CANDU 600 REACTOR OUTPUT OPTIMIZATION

by

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ABSTRACT

As part of a program to reduce capital cost and construction schedule, Atomic Energy of Canada Ltd. - CANDU Operations has identified design improvements to increase the output of a CANDU-600 unit by 17%. These improvements are based on proven technology thereby ensuring that the uprated plant will perform to the same high standards of reliability and safety as other CANDU designs.

This report presents comparisons of the uprated design to other CANDU designs and shows that the choice of design alternatives is based on the design, manufacturing, commissioning and operating experience gained through other CANDU designs which are already operating or in the process of being constructed.

1.0 INTRODUCTION

Atomic Energy of Canada Limited - CANDU Operations has undertaken a wide ranging development program with the primary aim of reducing specific capital cost and reducing plant construction duration. One of the most attractive and effective ways of reducing specific capital cost is to increase the plant output, particularly when the increased output can be accomplished with minimal increased plant cost and no effect on plant reliability or safety. The results of study of this concept for CANDU-600 showed that increases in output of up to 17% relative to the Wolsung-1 and Lepreau-1 designs are readily achievable based on technology proven through design, construction and operation of other CANDU designs.

The following sections review the design changes required to uprate by 17% and compare the design parameters for the uprated plant to experience with other CANDU designs.

2.0 DESIGN CHANGES REQUIRED TO UPRATE BY 17%

The design changes required to uprate the nuclear steam plant output by 17% fall into three categories:

1. a) Changes that increase output while maintaining the same maximum fuel bundle power.
   b) Increase in bundle power.
2. Process changes required to deal with the greater heat output.

   and
3. Consequential changes required because of changes in the process design.

Each of these categories is described in turn below.

Before discussing these changes, attention needs to be given to the design approach used.

2.1 DESIGN APPROACH

The approach taken during the design of the uprated CANDU 600 was to keep the design parameters to the extent possible within the range of proven CANDU design, manufacturing, testing and operating experience. Section 3 of this report reviews the design parameters within this context. Improvements beyond this range are feasible but a conservative approach was taken to maximize plant reliability and safety. Two parameters are of major importance and are discussed in the following sections.

2.1.1 MAXIMUM FUEL BUNDLE POWER

At present, maximum fuel bundle powers in CANDU are experienced in the Bruce A Nuclear Generating Station. Bruce A has a licence limit of 1035 kW for bundle power with maximum bundle powers up to about 975 kW for full power operation. The Bruce plant has had good fuel performance at these power levels. As good fuel performance is key to minimizing radioactive contamination during normal operation and hence is a key to good plant availability, the same bundle power limits were chosen for the uprated CANDU 600 reactor.

2.1.2 OUTLET HEADER QUALITY

Average reactor coolant steam quality at the outlet headers is one of the key parameters in heat transport system design. Increasing outlet header qualities allow reduced capital cost (smaller pumps and steam generators) but with an increased potential for corrosion of the system materials and increased complexity for system control. It was felt to be prudent at this time to use the same design basis as in Lepreau-1. The end-of-life conditions for the heat transport system were chosen to be 4% average outlet header quality for the uprated plant as is the case for Lepreau-1. These conditions were maintained without changes to steam pressure or feedwater temperature.

2.2 CHANGES THAT INCREASE OUTPUT WHILE MAINTAINING THE SAME MAXIMUM FUEL BUNDLE POWER

There are four changes that fit this categorization. Altogether they allow core output to increase by 6% while maintaining the same maximum fuel bundle power output. The other 11% (of the total of 17%) of the power increase requires a uniform increase in core power output which increases the required bundle power licence limit to

1035 kW (from 935 kW in Lepreau-1) and increases typical maximum bundle power to 975 kW (from about 880 kW in Lepreau-1). The four changes that allow the 6% power increase with no increase in maximum bundle powers are:

2.2.1 EIGHT MORE CHANNELS

Eight channels have been added to the uprated design. Their location is shown in Figure 1. These channels can be added without redesign of the feeder layout for the other 380 channels. Altogether these channels contribute slightly more than 1% to the core output.

2.2.2 FOUR BUNDLE SHIFT IN CENTRAL REGION

A four bundle shift refuelling scheme has been adopted in the uprated design for the central 188 channels. The region affected is shown in Figure 2. This proven mixed four bundle/eight bundle scheme has been adopted by Ontario Hydro for Pickering B, Bruce A and Bruce B as it has some beneficial effects on burnup and it reduces the changes in fuel bundle and fuel channel power that occur when the channel is refuelled. In the uprated plant it allows a core power output increase of slightly less than 3% while maintaining the same maximum fuel channel and bundle powers.

2.2.3 RADIAL FLATTENING

By fuelling the outer regions of the core at a greater rate, it is possible to increase the power output in the outer regions of the core while maintaining the central high power region essentially unchanged. The proposed exit irradiations are shown in Figure 3. Fuelling in this way in combination with the next change (Section 2.2.4) allows about 2% greater power output while maintaining the bundle maximum powers as is.

2.2.4 REACTIVITY MECHANISM CHANGES

By substituting the guide tube tensioning springs with a shorter titanium design and replacing the locator with a Zircaloy locator, it is possible to reduce the amount of neutron absorbing material near

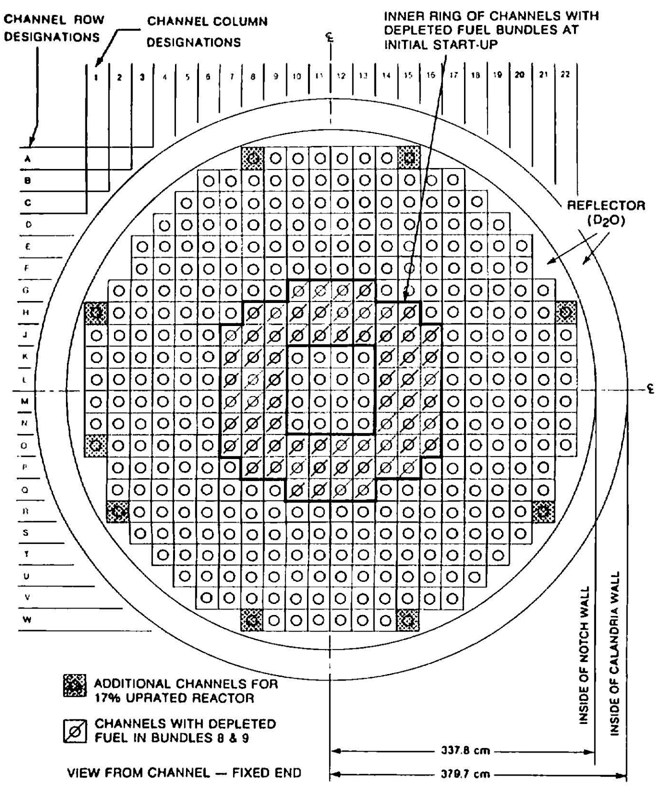

FIGURE 1 END VIEW OF REACTOR SHOWING ADDITIONAL CHANNELS AND INNER RING OF 68 CHANNELS WITH DEPLETED FUEL

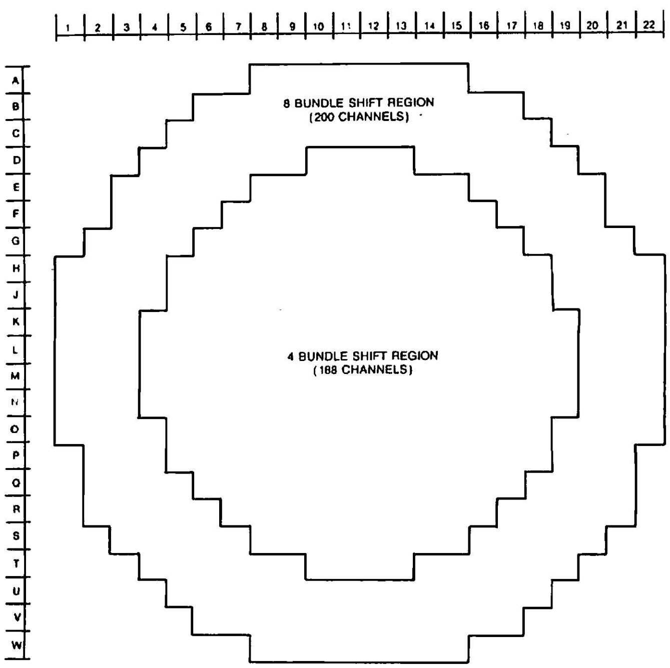

FIGURE 2 4-BUNDLE AND 8-BUNDLE SHIFT REGIONS

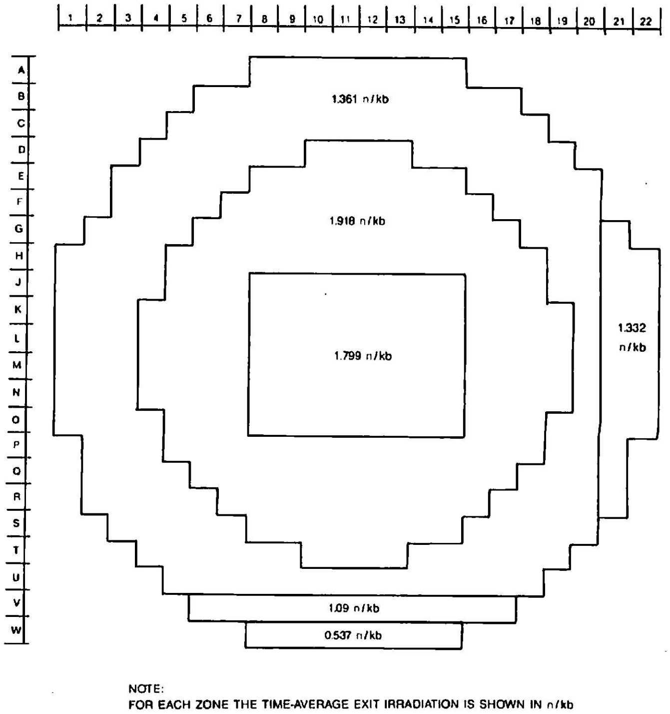

FIGURE 3 BURNUP ZONES

the edge of the core. A similar improvement has been incorporated for the Darlington plant. These changes allow a greater output from the edges of the core and a small burnup increase.

2.3 PROCESS CHANGES TO DEAL WITH THE GREATER HEAT OUTPUT

To deal with the 17% greater output from the unitwhile maintaining the same steam pressure and feedwater and service with temperatures, various straight forward modifications to the process equipment are required. Table #1 provides a list of the required changes. Comparisons of the proposed component designs to designs used elsewhere in CANDU stations follows in Section 3.

TABLE 1
CANDU 600 UPRATING
DESIGN CHANGES FOR 17% MORE OUTPUT
NSP PROCESS CHANGES

1. LARGER STEAM GENERATORS - SMALLER THAN DARLINGTON
2. LARGER HEAT TRANSPORT PUMPS - SIMILAR TO BRUCE/DARLINGTON
3. SMALL INCREASE IN PRESSURIZER CAPACITY - INCREASE IN LENGTH
4. FEEDER ORIFICE RESIZING
5. MODERATOR NOZZLE RELOCATION, PUMP AND HEAT EXCHANGER CAPACITY INCREASE.
6. SHUTDOWN COOLING HEAT EXCHANGER CAPACITY INCREASE
7. MAIN STEAM SAFETY VALVE CAPACITY INCREASE
8. ATMOSPHERIC STEAM DISCHARGE VALVE CAPACITY INCREASE

2.4 CONSEQUENTIAL CHANGES

As a consequence of the changes in process design parameters, some further changes to the plant are required.

2.4.1 INSTRUMENTATION AND CONTROL

With a modified pressurizer and steam generator design, some adjustments will be required in the heat transport and steam generator pressure and inventory control systems. These changes will be in setpoints and gain constants and no changes in technology will be required.

The regional overpower protection (ROP) trip setpoint will be increased to reflect the new operating conditions and better understanding of dryout phenomena for CANDU operating conditions. The new setpoints will allow up to 4% more operating margin at the uprated power than presently exists for Lepreau-1.

Some reduction in the heat transport high pressure trip setpoint will be required to meet Level B stresses following a loss of Class IV power. The change required will be small and the trip setpoint will remain above the liquid relief valve setpoint.

Other reactor trips will in principle be unchanged. Some adjustments to calibrate the trips to the new equipment or operating conditions will be required but the safety margins and the operating margins will be unchanged.

2.4.2 SHIELD PLUG

An improved shield plug was developed, tested and proven as part of the CANDU 950 design effort. As the uprated plant has somewhat higher channel flows, the improved shield plug has been adopted to ensure that vibration of the fuel will not damage the pressure tubes by fretting.

2.4.3 STEAM MAIN FAILURE RESPONSE

The uprated design features larger steam generators which will contain significantly more light water inventory. In the unlikely event of a steam main failure inside containment, the extra inventory would cause higher containment pressures than have been analyzed in the past. We are presently investigating the significance of these higher pressures and the impact on the design.

3.0 COMPARISON OF UPRATED DESIGN WITH OTHER CANDU DESIGN

3.1 NUCLEAR STEAM PLANT PROCESS DESIGN

3.1.1 INTRODUCTION

In this section the design bases used for determining the changes to the unit process equipment are described. The goal of the changes to the unit process equipment is to maintain process conditions that are similar to those in the standard CANDU 600 design.

All of the proposed increases in component capacities are within the range of proven CANDU design and manufacturing experience. Table 2 presents comparative component data of Lepreau-1 and

TABLE 2
COMPARISON OF CANDU 600 COMPONENT DATA FOR STANDARD AND UPRATED OUTPUT

COMPONENT	LEPREAU-1	UPRATED (17%)	DARLINGTON
Fuel Channels	380	388	480
Fuel Bundles	4560	4656	6240
Feeder Pipes	760	776	960
Steam Generator Heat			
Transfer Area (m^2)	3177	3437	4700
Heat Transfer (MWth)	516	606	664
Heat Transport Pump			
Flow Rate (L/s)	2228	2628	3100
Head (m)	215	263	224
Motor Operating Power (kW)	4780	6910	6700
Pressurizer Capacity (m^3)	45.3	46.8	63.7
Shutdown Cooling System			
Heat Transferred (MWth)	13.3	15.6	20.8
Moderator System Heat Exchangers Total Heat Load (MW)	96.1	115.0	138.0
Flow Rate (L/s)			
Moderator System Pumps	470	495	508

uprated designs for the CANDU 600 Nuclear Steam Supply System (NSSS) along with comparative Darlington data.

3.1.2 PRIMARY HEAT TRANSPORT SYSTEM

3.1.2.1 Design Operating Conditions

The design operating conditions for reactor coolant in the uprated PHT system have been chosen to be the same for Lepreau-1 design that is, 4% quality at the outlet header for 310°C and 10.0 MPa.

The 17% increase in power output from the core will be delivered at the same steam pressure (4.79 MPa at NSSS outlet) and with the same feedwater temperature (187°C at NSSS inlet) as for the Lepreau-1 design.

The core flow design criteria used for the Lepreau-1 design are maintained:

- maintain equal enthalpy rise across the core
- maintain similar ratios of pressure drop between inlet and outlet feeders of each channel.

In addition, flow distributions to the fuel channels will consider the flattening of the radial flux distribution at higher power. Feeder resizing increases the flow resistance for some of the high flow channels in the Lepreau-1 design which in retrospect do not need as high a proportion of the core flow as originally anticipated. In addition, flow through some channels will be increased. Feeder resizing will be limited for the most part to resizing of orifices. The redistribution of core flow due to the small increase in diametral creep for the fuel channels with an increase in fission power is also considered.

The pressures and temperatures remain essentially the same, relative to the design conditions for Lepreau-1.

- reactor coolant temperature at the inlet headers increased by 1°C to 267°C.
- inlet header pressure has increased from 11.33 MPa to 11.69 MPa.
- core flow has increased by 11% relative to as-built Lepreau-1 to 8970 kg/s.
- maximum channel flow at full power has increased to about 30 kg/s compared to 28 kg/s for Lepreau-1 as-built.
- steam flow has increased by 16% to 1213 kg/s (partly offset by the deletion of the reheater drain return).

Detailed operating conditions for the beginning and end of plant life are shown in Table 3.

For uprated design conditions, homogeneous fluid velocities in the various portions of the heat transport system will be within the guidelines established for the reference design, (i.e. 17 m/s) due to erosion/corrosion considerations.

3.1.2.1.1 PRESSURE TUBE PRESSURE - TEMPERATURE PROFILE

The design conditions of the 17% uprated CANDU 600 are an envelope of the pressure/temperature profiles of the maximum power, maximum flow, maximum quality and average power channels.

Relative to the design pressure/temperature profile of the reference channel for the standard CANDU 600.

- peak temperature in the channel has increased by 2.7°C to 319.5°C (located 2.45 m from channel exit).
- peak pressure in the channel has increased by 0.26 MPa to 11.94 MPa (located at channel entrance).

A stress assessment of the bounding pressure tube pressure/temperature profiles for 17% uprating has confirmed that the available margins for the pressure tube of the standard CANDU 600 can accommodate the small increase in pressure and temperature for uprated conditions with no design change.

3.1.2.1.2 Heat Transport System D_2O Inventory

The net increase in D_2O hold-up is 6.89 Mg due to eight additional channels and larger steam generators.

3.1.2.2 Additional Fuel Channels

The uprated reactor has 8 additional channels for a total of 388 channels. This increase is to provide some of the extra power generated by the reactor. Sixteen additional feeder pipes (8 inlet and 8 outlet feeders) are required corresponding to the increase in the number of fuel channels. The additional feeder pipes are all located within the existing feeder array. The sizes of the inlet and outlet headers do not change because the additional

TABLE 3

PHTS OPERATING CONDITIONS FOR 17% UPRATING

CASE	COLD	BEGINNING LIFE	END OF LIFE
Reactor Power (%)	0	117	117
RIH Temperature (°C)	37.8	262.9	267.0
Pressure (MPa)	12.1	11.71	11.69
ROH Temperature (°C)	37.8	310.0	310.0
Pressure (MPa)	10.0	10.0	10.0
Quality (%)	-	1.05	4.0
RIH-ROH P (MPa)	2.15	1.71	1.70
Core Flow (kg/s)	11581.6	9512.0	8972.0
PHT Pump Head (m)	263.32	259.92	263.23
Flow (ℓ/s)	2630.3	2759.8	2628.0
Steam Generator			
Heat Transfer Area (m^2)	-	3437.0	3437.0
Fouling Factor (m^2°C/kW)	-	0.0	0.0352
Inlet (D_2O) Temp. (°C)	-	309.4	309.1
Pressure (MPa)	-	9.91	9.86
Quality (%)	-	1.332	4.42
Outlet (D_2O) Temp. (°C)	-	262.6	266.8
Pressure (MPa)	-	9.52	9.50
Steam (H_2O) Flow (kg/s)	-	303.25	303.25

feeder nozzles are accommodated within the existing range of nozzle locations along the inlet and outlet headers.

3.1.2.3 Steam Generator

The PHT pumps and steam generators have been modified to increase their capacities in order to accommodate the 17% increase in plant output while maintaining the reference design basis.

Heat transfer area of the steam generator is increased to 3437 m^2, an 8% increase over the reference design. Improvement in steam generator sizing methods, the increase in design coolant flow and a $1^\circ C$ increase in coolant temperature at the steam operator outlet permit the transfer of 17% more heat than in the Lepreau-1 design. The increase in steam generator capacity is based on the same fouling factor (0.0352 $m^2{}^\circ C/kW$) used for previous CANDU 600 designs. The surface area margin provided by this factor caters for steam generator tube fouling and/or plugging over the life of the plant.

The secondary side H_2O inventory in the steam generator provides a heat sink capacity after reactor trip. The heat sink capacity of at least $1\frac{1}{2}$ full power minutes caters for the integrated decay power generation after reactor trip. In order to obtain the same safety margin as for the standard CANDU-600, the steam generator H_2O inventory is increased 17% corresponding to the increase in fission power. The larger inventory will provide the same time after reactor trip before the steam generator runs dry to allow for operator action in restoring feedwater flow to the steam generator.

The increase in steam generator capacity can be accommodated within the existing boiler box in the reactor building.

CANDU design and manufacturing experience is proven for larger steam generators (4700 m^2) which are used in Darlington (heat transfer for each Darlington steam generator is 664 MW which is about 10% larger than in the uprated CANDU 600 design.)

3.1.2.4 Heat Transport Pump

Most of the fission power increase of the uprated design is accommodated through a larger proportionate increase in pump capacity than steam generator capacity.

An upgraded Bruce B type pump, represented by its head/flow characteristic curve, has been used for calculating the operating conditions of the 17% uprated CANDU 600.

The increases in flow rate and head are approximately 17% and 22% respectively, relative to the Lepreau-1 design basis but the flow is only 11% higher than Lepreau as built. The new head is 263.4 m and the flow is 2628.0 ℓ/S with a motor operating power of less than 6.9 MWe.

The larger heat transport pump can be accommodated within the space allowed for the original pump.

The pumps used in the uprated plant are very similar in size to the Bruce/Darlington pumps, which have a motor operating power of 6.7 MWe.

3.1.2.5 Pressurizer

The pressurizer capacity is increased from 45.3 m^3 to 46.8 m^3 in order to accommodate the increased D_2O swell due to the increased hold-up in the steam generator and the 8 additional fuel channels (plus 16 feeder pipes).

The inside diameter of 2 m is retained and, the overall height of the vessel is increased 0.5 m to provide the larger capacity.

The larger pressurizer is readily accommodated within the vessel position in the reactor building of the standard CANDU 600.

3.1.3 MODERATOR SYSTEM

The locations and orientation of the inlet and outlet nozzles for moderator flow into the calandria are modified, to achieve better circulation and more uniform moderator temperatures within the calandria.

The inlet nozzles (4 on each side of the calandria) are relocated lower down on the calandria shell (approx. $120^\circ C$ from the vertical axis) and the nozzles are directed downwards (upwards in the standard design). The outlet nozzles are relocated to the upper half of the calandria (approx. $61^\circ C$ from the vertical axis) and there are more outlets of smaller diameter than in the Lepreau-1 design.

The layout of the moderator system piping to the relocated moderator inlet and outlet nozzles on the calandria is changed accordingly.

As part of a cost improvement program for the CANDU 600, the moderator shell/tube heat exchangers are replaced by plate heat exchangers and the 2 x 100% moderator pumps are replaced by smaller pumps both of which are required for full power operation. These changes take advantage of excess design margins in the original moderator system components.

For 17% uprating, the capacity of each moderator plate heat exchanger is increased to accommodate the larger heat load due to 17% increase in fission power.

The flow capacity of the moderator system is increased to accommodate the larger heat load to the heat exchangers, as well as reflecting heat transfer area versus pumping power considerations. Each pump is in fact smaller than in past CANDU 600 designs because of the change from a 2 x 100% pump configuration to the new configuration. Experience with moderator pumps indicates that the change to the new configuration will not have a significant impact on plant availability.

The Darlington main moderator circuit is similar to the uprated CANDU 600 main moderator circuit. However, Darlington has 2 x 100% main pumps (rather than using two smaller pumps both of which would be required for full power operation and 2 x 50% plate-type heat exchangers on a parallel - series arrangement such that either or both pumps can be utilized with either or both heat exchangers. The heat load for Darlington is 138 MW compared to 115 MW for the uprated CANDU 600. Therefore the proposed design is well within the range of CANDU design and manufacturing experience.

3.1.4 SHUTDOWN COOLING SYSTEM

The capacity of the shutdown cooling heat exchangers are increased to accommodate the additional decay heat due to 17% increase in fission power. The heat load per heat exchanger for the uprated CANDU 600 is smaller than for Darlington. Therefore the proposed design is within the range of CANDU design and manufacturing experience.

No change to shutdown cooling pump capacity is required.

3.2 FUEL PERFORMANCE

3.2.1 INTRODUCTION

The 17% uprated CANDU 600 uses the same 37-element fuel bundle that is presently used in all CANDU 600s. Over 60,000 of these 37-element CANDU 600 bundles have been irradiated with excellent results. The 37-element fuel design for the Bruce reactors is essentially the same as for the CANDU 600s. The only difference between the bundle designs are some minor geometry differences in end-cap shape and bearing pad position required for compatibility with the fuel handling systems. Ontario Hydro has irradiated over 230,000 Bruce bundles with results that have also been very good. The Bruce reactors are licensed for bundle powers up to 1035 kW, well above the licence limit for the present CANDU 600s. Actual maximum fuel bundle powers in Bruce A have generally been in the range 950 to 975 kW in the last few months with the reactors operating at or close to 100% full power. In the CANDU 600s actual maximum bundle powers have generally been about 875 kW. Uprating by 17% raises the licence limit to 1035 kW as in Bruce and raises the typical actual maximum bundle power to approximately 975 kW, about the same as in Bruce A.

3.2.2 FUEL DESIGN

Fuel design calculations and Bruce and test reactor experience show that the standard 37 element CANDU 600 fuel bundle will perform well under uprated conditions without any design change. Major performance parameters for the fuel in both Bruce and the CANDU 600 are compared in Table 4. This comparison demonstrates the very similar nature of Bruce and uprated CANDU 600 conditions.

3.2.3 FUEL DEFECT PROBABILITIES

In general fuel defects are caused by one of three mechanisms: debris in the heat transport coolant, manufacturing defects or stress corrosion cracking. Of these, the first two mechanisms are not affected significantly by the operating power level of the fuel and experience has shown that defect rates due to these mechanisms have been very low in the past. The third mechanism is reviewed in greater detail below.

There have been virtually no cases of stress corrosion cracking of the sheath in CANDU fuel since the introduction of CANLUB graphite coatings in 1973. The coatings protect the sheath by providing a barrier that absorbs fission products and prevents stress corrosion. We have done a large number of power boost experiments at CRNL over many years. As a result, we have established an understanding of these defects and a methodology for predicting defects. Over the years we have made improvements in the thickness of the graphite CANLUB and our assessment is that in the 17% uprated CANDU 600 we do not expect defects due to SCC in graphite CANLUB fuel.

In a reactor with a flattened axial power distribution the power boost associated with 4-bundle shifting is less severe than 8-bundle shifts because the power boost in bundles 1, 2 and 3 occurs at much lower burnup than it would in 8-bundle shifting. The low burnup means that at the time of boost the fuel sheath is more ductile, less embrittled and is therefore less susceptible to stress corrosion cracking. In the CANDU 600 the refuelling is in the opposite direction to Bruce. Thus the bundle experiences less corrosion and hydrogen pick up at the cool end of the fuel channel before being boosted in power.

The overall conclusion is that stress corrosion cracking failures will be less likely than in the Bruce 'A' reactors where experience has shown this defect mechanism to not be significant.

3.2.4 FUEL VIBRATION AND FRETTING, SLIDING WEAR AND STRENGTH

Fretting within the fuel is not a significant concern but the risk of fretting damage to the pressure tube must be minimal. The fuel dwell time in one channel position is much reduced because of 4-bundle shifting instead of 8-bundle shifts. This reduces the likelihood of pressure tube damage. The range of flow conditions for the uprated reactor

TABLE 4
SUMMARY FUEL PERFORMANCE PREDICTIONS

OUTER ELEMENTS	STANDARD CANDU 600 REACTOR	17% UPRATED REACTOR	BRUCE REACTOR
Bundle power kW	900 (50)*	1035 (50)*	1035 (50)*
Linear power rating (kW/m)	57.3 (50)	65.9 (50)	65.9 (50)
Fuel Centreline maximum temperature °C	1829 (0)	2152 (50)	2154 (50)
Sheath maximum inside temperature °C	373 (50)	382.5 (50)	382.5 (50)
Maximum value of gas release volume cm^3	11.9 (360)	38.6 (360)	35.4 (320)
Maximum % of gas release	9.6 (120)	26.3 (150)	25.9 (140)
Maximum value of internal gas pressure MPa	5.7 (310)	12.2 (210)	12.2 (170)
Maximum value of circumferential sheath ridge plastic strain %	.51 (0)	0.823 (0)	0.81 (0)

* The burnup values are in MW.h/kg(U) and written between brackets.

extends to the range of successful 5000 hour single-phase fretting tests at 30.4 kg/s flow using a flow through type of shield plug. Two-phase tests have shown that we can expect vibration levels similar to those in single-phase flow. Some additional testing is planned to confirm that the probability of pressure tube damage is extremely low. Testing and analysis is also planned to demonstrate that ample fuel bundle strength margins exist and that sliding wear on the pressure tube is within limits.

Fuel channel flow at full power for the uprated design will be about 30 kg/s. This compares to about 28 kg/s for the as-built Wolsung reactor and about 25 kg/s for the as-built Bruce A station. Hence the increase in maximum channel flow is quite small and the associated technical risk is consider insignificant based on the fretting tests due to date with further assurance supplied by the additional tests to be done.

3.2.5 OVERALL SUMMARY OF FUEL PERFORMANCE

Fuel defects due to debris and manufacturing flaws can be expected but at an extremely low rate. Defects due to other causes such as stress corrosion cracking should not occur based on extensive test data. An overall element defect rate similar to the current 0.004% is expected. CANDU reactor operators can detect and locate defects and remove them from the reactor at any power level and thereby maintain a clean core.

Vibration and fretting performance data at very high flows will be supplemented by some further testing. Fuel bundle strength and acceptable fuel channel sliding wear will also be confirmed by some further testing or analysis.

3.3 SAFETY AND LICENSING

The design of the special safety systems and their ability to mitigate the consequences of postulated accidents has been reviewed. The following section summarizes the key conclusions.

The reactor trip systems have been examined for the limiting scenarios and the only trips that required changes other than calibration to the new operating parameters are the regional overpower protection (ROP) trips and possibly the shutdown system #1 high pressure trip. The new ROP trip setpoint will yield about 4% more operating margin than the present Lepreau-1 setpoints. The high pressure trip may require some reduction in setpoint depending on other design variables such as the heat transport pump inertia. The final setpoint will remain above the liquid relief valve setpoint so the possibility of spurious trip will remain low. All the trip parameters remain unchanged in principle and continue to provide appropriate safety and operating margins.

Emergency Core Cooling System (ECCS) performance has been examined for a critical break large loss of coolant. The results show that sheath temperatures will be reduced quickly with fuel channel refilling also occurring quite quickly.

Containment performance has also been examined. For some events releases of radioactivity have increased. Better modelling of containment has been used in the uprating analysis, including more realistic models for organic iodine formation and inorganic iodine physical chemistry. The more realistic modelling more than compensates for the increased iodine source term. Further reductions are possible for cases involving impairments of the containment boundary through further improved modelling of iodine bearing water droplets aerosols.

The analyses in the review report have been reviewed by the Atomic Energy Control Board. On the basis of the AECB staff review of the analysis reports and other information, AECB staff believe that the final design can be meet the requirements for a construction licence as of December 31, 1985 and that given adequate completion of other operating licence requirements, any design changes required to get the final operating licence would not be major.

3.4 FUEL HANDLING SYSTEM

Conservative projections of the time required for refuelling and fuel handling system maintenance based on Pickering and Lepreau experience indicate that about 100 hours per week will be required to refuel and maintain the fuel handling system for the uprated reactor. A similar estimate for Wolsung-1 indicates about 55 hours per week are required. It should be noted that the Bruce stations have similar fuelling times and maintenance requirements on a per channel basis but only have three sets of fuelling machines to service four 480 channel 825 MW units. The Bruce stations have also adopted a mixed 4 bundle/8 bundle shift fuelling scheme with no resulting significant contributions to station unavailability from the fuel handling system.

3.5 FUEL CHANNELS

The higher flux in the uprated reactor will cause increased channel creep. To a first order, the channel creep increase is proportional to the power increase in the channel. Design changes have been proposed to accommodate the increased creep rate for a forty year period.

Operation for forty years is an extrapolation of the existing data base for channel behaviour. Material properties including corrosion, hydriding, critical crack length, etc. at high integrated flux are being investigated by research programs at Chalk River Nuclear Laboratories and by in-service monitoring programs at the operating CANDU units.

3.6. BALANCE OF PLANT

Turbine generators of the size required for the uprated plant are already in operation at the Bruce nuclear operating stations. These units operate at similar steam conditions. Their performance to date has been on par with other turbine generators in CANDU plants.

4.0 PRESENT STATUS

The Improved CANDU 600 design (which includes 17% uprating) has been offered to two prospective clients in Korea and Yugoslavia. Some other aspects of the Improved CANDU 600 design are discussed in another paper in this session. (Reference 1). Estimates made for these projects indicate that the 17% uprating is responsible for an 11% reduction in specific capital cost (dollars per kilowatt) for a unit.

5.0 FUTURE DIRECTIONS

The main roadblock to further levels of operating is the limit on fuel bundle power output imposed because of lack of large scale experience at bundle powers above 1000 kW. In order to allow higher bundle powers, while keeping fuel element powers at their present levels, a program to develop a new fuel bundle has been initiated at CRNL. Some of the key considerations relating to further levels of uprating are discussed in another paper at this conference (Reference 2).

6.0 CONCLUSIONS

Our review of the design parameters for the uprated CANDU 600 has shown that for the most part these parameters are within the range of design and operating experience accumulated on the Pickering, Bruce, CANDU 600 and Darlington stations. Any extrapolations beyond that data base attributable to uprating (channel flow is the main example) are small and the technical risk associated with these extrapolations is also small.

AECL is therefore confident that the uprated CANDU 600 will meet the same high standards of performance and safety that have been exhibited by Lepreau-1 and other CANDU plants.

7.0 REFERENCES

1. N.G. Craik, "Improved CANDU 600 Designability", Proceedings of the Canadian Nuclear Society 1987 Annual Conference.

2. N.J. Spinks and D. Groeneveld, "Thermal Power Uprating of CANDU Reactors", Proceedings of the Canadian Nuclear Society 1987 Annual Conference.