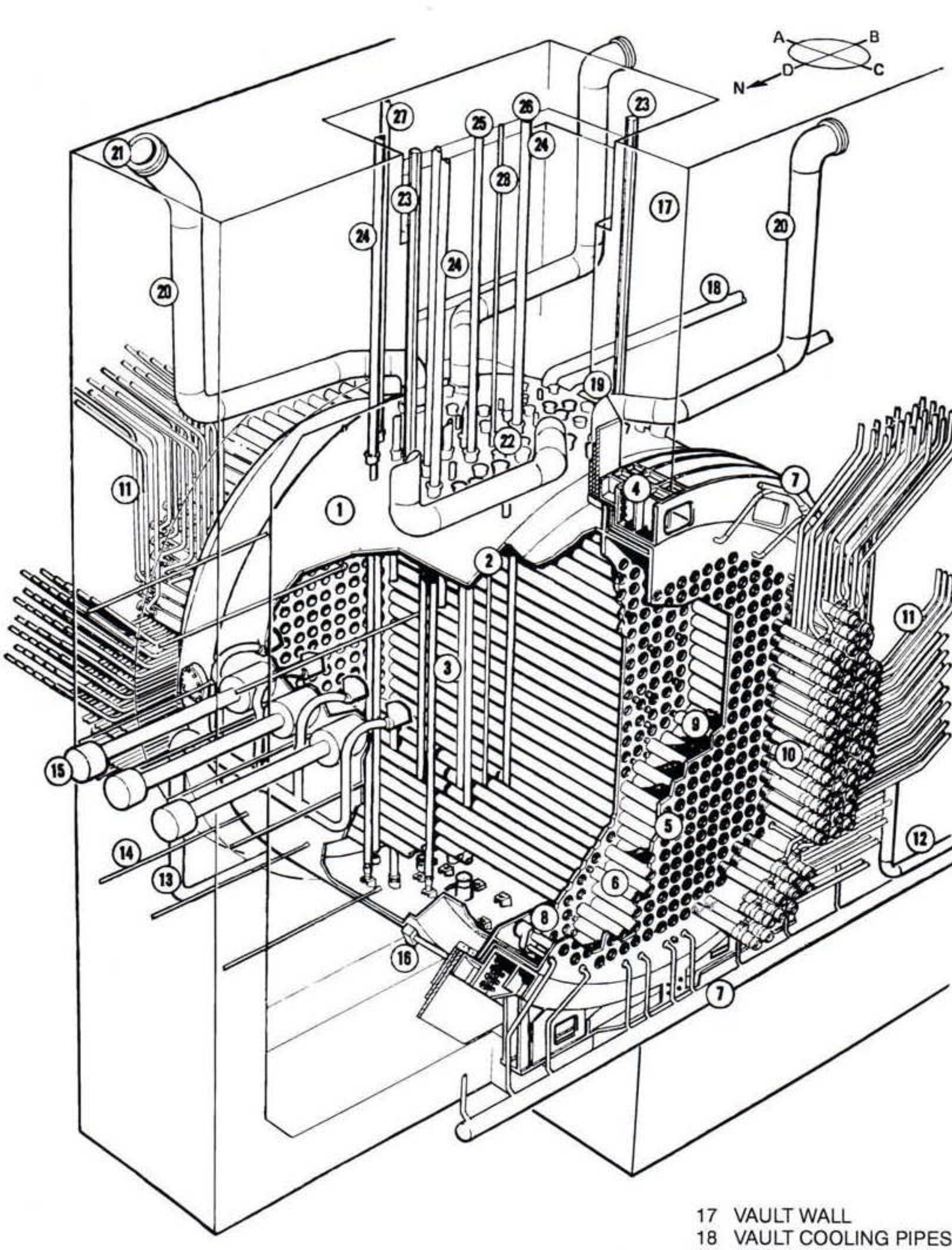
Figure 5 37-element fuel bundle

Control of the reactor is achieved by varying the amount of light water in special liquid level tubes in separate control zones in the calandria and by manipulation of control rods which are inserted into special vertical channels within the calandria and pass at right angles between the calandria tubes. Typical parameters for 600 MWe (electrical) and 950 MWe (electrical) CANDU reactor cores are given in Table 5.

Table 5 CANDU core parameters

Parameter	CANDU 600	CANDU 900
Number of fuel channels in core	380	600
Number of fuel bundles per channel	12	12
Number of fuel elements per bundle	37	37
Number of pellets per rod	30	30
Fuel pellet material	UO_2 sintered	UO_2 sintered
Fuel cladding material	Zircaloy-4	Zircaloy-4
Fuel channel array	square	square
Fuel channel lattice pitch	286 mm	286 mm
Fuel element configuration	circular	circular
Fuel bundle length	495 mm	495 mm
Fuel bundle diameter	102.4 mm	102.4 mm
Fuel element diameter	13.08 mm	13.08 mm
Fuel pellet diameter	12.16 mm	12.16 mm
Mass of uranium dioxide in core	95 Mg	153 Mg
Mass of uranium in core	84 Mg	135 Mg
Fuel type	natural U	natural U
Average core power density	~ 11 MW/m^3	~ 11 MW/m^3
Active core length	5.94 m	5.94 m
Extrapolated core length	6.06 m	6.06 m
Equivalent core diameter	6.27 m	7.90 m
Total core fission power	2 180 MW	3 394 MW
Total core thermal power	2 060 MW	3 237 MW
Maximum channel power	6.5 MW	6.5 MW
Maximum bundle power	0.8 MW	0.8 MW
Active heat transfer area	~ 3 430 m^2	~ 5 420 m^2
Average heat flux	~ 600 kW/m^3	~ 600 kW/m^3
Maximum heat flux	~ 1 000 kW/m^3	~ 1 000 kW/m^3

The reactor core includes monitoring and control devices to measure neutron flux and to modify the flux profile if required to obtain the best power distribution. Safety devices to bring about rapid shutdown, such as the shutoff rods and poison injection system, are also built into the core. The general arrangement of these is shown in Figure 6, and their locations are shown in the three views given in Figures 7, 8, and 9.



1 CALANDRIA
2 CALANDRIA SHELL
3 CALANDRIA TUBES
4 EMBEDMENT RING
5 FUELLING TUBESHEET
6 END SHIELD LATTICE
7 END SHIELD COOLING PIPES
8 INLET OUTLET STRAINER
9 STEEL BALL SHIELDING
10 END FITTINGS
11 FEEDER PIPES
12 MODERATOR OUTLET
13 MODERATOR INLET
14 FLUX MONITOR AND POISON INJECTION
15 ION CHAMBER
16 EARTHQUAKE RESTRAINT
17 VAULT WALL
18 VAULT COOLING PIPES
19 MODERATOR OVERFLOW
20 ACCIDENT DISCHARGE PIPE
21 RUPTURE DISC
22 REACTIVITY CONTROL ROD NOZZLES
23 VIEWING PORT
24 SHUTOFF ROD
25 ADJUSTER ROD
26 CONTROL ABSORBER ROD
27 ZONE CONTROL ROD
28 VERTICAL FLUX MONITOR

Figure 6 CANDU 6 reactor assembly

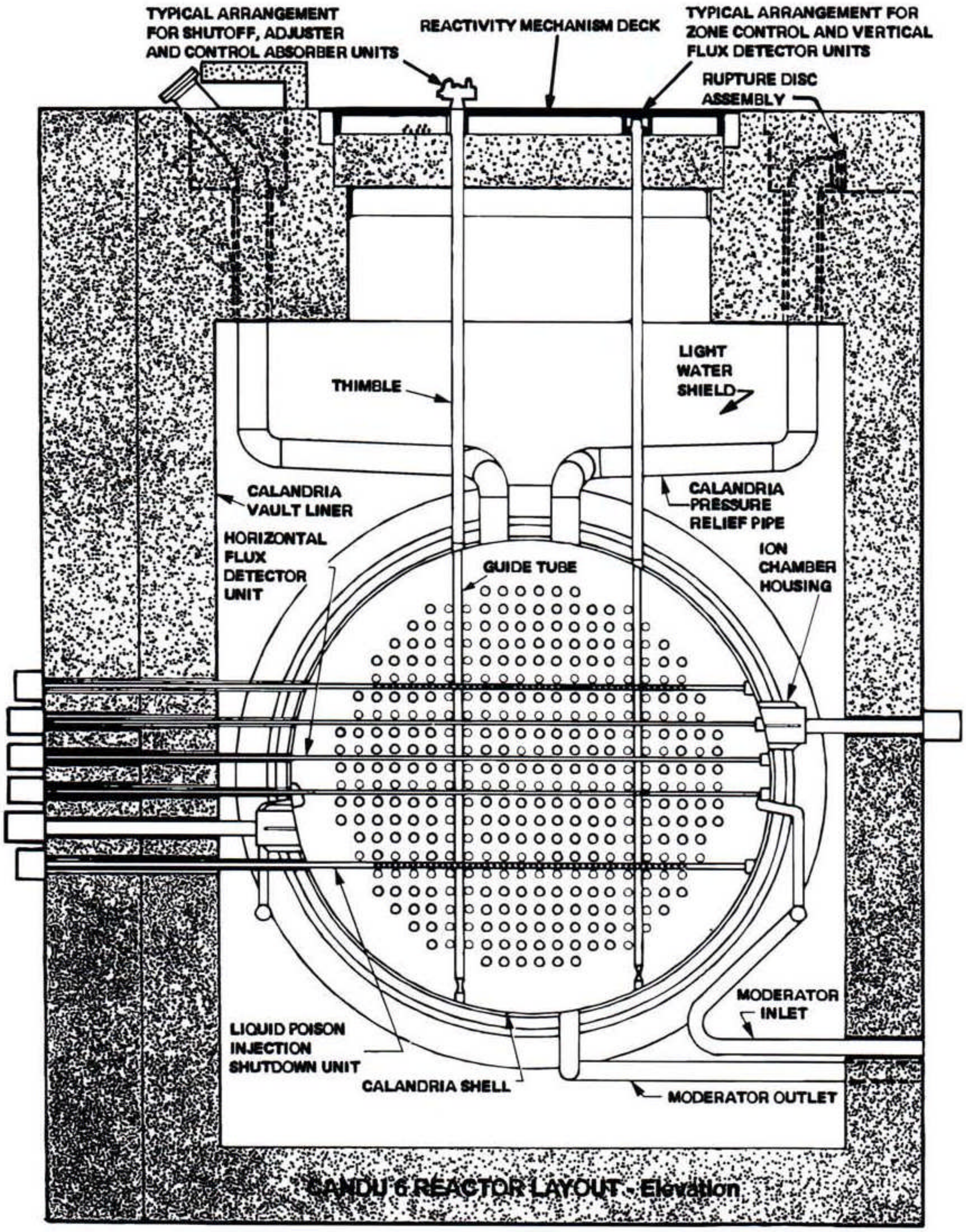


Figure 7 Reactor cross section

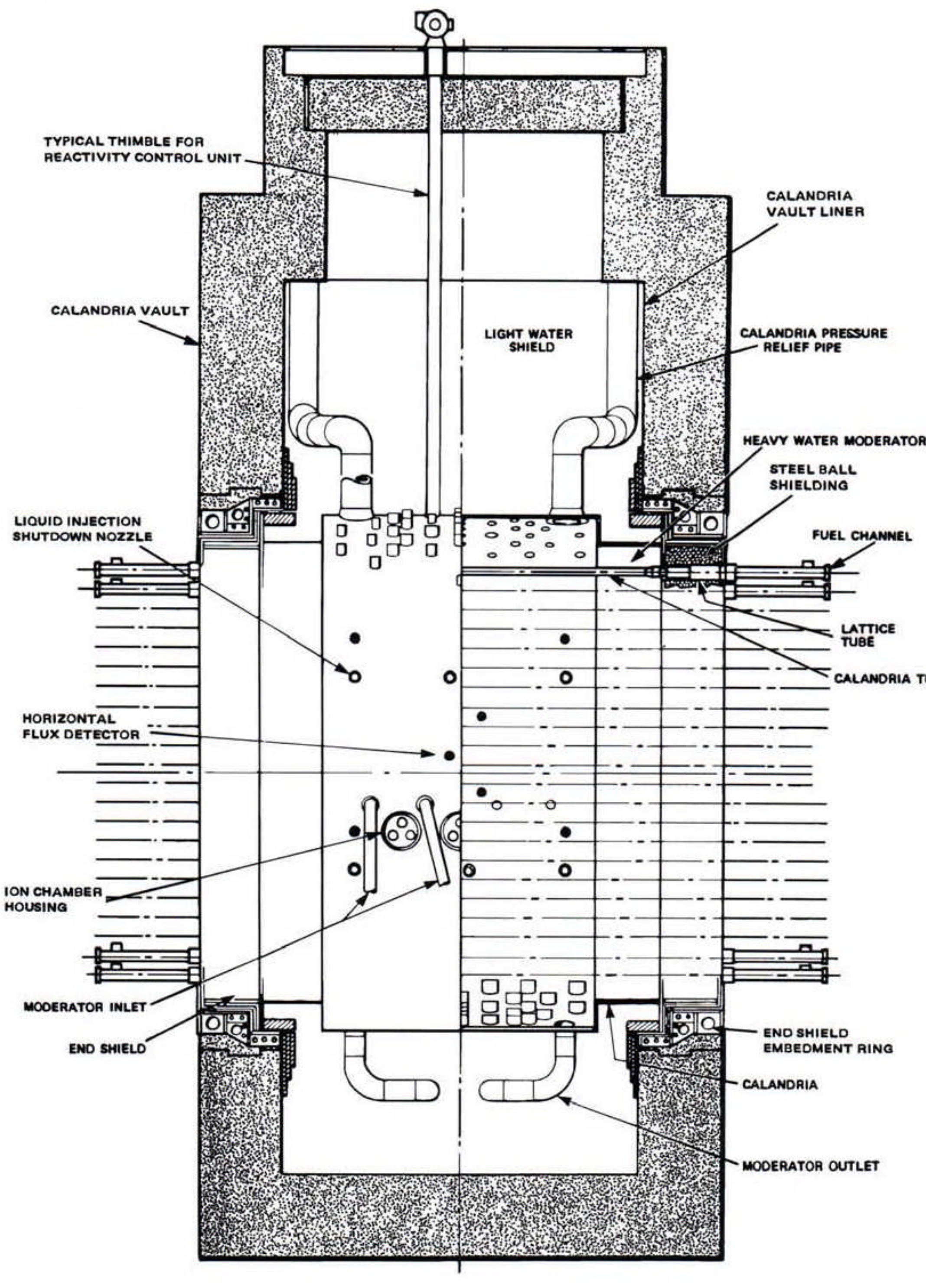


Figure 8 Reactor longitudinal section

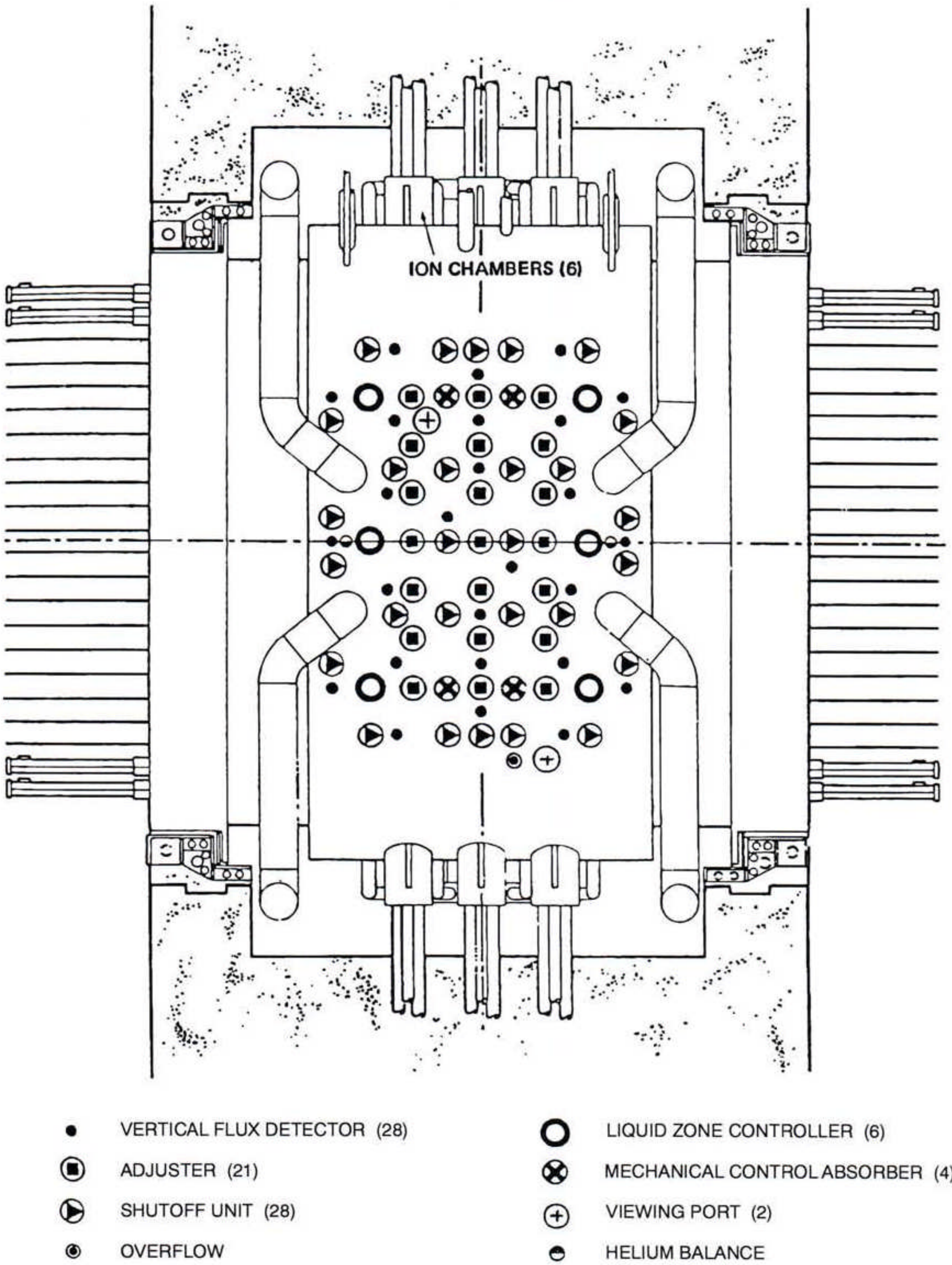


Figure 9 Reactor plan view

6.3 Coolant Loop Arrangement

The pressure tubes contain heavy water coolant at about 10 MPa and form part of the coolant loop. Flow in the pressure tubes is arranged to be in opposite directions in adjacent fuel channels. Thus, at each end of the reactor, there are sets of inlets and outlets to and from each fuel channel, as shown in Figure 10. To carry the coolant to and from each of these inlets and outlets, small diameter feeder tubes are used, as shown in Figure 11. These are linked to twin common headers above each end of the reactor core. Coolant from the outlet headers at each end passes to the steam generators, then through the coolant pumps and back to the inlet headers at the same end of the reactor core. After a second pass through the fuel channels, the coolant passes to steam generators and coolant pumps at the other end of the reactor core. The complete coolant loop thus has a double figure-of-eight configuration. Typically, there are two steam generators and two coolant pumps at each end of the reactor core, making a total of four of each. Although each pair of steam generators and its associated coolant loop has a separate figure-of-eight configuration, the headers are cross-connected so that the whole system operates at the same pressure. Pressure is maintained by a single pressurizer connected to one of the coolant loops. The pressurizer is a tall cylindrical vessel containing half water and half steam and maintained at saturation conditions. By varying the temperature in the vessel with heaters or water sprays, the pressure in the entire coolant system can be controlled. Typical parameters for 600 MWe (electrical) and 950 MWe (electrical) CANDU coolant systems are given in Table 6. Full technical details are given in Appendix C Single Unit Station Data.

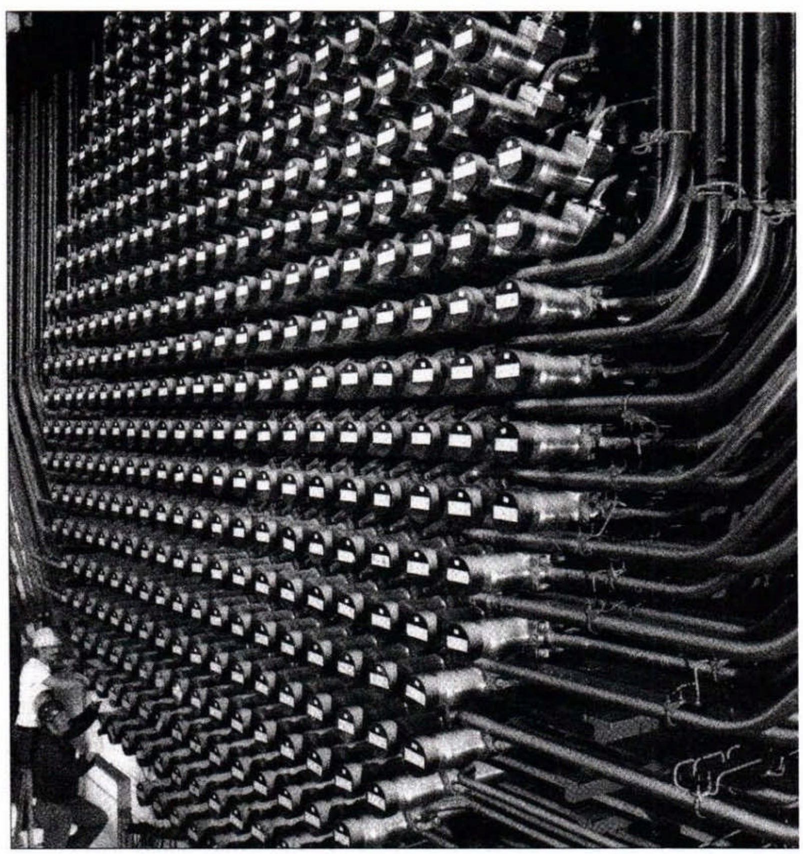


Figure 10 Fuel channel end fittings on reactor face

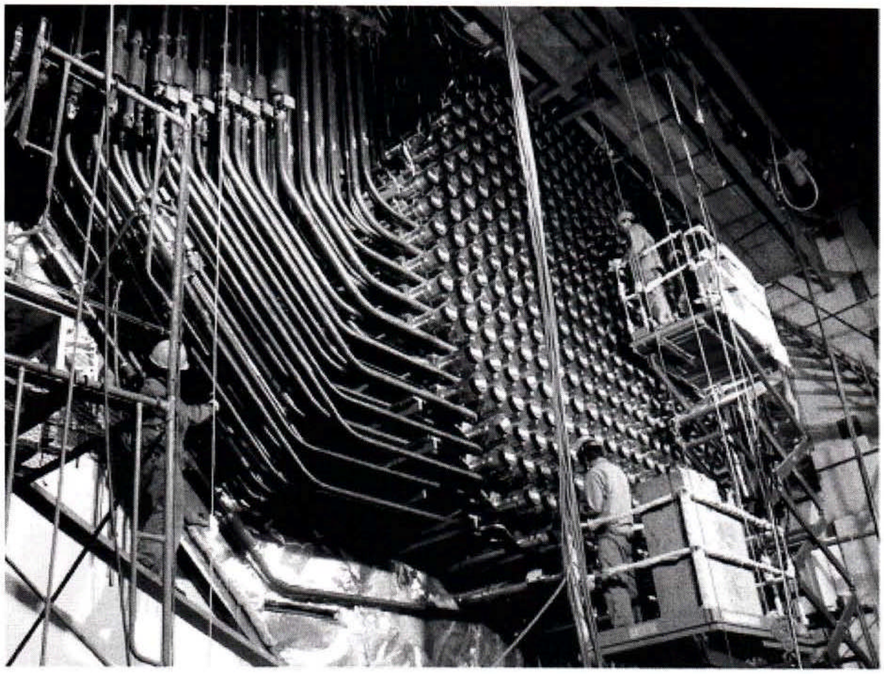


Figure 11 Feeder tube assembly on reactor face

Table 6 CANDU coolant system parameters

Parameter	CANDU 600	CANDU 900
Number of pressure tubes	380	600
Pressure tube material	Zirconium-niobium alloy	Zirconium-niobium alloy
Pressure tube diameter (ID)	103.38 mm	104 mm
Number of primary pumps	4	4
Pump flow rate (each)	2.228 m^3/s	3.959 m^3/s
Pump total heat (each)	215 m	245 m
Coolant flow rate through core	7 600 kg/s	13 500 kg/s
Coolant inlet temperature	267°C	266°C
Coolant outlet temperature	312°C	312°C
Coolant inlet pressure	11.04 MPa	11.17 MPa
Coolant outlet pressure	10.03 MPa	10.29 MPa

6.4 Steam Generators

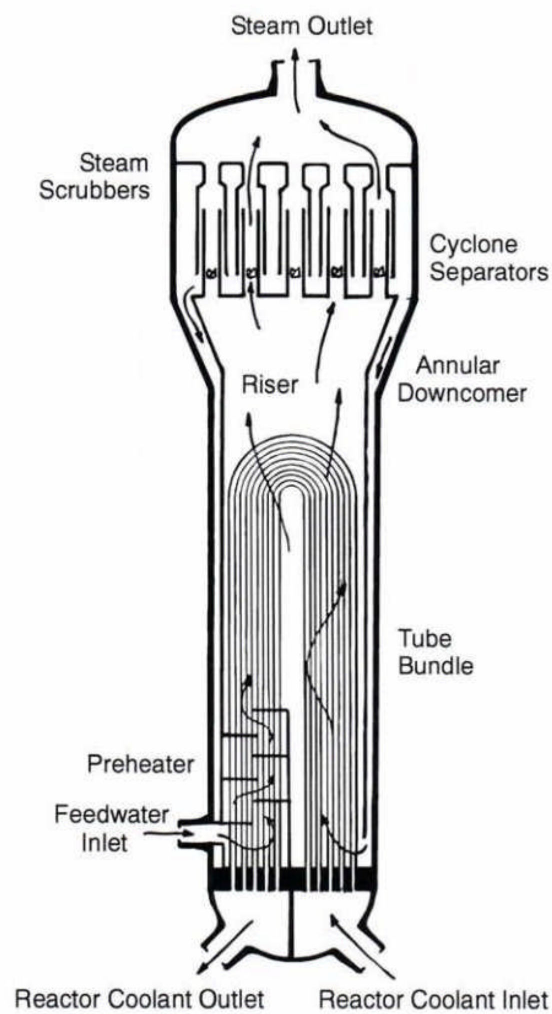


Figure 12 Steam generator for CANDU system

The purpose of the steam generators is to transfer heat from the primary coolant system to the secondary steam system and thus supply the required heat to operate the steam cycle. Like those of the PWR, the steam generators are tube or surface heat exchangers, with high pressure coolant passing inside the tubes and lower pressure steam being generated on the outside of the tubes. The steam pressure is typically 5 MPa. The usual configuration is a vertical cylinder with inverted U-tubes in the lower part and steam-water separators in the upper part, as shown diagrammatically in Figure 12. Steam generated within the tube bundle rises and promotes natural circulation. At the top of the vessel, the steam-water mixture passes through cyclone type primary separators where the steam is separated from the circulating water. The water returns down an annulus between the tube bundle and the steam generator shell. The separated steam passes through secondary separators or steam dryers to remove moisture and so improve its quality before passing out at the top of the vessel. Incoming feedwater enters at the bottom of the steam generator. This cooler feedwater is confined within baffles and made to flow in a criss-cross manner over that part of the tube bundle near the primary coolant outlet. In this way, the feedwater temperature is raised to saturation conditions before mixing with the circulating water in the steam generator. This arrangement improves thermodynamic performance, but limits the range of temperature at which feedwater can be introduced to the steam generator. A sudden drop in feedwater temperature during normal operating conditions could cause thermal shock to the vessel and tubes. Typical parameters for 600 MWe (electrical) and 950 MWe (electrical) CANDU reactor steam generators are given in Table 7.

Table 7 CANDU steam generator parameters

Parameter	CANDU 600	CANDU 900
Number of steam generators	4	8
Heat transfer capacity (total)	2 064 MW	3 258 MW
Primary side water flow (total)	7 600 kg/s	13 500 kg/s
Primary side operating pressure	~ 10 MPa	~ 10 MPa
Primary side inlet temperature	312°C	312°C
Primary side outlet temperature	~ 267°C	~ 266°C
Secondary side steam flow (total)	1 047 kg/s	1 612 kg/s
Secondary side feedwater flow (total)	959 kg/s	1 536 kg/s
Secondary side operating pressure	4.69 MPa	5.07 MPa
Secondary side inlet temperature	187°C	177°C
Secondary side outlet temperature	260°C	265°C
Steam outlet wetness	0.25%	0.25%
Number of tubes (each)		4 663
Heat transfer area per steam generator		~ 3 066 m^3
Tube diameter (OD)	22.23 mm	15.9 mm
Tube material	Inconel 600	Incoloy 800

6.5 Steam Turbines

Each CANDU unit has a single steam turbine and electrical generator on a single shaft. The steam turbine consists of a high pressure turbine receiving saturated steam from the steam generators and three low pressure turbines receiving steam, after moisture separation and reheating, from the high pressure turbine. Saturated steam from the steam generator is used for reheating, and the reheated steam becomes superheated at the lower pressure. This is required to avoid too high a moisture content in the turbine exhaust. Three low pressure turbines are required due to the large increase in steam specific volume as it expands to condenser conditions. Figure 13 shows the 600 MW steam turbine at Point Lepreau viewed from the steam inlet end, with the steam inlet valves and high pressure turbine in the foreground and the low pressure turbines further back. Figure 14 shows the same turbine from the electrical output end, with the exciter and generator in the foreground and the low pressure turbines behind them. Typical parameters for 600 MW (electrical) and 950 MW (electrical) steam turbines are given in Table 8.

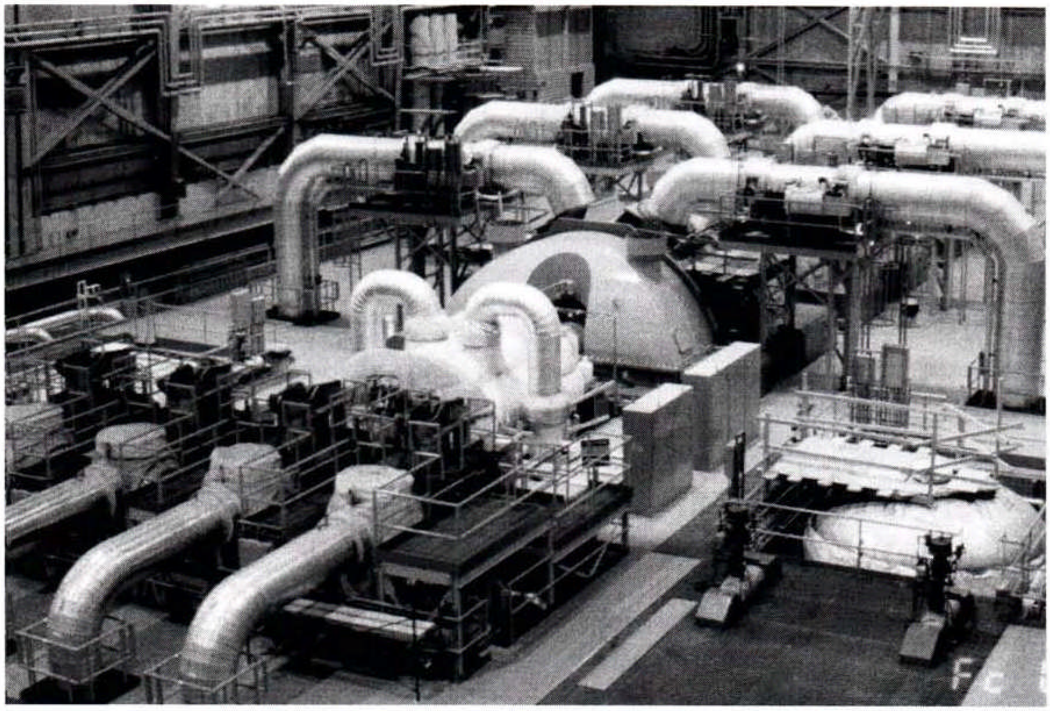


Figure 13 600 MW steam turbine for nuclear unit

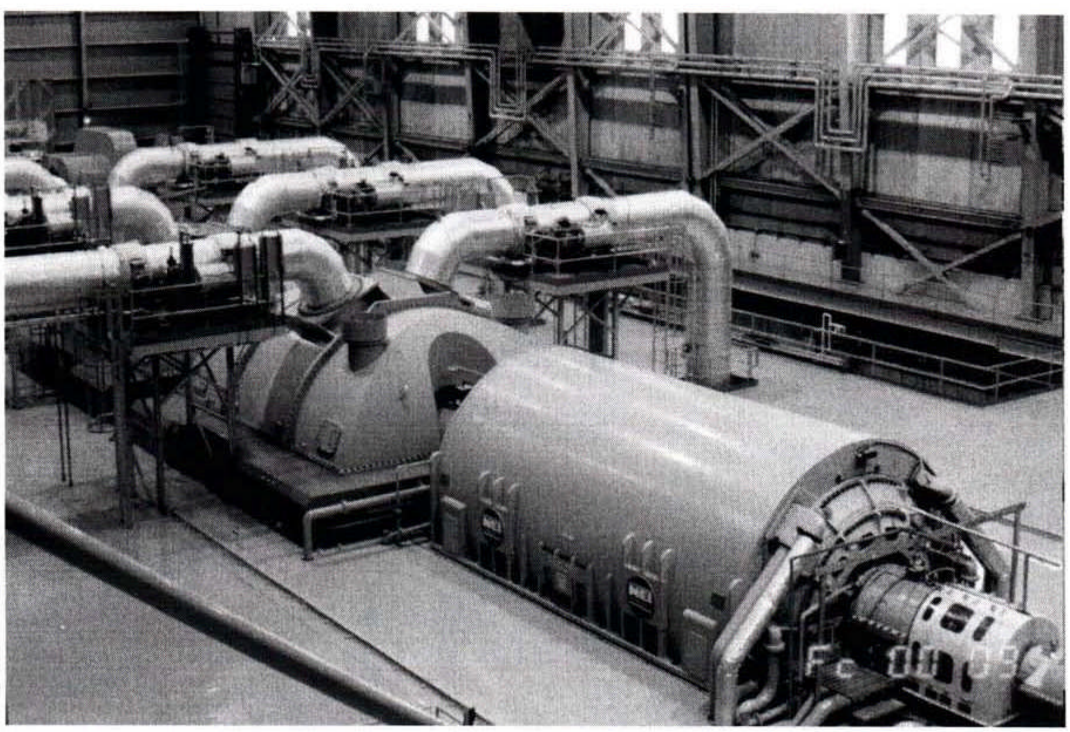


Figure 14 Turbine generator for nuclear unit

Table 8 Steam turbine parameters

Parameter	CANDU 600	CANDU 900
Number of HP cylinders	1 double flow	1 double flow
Number of LP cylinders	3 double flow	3 double flow
Main steam flow rate	1 047 kg/s	1 612 kg/s
Live steam flow to valves	957 kg/s	1 533 kg/s
Live steam flow to reheater	90 kg/s	77 kg/s
HP steam inlet pressure	4.55 MPa	4.93 MPa
HP steam inlet temperature	258°C	263°C
HP steam exhaust pressure	0.665 MPa	0.545 MPa
HP steam exhaust quality	88.2%	86.5%
LP steam inlet pressure	0.588 MPa	0.500 MPa
LP steam inlet temperature	242°C	247°C
LP steam exhaust quality	89.5%	91.0%
Steam cycle efficiency	32.8%	34.4%
Electrical generator gross output	678 MW	1 121 MW

6.6 Technical Data

The technical data in this chapter were obtained from various sources, including references in the bibliography and public relations brochures from Atomic Energy of Canada Limited, and therefore are not necessarily applicable to a particular nuclear power plant. Some data may be inconsistent or in conflict due to evolution of the design or to differences between design and operational data. Furthermore, some missing data were deduced by simple calculation for illustrative purposes and are indicated as an approximate value, meaning that what is presented is typical rather than specific.

7 Possible CANDU Reactor Development

7.1 CANDU Advantages

Refuelling can be done while the reactor is on power by pushing new fuel bundles in at one end of a fuel channel while spent fuel bundles are removed at the other end. This allows the CANDU reactors to achieve high capacity factors compared with those reactors which have to be shut down periodically for refuelling. Figure 15 shows a typical annual analysis of capacity factors for CANDU reactors and other light water reactors in the early years of their life before major maintenance or refurbishment became necessary. This shows that in that year in the top 25 reactors worldwide, with regard to capacity factor performance, there are proportionately twice as many CANDU reactors listed as in the general population of water cooled reactors. One year within the first ten years of operation Point Lepreau was actually ranked first on this list. This was due to only one short planned annual shutdown for routine maintenance of systems not able to be accessed during normal operation during the initial years of operation.

Furthermore the relative simplicity of the fuel bundles and the use of natural uranium leads to low fuel costs for CANDU reactors. With no heavy reactor pressure vessel and the possibility of modular construction the erection time of a CANDU reactor can be shorter than that of some other reactor systems. Having been developed to use natural uranium in conveniently small fuel bundles, the CANDU reactor has the ability of utilizing low enriched fuel from other sources such as light water reactors and also blended fuel from nuclear weapons. This makes it attractive as part of a fuel reprocessing and recycling scheme involving different nuclear plants and facilities.

The refurbishment of CANDU reactors has been proven. Steam generators and reactor pressure tubes can be replaced. This has the potential of effectively doubling the life of the plant enabling it to operate efficiently for 50 to 60 years. The overall capital investment including the cost of refurbishment can make the CANDU reactor an attractive investment for bulk base load power production.

AECL EACL

The Top Twenty-five

Lifetime World Power Reactor Performance to September 30, 1995* *from among 370 reactors over 150 MW*

Rank	Country	Unit	Type	Year of First Power	Capacity Factor %†
1	Germany	Emsland	PWR	1988	91.4
2	Germany	Neckar 2	PWR	1989	89.2
3	Canada	Point Lepreau	CANDU	1982	88.3
4	Germany	Grohnde	PWR	1984	88.1
5	Canada	Pickering 8	CANDU	1986	88.0
6	Canada	Pickering 7	CANDU	1984	87.6
7	Belgium	Tihange 3	PWR	1985	87.5
8	Finland	Loviisa 2	PWR	1980	86.8
9	Hungary	Paks 2	PWR	1984	86.2
10	Switzerland	Beznau 2	PWR	1971	85.7
11	Germany	Philippsburg 2	PWR	1984	85.5
12	Hungary	Paks 4	PWR	1987	85.4
13	Canada	Darlington 3	CANDU	1992	85.4
14	Canada	Darlington 4	CANDU	1993	85.2
15	Hungary	Paks 3	PWR	1986	85.2
16	Canada	Pickering 6	CANDU	1983	84.6
17	Switzerland	Gösgen	PWR	1979	84.2
18	Germany	Grafenrheinfeld	PWR	1981	83.9
19	Finland	TVO 1	BWR	1978	83.8
20	Spain	Cofrentes	BWR	1984	83.7
21	Spain	Trillo 1	PWR	1988	83.3
22	Spain	Almaraz 2	PWR	1983	83.2
23	Finland	Loviisa 1	PWR	1977	83.2
24	Korea	Wolsong 1	CANDU	1982	83.2
25	Finland	TVO 2	BWR	1980	83.1

*Source: Nuclear Engineering International

†Capacity Factor = $\frac{\text{actual electricity generation}}{\text{perfect electricity generation}}$

Figure 15 Typical capacity factors of CANDU and other reactors

7.2 Future Prospects

Future prospects for the CANDU reactor are good. About one third of CANDU reactors in operation are of the CANDU 6 type and all have performed well. The few larger reactors have also provided good service. A CANDU reactor would be a good choice for any future nuclear reactors built in Canada. However it would be in competition with light water reactors especially the PWR where later designs are under construction.

Of particular note is the modular construction proposed for new reactors. The advantage that this modular construction offers is a decreased construction time and hence lower costs. It also allows specific parts of the reactor to be more easily replaced during the life of the plant.

New fuel bundles with 43 fuel elements have been developed and tested. The increased surface area and smaller diameter elements resulting from this change allow for increased heat transfer and hence more power per bundle. Such design developments are able to reduce the capital and operating costs of the current CANDU system.

7.3 Safety Aspects

The primary way to avoid reactor accidents is through use of duplicate safety systems, core cooling systems, and engineered safeguards built into the reactor design. The CANDU reactor has evolved to be inherently safe, with consideration given to all conceivable accidents. Accidents to early reactors, as mentioned earlier, have highlighted the need for very conservative and safe design requirements.

In case of an adverse transient which could lead to a potential accident, there are two independent shutdown systems. The SDS1 system consists of mechanical shutoff rods which drop into the core by gravity when a trip signal is received. The SDS2 system consists of an array of nozzles which inject gadolinium nitrate under gas pressure into the moderator on receipt of a trip signal.

In the event of loss of coolant from the reactor heat transport system, an emergency core cooling system injects cooling water into the headers to ensure a water supply to maintain fuel cooling. This is backed up by alternative systems to maintain cooling for an extended period.

Use of multiple pressure tubes instead of a single large pressure vessel permits thinner walls and simpler manufacture to the required pressure threshold. Any leaks can be detected by monitoring moisture content and pressure in the gap between the pressure tube and the calandria tube, which is done on a continuous basis. When detected, a faulty pressure tube can be readily replaced.

The heavy water moderator and reflector in the calandria surrounding the pressure tubes are at a relatively low temperature compared with that of the coolant flowing in the pressure tubes. Because of this lower temperature, the moderator and reflector act as an energy sink in case of certain reactor accidents. This heat sink is an important feature in the CANDU design because it means that if the emergency core cooling system fails, some heat can be transferred from the fuel to the moderator. The calandria itself is surrounded by light water in the reactor vault, thus creating an additional short term heat sink for the moderator.

Most important are the engineered safeguards that protect the public from possible release of fission products in the event of a component failure. As in most nuclear power plants, there are four barriers that prevent the release of significant quantities of fission products to the environment. These barriers are: the fuel itself, the fuel cladding, the primary heat transport system boundary, and the containment building.

The first barrier is the uranium dioxide fuel, which is chemically inert even in high temperature water and has a high melting point. Even under high temperature conditions, about 99% of all radioactivity is trapped in the uranium dioxide matrix. This means that almost all solid fission products are contained in the fuel under normal non-melting conditions.

The second barrier is the fuel cladding, which is a zirconium alloy sheath. This sheath is designed to withstand the stress associated with fuel expansion and buildup of trapped fission gases. Zirconium would be subject to damage should dryout and elevated temperatures occur in the fuel channel. Provided the fuel bundles are kept flooded with coolant during accident conditions, this barrier will remain intact.

The third barrier is the primary heat transport system boundary. This provides containment for the coolant, which may contain fission products in the event of fuel cladding leakage. The primary circuit is a closed loop and does not allow any fission products to go any further unless it in turn has a leak.

The fourth and final barrier to fission product release is the prestressed low leakage concrete containment building. The building is maintained under a slightly negative pressure and is lined with a plastic coating which limits the leakage of fission products in the event of overpressure due to an accident. In addition, air exchange filters in the ventilation system remove any fission products in the circulating air which is discharged to the atmosphere.

7.4 The Advanced CANDU Reactor

A possible new development of the CANDU reactor is the Advanced CANDU Reactor (ACR 1000), with a nominal net electrical output of 1000 MWe. The general arrangement is very much the same as the basic CANDU reactor, as shown in Figure 16, but some significant technical changes have been included.

These reactors have been designed so that they can be built within a four-year period and have an expected plant life of 60 years, with an overall lifetime capacity factor of over 90%. They use light water instead of heavy water as coolant, thus simplifying several supporting auxiliary systems. This necessitates use of low enrichment fuel to compensate for increased neutron absorption in the coolant. The fuel bundles are of the CANFLEX 43 element design, as shown in Figure 17, which provides increased power output per bundle and hence per channel. This design has fuel elements of two different sizes. There are 8 central elements 13.5 mm in diameter and 35 outer elements 11.5 mm in diameter. A quick comparison between the CANDU 6 and Darlington Generating Station shows the general trend of the conventional CANDU design towards larger capacities, while a comparison between Darlington and the ACR 1000 shows somewhat of a reversal of this trend with current technical changes. Table 9 shows some key technical parameters which illustrate this trend.

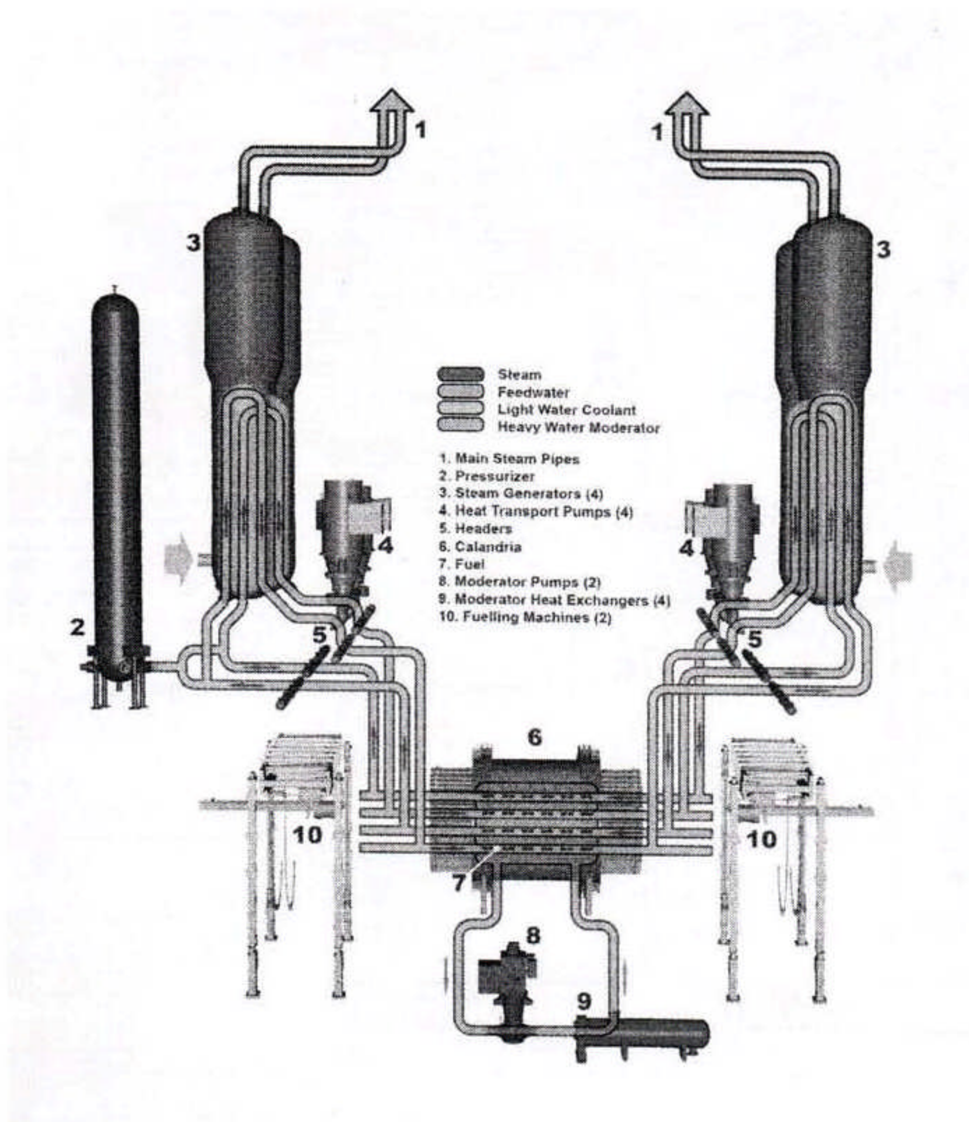


Figure 16 ACR 1000 nuclear systems schematic

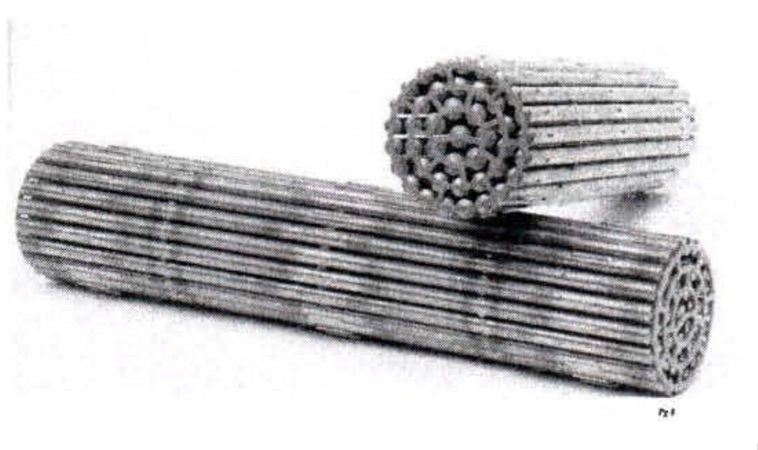


Figure 17 ACR CANFLEX fuel bundles

Table 9 Advanced CANDU reactor parameter comparison

Parameter	Units	CANDU 6	Darlington	ACR 1000
Reactor power output	MWt	2 064	2 657	3 187
Moderator		D_2O	D_2O	D_2O
Coolant		D_2O	D_2O	H_2O
Fuel		Nat UO_2	Nat UO_2	Enriched UO_2
Heavy water inventory in moderator	Mg	265	312	250
Heavy water inventory in coolant	Mg	192	280	0
Number of fuel channels		380	480	520
Number of bundles per fuel channel		12	13	12
Number of elements per bundle		37	37	43
Fuel burnup	MWd/tU	7 500	7 791	20 000
Calandria diameter	m	7.6	8.5	7.5
Lattice pitch	mm	286	286	240
Pressure tube wall thickness	mm	4	4	6.5
Inlet header pressure	MPa	11.2	11.3	12.5
Inlet header temperature	°C	260	267	275
Outlet header pressure	MPa	9.9	9.9	11.1
Outlet header temperature	°C	310	310	319
Maximum channel flow	kg/s	28	27	28
Number of heat transport pumps		4	4	4
Pump motor rating (each)	MWe	6.7	9.6	10.0
Pump rated flow (each)	m^3/s	2.228	3.240	4.300
Number of steam generators		4	4	4
Tube diameter	mm	15.9	15.9	17.5
Steam pressure	MPa	4.6	5.0	5.9
Steam temperature	°C	260	265	276
Steam quality	%	99.75	99.75	99.90
Net power to turbine generator	MWt	2 060	2 650	3 180
Steam cycle efficiency		35.3	35.3	~ 36.6
Gross electrical power output	MWe	728	935	1 165
Net electrical power output	MWe	666	881	1 085
Turbine inlet steam temperature	°C	258	263	273
Final feedwater temperature	°C	187	177	217
Condenser vacuum	kPa	4.9	4.2	4.9

The table clearly shows the evolution of the ACR 1000. The 43-element fuel bundle provides an average channel power of 6.13 MW (12 bundles) as opposed to 5.43 MW (12 bundles) and 5.54 MW (13 bundles). This in turn enables fewer channels to be used for an equivalent output, resulting in a smaller reactor for the same output. Furthermore, the channel pitch has been reduced by 16% resulting in an even smaller reactor. The overall result is that the ACR 1000 reactor is about the same size as the CANDU 6, as illustrated in Figure 18. The heavy water inventory has been reduced accordingly. The light water coolant eliminates a pressurized heavy water circuit altogether, thereby effecting a further and major saving in inventory and heavy water support systems. The low enriched uranium increases the burnup by a factor of about 2.5 and consequently decreases the amount of spent fuel. An additional reserve water system provides a passive safety feature, making this a Generation III+ design. Other than these important changes, there are many small improvements in the general operating conditions of the heat transport system and steam cycle, giving an improvement in overall efficiency from 35.3% to 36.6%, which ultimately saves on fuel consumption.

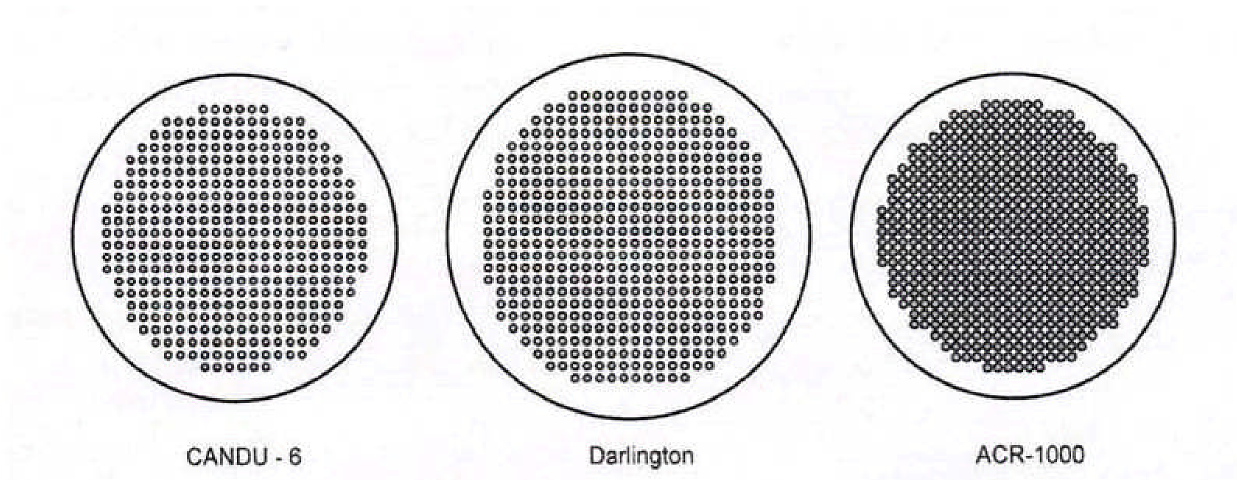


Figure 18 Comparison of core sizes

8 Problems

1 Sketch a typical figure-of-eight CANDU heat transport system showing all key components and describe the functions of these components.

2 Sketch a typical CANDU reactor core and show in the sketch where, with respect to the fuel channels, the various control devices are installed. Identify the devices and state their purpose.

3 Describe the structure and characteristics of a CANDU fuel channel and the fuel within the channel. Describe also what happens in the fuel channel as the coolant flows through it.

4 Sketch a CANDU steam generator showing the key components and explain the function of each component and why they are arranged in this particular configuration.

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