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EVOLUTION OF CANDU REACTOR DESIGN

by

G.A. PON

Presented at the Canadian Nuclear Association 18th Annual Conference, Ottawa, Ontario, 11-14 June 1978

Chalk River Nuclear Laboratories

Chalk River, Ontario

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Evolution de la conception du réacteur CANDU*

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Résumé

La filière CANDU (Canada Deutérium Uranium) a vu le jour au début des années 1950 alors que des études d'ingénierie préliminaires ont été faites lesquelles ont conduit au NPD (Nuclear Power Demonstration) de 20 MWe et à la centrale Douglas Point de 200 MWe. La décennie suivante a vu le premier fonctionnement de ces deux centrales et la décision de construire la centrale Pickering de 2000 MWe et la centrale Bruce de 3000 MWe. La décennie actuelle est marquée par l'excellente performance de Pickering et de Bruce et par la décision de construire Gentilly-2, Cordoba, Lepreau, Wolsung, Pickering B, Bruce B et Darlington.

Dans la plupart des cas, les concepts successifs ont donné lieu à une augmentation de la capacité de la centrale. Des développements évolutionnaires ont été effectués pour répondre aux nouvelles exigences en matière de taille, de rendement, de sûreté, de fiabilité, d'entretenabilité et de coût. Ces changements, que l'on décrit système par système, ont été apportés au cours des études d'ingénierie effectuées pour des projets parallèles de réacteurs réalisés avec un décalage de temps - circonstance qui permet d'avoir un contact avec les réalités pratiques des facteurs économiques, des fonctions manufacturières, des activités de construction et de la mise en service. Les caractéristiques d'un certain projet ont permis d'élaborer des concepts différents pour d'autres projets se trouvant encore sur la planche à dessin et l'expérience acquise dans une première application a fourni une base solide pour son nouvel emploi dans des projets ultérieurs. Ainsi, les expériences acquises dans le NPD, à Douglas Point, à Gentilly-1 et à KANUPP ont contribué à la performance de Pickering et de Bruce qui à leur tour ont contribué à la conception de Gentilly-2.

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ABSTRACT

The CANDU (CANada Deuterium Uranium) design had its beginnings in the early 1950's with the preliminary engineering studies that led to the 20 MW(e) NPD (Nuclear Power Demonstration) and the 200 MW(e) Douglas Point station. The next decade saw the first operation of both these stations and the commitment of the 2000 MW(e) Pickering and 3000 MW(e) Bruce plants. The present decade has witnessed the excellent performance of Pickering and Bruce and commitments to construct Gentilly-2, Cordoba, Pt. Lepreau, Wolsung, Pickering B, Bruce B and Darlington.

In most cases, successive CANDU designs have meant an increase in plant output. Evolutionary developments have been made to fit the requirements of higher ratings and sizes, new regulations, better reliability and maintainability and lower costs. These changes, which are described system by system, have been introduced in the course of engineering parallel reactor projects with overlapping construction schedules - circumstances which ensure close contact with the practical realities of economics, manufacturing functions, construction activities and performance in commissioning. Features for one project furnished alternative concepts for others still on the drawing board and the experience gained in the first application yielded a sound basis for its re-use in succeeding projects. Thus the experiences gained in NPD, Douglas Point, Gentilly-1 and KANUPP have contributed to Pickering and Bruce, which in turn have contributed to the design of Gentilly-2.

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1.0 The CANDU (Canada Deuterium Uranium) design had its beginnings in the early 1950's with preliminary engineering studies on a 20 MW(e) and a 200 MW(e) plant. These studies eventually culminated in commitments to the construction of NPD (Nuclear Power Demonstration) and Douglas Point. The 1960's resulted in the operation of NPD in 1962 and Douglas Point in 1966. At the same time, commitments to construct Pickering were made in 1964 and Bruce in 1969. The 1970's have witnessed the excellent operating performance of Pickering and Bruce and the commitments to construct Gentilly-2, Cordoba, Pt. Lepreau, Wolsung, Picking B, Bruce B and Darlington.

In most cases, successive plants have meant an increase in plant output. Evolutionary developments have been made to fit the requirements of higher ratings and sizes, new regulations, better reliability and maintainability, and lower costs. These evolutionary changes have been introduced in the course of engineering parallel reactor projects with overlapping construction schedules - circumstances which provide close contact with the practical realities of economics, manufacturing functions, construction activities, and performance in commissioning. Features for one project furnished alternative concepts for other plants on the drawing board at that time, and the experience gained in first application yielded sound basis for re-use in succeeding projects. Thus the experience gained in NPD, Douglas Point, Gentilly-1 and KANUPP (Karachi Nuclear Power Project) have contributed to Pickering and Bruce. In turn, all of these plants have contributed to the design of Gentilly-2*.

The evolutionary changes that have taken place are discussed below.

2.0 Primary Heat Transport System

There has been a continuing quest for higher reliability, better maintainability of equipment, and a reduction of radiation dose to operating staff. This is manifested in the dramatic reduction in the number of components. For example, NPD had approximately 100 valves per MW in the nuclear steam supply system. This has been reduced to less than 1 valve per MW in the Bruce and Gentilly-2 designs. The number of pumps has gone from 16 in Pickering to 4 in Bruce and Gentilly-2. The number of steam generators has gone from 12 in Pickering to 8 in Bruce to 4 in Gentilly-2.

All materials in the heat transport circuit are now

*Gentilly-2 is the first of the 600 MW(e) designs; others are Lepreau, Cordoba and Wolsung.

being specified for very low levels of cobalt in order to keep radiation fields to a minimum.

2.1 Steam Generators

Steam generator size has been generally limited by the industrial capability to produce them. We are now down to 4 in the 600 MW(e) Gentilly-2 design. Monel was used as the tubing material for Douglas Point, RAPP (Rajasthan Atomic Power Project), KANUPP and Pickering. This material has been proven to be quite satisfactory for the non-boiling coolant conditions of those plants. Inconel 600 has been used in NPD and in Bruce. This is a more costly material than Monel; however, its corrosion resistance in a boiling environment (as in Bruce) is much superior. We are using Incoloy 800 in all of the 600 MW reactors (Gentilly-2, Pt. Lepreau, Cordoba and Wolsung) as it is about equal in most respects to Inconel 600, has greater resistance to intergranular attack, and is somewhat lower in cost. Table 2.1 gives a more detailed comparison of the features of different steam generators.

2.2 Heat Transport Pumps

Pump-motor sets have remained essentially of the same configuration for all of the CANDU stations, i.e., vertical electric motor driven, centrifugal, volute type casing, one radial guide bearing in the pump with pumped fluid as lubricant, tilting pad type guide and double acting thrust bearing in the motor, and mechanical shaft seals.

As indicated previously, the trend has been toward a fewer number of larger pumps.

Maintainability has been improved with the provision of interchangeable sub-assemblies. The appropriate placement of shielding has permitted the changing of a pump motor on Bruce while the reactor continued to operate at 60-70% power.

There has been a recent trend away from solid rotor flywheels (Douglas Point to Gentilly-2) to additional packages of rotor laminations located just outboard of the main rotor (Pt. Lepreau, Bruce "B"). This manner of fabrication precludes the requirement for inservice inspection for that component as it is highly unlikely that a defect could grow from one lamination to another.

Regulatory requirements for pumps have grown from very little in the beginning to the present time where the pump pressure boundary is considered in the same way as nuclear pressure vessels (ASME Section III Class 1). Consequently, NDE (Non-Destructive Examination) and quality assurance requirements have increased considerably.

A detailed comparison of pump characteristics is given in Tables 2.2 and 2.3.

3.0 Reactor Core Design

In 1955, a detailed design of a demonstration natural uranium reactor was initiated. It was called NPD and was based on a vertical pressure vessel concept. In 1957, this was changed to a horizontal pressure tube configuration - a configuration which has remained in succeeding heavy water cooled reactors.

3.1 Evolutionary changes have been in the direction of achieving

(a) large increases in core rating with the minimum increase in reactor size (the higher the power density, the lower the capital cost);

(b) reduction in shop fabrication costs through simplification;

(c) reduction in field assembly through more shop fabrication.

3.2 The major impact of higher power densities on capital costs is in the reduction of heavy water inventory. The amount of heavy water in the reactor core per MW produced in the reactor is listed below.

Heavy Water in Core per MW Thermal

	$m^3/MW(t)$
NPD	.410
Douglas Pt.	.169
KANUPP	.182
Pickering A	.157
Bruce A & B	.112
Gentilly-2	.105

3.3 Higher power densities require more MW's produced per metre length of fuel channels. The table below indicates the achievements to date.

MW Thermal per Metre Length of Fuel Channel*

	MW(t)/m
NPD	.163
Douglas Pt.	.453
KANUPP	.443
Pickering A	.752
Bruce A & B	.881
Gentilly-2	.931

The above increase in rating has been achieved by

(a) increasing the pressure tube diameter from 3¼" (NPD, Douglas Pt. and KANUPP) to 4" (Pickering, Bruce, Gentilly-2);

(b) increasing the number of fuel pencils per bundle from 19 in NPD to 37 in Bruce and Gentilly-2;

(c) increasing the fuel rating from 24.9 kW/m in NPD to 50.9 kW/m in Gentilly-2.

3.4 The above increase in performance has been accompanied by simplification of the calandria assembly by

(a) elimination of the reflector as a separate compartment of the calandria;

(b) integration of the end shields with the calandria tube sheets;

(c) substitution of shielding balls for machined slabs in the end shield assemblies;

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*Calculated by taking total MW thermal divided by total length of fuel channels.

(d) integration of assembly supports with the end shield rings.

All of these changes were made to reactors following Pickering A.

The water-filled reactor vault concept developed for KANUPP enabled elimination of complicated and costly provisions for cooling and shielding in Douglas Point and Pickering. The Gentilly-2 reactor follows this approach.

3.5 The concepts of vessel integration, water-filled vaults and shop assembly were advanced further in the Bruce design in which the calandria, end shields and shield tank are preassembled to provide effectively a shop fabricated reactor core, complete save fuel channels and components for reactivity mechanisms. Studies show that it would be feasible to consider incorporating these componenets in the shops as an alternative to field installation.

4.0 Control and Safety Systems

In the broadest terms, nuclear power station control embraces several different functions:

1) Control of the fission process in the reactor itself, coupled with control of the station's electrical output.

2) Control of the thermal processes and their auxiliaries.

3) Safety-related control, i.e., power reduction, reactor shutdown, initiation of containment, emergency cooling, or emergency power, to protect the station and the public against any process or control system failure.

4) Display annunciation, and recording of the data to operate the station efficiently and diagnose any malfunctions that may occur.

Design of the systems that perform these functions has undergone steady evolutionary changes in response to changes in regulatory requirements, increases in the size and power output of CANDU stations, and the evolution of the instrumentation and control technology itself.

4.1 Plant Control Mode

In Douglas Point and Pickering, the reactor leads the turbine, that is, the thermal power output of the reactor is set and controlled to a desired value and the overall plant control system adjusts the turbine output to accept the reactor output. With Bruce A, Gentilly-2, and later CANDU stations, the reactor follows the turbine. This reflects a strong utility preference. In addition, as nuclear power begins to supply an increasing fraction of utility generating capability, it becomes necessary for the nuclear plants to contribute to grid stability by responding to system frequency variations.

In Bruce A and Gentilly-2, the desired turbine generator load is set, and variations in this load, acting through the boiler pressure control system, adjust the reactor power to match the desired generator output.

4.2 Control System

Douglas Point uses an analogue system for control with a single digital computer serving limited duties in the areas of monitoring, data recording, and control. The computer is not essential for the operation of the station.

The larger plants after Douglas Point required more control functions of a more sophisticated nature (e.g., spatial control). One of the better ways to meet this requirement was to use a dual digital computer system to control the plant. Hence, on all stations following Douglas Point, this has been the practice. Atomic Energy Control Board requirements call for a separation of the regulating systems from the safety systems. The safety systems, therefore, are not implemented in the dual digital computer system but in separate independent systems.

4.3 Shutdown Systems

The shutdown systems that are incorporated in our plants are listed as follows:

Douglas Point	Pickering A	Bruce A	Gentilly-2
Moderator Dump	Moderator Dump plus 11 Shut-off Rods	30 Shutoff Rods	28 Shutoff Rods
		Poison Injection	Poison Injection

Moderator dump is adequate for the Douglas Point design but by itself is not fast enough for the most severe accident postulated for Pickering. In Pickering, the moderator dump was complemented by 11 shutoff rods to provide a composite system that would accommodate the most severe accident. For plants after Pickering A, we have gone to two completely independent shutdown systems - (a) shutoff rods and (b) poison injection. Each of these two systems is capable by itself of accommodating the most severe accident. Each of the two systems is activated by an independent set of parameters.

4.4 Reactivity Control Devices

The following indicates the various devices used to control the reactor:

	Fine Control	Spatial Control	Coarse Control
Douglas Point	4 Absorbers* (S.S.)	4 Absorbers* (S.S.)	8 Boosters Moderator Level Boron Addition
Pickering A	14 Absorbers (H_2O)*	14 Absorbers (H_2O)*	18 Adjusters (Co) Moderator Level Boron Addition
Bruce A	14 Absorbers (H_2O)*	14 Absorbers (H_2O)*	16 Boosters 4 Control Absorbers Boron Addition
Gentilly-2	14 Absorbers (H_2O)*	14 Absorbers (H_2O)*	21 Adjusters (S.S.) 4 Control Absorbers Boron Addition

*Same units for both functions.

The elimination of moderator dump as a shutdown mechanism in Bruce and Gentilly-2 has also eliminated moderator level as a reactivity control device for these plants.

The reactors after Douglas Point have a power output that is at least 2½ times higher. The larger size requires spatial control of the power and this is achieved by 14 absorber tubes that use light water.

Enriched fuel rods (boosters) are used in Douglas Point and Bruce A. Their main function is to provide a start-up capability following a shutdown. This can also be achieved by using absorber rods of either cobalt or stainless steel that reside in the core during operation and are removed to give extra reactivity during startup. The trend is towards the use of adjuster rods, as booster rods bring with them added complexity.

Four control absorbers are provided in Bruce and Gentilly-2. They are mechanically the same as shutoff rods. Provision of these control absorbers allows fast power reductions at Bruce and Gentilly-2 without reactor trips.

4.5 Power Measurements

The following indicates the various measurements that are provided in the reactors to aid in determining local and gross reactor power:

Douglas Point	Pickering A	Bruce A	Gentilly-2
6 Ion Chambers	6 Ion Chambers	9 Ion Chambers	9 Ion Chambers
T_{out}-all channels	T_{out}-all channels	T_{out}-all channels	T_{out}-all channels
	28 Co flux detectors	28 Pt flux detectors	28 Pt flux detectors
	22 channels instrumented for channel power measurement	22 channels instrumented for channel power measurement	102 V flux detectors
		54 V flux detectors	

Pickering was the first reactor to require spatial control of power. Local power determinations are generated from measurements on 28 flux detectors and 22 instrumented fuel channels. This has been repeated on Bruce. Gentilly-2 is provided with 102 vanadium flux detectors to generate a "flux map" of the core. These detectors, together with the 28 platinum detectors, give the operator a complete story on power generation at any point in the core.

5.0 Fuel Handling Systems

AECL first adopted fuel changing during on-power conditions on our NRU 200 MW(t) experimental reactor located at CRNL which went critical in November 1957. The single length irradiated natural uranium fuel rods are extracted into a lead-shielded flask and replaced with new rods while the reactor is still on power. The flask itself weighs approximately 240 t and is positioned by a gantry and trolley system located above the reactor. A rotating drum inside the shielded shell is used to align either of its two internal barrels with the flask "snout" for the insertion or removal of the new or irradiated bundles. Irradiated bundles are transferred to a temporary storage block and after a ten-day decay period are moved to the storage bay.

5.1 From this beginning evolved the approach to our on-power fuelling capability for the Canadian-designed heavy water moderated natural UO_2 nuclear power stations. The significant differences from the early NRU approach were:

(a) the use of much smaller unshielded machines remotely operated by automatic controls;

(b) the use of short length fuel bundles (½ m), which allowed the insertion and replacement of fuel in the reactor core in finite steps at spaced intervals. This type of movement approximated the ideal of a continuous flowing process to maximize fuel economy.

5.2 The NPD demonstration unit was the first Canadian design to utilize the horizontal channel concept. Two identical fuelling machines are used at each end of the channel, one of which pushes new fuel into the channel while the other machine receives irradiated fuel from the opposite end of the channel. To accomplish this, the machines travel to and locate on the selected channel end fitting. Here they automatically seal to the end fitting, pressurize to reactor pressure, remove the closure plug and insert or remove the required number of fuel bundles. The normal fuelling process is against the channel coolant flow, a feature which applies to KANUPP and Bruce. The fuel string is restrained by a fuel latch assembly.

There is no shield plug assembly used in NPD since the machines operate inside of the biological end shield.

5.3 The Douglas Point channel and its method of fuelling, although similar in appearance to that of NPD, has several fundamental differences:

(a) Fuelling takes place with the flow.

(b) Shield plugs are used since the coolant tube end fittings are outside of the end shield.

(c) Fuel latches are not used in the Douglas Point channel. Fuel separation is accomplished in the fuelling machines.

(d) Axial push-pull closure plugs locking is used as opposed to the rotary locking of the NPD closure plugs.

These differences have evolved primarily because of the different design teams concerned and have resulted in the divergent paths indicated.

- NRU
  - NPD
    - KANUPP
      - BRUCE A
        - BRUCE B
  - DOUGLAS POINT
    - RAPP
      - PICKERING A
        - PICKERING B
          - GENTILLY-2

5.4 Typical equipment for these on-power fuelling systems incorporates a wide range of electrically, hydraulically and pneumatically driven mechanisms with feedback control based on both analogue control and digital computer systems. The choice of these systems has essentially been that of the design teams involved. The cost, however, of station outages with the increasing reactor size has necessitated continual improvement to ensure a high standard of reliability from all equipment. This is particularly true for the fuelling machines and the fuel channel internal components handled by the heads during a fuel change on an operating reactor.

5.5 Although details vary, heads for all these stations are basically similar and consist in the main of a rotating magazine and linear motion rams contained in a pressure casing. Magazine size and ram strokes are, of course, designed to suit the fuel bundles and fuel channel internal components being handled. These operating heads are transported to and from the fuel channel ends by a variety of designs of transport carriages.

6.0 Heavy Water Management

One of the obvious consequences of the use of heavy water in the high pressure heat transport system of CANDU is the requirement to retain the heavy water and to keep upkeep or makeup costs to a minimum. This requirement is just as stringent today as it was 20 years ago. Two factors will continue to push us toward keeping heavy water losses and upkeep to a minimum:

(a) The monetary penalty of heavy water losses has always been with us. Projected costs of heavy water indicate that losses will be a continuing financial burden.

(b) The radiation dose resulting from uptake of tritium by operating staff is becoming a significant fraction of the total radiation dose to station personnel.

The fundamental principles of heavy water management are:

(a) to prevent the leakage of heavy water from the systems, and if there is leakage, then

(b) to prevent the mixing of the leaked heavy water with ordinary water.

6.1 Leakage Control

The replacement of mechanical components or connections with all-welded systems is an area that can and has paid off. The progress that has been made is indicated as follows:

	Valves/Unit		Non-welded Joints/Unit (exclude fuel channel closures and feeder connections)
	Packed Stem	Bellows Sealed	
NPD	1500	0	4000
Douglas Point	2000	0	3000
Pickering A	170	570	1000
Bruce A	75	500	250
Gentilly-2	90	300	200

It has been recognized with operating experience that large capacity water vapour recovery units can provide a net benefit.

The water vapour recovery equipment capability has been increased from about 5 kg/h for NPD and Douglas Point to about 18 kg/h for Pickering, Bruce and Gentilly-2.

6.2 Atmospheric Separation

The separation of D_2O operating areas from those containing H_2O vapour or liquid is most important in keeping the upgrading of collected D_2O leaks to a minimum. This requires careful attention to design, construction and maintenance of such items as door seals and pipe penetrations.

In addition to separating D_2O and H_2O atmospheres, we have also introduced with the Bruce A and Gentilly-2 designs separation between high tritium D_2O areas and other D_2O areas. This is accomplished by having separate recovery, collection and clean-up facilities for the heat transport system D_2O and moderator system D_2O. The tritium level in the high pressure heat transport system is thereby kept to a minimum. This in turn helps to keep the radioactivity dose to the operating staff to a minimum.

7.0 Changing Emphasis Applicable to All Design Areas

Changes in emphasis that have been applicable to all areas of CANDU design are identified below.

7.1 Quality Assurance

Quality in all phases of a nuclear power plant from concept to operation has always been recognized as a requirement. In recent years, a more formalized approach has been necessary. This had its beginnings in the manufacturing industries and we now have a series of four standards (CSA Z299 series) for the quality assurance (Q/A) of manufacture of nuclear equipment. Other standards are being initiated under the CSA N286 series. They consist of

CSA N286.0	General Requirements for Q/A of Nuclear Power Plants
N286.1	Q/A of Procurement
N286.2	Q/A of Design
N286.3	Q/A of Construction & Installation
N286.4	Q/A of Commissioning
N286.5	Q/A of Operation
N286.6	Q/A of Decommissioning

However, standards by themselves are not sufficient. There must be an attitude which recognizes the necessity and importance of Q/A by all those who must comply by these standards. With the proper attitude, there will come commitment and with commitment quality will result. There is the danger, already voiced by many, that we are being swamped by paper and bureaucracy in our Q/A programs. It is clear we must show common sense in minimizing the paper and bureaucracy and yet retaining the quality.

7.2 Reliability and Maintainability

Reliability and Maintainability, or simply R&M, has been formally identified as a separate discipline in 1970. Two of the main objectives of an R&M program are

(a) higher capacity factors for the operating plant;
(b) lower radiation exposure for the operating staff.

The requirement for the first objective is obvious - a reliable source of economic electrical energy. The attainment of the latter objective is discussed in more detail in the next section.

R&M is a discipline that does not just happen. Conscious and persistent efforts on the part of many people are required to achieve them. The engineering organization must take the lead to design R&M into the components and systems that make up the nuclear power plant. The job does not end there. The fabricators and construction forces must fabricate and assemble to specification. The operator must operate to give a reliable source of economic energy.

7.3 Reduction in Radiation Exposure

Recommendations have been made by the International Commission on Radiological Protection (ICRP) on maximum permitted doses for occupationally exposed persons. Continued exposure at these limits is expected to have a risk of fatality comparable to, or less than, conventional fatality risks facing occupational groups in industry in general. Canada has accepted the recommended limits of the ICRP which are 5 rem/year whole body exposure for Atomic Energy Workers. In practice, we have taken a design target of 2.5 rem/year per man as the average.

The major factors which affect the radiation dose incurred by a worker are:

1. Amount of equipment.

2. Frequency of failure, servicing, inspection.

3. Time required to repair, service, inspect.

4. Radiation conditions (fields and airborne concentrations).

Since radiation dose is proportional to the product of these four factors, a reduction in any factor will reduce the dose received.

It became very evident in the late 1960's with the operation of Douglas Point that a formal program of radiation dose reduction was required to prevent future problems. For Douglas Point, the major emphasis was on the reduction of radiation fields by chemistry control and the removal of high activity materials (Item 4 above). For new stations not yet operated, the emphasis was on all four items listed above. This has taken the form of detailed design reviews. From these design reviews a general classification of solutions in the design stage has emerged:

1) Stop adding equipment.
2) Eliminate equipment.
3) Simplify equipment.
4) Provide necessary equipment of high reliability.
5) Relocate equipment to lower radiation field.
6) Eliminate materials such as cobalt which could become highly radioactive.
7) Provide better chemical control and purification.
8) Extend interval between maintenance periods.
9) Arrange for quick removal for shop maintenance.
10) Reduce in-situ maintenance times.
11) Provide adequate space around equipment.
12) Provide adequate shielding in order that maintenance can take place in low fields.

8.0 The Future

The future will see continuing emphasis on reliability and maintainability, quality assurance, reduction in radiation dose, and capital cost reduction. The excellent performance record of Pickering A and Bruce is to be maintained in future stations through a vigorous program of R&M and a common sense approach to Q/A. Radiation dose to the operating staff must continue to be kept to a minimum. A renewed effort on capital cost reduction must be instituted. All areas of cost, from engineering, to fabrication, to construction, and to commissioning, must be carefully scrutinized to bring about real savings.

The overall schedule should be critically examined with a view to shortening it.

The Canadian Nuclear Power Program has been most successful up to this point in time. Let us all work together to keep it so for the future.

9.0 Acknowledgements

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TABLE 2.1

STEAM GENERATORS

	DOUGLAS POINT	PICKERING	BRUCE A	GENTILLY-2
Power MW(e)/boiler	2.5	45	95	150
No. of Boilers	80	12	8	4
Tubesheet Diameter mm	250/360 (10"/14")	1730 (5'-8¼")	2520 (8'-3 1/8")	2770 (9'-1")
Tubesheet Thickness mm	79/110 (3 1/8"/4½")	280 (11 1/16")	360 (14¼")	390 (15 3/8")
Tube Size OD/Wall mm (in.)	12.6/1.2 (0.496/0.049)	12.6/1.2 (0.496/0.049)	13.0/1.13 (0.51/0.0445)	(15.9/1.13) (0.625/0.0445)
Material	M-400	M-400	I-600	I-800
No. of Tubes	196	2600	4200	3550
Steam Drum Diameter mm	1680 (5'-6")	2500 (8'-2 3/8")	3560 (11'-8¼")	4010 (13'-1 3/4")
Shell Thickness mm (in.)	12.7 (1/2)	41.3 (1.625)	57.2 (2.25)	49.4 (1.943)
Overall Height mm	9750 (32')	14 200 (46'-7")	15 500 (50'-10 5/16")	19 310 (63'-4¼")
Overall Weight (Dry) kg (lb)		83 900 (185 000)	145 000 (320 000)	193 000 (426 000)
Heating Surface Area m^2 (ft^2)	1040 (11,190)	1860 (20,000)	2420 (26,000)	3180 (34,200)
Recirculation Ratios	3.71	5.5:1	5.4:1	5:1

TABLE 2.2

HEAT TRANSPORT PUMPS

STATION	DOUGLAS POINT	PICKERING	BRUCE	GENTILLY-2
Pump Type	Vertical Centrifugal Single Stage	Vertical Centrifugal Single Stage	Vertical Centrifugal Single Stage	Vertical Centrifugal Single Stage
Head m (ft)	143 (469)	146 (480)	213 (700)	215 (705)
Flow m^3/s (Igpm)	0.43 (5670)	0.77 (10,100)	3.307 (43,600)	2.23 (29,400)
Power per Pump kW (hp) (hp)	600 (800)	1,170 (1560)	8250 (11,000)	5250 (7000)
Discharge Pressure MPa (psia)	9.577@ 249°C (1389 @ 480°F)	9.715@ 249°C (1409 @ 480°F)	10.625@ 265°C (1541 @ 509°F)	11.342@ 266°C (1645 @ 512°F)
Number of Pumps operating per Reactor	8	12	4	4
Speed (rpm)	1800	1800	1800	1800

TABLE 2.3

HEAT TRANSPORT PUMPS

	DOUGLAS POINT	PICKERING	BRUCE 'A'	GENTILLY-2	POINT LEPREAU	BRUCE 'B'
ASME CODE	Sect.VIII	Sect. VIII	Preliminary Sect.III Cl.1 1969	Sect.III Class 1	Sect.III Class 1	Sect.III Class 1
VOLUTE MATERIAL	SA-216-WCB	SA-216-WCB	SA-216-WCB	SA-216-WCC	SA-216-WCC	SA-216-WCC
FLYWHEEL	Solid in Motor	Solid in Motor	Solid in Motor	Solid in Motor	Rotor Laminations	Rotor Laminations
ROTATIONAL $kg \cdot m^2$ INERTIA ($lb \cdot ft^2$)	300 (7,000)	630 (15,000)	2000 (50,000)	1300 (30,000)	1300 (30,000)	2000 (50,000)
SEISMIC CLASSIFICATION	None	None	None	D.B.E. Cat. 'A'	D.B.E. Cat. 'A'	D.B.E. Cat. 'A'
PUMP BEARINGS	Hydro-dynamic Carbon	Hydro-dynamic Carbon	Hydro-static D_2O Ener-gized	Hydro-static D_2O Ener-gized	Hydro-static D_2O Ener-gized	Hydro-static D_2O Ener-gized
MOTOR BEARINGS	Oil Lubri-cated Tilting Pad Type	Oil Lubri-cated Tilting Pad Type	Oil Lubri-cated Tilting Pad Type	Oil Lubri-cated Tilting Pad Type	Oil Lubri-cated Tilting Pad Type	Oil Lubri-cated Tilting Pad Type

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