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COMMISSIONING AND OPERATING EXPERIENCE WITH CANADIAN NUCLEAR-ELECTRIC STATIONS

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ABSTRACT

Fourteen Canadian-designed nuclear-electric generating units are operating or under construction, with a total capacity of more than 6 000 megawatts:

	MW(e)	Start-up
NPD	22.5	1962
Douglas Point	208	1967
Gentilly	250	1971
Pickering	4 x 508	1971
Bruce	4 x 752	1975
KANUPP (Pakistan)	125	1971
RAPP (India)	2 x 200	1972

NPD has operated reliably for nine years both as an energy producer and as a facility for training and technical development. An average capacity factor of 96% was achieved in five successive winters. In 1968 the NPD heat-transport system was converted from pressurized heavy water to boiling heavy water.

During the four years of operating experience at Douglas Point, data have been collected on heavy-water upkeep, heat-transport chemistry, radiation fields, performance of mechanical components (pumps, valves, turbine) and the main causes of lost production. Fuel performance is described in a separate paper at this Cᴄnference.

Gentilly, the first CANDU reactor with a direct-cycle heat-transport system (boiling light water), went critical on 12 November 1970. Low-power reactivity characteristics and early experience at high power, particularly transient behaviour, will be described.

Discussion of Pickering and KANUPP will concentrate on the experience obtained during commissioning, particularly the containment system operating under negative pressure at Pickering, and the initial operating experience.

At both NPD and Douglas Point, we have experienced severe failures of pumps in the heat-transport systems. Corrective action has, however, been very effective.

On-power refuelling has been routine at NPD since November 1963 and at Douglas Point since March 1970. In November 1969 improved "Mark II" fuelling machines were installed at NPD and resulted in easier maintenance and greatly improved reliability.

Heavy-water management has continued to receive priority attention. Experience with mechanical equipment, containment methods and recovery equipment at NPD and Douglas Point has provided the information on which to base improved systems for Pickering and Bruce.

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Fourteen Canadian-designed nuclear-electric generating units [1] are operating or under construction, with a total net capacity of more than 6 000 MW(e).

* Abbreviated to "Ontario Hydro" in the text of this paper.

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	MW(e) net	First Electricity
NPD	22.5	1962
Douglas Point	208	1967
Pickering	4 x 508	1971
Gentilly	250	1971
KANUPP (Pakistan)	125	1971
RAPP (India)	2 x 200	1972
Bruce	4 x 752	1975

Except for Gentilly, each unit is characterized by

- heavy water for moderator and heat transport, giving best possible neutron economy;
- natural-uranium oxide as fuel, giving extremely low fuelling cost (0.7 m$/kWh for Pickering) with little dependence on the cost of uranium and no requirement to reprocess the spent fuel;
- pressurized heavy water in the primary heat-transport system; natural-water steam generated through an indirect cycle;
- Zircaloy pressure tubes to contain the fuel; hence no massive pressure vessel; and
- refuelling at full power.

At Gentilly, the reactor follows a prototype design, with boiling natural water as the heat-transport medium. The steam produced in the reactor pressure-tubes is separated in steam drums and then passed directly to the turbine.

This paper summarizes our experience in operating or commissioning NPD, Douglas Point, Pickering and Gentilly. The current status for KANUPP, RAPP and Bruce is treated more briefly. Fuel performance at NPD and Douglas Point, which has been very good, is described in companion papers [2,3] at this Conference.

NPD (Nuclear Power Demonstration)

NPD is a 22.5-MW(e) pilot plant; its early performance was reported to the Third Conference [4]. The station was designed by the Canadian General Electric Company Limited and Ontario Hydro, is owned jointly by Atomic Energy of Canada Limited (AECL) and Ontario Hydro, and is operated by Ontario Hydro.

In excellent fashion, NPD has fulfilled three main missions:

- demonstration of the concept;
- development of staff for the on-going program, in Canada and abroad; and
- research and development to improve existing and future designs.

Operating summary

NPD became critical on 11 April 1962, produced full power on 28 June 1962 and was declared in-service on 1 October 1962. In May 1966 its output was increased from 19.5 to 22.5 MW(e) net.

Because NPD is used both for demonstrating performance and for executing development programs, its operating schedule has been divided between "Capacity Runs" and "Improvement/Test Periods".

In 13 Capacity Runs, totalling 61 months, the net capacity factor[1] has averaged 82%. In its original form, NPD reached maturity in 1966 with a net capacity factor of 88% for the full year (of which only three months were occupied by a Capacity Run). Since then, less emphasis has been placed on demonstrating performance, and more has been placed on technical development and staff training. In the period from 1 October 1962 to 31 March 1971 the net capacity factor was 60.2%.

In the summer of 1968, the heavy-water heat-transport system was converted from the pressurized (non-boiling) regime to a boiling regime with an average outlet steam quality of 12 mass %.

In 1969 the original prototype fuelling machines were replaced by a "Mark II" design, with major improvements based on experience with the original machines.

On-power fuelling

Refuelling at full power has been carried out safely and reliably at NPD since 24 November 1963. To the end of 1970, 1 680 fuel bundles had been loaded at high power.

NPD was the first pressurized-water power reactor in the world to have the capability of on-power fuelling. Because of their prototype nature, the original fuelling machines required excessive maintenance to retain adequate reliability. Nevertheless, capacity factor reduction due to fuelling-machine problems, which had averaged 4.3% in 1964—1966, had dropped to 1% in 1966 and was zero in 1967—1969.

In 1966, the design of the new "Mark II" machines was started and was based on nearly five years of experience with the original machines. The new machines, which were installed in 1969, have met our reliability and maintenance requirements; the same basic improvements combined with lessons from Douglas Point will be included in the design of the machines for Bruce.

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[1] Net capacity factor (%) = $\frac{\text{net MWh generated}}{\text{net rated MWh}} \times 100$. When the station is shut down, the capacity factor is negative because electricity is still consumed by auxiliary equipment. The method of calculation penalizes for <u>all</u> causes of lost production, including deliberate lost production for research, development and training.

Heat-transport pumps

These are vertical centrifugal pumps with double face-type mechanical seals to contain the heavy water. They have performed well except that, initially — as reported in 1964 [4] — the seals failed after an intolerably short life (typically 200 h). These premature failures were caused by gas coming out of solution at the seals, thus destroying the cooling and lubricating effects of the heavy water.

By a combination of careful venting, de-gassing, design improvements, and minimizing the start—stop cycles, seal failures have been virtually eliminated as a cause of lost production; indeed, no outages have been caused by seal failures since 1968. Several modified seals have been developed, and those currently in service are in good condition after running for periods of 9 000—27 000 h.

Heavy-water upkeep

The cost of heavy-water upkeep is the sum of the costs of

— replacing heavy water that is lost, and

— upgrading heavy water that is recovered at less than reactor-grade isotopic purity,

but it is convenient to treat the costs of upgrading in terms of a "downgrading equivalent loss". Hence, we use the equation

Upkeep Rate = Loss Rate + Downgrading Equivalent Loss Rate.

In the eight years 1963—1970, the loss rate has averaged 0.4 kg/h and the "downgrading equivalent loss rate" has averaged 0.41 kg/h, giving a total upkeep rate of 0.81 kg/h. In 1970 the figures were 0.33 kg/h for the loss rate, 0.41 kg/h for the "downgrading equivalent loss rate" and 0.74 kg/h for the total upkeep rate.

Progressive application and improvement of driers that recover heavy-water vapour and improvements in the arrangements for sealing the rooms that contain heavy-water systems have increased recovery efficiency from 68% in 1963 to over 96% in 1970. When recovery efficiencies are high and there is spare drier capacity, the upkeep costs become relatively insensitive to the rate at which heavy water escapes into various rooms — provided, of course, that ingress of natural water is kept small. This fact is being exploited at Douglas Point and NPD and in the design of Pickering and Bruce.

A total upkeep rate of 0.8 kg/h for heavy water costing $60/kg represents a unit energy cost of 0.12 m$/kWh for a 500-MW(e) unit operating at 80% net capacity factor. With known improvements in mechanical joints, room sealing, vapour-recovery equipment, and the segregation of heavy-water and natural-water systems, we expect that large commercial stations such as Pickering and Bruce, will have, at maturity, heavy-water upkeep costs representing not more than 0.2 m$/kWh.

Conversion to boiling

In the summer of 1968, the heavy-water heat-transport system of NPD was converted from the pressurized (non-boiling) regime to a boiling regime in order to gain experience that might be applied in future large stations.

The conversion was made without unloading the fuel and without changing the thermal power, the turbine steam conditions or the electrical output. The heavy water leaving each of the 132 pressure tubes is partially boiling — the maximum and average steam qualities at full-power being 22 mass % and 12 mass %, respectively. On-power refuelling continues, and return to the non-boiling regime (planned for 1971) is easily carried out. The main changes were

- a reduction in pressure to permit boiling;
- a reduction in heavy-water flow rate (by changing the channel orifices);
- a larger pressurizer tank to handle the swell caused by steaming;
- changes to the reactor regulating and protective systems (e.g., to cope with the positive reactivity effect of boiling); and
- the use of ammonia instead of lithium hydroxide for pD control.

The conversion was started in June 1968 and high-power operation was achieved in September. The station operated in the boiling mode at full power in the winter peak period from 1 December 1968 to 28 February 1969 with a net capacity factor of 87%.

Manpower development

Throughout its life, and particularly since 1966, NPD has been used extensively as a "training laboratory" for Ontario Hydro's Nuclear Training Centre (NTC) which is located at the same site.

Almost all the 260 engineers, 255 operators and 315 maintenance staff recruited by Ontario Hydro to operate nuclear-electric generating stations in Ontario have spent periods from a few weeks to several years at NPD. In addition, key staff from Hydro-Québec (Gentilly), India (RAPP) and Pakistan (KANUPP) have received extensive training at NPD/NTC.

As an indication of the magnitude of the training effort in this very small station, it may be noted that a staff of 85 could carry out the current work program, exclusive of training, while fewer than 60 would be required for operation solely to produce electricity.

DOUGLAS POINT

Douglas Point is a full-scale 208-MW(e) prototype owned by AECL and operated by Ontario Hydro. It was the first unit designed jointly by AECL and Ontario Hydro and is the forerunner for RAPP, Pickering and Bruce.

Douglas Point is now in the fourth year of what is expected to be a six-year period of immaturity. In the first three years, many major problems that affected production reliability were identified and related to faults in design, manufacture or construction, for which solutions have been or are being developed.

Early performance has not been at a level that would be acceptable for large commercial stations over a long term. However, Douglas Point is effectively a "first design" — the team that designed it was not the team that designed NPD, and the design did not benefit significantly from operating experience at NPD. It therefore represents an early step on the "learning curve".

However, operating experience at Douglas Point has had significant effect on the design of Pickering, even though Pickering was committed three years before Douglas Point started up; it has had even more impact on the design of Bruce.

Operating summary (to 31 March 1971)

Douglas Point became critical on 15 November 1966, generated electricity for the first time on 7 January 1967 and was declared in-service on 26 September 1968. Production performance since then is indicated by the following performance factors:

	Net Capacity Factor	Operating Factor[2]
	%	%
1968	42.7	50.6
1969	20.7	31.5
1970	45.5	59.0
1971 (to 31 Mar)	60.3	68.5
26 Sep 1968 — 31 Mar 1971	36.7	47.8

The reduction of power output — 19 MW(e) — that resulted from the removal of faulty turbine blades is one of the main reasons for the difference between the net capacity factor and the operating factor; another was the inefficient fuelling program that gave an undesirable power distribution pattern until on-power fuelling began in March 1970.

Before on-power fuelling began, the station was often shut down at weekends for refuelling and for tests and modifications in preparation for on-power fuelling. After it began (on 1 March 1970), shutdowns for refuelling became unnecessary and the power-generating performance has consequently improved. The net capacity factor from 1 March 1970 to 31 March 1971 has been 51.5%, and the operating factor has been 63.1%. In the same period, 1 948 new fuel bundles have been loaded into the reactor, 79% of them when the reactor was at high power.

[2] Operating Factor = $\frac{\text{Hours producing electricity}}{\text{Total elapsed hours}}$ x 100. The method of calculation penalizes for all non-operating time, for any cause including research, development and training.

Of the total lost production since 1 March 1970, about 55% has been due to the nuclear steam-supply systems and 45% due to the conventional systems.

Reactor components

Until now (April 1971) only one reactor component has failed significantly. In 1967, the rubbing of a vertical reactivity control mechanism (booster rod) flow tube against a horizontal calandria tube caused a hole in the calandria tube, permitting leakage of moderator heavy water. The reason was determined and a simple positive cure was provided. The failed calandria tube (and the associated pressure tube and end-fittings) were removed and replaced.

The ease of replacement of whole channels is, of course, a significant advantage of the pressure-tube reactor. But, because the necessary tools and trained staff were not immediately available, the total shutdown time — to identify the failed tube, develop and procure special tooling, and replace the channel — was about nine weeks. Once the tools were available, however, the time at the reactor for replacing the channel was only 22 h, 35 min.

Heat-transport circulation pumps

In 1967, major failures of the ten heat-transport pumps caused two extensive periods of lost production, one lasting 8 weeks and the other lasting 21 weeks, for redesign and rebuilding. There was no single cause for these failures; the multiple causes included:

- excessive radial thrust causing the pump bearings to fail; this was due to lower-than-expected flow resistance in the heat-transport system, and hence a higher flow in the pump than it had been designed for; the problem was overcome by trimming the pump impellers;
- inadequate design and lubrication of motor bearings;
- pump-to-motor misalignment;
- unbalance of the motor flywheel;
- material distortion caused by excessively hot heavy water in the pump glands and bearings; this was caused by inadequate design of the gland-cooling systems; and
- unexpected upward shaft movement due to hydraulic forces and resulting in excessive wear of seal faces; seal redesign was necessary.

The action taken at the time and subsequent improvements to the seals have been successful. Pumps are now in service after operating for more than 20 000 h, and inspection indicates that at least 30 000 h can be expected without maintenance. However, the rebuilt pumps contain Stellite components (the heavy-water-lubricated guide bearings and the shaft seal faces). Since wear of the Stellite is believed to be a significant cause of high ^{60}Co radiation fields, a systematic program is underway to replace it with AISI-type 420 stainless steel for the guide bearings and with titanium carbide for the shaft seal faces. At the same time that these changes are made, the motors are being rewound as the original motor insulation has proved to be short-lived.

Sudden forced outages

A sudden forced outage is one for which no warning can be given to Ontario Hydro's System Control Centre. In 1969 there were 31 such events, and in 1970 there were 23. This is an intolerable frequency (one or two per year would be acceptable).

Many of the outages were caused by unnecessary reactor trips automatically produced by signals indicating high temperature plus low flow in any of the 306 fuel channels. Trips of this kind were provided to respond to a "loss of cooling" condition that might lead to an accident, but, in all cases, they were in fact caused by instrument faults — often transient. The need for this protection has been re-evaluated; since September 1970, with the approval of the Atomic Energy Control Board, signals from these sources provide only an indication or an alarm and no longer cause the reactor to trip.

Most of the other outages were caused by signals that trip the reactor because of low pressure in the heat-transport system. The heat-transport system pressure is controlled by addition (feed) and removal (bleed) of heavy water. Thermally it is tightly coupled to the secondary steam system and its feedwater supply; it is therefore very sensitive to changes (as, for example, in feedwater flow or turbine steam flow) on the secondary side. Despite several improvements in pressure control, trips for low pressure in the heat-transport system have continued at an unacceptable frequency.

Since this low-pressure trip was provided to respond to conditions arising in a "loss of coolant" accident, particularly a major pipe rupture, and since the pressure transients that are experienced in normal operation are not harmful, alternative methods of sensing major ruptures have been sought. Recently we decided to eliminate the low-pressure trip and to respond to "loss of coolant" by sensing large changes in the gross flow rate of the heat-transport heavy water. Since gross flow is very stable, we expect the frequency of sudden forced outages to be acceptable in future.

Fuel defects, identification, and removal

To 31 March 1971, only 39 fuel bundles (about 0.6% of the total irradiated) have been removed from the reactor because of cladding defects [3].

The defective bundles are identified by a delayed-neutron monitoring system and are promptly removed from the reactor while it is at high power. As a result, fuel defects have had very little effect on electricity production, and the radiation fields and radioactive contamination levels due to fission-product release are minimized.

Heavy-water upkeep

The design of Douglas Point was based on the assumption that heavy-water systems could be made leak-tight and could be kept leak-tight. As a result, little attention was paid to providing efficient means for recovering chronic escapes, or to minimizing the downgrading that is caused by natural water escaping into areas occupied by heavy-water systems. In practice, the escape of heavy water from the high-pressure high-temperature parts of the heat-transport system has been significant.

In 1967 it was decided to apply the experience obtained at NPD and to convert areas containing heavy-water equipment into sealed rooms from which vapour could be recovered with molecular-sieve air driers, and to improve the leak-tightness of both natural-water and heavy-water equipment. This work — which mainly involves the fuelling-machine vaults and the boiler room — continues as more experience is gained.

The fuelling-machine vaults contain the 612 pressure-tube closure plugs, 612 feeder-to-pressure-tube connections, and several hundred connections for flow, pressure and fuel-defect-location measurements. Typically the heavy water escapes at 15—20 kg/h, and it is recovered at an isotopic purity of 75—99 mass %.

The boiler room contains a very large number of potential leak points for both heavy water and natural water. While the heavy-water equipment was built to minimize escape, the natural-water equipment was not. While, typically, the heavy water escapes at only about 5 kg/h, the isotopic purity on recovery is generally low, typically 20 mass %.

Major chronic leaks of both heavy water and natural water in the boiler room have come from packed-stem valves (at glands and bonnet joints) and from flanged pipe joints. During 1970 a new problem — the failure of boiler blowdown lines (natural water) — caused several outages and severe downgrading of the recovered heavy water. There are eight boilers, each with ten heat-exchanger sections and six blowdown lines per section — a total of 480.

In these, more than 40 poor-quality welds between carbon steel and Monel have failed and, potentially, all 480 lines may fail. The failed welds have been repaired or plugged until a better remedy can be found, but because radiation fields are high and accessibility is poor, repairs are costly in terms of radiation exposure. In April 1971 we plan to remove all blowdown lines with special rapid techniques, and at a later date to install a modified blowdown system.

A continuing program to improve or eliminate poor components and to improve the capability to recover heavy water is resulting in a steady reduction of the upkeep costs. Currently the overall recovery efficiency is about 95% and the total upkeep rate is about 4 kg/h, or about 1.5 m$/kWh at 80% capacity factor and today's price of $60/kg for heavy water.

In 1970 the average total upkeep rate was 4.15 kg/h made up of a loss rate of 1.65 kg/h and a "downgrading equivalent loss rate" of 2.5 kg/h. While this is a factor of six higher than desired at Douglas Point, the principal causes are known and believed to be largely overcome at Pickering and Bruce.

Chemistry of heat-transport system

The heat-transport system is constructed of several metals:

Main piping	Carbon steel
Pressure tubes	Zircaloy-2
Steam-generator tubes	Monel
Pressure-tube end-fittings	Stainless steel (AISI types 403 and 410)
Pump bowls	Stainless steel (AISI type 316).

"Hot conditioning" with the heat of the pumps before start-up formed a hard adherent corrosion-resistant layer of magnetite, and resulted in suspended solids (crud) concentrations of less than 0.09 mg/kg.

Difficulties were experienced with chemical purification during 1967 and 1968. The effective flow rates in the purification systems were less than had been intended in the design and less than were needed. Faulty components allowed ion-exchange resin to enter the main system, and resulted in a pD as low as 7.3 and chloride concentrations up to 3 mg/kg for short periods. Attempts to maintain reducing conditions by providing excess dissolved deuterium were initially unsuccessful.

With experience, equipment and procedures were changed, and chemical control improved greatly in 1969. During operation, crud suspensions are now typically less than 0.01 mg/kg, and dissolved deuterium is maintained at 5—7 ml/kg to ensure reducing conditions. However, the large amount of Monel and Stellite in the system — coupled with poor chemical control in previous years — has resulted in high ^{60}Co radiation fields.

Extensive studies are underway by AECL and Ontario Hydro to provide a better understanding of the transport of ^{60}Co and other activated corrosion products and to develop decontamination methods. It is hoped that decontamination and techniques to control the transport of activation products will allow us to greatly reduce radiation fields during major maintenance periods in the future.

Conventional plant

Forty-five percent (45%) of lost production since 1969 has been due to problems associated with the conventional portion of the station. The major problems have been with the turbine, the live-steam reheater, electrical motors and the water-treatment plant.

Turbine

The turbine was opened for inspection and overhaul in 1969, and again in 1970. Significant erosion damage was found in the high-pressure cylinder and a number of blading failures were found in the three low-pressure cylinders.

The erosion damage resulted from inadequate design, including poor choice of materials, to cope with saturated-steam conditions. The low-pressure cylinders are not exposed to such damage as the steam is dried and superheated in live-steam reheaters before entering the low-pressure sections.

Erosion damage occurred on the cylinder horizontal joint and diaphragm half joints, and these were repaired by filling with AISI-type-308 stainless steel. The diaphragm liners and rims also suffered damage.

All rotating blading has been removed from the second stage of each of the three low-pressure cylinders because of fatigue failures caused by the diaphragm impulse frequency being too close to the natural frequency of the blading. New blading of modified design is being procured.

Reheaters

On three separate occasions, tubes have failed in the live-steam reheaters. The failures have been attributed to score marks made by dies and to thinning of the metal at bends.

Motors

Because of poor-quality insulation, windings have failed in the drive motors for boiler-feed pumps and condensate-extraction pumps.

Make-up water system

Failure of a plastic distribution header in an ion-exchange tank permitted sulphuric acid used for regeneration to be discharged into the feedwater system and boilers. We believe no damage was done to the boilers or associated equipment, but the station had to be shut down while the feedwater system and the boilers were drained, flushed and refilled.

PICKERING

Pickering is Canada's first commercial nuclear-electric station and is expected to be fully competitive with contemporary fossil-fuelled stations when the annual capacity factor is 65% or more. It was designed by AECL and Ontario Hydro; it is owned, constructed and operated by Ontario Hydro. Each of the four units is to produce 508 MW(e) net. The station is located on the north shore of Lake Ontario, about 35 km east of the centre of Toronto.

Construction began in September 1965, but was interrupted in 1967 by a 10-month strike of construction workers. The first reactor was made critical on 25 February 1971; it supplied the first steam to its turbine on 16 March 1971 and electricity was generated for the first time on 4 April

1971 when the output reached 80 MW(e) gross. Full power from the first unit is expected by 1 August 1971.

Commissioning of the second unit has started and electricity is expected in December 1971. Construction of the third and fourth units is on schedule with commissioning to start in 1972.

Precritical commissioning of the first unit

Commissioning of auxiliary and service systems began in late 1969, to the extent permitted by construction activities.

Natural water was used to commission the two heavy-water circuits (the moderator system and the heat-transport system). We decided on this mainly because our initial supplies of heavy water were expected to contain large concentrations of tritium (some in excess of 5 Ci/kg). If we had used highly tritiated heavy water, we would have needed to provide radiological protection for several hundred construction personnel still working nearby.

Of course, the reactivity is very sensitive to the isotopic purity of the heavy water in the moderator system (downgrading with 0.1% H_2O is worth about 3.5 mk). However, it is sixty times less sensitive to the isotopic purity of the heavy water in the heat-transport system. As intended, we were able to make the changes from natural water to heavy water without elaborate drying procedures.

The moderator system was the first to be commissioned and it was filled with natural water in February 1970. This early start was made so that various prototype reactivity controls (shutoff rods, cobalt adjuster rods, liquid zone-control system) that were being introduced for the first time at Pickering could be given extensive study and testing well before reactor start-up. The transition from natural water to heavy water was started in December 1970. After the natural water had been drained, the system was first flushed with 30 Mg of heavy water which was thereby downgraded from 99.75 to 98.86 mass %. This was then also drained and replaced by the full charge of 300 Mg. The final isotopic purity was 99.73 mass %. The 30 Mg of 98.86 mass % purity heavy water was later used in the final filling of the heat-transport system.

The introduction of natural water into the heat-transport system began in August 1970. In October heat generated by the main pumps was used to bring it to its operating temperature (265°C) so as to create a dense adherent protective film of magnetite on all carbon-steel surfaces. The system was held at this temperature for 14 days to study its performance and the performance of associated equipment. In particular, leak-tightness, the operation of the main pumps, chemical control and pressure control were checked — all with gratifying results.

In January 1971 the natural water was drained from the heat-transport system. The horizontal pressure tubes were opened, drained and wiped, the fuel was loaded and finally the heat-transport system was filled with heavy water. The final isotopic purity was 99.03 mass %.

Since the fuelling machines had not then been commissioned, the first fuel was loaded with hand tools. The 4 680 fuel bundles (containing about 93 Mg of natural uranium) were loaded during 13 days in January 1971.

Commissioning of most other systems proceeded in parallel with that of the heavy-water systems. Of particular interest and importance is the negative-pressure containment system, which serves all four units of the station. In the event of a loss-of-coolant accident at Pickering, radioactive material escaping from a reactor would be contained in a very large "vacuum building" (50 m high, 50 m in diameter), which is normally held at less than 0.3 atm and is automatically connected to any reactor building where there is a large pressure increase. The building was first evacuated in May 1970 to a pressure of about 0.07 atm, which was then easily maintained. The twelve large (2-m diameter) pressure-relief valves that connect the vacuum building to the reactor buildings successfully passed a series of tests, including tests at full-flow conditions, starting in September 1970. These tests also confirmed the operation of the very powerful natural-water dousing system which would condense steam arriving in the vacuum building after an accident.

One of the important differences between Pickering and the previous Canadian nuclear power stations is in the extensive use of digital computers for continuous process control, including reactor start-up, reactor-power regulation and boiler steam-pressure regulation. Although some control functions can be handled manually, the reactor is automatically shut down if at least one computer is not available. To ensure adequate reliability two computers with a data link are supplied for each of the four units.

In-service reliability of the computers remains to be proved, but early operation has been satisfactory. The flexibility of this type of control was apparent during low-power operations when necessary minor program adjustments were made easily and with a minimum of delay.

Low-power operation

The reactor was brought to critical for the first time on 25 February 1971. The approach to critical was carried out by raising the level of the heavy-water moderator in the core while monitoring the neutron flux with special start-up instruments. Two sets of instruments were used, one in the centre of the core (in a fuel channel that had been isolated and left empty for this purpose) and the other at the normal location of the control-system ion-chambers outside the core. The time from first admitting heavy water to the core to criticality was about 2½ h. Following criticality, the reactor was operated at low powers (in the range from 10^{-6} to 10^{-3} of full power) to test the reactivity-control equipment and to make a series of physics measurements. No major problem was encountered and the reactivity worths of the control mechanisms were confirmed. The reactor was then shut down to rehabilitate the fuel channel used for start-up instruments and to prepare all systems for high-power operation.

High-power operation

On 14 March 1971, the satisfactory response of those instruments that had not been on-scale during the low-power measurements was confirmed with the reactor operating at 2% of full thermal power.

Steam was admitted to the turbine on 16 March 1971, but synchronous speed could not be attained because the turbine exhaust hood cooling sprays were inadequate. The spray nozzles were replaced, and synchronous speed was reached on 23 March 1971.

Several time-consuming but minor problems prevented our synchronizing the generator to the Ontario Hydro network until 4 April 1971. One involved a turbine intercept valve that stuck in the closed position and had to be replaced. Another delay came during short-circuit saturation measurements of the generator when the housing of the 24-kv isolated-phase bus duct over-heated because of induced currents.

The power run-up of the first unit involves several weeks of testing at successively higher power settings. Major tests, such as generator load rejection, are being carried out at 25%, 50%, 75% and 100% of full power.

By 27 April 1971, a power output of 250 MW(e) gross had been attained, with only minor problems up to that time. Further power increases, however, will be delayed until modifications are made to the turbine intercept valves, which are continuing to stick and are therefore unreliable to prevent turbine overspeeding after large load rejections.

We are naturally pleased that commissioning has so far proceeded on schedule without major problems. However, as with any large complex plant of new design, including modern fossil-fuelled plants, we recognize that it is only realistic to expect that significant problems will be recognized and will have to be solved before our objectives of reliability and economy are satisfied.

GENTILLY

Gentilly is Canada's first nuclear-electric station to use boiling natural water as the medium for heat transport. Construction started in late 1966 and the reactor was made critical on 12 November 1970.

The station is owned by AECL, and was designed by AECL in co-operation with Hydro-Québec and several consulting engineering organizations. Hydro-Québec acted as prime contractor and is responsible for commissioning and operation.

Reactor-physics measurements

Gentilly is the world's first reactor of its type, and it has unique reactivity and control characteristics. To allow early and ample tests of the predictions on which its design had been based, the reactor was made critical several months before we intended to generate electricity.

The first approach to critical was made by raising the level of the moderator, but with no water in the heat-transport system and with no poison (boron) in the moderator. This provided a very clean measurement of reactivity for comparison with computer analysis of the reactor physics; it also represented the most reactive condition, which is of interest for safety studies. The observed critical height agreed closely with the predicted value.

When water was added to the heat-transport system, its reactivity worth was determined. This, of course, is an important parameter in all calculations of the regulating system, which must cope with the transitions from liquid to partial boiling (at full load, 20 mass % of the water is steam when it leaves the core). Also the thermal-flux distributions, both radial and axial, were measured in detail by the activation of copper wires passing through the core.

Enough boron was added to the moderator to bring its critical height up to the top of the calandria, which is the normal operating position; then the reactivity worths of the control rods and the booster rods were measured. The results agreed well with predictions, with the control rods having slightly more reactivity and the boosters slightly less than expected.

High-power operation

After the reactor-physics measurements, the reactor was kept shut down until the end of January 1971 while construction was finished and while the commissioning needed for high-power operation was carried out.

With the main turbine condenser acting as a heat sink, the first nuclear steam was produced on 6 February 1971, and the reactor was operated at up to 25% of full thermal power in February and March. During this period, one of the six pumps in the heat-transport system suffered mechanical damage, including a fractured shaft. The cause has been diagnosed as torsional fatigue and modifications to the impellers are under way.

Spatial flux instability was expected, but became apparent at a power lower than that at which it was originally expected. The control system for spatial flux perturbations has now been modified, so that it begins to function at 10% of full power — previously it was designed to function only above 35% of full power.

On 5 April 1971 the generator was synchronized to the Hydro-Québec system for the first time. It is intended that output will be progressively increased to full power over the next few months.

KANUPP (Karachi Nuclear Power Project)

KANUPP is Pakistan's first nuclear-electric station. It was designed and constructed by the Canadian General Electric Company Limited (CGE) for the Pakistan Atomic Energy Commission (PAEC).

Construction started in late 1966 and was complete in late 1970. The dry, warm climate at Karachi permitted work to continue in all seasons and, in particular, it permitted what almost amounted to a reversal of the sequence that we have followed in Canada for construction of the reactor containment building and nuclear systems. After the foundations and support steel had been erected, all major equipment was permanently put in place, and only then did we pour the concrete for the pre-stressed walls and dome. This sequence saved considerable time and obviated the need for a large temporary entrance through the containment wall.

CGE engineers and PAEC operating staff are jointly commissioning the station. The key PAEC staff received training in Canada at CGE and NPD. As of 31 March 1971, the service systems, turbine-generator, auxiliary nuclear systems, and moderator system had been commissioned; pressure tests of the heat-transport system are under way and these will be followed by performance tests of the pumps and "hot conditioning".

First electricity production is scheduled for August 1971.

RAPP (Rajasthan Atomic Power Project)

RAPP was authorized by the Government of India in December 1963. The nuclear steam-supply systems were designed by AECL with the Montreal Engineering Company acting as consulting engineer for the non-nuclear systems. The Indian Department of Atomic Energy (DAE) owns the station, is constructing it and will operate it.

AECL is responsible for commissioning, but has contracted Ontario Hydro to provide a team of specialized staff to manage the work for the first unit. The team, which began to arrive in February 1970, now amounts to seventeen men. The contract also gives Ontario Hydro responsibility to assist in training the Indian operating staff, and key DAE personnel have already received this training at NPD and Douglas Point.

As of 31 March 1971, the turbine-generator had been erected and mothballed, the service water and compressed-air systems were essentially complete and their commissioning had begun, the moderator system was about to be commissioned with the 70 Mg of heavy water that are available, the heat-transport system was being commissioned with natural water and leak tests were under way, parts of the electrical system had been commissioned but other parts were still under construction, and the auxiliary nuclear systems were under various stages of construction. Half of the first fuel charge was supplied from Canada and is at the site; the remainder is currently under construction in India.

The first unit is scheduled to achieve criticality on 15 October 1971 and the second unit in January 1974.

BRUCE

Bruce is Canada's first opportunity to apply significant operating experience (from NPD and Douglas Point) to the design of a new nuclear-electric station. Like Pickering, it is being designed by AECL and Ontario Hydro; it is owned, is being constructed and will be operated by Ontario Hydro.

Design started in 1968 and electricity is expected from the first of the four 750-MW(e) units in 1975. While the basic features of Douglas Point and Pickering are retained, several major design changes are being made to improve reliability or economy. Some of the most important considerations are

- As reactor size increases, adjusting the moderator becomes a relatively less effective method of controlling reactivity. At Bruce we shall depart from the previous designs in which reactivity regulation was obtained by varying the moderator level and emergency shutdown was obtained by dumping the moderator. Abandoning these features will result in a simpler containment building and a simpler moderator system.
- At Bruce the calandria will be integrated with the biological and thermal shields, resulting in a single assembly that can be manufactured in the shop to reduce cost and speed up construction.
- Many major components are to be moved out of the containment building with the advantages of accessibility at high power, greatly reduced radiation dose, and a virtually complete separation of the heavy-water heat-transport system from the natural-water system (in particular the high-pressure high-temperature steam and feedwater systems). As a result we expect that downgrading of recovered heavy water will be greatly reduced. The main components that have been moved out of the containment building are the steam generators (except D_2O connections), the steam drums, the main heat-transport pumps (except D_2O connections), the moderator pumps and heat exchangers, and the drive assemblies for the reactivity-control mechanisms. The containment building will contain, essentially, only the reactor and the high-temperature high-pressure parts of the heat-transport system.
- In previous designs each reactor had its own self-contained set of fuelling machines. At Bruce there will be two sets of fuelling machines for the four-unit station. Each set will be able to refuel any of the four reactors. The fuelling machines, and the fuel channels, are being designed to include the best features of NPD and Douglas Point, as known at this time.
- A determined effort has been made, not only to improve component quality, but also to reduce the number of critical components and thereby achieve simplification. For example, at Pickering there are 16 heat-transport pumps (each 1500 kW) and 12 boilers. At Bruce there will be four pumps (each about 9000 kW) and eight boilers.

SUMMARY

The pilot plant (NPD) has fulfilled its missions and continues to play a vital role in development of people and technology.

The prototype station (Douglas Point) has not yet demonstrated acceptable performance, but is steadily improving and has contributed invaluable operating experience to the design and operation of Pickering and Bruce.

The first commercial station (Pickering) has experienced favourable commissioning progress with no significant problems so far.

The prototype boiling-natural-water station (Gentilly) is at a very early stage of high-power operation. No significant conclusions can be drawn at this time.

KANUPP and RAPP — These overseas projects have not yet produced electricity. They represent major contributions to the technological development of Pakistan and India.

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