# Canadian CANDU Fuel Development Program and Recent Fuel Operating Experience

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## Abstract

This paper reviews the performance of the CANDU® fuel in the Canadian CANDU reactors in 1997 and 1998. The operating experience demonstrates that the CANDU fuel has performed very well. Over the 2-year period, the fuel-bundle defect rate for all bundles irradiated in the Canadian CANDU reactors has remained very low, at between 0.006% to 0.016%. On a fuel element basis, this represents an element defect rate of less than about 0.0005%.

One of the reasons for the good fuel performance is the support provided by the Canadian fuel research and development programs. These programs address operational issues and provide evolutionary improvements to the fuel products. The programs consist of the Fuel Technology Program, funded by the CANDU Owners Group, and the Advanced Fuel and Fuel Cycles Technology Program, funded by Atomic Energy of Canada Ltd. These 2 programs, which have been in place for many years, complement each other by sharing expert resources and experimental facilities. This paper describes the programs in 1999/2000, to provide an overview of the scope of the programs and the issues that these programs address.

## 1.0 Introduction

Fuel development activities in Canada have been, for a number of years, driven by 2 requirements. The first requirement is to address issues that are related to fuel operation and performance. To meet this need, the Canadian utilities — Hydro-Québec, New Brunswick Power, Ontario Power Generation Inc. — and Atomic Energy of Canada Limited (AECL), jointly or in part, support work programs under the CANDU Owners Group (COG). The focus of these COG programs is to maintain and improve the reliability, economics, and safety of the 28- and 37-element natural-uranium CANDU fuel bundles in operating stations. The second requirement is to improve the CANDU fuel products. The focus of these programs is to develop advanced fuel bundles and fuel cycles that will reduce capital and fuelling costs, increase the operating and safety margins, improve natural-uranium utilization, and provide synergy with other reactor systems to improve resource utilization and spent fuel management (Reference 1). These programs are supported by AECL in its Advanced Fuel and Fuel Cycles Technology Program.

The COG and the AECL fuel development programs have complemented each other for many years. The programs have shared technical expertise as well as the experimental facilities. The results from one program have also been valuable for the performance and direction of the other, and vice versa.

Since the review of the Canadian fuel development programs at the CANDU Fuel Conference in 1997 (Reference 2), the AECL Advanced Fuel and Fuel Cycles Technology Program has met a number of important milestones. One of them is the demonstration irradiation of 24 CANFLEX® bundles at the

Point Lepreau Generating Station (PLGS). This is a prerequisite to the full-core implementation of the CANFLEX bundles in a power reactor. During the same period, the COG Fuel Technology Program has also been making important achievements. For example, the irradiation of 37-element bundles with different design parameters was completed at the PLGS. The post-irradiation examination (PIE) of the irradiated fuel elements has since been conducted in the hot-cells at Chalk River Laboratories (CRL) and has provided valuable information on the behaviour of fuel elements that have different $UO_2$ densities and $UO_2$-sheath clearances.

Since 1997, the size of the COG Fuel Technology Program has been significantly reduced for several reasons. Many of the existing programs that address issues of common interest to the utilities have been successfully completed. The fuel has also been performing very well in reactors. Additional issues of common interest are therefore fewer and are considered to be of lower priority. Finally, the re-structuring of COG and the attendant budget changes have effected a funding reduction in the Fuel Technology Program. It is anticipated that the effort in the COG Fuel Technology Program would be augmented in the future by direct utility support, for fuel-related issues that are specific to each utility.

In this paper, the fuel performance data for the Canadian CANDU stations is first presented. As noted earlier, the fuel performance has been very good. The Canadian fuel development programs in 1999/2000 are then described in the paper, to provide an overview of the scope of the programs and the issues that these programs address.

## 2.0 Fuel Performance in Canadian CANDU Stations

One of the typical measures of fuel performance is the fuel-bundle defect rate, expressed as the percentage of the bundles in which sheath failure has occurred during irradiation. The means of fuel defect detection, and the convention of reporting the number of fuel defects differ among the Canadian CANDU stations. The Bruce A, Bruce B, Point Lepreau and Gentilly-2 nuclear generating stations (NGSs) have the gaseous fission-product monitoring system (GFP) to monitor the activity level in the heat-transport coolant, and the delayed-neutron scan system to locate the channel that contains the failed fuel. The Darlington NGS has the GFP system only. The Pickering NGS A and NGS B monitor the activity in the heat transport coolant using grab-samples. In addition to the coolant activity level, the other indicators of fuel defects include coolant activity changes upon fuelling, gamma signals and alarms when the fuel is discharged, wet sipping performed on the suspect discharged fuel, and inspection of the suspect bundle in the fuel bay. Some stations report a fuel defect when there is strong collaborative evidence from the defect indicators. Some stations report the number of estimated fuel defects, and report confirmed defects only when the defects are visually confirmed during inspections in the fuel bay. Therefore, depending on the monitoring system available at the stations and the reporting convention, the number of estimated fuel defects may be an overestimate, whereas the number of confirmed fuel defects may be an underestimate of the actual number of fuel failures.

Table 1 summarizes the number of estimated and confirmed fuel-bundle defects reported in the Canadian CANDU stations for the years 1997 and 1998. Note that the number of estimated defects include the confirmed defects. For the confirmed fuel defects, the causes are also indicated in the table. The causes of the fuel defects are classified into 4 types:

i) Debris fretting, caused by debris in the heat transport system, that lodges within the fuel bundle and frets through the sheath;

ii) Manufacturing-related defects;

iii) Stress corrosion-cracking defects (SCCs), normally associated with power ramps;

iv) Unassigned, i.e., defects for which the root causes were not identified.

Of the stations listed in Table 1, the Gentilly-2 NGS and the PLGS each has a single reactor unit. The Bruce A and B, Darlington, and the Pickering A and B NGSs all have 4 nuclear reactors per station. The Pickering A and Pickering B stations use the 28-element fuel bundles, and the other stations use the 37-element fuel bundles. Some reactor units were shut down, or outages have occurred in others at some time during the 2-year period. The operating history has been reflected in the number of bundles discharged from the stations.

The cumulative fuel defect rates over the 2-year period for all Canadian CANDU stations, expressed as the percentage of the number of bundle defects to the total number of bundles discharged, were very low. The fuel defect rate is 0.006% for confirmed defects, and 0.016% for all estimated defects. The actual fuel defect rate is therefore between 0.006% and 0.016%. This operating record demonstrates excellent performance for CANDU fuel. Because most failed bundles involve only 1 fuel element, the element defect rate over the 2-year period for all discharged bundles in Canada is on the order of 0.0002% to 0.0005%.

Five out of the eight confirmed fuel defects were attributed to debris fretting. This category of fuel defect should ideally have been excluded in the calculation of the defect rate because debris fretting is beyond the control of fuel design and fuel manufacturing, and is therefore not related to fuel performance. Two out of the eight confirmed fuel defects were attributed to unassigned causes, and one was a manufacturing-related defect. Between 1997 and 1998, fuel performance has not been a concern and has not caused any loss of power generation in any of the Canadian CANDU stations.

## 3.0 COG Fuel Technology Programs

Since the last review of the COG Fuel Technology Program at the 1997 International Fuel Conference, a number of activities funded under this program have been completed. Four activities that deal with issues of common interest to all COG partners continue into 1999/2000. They include a review of fuel specifications, PIE of fuel, compilation of an irradiated fuel database, and validation of the ELESTRES - IST code.

### 3.1 Fuel Specifications Review

The review of the fuel technical specifications and the documentation of the rationale for the specification limits continue into 1999/2000. The review identifies any needs for updating the specifications, in order to capture new information from research and development, manufacturing, and operating experience. By documenting the rationale, the reasoning behind the specifications can also be understood by the users, and this understanding provides additional quality assurance in the fuel procurement process. The documentation of the rationale is also necessary for the training of newcomers to the nuclear fuel industry.

As of today, the reviews of all the specifications for fuel materials and parts have been completed and documented. A result of this review, which deals with the hydrogen gas limit in fuel elements (Reference 3), will be described in this conference.

### 3.2 Post-irradiation Examination of Fuel

The PIE of fuel that is of common interest to the Canadian CANDU stations is being continued in the 1999/2000 program. In 1998, 2 fuel bundles from the Gentilly-2 NGS that were in a channel with diametrally crept pressure tube were examined. The purpose of this examination was to determine whether creep, which is a part of reactor aging, has any visible effects on fuel behaviour and performance (Reference 4). In 1999, it is planned to examine fuel bundles that resided near the liquid zone controllers and hence had been subjected to many power changes, to see whether power changes affect fuel performance.

The irradiation and PIE of six "special" bundles were also of generic interest. These were bundles specially fabricated to (i) high and low $UO_2$ densities, (ii) large and small diametral clearances, and (iii) with and without a CANLUB coating. These "special" bundles were all irradiated at the PLGS. Bundles with high and low densities and with large and small clearances were examined at the hot-cells at CRL in 1998. The objective of these PIE was to provide the designers with in-reactor data, to confirm the technical specifications of density and clearance. A paper describing the results of the PIE will be presented at this conference (Reference 5). In 1999, irradiated bundles with and without CANLUB coating will be examined to investigate any effect of the CANLUB coating on the fuel chemistry and fuel temperature.

### 3.3 Irradiated Fuel Database

An electronic database of fuel behaviour parameters including fission-gas release, sheath strain, power-burnup history, etc. has been compiled, using the PIE results of CANDU fuel elements irradiated in the power reactors and test reactors. This database will be used extensively for the validation of the fuel behaviour code ELESTRES-IST. In 1999, the database will be updated to include additional data that have since been collected. At present, the database consists of about 337 cases, about one third from power reactors, and the rest from irradiation in test loops or reactors.

### 3.4 ELESTRES-IST Code Validation

The ELESTRES-IST code is supported by the COG partners as the industry standard toolset (IST) for modelling fuel-element behaviour during reactor operation. The code, based on the finite-element technique, combines the features in the ELESIM (Reference 6) and the ELESTRES (Reference 7) codes. The IST version has been subjected to vigorous tests to ensure the stability of the numeric schemes, and to examine any discontinuities in the code predictions.

Verification and validation of the ELESTRES-IST code will start in 1999. The validation will be performed on a phenomenological basis, in which key phenomena that the code models will be identified, appropriate irradiation data for validation of the phenomena will be compiled, and the code's predictions will be compared with the compiled analytical data. Upon completion of the phenomenological validation, the code will be validated by comparing the code's predictions with irradiation data on an integral basis.

## 4.0 AECL's Advanced Fuel and Fuel Cycles Technology Program

The Advanced Fuel and Fuel Cycles Technology Program at AECL has 3 categories of activities. The first category deals with the development of new fuel bundle designs that can be used for the natural-uranium fuel as well as other advanced fuel and fuel cycles. For example, the 43-element CANFLEX bundle has been developed as a successor to the 28-element and 37-element bundle designs. Incorporating the critical heat flux (CHF) enhancement buttons, the CANFLEX bundle offers higher critical channel power (CCP) performance compared with that of the 28-element and the 37-element fuel bundles. Hence the use of the CANFLEX bundles in reactors will increase the operating margins, or permit power uprating (Reference 8). At the same time, by subdividing the bundle into a larger number of elements, and by using two different element sizes, the CANFLEX bundle has a 20% lower peak linear element rating compared with that of the 37-element bundle for the same bundle power output. These features make the CANFLEX bundle an ideal carrier for the advanced fuel cycles that demand operation at higher fuel burnup.

The second category of programs deals primarily with the fuel matrix, and includes activities in plutonium mixed-oxide fuel, DUPIC fuel, thorium fuel, and other alternate fuel and fuel cycles. Using the CANFLEX bundle as the carrier for the advanced fuel and fuel cycles, these programs focus on the fuel element performance, fabrication optimization, thermalhydraulic performance, as well as the core physics, fuel management, and economic aspects of these advanced fuel cycles.

The third category consists of programs in fuel-supporting technologies. These programs deal with the maintenance and improvement of fuel design technologies, such as the update of fuel specifications and the validation of fuel-design computer codes. These programs also support the development of improved design features, such as the optimized fuel-element design for high burnup operation, the enhancement in CHF performance, and the improved method of appendage attachment. Upon completion of the development, these features will be deployed in the advanced fuel bundle design.

## 4.1 CANFLEX Fuel Bundles

The CANFLEX-NU (natural uranium) bundle program has reached an important milestone since the last review at the 1997 conference. On September 3, 1998, 8 CANFLEX bundles were loaded into the PLGS to begin the 24-bundle demonstration irradiation program. A report of the demonstration irradiation program and its status is given in Reference 9. To date, 24 bundles had been loaded into 2 channels under normal fuelling, and 8 bundles have since been discharged, 4 from each channel. The immediate plan is to select one of these 8 discharged bundles for shipment to the hot-cells at CRL for a detailed destructive examination to confirm the fuel's performance.

The 8 discharged bundles successfully completed their normal residence times in the reactor. During the irradiation, the fuel performance was monitored, as usual, with the failed fuel monitoring systems, and there was no indication of any fuel failure. The preliminary inspection of these bundles in the fuel bay shows no abnormal behaviour. Some markings were reported on the end plates during the bundle inspection at the fuel bay. However, it has since been confirmed that these markings were made during the manufacture of the end plates and were not caused by interactions with the fuel handling systems.

During the past year, the focus of the CANFLEX-NU program is to complete all the necessary design work, to allow for full-core implementation of the CANFLEX bundles. The CHF measurements in Freon have shown that the CANFLEX bundle will provide a significant increase in the CCP because of the use of the CHF enhancement buttons. This has since been confirmed by water CHF tests performed at the STERN Laboratories during 1999. The water CHF tests were done in a full-size channel rig with 12 CANFLEX-bundle simulators that were heated by internal heaters. The CHF locations were measured by sliding thermocouples. The CHF test program is now complete and is described in a paper (Reference 10) presented at this conference.

Another achievement in the past year is the completion of the second cross-flow test on the CANFLEX fuel bundle, performed at the Korean Atomic Energy Research Institute (KAERI)*. The cross-flow test simulates the bundle when it resides in the liner tube region of the end-fitting during on-power refuelling. The original cross-flow test, performed in 1996, confirmed that the CANFLEX fuel bundle can withstand cross-flow conditions in excess of 4 h. This time limit exceeds the design requirement that specifies that bundles would reside in the cross-flow region for up to 4 min during refuelling. However, from an operational standpoint, it is desirable to know the time limit that the bundle can remain in the cross-flow region without incurring any damage. This time limit will then define the allowable recovery time if an abnormal fuelling event occurs in which bundles are stuck under the cross-flow conditions. The follow-up cross-flow test was successfully performed at KAERI in early 1999 to quantify this time limit.

In addition to the above tests, further design analyses were completed in the past year to provide further confirmation and support to the in-reactor and out-reactor fuel qualification tests. These analyses include, for example, mechanical analysis of the stresses on the bundles during fuelling, end flux peaking, etc. These analyses are discussed in References 11 and 12.

By April 2000, the supporting design work for the full-core implementation of the CANFLEX-NU bundles will be complete. An exception would be the PIE of one of the CANFLEX bundles that will be discharged from the PLGS in the year 2000. The remaining work required for full-core implementation, which is not a part of the supporting design activities, includes the safety analyses that are required to obtain license approval, the activities at the station associated with implementing a new fuel type, and the procurement of CANFLEX bundles. Although AECL is expected to provide significant effort to support these remaining tasks, the lead will be provided by the staff of the PLGS.

With the CANFLEX-NU design work mostly completed, AECL's development effort will now focus on the CANFLEX-SEU (slightly enriched uranium) fuel. One of the advantages of enrichments of up to 1.2% in a CANDU reactor is to allow flattening of the core flux distribution, which leads to a higher core power for the same size core. This is a major benefit because it can reduce the unit capital cost of new reactors. By using enriched fuel, it also has the advantage of reducing the volume of spent fuel, and hence reducing the back-end costs associated with spent fuel storage and disposal (Reference 13). Recovered uranium (RU) from the reprocessing of spent LWR fuel is a variation of the SEU cycle. RU, which has an enrichment of about 0.9%, is expected to be a low-cost alternative to the conventional SEU. The CANFLEX-RU work is a joint program by AECL, British Nuclear Fuels Limited (BNFL) and KAERI (Reference 14).

In 1999/2000, the AECL program on CANFLEX SEU/RU fuel is to begin the design verifications that are necessary to qualify the CANFLEX SEU/RU fuel bundle. The design verification will include all the tests and analyses that would support the demonstration irradiation of CANFLEX SEU/RU bundles in a power reactor. Reactor physics simulations are also being performed to establish a reference CANFLEX SEU/RU core design. In the areas of safety analysis, a review of the implications of the use of CANFLEX RU/SEU fuel in the consequences of all design-basis accidents will be performed, to define the work scope for more detailed assessment.

The CANFLEX bundle has been qualified for the channel flow rates in present CANDU reactors. However, for advanced core designs, it would be desirable to define the high flow limit to which the CANFLEX bundle can operate. A higher channel flow will allow up-rating of the channel power, which in turn can reduce the capital cost of a reactor. In 1999/2000, an investigative program is planned to

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* The CANFLEX bundle has been developed jointly by AECL and KAERI since 1991, and before that by AECL since 1986.

measure the fuel vibration levels and the fuel and pressure tube fretting-wear rates of the CANFLEX bundles under high flow conditions. The test will be performed in the flow visualization rig at AECL's laboratory at Sheridan Park. The test rig consists of up to 4 CANFLEX bundles in an acrylic pressure tube section simulating the fuel channel inlet. The bundles will be subjected to channel flow rates in increasing increments, and the vibration levels of the bundle will be measured.

## 4.2 Advanced Fuel and Fuel Cycles

For years, AECL has maintained a number of advanced fuel and fuel cycles programs. These programs aim to establish the technical feasibility, to improve the fabrication processes, and to provide process data for economic evaluation of the various advanced fuel and fuel cycles. These research and development programs (Reference 2) — including low void reactivity fuel, DUPIC fuel cycle, thoria fuel, Pu mixed-oxide fuel, inert-matrix fuel—are continuing into 1999/2000. Some of these programs are discussed below:

### 4.2.1 DUPIC Fuel

The DUPIC fuel cycle (Direct-Use of spent PWR fuel in CANDU) involves converting the pressurized-water reactor (PWR) spent fuel into the CANDU fuel using a dry process called OREOX. By subjecting the PWR spent fuel to repeated oxidation and reduction processes (OREOX), selected fission products that are highly neutron-parasitic can be removed. The resulting powder, when fabricated into CANDU fuel, can be used directly in a CANDU reactor. The OREOX process provides a higher degree of proliferation resistance than conventional reprocessing. The DUPIC program is jointly sponsored by AECL, KAERI, and the US Department of State. At AECL, 3 DUPIC elements were fabricated at AECL's Whiteshell Laboratories (Reference 15). These elements are now being irradiated in the NRU reactor at CRL. The activities in fabrication and irradiation are aimed at confirming the technical feasibility of the cycle, optimizing the process, and obtaining data for economic assessment.

### 4.2.2 Thorium Fuel Cycles

For the thorium fuel cycles, AECL maintains an ongoing program in thoria fuel fabrication, test irradiation, and fuel management studies of thoria-fuelled CANDU cores (References 16 and 17). In 1999/2000, further fabrication development will be carried out, which would lead to a test irradiation in order to provide performance feedback to qualify the improved fabrication processes.

### 4.2.3 Pu Mixed-oxide Fuel

The higher initial enrichment and discharge burnup of light-water reactor (LWR) fuel, compared with the CANDU fuel, result in a higher concentration of plutonium in the spent fuel. The plutonium from the reprocessing of LWR spent fuel can be fabricated into mixed-oxide fuel and used in the CANDU reactor. Because of the higher neutron economy, twice as much energy can be derived from the plutonium in a CANDU reactor than in a LWR. AECL maintains a program of mixed-oxide fuel fabrication and irradiation, to acquire additional data in performance behaviour and fabrication processes. Some of the results were presented in Reference 18.

A variation of the application of Pu mixed-oxide fuel is the dispositioning of weapons-grade plutonium. This involves the fabrication of mixed-oxide fuel, using weapons-grade plutonium, and utilizing this fuel in a CANDU reactor. Canada, through AECL, is participating in the weapons-grade Pu disposition program with the United States and the Russian Federation (Reference 19).

### 4.2.4 Inert-matrix Fuel

AECL has an ongoing development program in inert-matrix fuel. Inert-matrix fuel is fuel in which a fissile component, plus optionally actinide-wastes, are incorporated in an inert-matrix, i.e., a matrix which does not produce Pu or higher actinides. Actinide wastes, especially $^{241}Am$, $^{237}Np$ and $^{244}Cm$, which are concentrated during reprocessing and are the most carcinogenic can be annihilated. The destruction of weapons-grade or reactor-grade plutonium is another application. Several candidates for use as the inert-matrix are studied worldwide, including zirconia, spinel, and other materials. Over the last 4 years, AECL has examined silicon carbide (SiC) as a potential inert-matrix. SiC has several material properties which make it an excellent candidate—extremely high thermal conductivity (leading to fuel centerline temperatures as low as 100 °C above coolant temperature), resistance to oxidation and low neutron absorption. The physics of annihilation of Pu and the actinides in CANDU reactors with this fuel have been studied and show that the versatility of the CANDU reactor can allow full-core loading. Consequently, disposition rates of the actinides are high (Reference 20). In addition, studies in fabrication, compatibility with coolant water, compatibility with Zr-based cladding under accident conditions, waste disposal and accelerator-simulations of in-reactor damage have been conducted (Reference 21). The results confirm that SiC is an excellent candidate for inert-matrix fuel.

## 4.3 Fuel-supporting Technologies

AECL maintains a program of technology development to support the advanced bundle design and the deployment of advanced fuel cycles. One technology is the development of high burnup fuel, which has a target burnup of about 3 times that of the present natural-uranium fuel. Currently, fuel elements designed for high burnup operation are being irradiated in the NRU reactor at CRL. A program has also been initiated to extend the stress corrosion cracking (SCC) thresholds to higher burnup, through analytical modelling, irradiation, and power ramp tests.

Another development activity is the improved technique of attaching appendages to the Zircaloy fuel sheath. Currently, the attachment of bearing pads and spacers in the CANDU 28- and 37-element bundles, as well as the CHF buttons in the CANFLEX bundles, are being done by beryllium brazing. Although this attachment technique has proven to be very reliable, as confirmed by the very low number of fuel defects in CANDU reactors, it has been known that the heat-affected zone in the braze region is the part of the sheath that is most susceptible to SCC. Although at present, SCC does not pose a fuel performance limitation, there is still an advantage to increase the margin to SCC, particularly for fuel that operates at high burnup. The improved attachment techniques aim to reduce or eliminate the heat-affected zone.

Another activity in fuel-supporting technologies is to maintain and improve AECL's fuel design capability. To this aim, AECL funds the revisions to the fuel specifications. At present, all fuel specifications for fuel parts and materials have been revised, and the specifications have been issued to all Canadian CANDU utilities, the Canadian fuel manufacturers and the uranium supplier for their comments. It is expected that these revised specifications will be formally issued by the end of 1999.

Another activity is to maintain and improve the fuel design codes (References 22 and 23). In addition to the ELESTRES-IST fuel-element code, AECL maintains a suite of fuel design codes that range from dedicated finite-element codes for assessing stress on end plates and end caps, to specialized codes for assessing SCC susceptibility. These codes are being subjected to a vigorous program of verification and validation, to bring them to the current QA standards recommended by the Canadian Standards Association.

The fuel design codes are useful in assessing whether operating power transients may incur a risk of fuel defects or whether certain fuel manufacture deviations may have an adverse effect on fuel performance. The codes are also useful to assess the sensitivity of fuel performance to design parameters. The codes can also be a means for design qualification, replacing some of the qualification testing that needs to be done if the codes are not available. In the latter case, analyses with validated codes may reduce the reliance on qualification testing, and in so doing, may provide a means to reduce the cost, and shorten the time period from product conception to commercialization.

To complement the development of fuel performance codes for high burnup application, an investigative program of obtaining physical properties of fuels, such as diffusion rates, thermal properties, and chemistry is also being conducted. A new technique to measure fission-product diffusion coefficient in $UO_2$ fuel using the accelerator and secondary-ion mass spectrometry was recently developed (Reference 24).

## 5.0 Concluding Remarks

The Canadian CANDU fuel development programs in 1999/2000 have been described in this paper. The programs consist of the COG-sponsored Fuel Technology Program and the AECL's Advanced Fuel and Fuel Cycles Technology Program. The programs cover operational issues related to the present 28- and 37-element fuel, and new advanced products such as the 43-element CANFLEX bundles and advanced fuel cycles. These programs, directly and indirectly, contribute to the remarkably good fuel performance that has been experienced by the Canadian CANDU stations. These programs also support advanced fuel bundles and fuel cycles that would lead to the benefits of capital and fuelling costs reduction, operating and safety margins increase, and improvement in natural-uranium utilization and spent fuel management.

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## Table 1: Fuel Bundle Defects in Canadian CANDU Stations in 1997 and 1998

| Station/Year | Estimated No. of Defects | Confirmed Defects | | | | | No. of Discharged Bundles |
|---|---|---|---|---|---|---|---|
| | | Fretting | Manu. | SCC | Unassigned | Total | |
| Bruce A | | | | | | | |
| 1997 | 0 | 0 | - | - | - | 0 | 7472 |
| 1998 | 1 | 1 | - | - | - | 1 | 1298 |
| Bruce B | | | | | | | |
| 1997 | 1 | 1 | - | - | - | 1 | 20550 |
| 1998 | 2 | 2 | - | - | - | 2 | 17602 |
| Darlington | | | | | | | |
| 1997 | 2 | - | - | - | - | 0 | 15508 |
| 1998 | 5 | - | - | - | - | 0 | 21349 |
| Gentilly 2 | | | | | | | |
| 1997 | 1 | - | 1 | - | - | 1 | 4324 |
| 1998 | 0 | - | - | - | - | 0 | 3900 |
| Pickering A | | | | | | | |
| 1997 | 2 | - | - | - | - | 0 | 8054 |
| 1998 | 0 | - | - | - | - | 0 | 0 |
| Pickering B | | | | | | | |
| 1997 | 1 | - | - | - | - | 0 | 9424 |
| 1998 | 3 | - | - | - | - | 0 | 11740 |
| Point Lepreau | | | | | | | |
| 1997 | 3 | 1 | - | - | 2 | 3 | 3488 |
| 1998 | 0 | - | - | - | 0 | 0 | 3828 |
| Total for All Stations | 21 | 5 | 1 | 0 | 2 | 8 | 128537 |
| Cumulative 2-yr. Bundle Defect Rate based on Estimated No. of Defects | | | | 0.016% | | | |
| Cumulative 2-yr. Bundle Defect Rate based on Confirmed No. of Defects | | | | 0.006% | | | |