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- CANDU Fuel Conference
- India's Fuel Program
- Advanced Fuels
- Maintenance Conference
- UDM Design
- Three Pioneers

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Cover Photo

The cover photograph shows the Pickering Generating Station, the focus of much media coverage following the release of the Epp report on Pickering A.

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EDITORIAL

Pickering A: a damning but incomplete report

Just as this issue of the *CNS Bulletin* was going to press the report of the Pickering "A" Review Panel was released. *(For those who have not yet read it, the report is available from the Ontario Ministry of Energy website <www.energy.gov.on.ca>)*

This is a damning indictment of the management of the rehabilitation program for the Pickering "A" units and confirms the suspicions many of us have had over the past few years. A statement in the "Introduction" sums up the conclusion of the three-person Review Panel:

> *While the analysis of what went wrong provides a catalogue of problems, ultimate responsibility must lie with the OPG Board and senior management.*

While the Panel acknowledges that Ontario Power Generation encountered an unexpected regulatory requirement - the need for a formal environmental assessment - they point out that OPG did not use the time for that assessment (19 months) to "ensure that critical activities such as design engineering were completed before construction began"

Cost estimates were particularly bad. Over the period from1999, when the OPG Board first approved the rehabilitation project, to mid 2003, there were 11 changes (all increases) in the overall estimate, from \$1.1. billion to \$2.5 billion. Although the report, rightly in our view, criticizes the Board, it does note that senior management presented the Board with different (increased) estimates every few months from late 2001 to mid 2003.

Although accepting all of the Panel's criticisms, there are, in our view, serious omissions in the report. Nothing is said about the lack of appreciation of the inadequate documentation of Pickering A. That inadequate documentation was partially because of the environment when it was built with plants being built while they were still being designed; partially because Ontario Hydro did not complete the documentation after the plant was built and put into operation; partially because the Atomic Energy Control Board never required it; and, largely because those who knew the systems took early buy-outs in the massive staff reduction at Ontario Hydro in the early 1990s under chairman Maurice Strong.

Nor is there any comment on the role of the American "advisers" who took over Ontario Hydro's nuclear program in 1997, and one of whose members was still in charge when the Pickering A rehabilitation program was begun.

The panel makes several recommendations, mostly on needed improvements in corporate governance. However, given the record of past inept interference by the provincial governments of the day, the recommendation that the [Ontario] government become more directly involved appears dangerous. The "shareholder", the provincial government, does have the responsibility and mandate to appoint the members of the OPG Board, and must do that responsibly. Nevertheless, it should stay out of the day-to-day operation of the utility.

The Panel also makes recommendations on project management and "company culture". On the latter, the Panel recommends "OPG place a priority on programs for staff development so that people of ability are available internally as strong candidates for senior positions".

Where was that perspective over the past decade ?

Fred Boyd

IN THIS ISSUE

First, the publication date for this issue was delayed by several unforeseen factors. Our apologies. The delay did, however, enable us to include a report on the CANDU Maintenance Conference in late November, a note on the Interim Report on the August 14th blackout, and our editorial comment, above, on the Pickering "A" report.

This issue is largely devoted to the **8th International Conference on CANDU Fuel,** which was held in September. Our report on the conference is followed by four papers that were presented. Two provide overviews of the programs in other countries for the production and use of fuel in CANDU or CANDU-type reactors. The first is: **An Overview of PHWR Fuel in India.** That is followed by **CANDU Type Fuel Activities in Argentina.**

The Canadian work is presented in **Advanced Fuel Development in AECL.** A specific proposal for the use of a new fuel design is given in **Bruce Power New Fuel Project.**

Then we switch to the other major conference, the **6th CNS International conference on CANDU Maintenance.** One paper from that conference is reprinted **Universal Delivery Machine - Design of the Bruce and Darlington Heads.**

As a change of pace, a paper from last summer's Annual Conference is presented, **Incorporating Human Factors into Design Change Processes - A regulator's perspective.**

A short report is offered on a meeting held in late November, the *Royal Society of Canada Symposium on Energy, Environment and Society,* under the theme title of that symposium **Making Choices.**

There is our typical, eclectic, selection of items under **General News,** and sadly, **Obituaries** of three more pioneers of our nuclear program who have passed away this fall.

Some information on current activities of the Society is presented in **CNS News,** there is an updated **Calendar** and, of course, the unique perspective of Jeremy Whitlock in **End Point.**

Your reaction is always welcomed.

8th CANDU Fuel Conference

Resort setting proves successful for delegates from eight countries despite weather

About 100 delegates from eight countries gathered at the Delawana Inn in Honey Harbour on Georgian Bay, Ontario, September 21 to 24, 2003, for the **8th International Conference on CANDU Fuel** held by the Canadian Nuclear Society. Whether by design or accident, it rained off and on most of the three days, but that did not diminish the enthusiasm of the delegates or prevent them from enjoying a boat trip down Georgian Bay. In fact it probably made it easier to stay indoors to attend the various sessions easier by removing the temptation to enjoy the attractive setting and the amenities of the venue.

The organizing team for the 8th International CANDU Fuel Conference poses for the camera. L - R: Phyllis Gutzman, Brock Sanderson (chair), Denise Rouben, Lawrence Dickson, David Caswell.

A dinner followed by a reception on the first evening (Sunday) gave delegates a chance to meet in a casual atmosphere. The conference proper began the next morning, after brief opening remarks by conference chairman **Brock Sanderson,** with a Plenary session focussed on *International Experience and Programs,* with papers from five countries.

Dr. C. Ganguly, head of the Nuclear Fuel Complex of the Department of Atomic Energy of India, was the first speaker with a paper titled *An Overview of PHWR Fuel in India* in which he gave a concise picture of the Indian pressurized heavy water reactor (PHWR) program and of the Nuclear Fuel Complex which has manufactured 280,000 bundles of fuel over its three decades of existence. *(Dr. Ganguly's paper is reprinted in this issue of the* CNS Bulletin.*)*

The report from Korea that was presented by **Dr. Ho Chun Suk** from the Korea Atomic Energy Research Institute (KAERI) had the long title of *Status of the Demonstration Irradiation of the CANDU New Fuel Bundle CANFLEX-NU in Korea.* Recognizing the successful test of 24 CANFLEX bundles in the Point Lepreau reactor KAERI and the Korea Electric Power Research Institute are conducting a three-year program to validate CANFLEX-NU fuel under Korean licensing requirements. Use of CANFLEX fuel in the Wolsong-1 reactor would help recover some of the primary system operating margins that have decreased due to ageing.

After the break **Dr. Peter Boczar,** of Chalk River Laboratories, gave a presentation entitled *Advanced Fuel Development in AECL.* He provided a broad view of the development of fuel for the Advanced CANDU Reactor (ACR) and the continuing work on MOX and DUPIC fuel. ACR fuel is based on the CANFLEX design but will use low enriched uranium to achieve a smaller reactor core, longer burn-up and a low void reactivity coefficient. *(The paper on which his talk was based is reprinted in this issue of the* CNS Bulletin.)

Dr. L. Alvarez, from the Comisión Nacional de Energia Atómica, Argentina, spoke on *Fuel Cycle Activities in Argentina.* Most of the fuel used in the Embalse reactor has been made in Argentina and the performance has been excellent, Alvarez reported. He went on to describe some of the fuel design and manufacturing activities in his country. *(Dr. Alvarez's paper is reprinted in this issue of the* CNS Bulletin.)

The final plenary paper, *Fuel Experience in China,* was presented by **Dr. M. Chen,** from the Third Qinshan Nuclear Power Company. Following the initial charge the fuel for the Qinshan CANDU units will be made in China using technology from Zircatec Precision Industries.

The afternoon and the next two days were devoted to three parallel sessions each under the following headings: *Fuel Performance; Fuel Safety; Fuel Bundle Thermalhydraulics; Manufacturing and Quality Assurance; defected Fuel Modelling; Fuel Model development; Fuel and Fuel Channel Behaviour; Fuel Management; Advanced Fuel Cycles,* with some of the topics filling two sessions.

At the close of the afternoon sessions on the second day delegates headed for the wharf to board a boat for a cruise of part of Georgian Bay. Again the weather did not cooperate, offering wind and rain interspersed with a few calm periods. This did not dampen the spirits of the group, many of whom braved the weather like seasoned sailors.

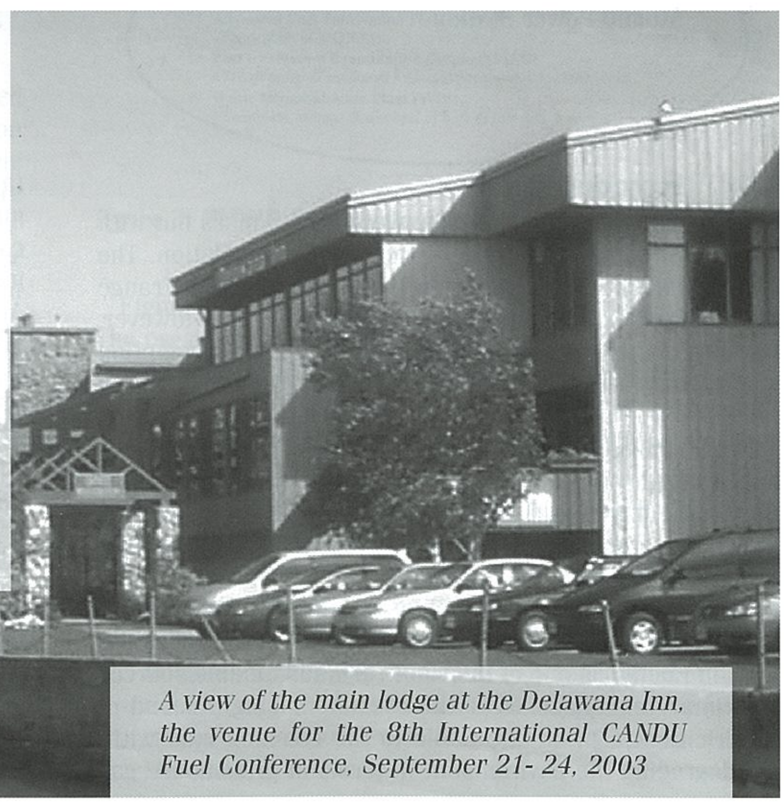
All meals were held in the attractive dining room of the lodge which has windows on three sides presenting an attractive view of the Bay. With the extensive buffets it is likely that some delegates went home somewhat heavier than they when they arrived.

The conference was organized by a small committee headed by Brock Sanderson of AECL Chalk River Laboratories and including Phyllis Gutzman, Denise Rouben, Lawrence Dickson and David Caswell.

A number of companies and organizations associated with the Canadian nuclear program provided valuable sponsorship: AECL, Bruce Power, COG, GE Canada, Hydro Québec, NB Power, NSS, OPG, Zircatec. The International Atomic Energy Agency assisted some of the delegates from other countries.

The proceedings will be issued on a CD, which can be obtained from the CNS office.

The delegates at the 8th International CANDU Fuel Conference at Honey Harbour, Ontario, managed to squeeze together on September 23, 2003 for this official photograph.



A few delegates brave the elements during the boat cruise at the 8th International CANDU Fuel Conference

A view of the main lodge at the Delawana Inn, the venue for the 8th International CANDU Fuel Conference, September 21- 24, 2003

An Overview of PHWR Fuel In India

by C. Ganguly[1]

***Ed. Note:** The following paper was the first presentation in the opening plenary session of the 8th International Conference on CANDU Fuel held in Honey Harbour, Ontario, September 21 - 24, 2003.*

Abstract

Pressurised Heavy Water Reactor (PHWR) is the first stage and the backbone of the nuclear power programme in India. Presently, twelve PHWR 220 MWe type units are in operation and two units of PHWR 540 MWe & four units of PHWR 220 are in different stages of construction. Design and development activities are underway for further augmenting the capacity of PHWR 540 to 680 MWe. Eight such reactors have been planned by Nuclear Power Corporation of India Limited (NPCIL).

The exploration and mining activities of uranium has been significantly enhanced by Atomic Minerals Directorate for Exploration and Research (AMD) and Uranium Corporation of India Limited (UCIL) respectively and a new underground mine has been opened in Turamdih in Jharkhand State. Nuclear Fuel Complex (NFC) has, so far, manufactured some 280,000 zirconium alloy clad natural uranium oxide fuel bundles and some 4,600 depleted and some 300 thorium oxide assemblies. These fuel bundles and assemblies are of the 19-element type. NFC has recently initiated the fabrication activity of 37-element fuel bundles for the 2 forthcoming PHWR 540 units of the Tarapur Atomic Power Project (TAPP 3&4). In recent years, NFC has introduced several modifications in the manufacturing and quality control processes, which have significantly improved the productivity, recovery and quality of PHWR fuel and in turn its performance in reactors. Large-scale introduction of depleted uranium oxide bundles has been planned not only for the initial core for neutron flux flattening but also for the subsequent equilibrium cores. As part of Plutonium recycling programme, 50 numbers of MOX-7 test fuel bundles have been manufactured by Bhabha Atomic Research Centre (BARC) in collaboration with NFC for irradiation-testing in one of the PHWR 220 units of Kakarapar Atomic Power Station.

1.0 Introduction

India occupies only 2% of the world's land mass but with 1.1 billion people has 16% of the world's population. The total installed electric power in the country is in the range of 130,000 MWe (including captive power plants). However, because of the large population, the per capita consumption of electricity is in the range of 600 kWh per year only, which is nearly 4 times lower than the world average and some 15 times lower when compared to the OECD countries. For improving the per capita consumption of electricity, apart from initiating several measures to control the population growth, a programme of installing approximately 10,000 MWe electric power every year has been launched in order to bring the total installed electric power in the country in the range of 300,000 MWe by the year 2020. 'Nuclear Fission Energy' is one of the viable and sustainable sources of primary energy to meet the ever-increasing demand of electricity at a price affordable to the common man without degrading the environment in terms of greenhouse gas (and in turn global warming) and acid rains. Presently, 14 nuclear power reactors are in operation with total installed capacity of 2720 MWe, which is only some 2.5% of the total installed electric power in the country. The Department of Atomic Energy (DAE) has set a target of installing 20,000 MWe nuclear power by the year 2020 of which water-cooled thermal reactors would account for more than 18,000 MWe as shown in Table 1. The electricity generated from nuclear power reactors in India has progressively increased from 11,174 million units (kWh) in 1998-99 to 19,242 million units in 2002-03 as shown in Figure 1.

Pressurised Heavy Water Reactor (PHWR), popularly known as CANDU all over the world, and its fuel cycle is the backbone and the first stage of the indigenous nuclear power programme in India. The three-stage programme, linking the fuel cycles of PHWR and Liquid Metal-cooled

1 Dr. Ganguly is Director of the Nuclear Fuel Complex, Department of Atomic Energy, Hyderabad, India

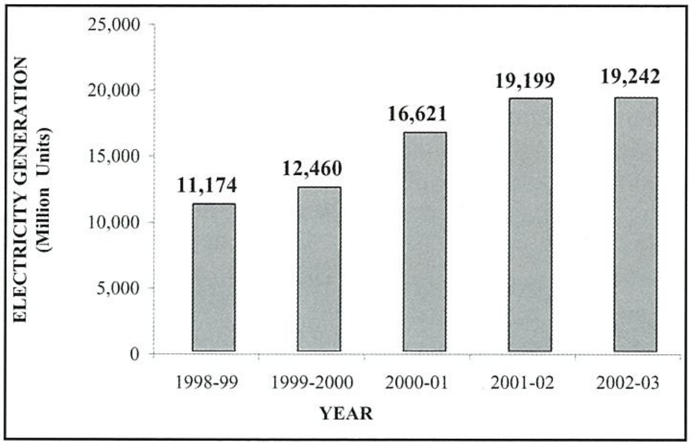


Figure 1: Electricity Generated By Nuclear Power Corporation of India Ltd. (NPCIL) During The Last Five Years.

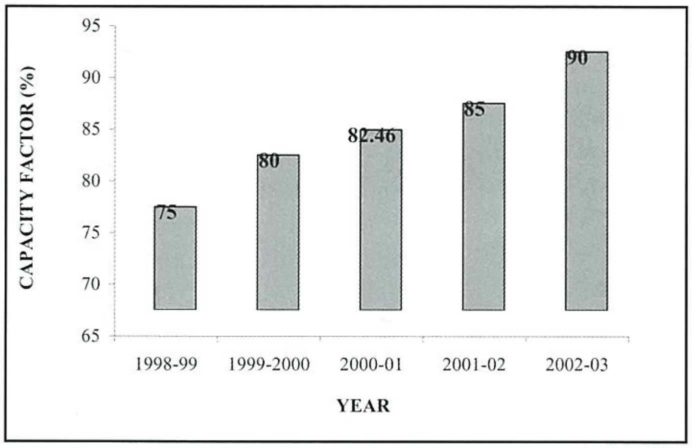


Figure 2: Capacity Factor Achieved By Nuclear Power Plants Operated By NPCIL During The Last Five Years.

Fast Breeder Reactor (LMFBR) and thorium-based advanced thermal reactors, is aimed at judicious utilization of modest uranium but vast thorium resources. For a viable long-term nuclear power programme in India, LMFBRs are essential. The PHWRs in the first stage would not only generate electricity but also produce sufficient plutonium for launching a large LMFBR programme, which would breed U^{233} from thorium blankets.

The first CANDU reactor in India, the Unit 1 of the Rajasthan Atomic Power Station (RAPS 1), was constructed and commissioned in collaboration with Canada and went into commercial operation on December 16, 1973. RAPS 1 was the forerunner of the 220 MWe type PHWRs. Thereafter, India has been pursuing a self-reliant PHWR programme. Presently, 12 PHWR 220 type units are in operation and 4 are under construction. The second unit of Rajasthan Atomic Power Station (RAPS 2) and Madras Atomic Power Station (MAPS 2) have undergone en-masse coolant channel replacement and are operating satisfactorily. In general, there has been progressive improvement in the 'capacity factor' of all the PHWR units in India as shown in Figure 2. Last year, one of the PHWR 220 units of Kakrapar Atomic Power Station operated with a plant load factor of 98% and reached the highest position among all the operating PHWRs in the world. Two PHWR 540 units are under construction as part of Tarapur Atomic Power Project (TAPP 3&4). These two reactors would be the first in the series of PHWR 500 type units in India. Design and development activities are underway for further augmenting the capacity of the PHWR 500 MWe units to 680 MWe by allowing partial boiling in the core. Eight such PHWR 680 MWe are in the planning stage.

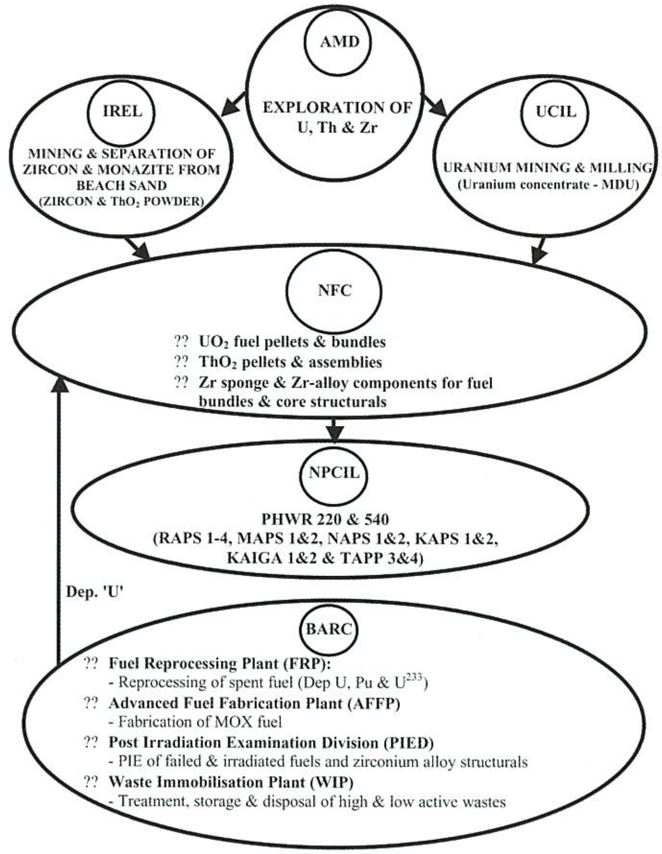


figure 3: PHWR Fuel Cycle Activities In India

Figure 3 describes the PHWR fuel cycle activities in India. The organizations involved are:

(i) Atomic Minerals Directorate for Exploration and Research (AMD) with Headquarters at Hyderabad – responsible for locating uranium, zirconium and thorium deposits.

(ii) Uranium Corporation of India Limited (UCIL) with Headquarters at Jaduguda, Jharkhand State – responsible for mining and concentration of uranium ore in the form of yellow cake [magnesium di-uranate (MDU)].

(iii) Indian Rare Earths Limited (IREL) with Headquarters at Mumbai (Bombay) – responsible for mining of heavy minerals, including zircon and monazite and for preparation of reactor grade thorium oxide powder from Monazite.

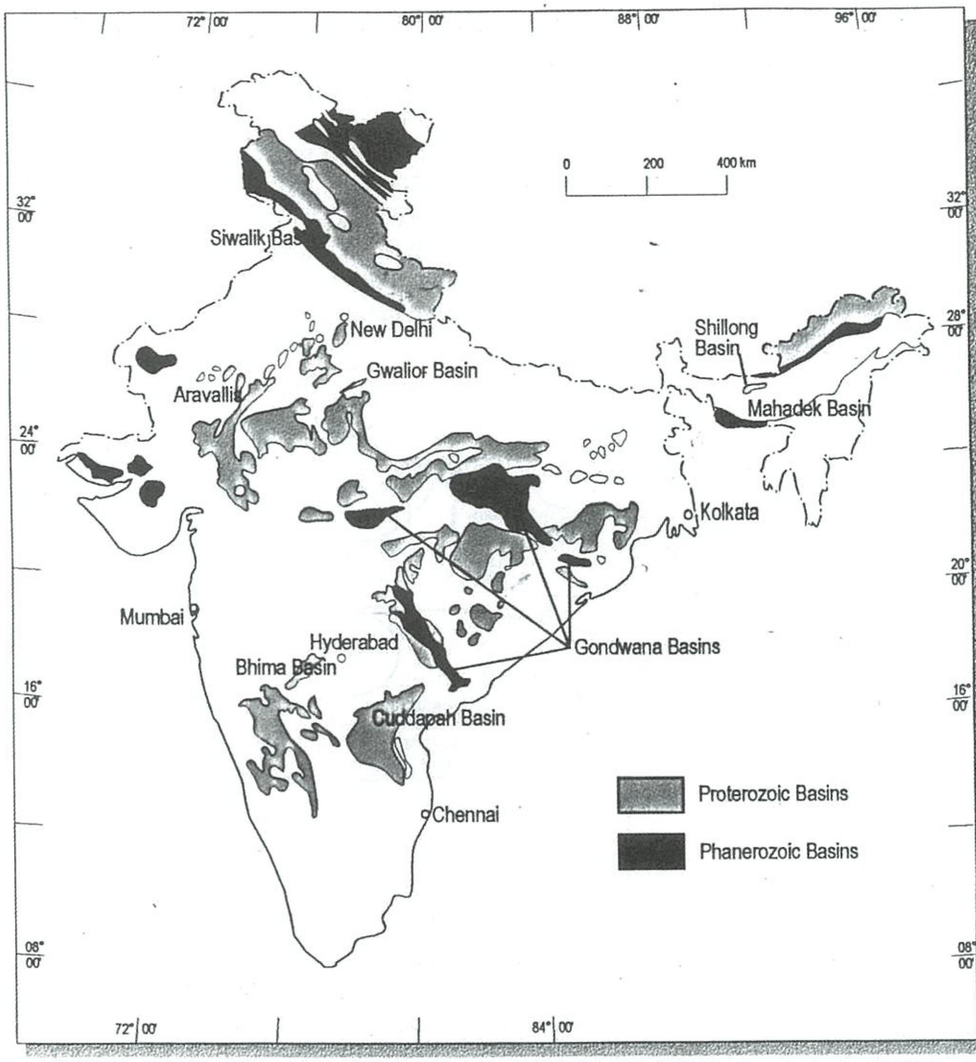


Figure 4: The Important Proterozoic & Phanerozoic Basins of India As Target Areas For Uranium Exploration.

(iv) Nuclear Fuel Complex (NFC) with Headquarters at Hyderabad – responsible for manufacturing uranium oxide powder, pellets & fuel bundles, thorium oxide pellets & assemblies, reactor grade zirconium sponge, zirconium alloy ingots and zirconium alloy tubes & components for fuel bundles and core structures.

(v) Nuclear Power Corporation of India Limited (NPCIL) with Headquarters at Mumbai – responsible for design, construction, operation and maintenance of nuclear power reactors in India.

(vi) Bhabha Atomic Research Centre (BARC) with Headquarters at Mumbai – responsible for reprocessing and waste management of PHWR spent fuel, fabrication of mixed uranium plutonium oxide fuel, in-service inspection of reactors and post-irradiation examination (PIE) of failed fuel.

NFC, an industrial unit of DAE, was set up at Hyderabad in the early 1970s, for manufacturing zirconium alloy clad natural uranium oxide fuel for PHWRs, using 'uranium concentrate' and 'zircon' as starting materials from UCIL and IREL respectively. During the last 30 years, NFC has manufactured more than 280,000 fuel bundles of the 19-element type for the 12 operating PHWRs. In addition, some 4,600 depleted uranium oxide bundles and nearly 300 thorium oxide bundles have been manufactured and utilized for neutron flux flattening of the initial cores during start-up. Depleted uranium was used in the first six PHWR 220 units at Rajasthan (RAPS 1&2), Madras (MAPS 1&2) and Narora (NAPS 1&2). Whereas some 350 to 550 depleted uranium oxide bundles were loaded in each core of these six PHWR 220 units, from the Unit 1 of Kakrapar Atomic Power Station (KAPS) onwards, some 35 ThO2 bundles were adequate for each reactor to carry out effectively the neutron flux flattening in the initial core without violating specified channel power and bundle power limits and at the same time operating the reactor at rated full power. So far, 232 thorium oxide bundles were successfully irradiated in the initial cores of KAPS 1&2, Kaiga 1&2, RAPS 3&4 and RAPS 2, after retubing. The maximum power and burn-ups of these thoria bundles were 408 kW and 13,000 MWd/t respectively. Recently, a decision has been taken to use depleted uranium oxide bundles again in all future PHWR units for judicious utilization of the large quantities of depleted uranium that has stock-piled in the spent fuel reprocessing plants. Accordingly, in the first quarter of 2003, depleted uranium oxide bundles have been manufactured for MAPS 2, after retubing for neutron flux flattening. The manufacturing activity of 37-element natural uranium oxide fuel bundles and depleted uranium oxide assemblies has been initiated for the forthcoming PHWR 540 units at Tarapur, namely TAPP 3&4. Tarapur 4 is likely to attain criticality in December 2004, for which the fuel should be in site by July 2004. The criticality date of Tarapur 3 is December 2005. NFC has already supplied the zirconium alloy calandria tubes, coolant tubes, garter springs, reactivity and shut-off mechanisms for TAPP 4. The manufacturing activities of zirconium alloy core structurals for TAPP 3 are underway.

The present paper summarises the developments in

PHWR fuel and fuel cycle in India during the last 2 years. The exploration and mining of uranium, the core loading concepts for large scale utilisation of depleted uranium oxide bundles in PHWR 220 & PHWR 540 units, the manufacturing experience of natural uranium & depleted uranium oxide and MOX fuel pellets and fuel bundles and in-core performance of PHWR fuel have been highlighted.

2.0 Exploration And Mining Of Uranium

India has modest uranium reserves of very low grade (0.04-0.06%), mostly in deep underground deposits. So far, 3 underground mines, namely Jaduguda, Bhatin and Narwapahar were in operation in the Singhbhum District of Jharkhand State. Since the beginning of 2003 the fourth underground mine has been opened in Turamdih in the vicinity of Narwapahar. The price of indigenous natural uranium concentrate is several times higher than the ones available in the international market because of the high cost involved in deep underground mining and treating very low grade ores. In recent years, the exploration and mining activities have been intensified in order to match the uranium demand of the rapidly expanding nuclear power programme. Figure 4 shows the important Proterozoic and Phanerozoic Basins of India, which are the target areas for uranium exploration. The summary of uranium exploration and mining activities are as follows:

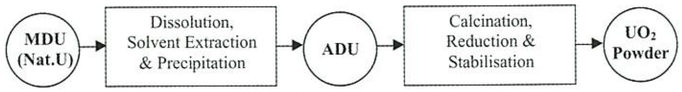
- Singhbhum Deposits (Jharkhand State): These are relatively low grade 'vein' type deposits in schistose host rock with uranium content in the range of 0.04 – 0.06% U_3O_8. Some 42,000 tons of uranium has been proven in this area. Presently, 4 deep underground mines, namely Jaduguda, Bhatin, Narwapahar and Turamdih are in operation and feeding the uranium Mill at Jaduguda. Soon commercial exploitation would be started in Bagjata underground mine and the Bandhuhurang open cast mine in eastern Singhbhum District. Construction activity of a second uranium mill in the vicinity of the Turamdih mines would also be initiated soon.
- Cuddapah Basin (Andhra Pradesh State): The experts are of the opinion that this area is likely to emerge as a major uranium province, based on the indicative 'unconformity'. So far, some 5,000 tons of uranium of average grade 0.09% has been proven in Lambapur and Peddagattu in Nalgonda District. Advanced geophysical survey based on magnetic and electromagnetic equipment has been planned. For data reliability, procurement action of hydrostatic drilling rig with deviation control system, for nearly zero deviation up to greater depths, is underway. This would pave the way to intercept deep and concealed high-grade uranium ore bodies in this area. A Uranium Mill has been planned adjacent to the mines.
- Mahadek Basin & Shillong Basin (Meghalaya State): In the State of Meghalaya, there are two distinct type of ore bodies. In the 'sandstone' type deposits in the Mahadek Basin, some 15,000 tons of uranium with average grade 0.1% U_3O_8 have been proven in Domiasiat and Wahkyn. These are mostly shallow type deposits with the ore body some 30 to 50 meters below the surface. An additional, 15,000 tons uranium is expected in the Mahadek Basin. The uranium deposits at Shillong Basin are indicative of 'unconformity' type based on geological set up and the age of the 'unconformity'. Intensive geophysical survey is also planned in this area for intercepting deep and concealed high-grade uranium deposits, if any.
- Bhima Basin (Karnataka State): Though 'unconformity' type deposit is not ruled out in this area, presently 'vein' type uranium deposits have been observed in depths of 200 – 250 meters in both granite base and dolomitic limestone overburden. So far, some 2,000 tons have been proven of average grade 0.2% in Gogi. An alkaline leaching technique has been developed for extracting uranium both from granite and limestone ore bodies.
- Rohil-Ghateshwar, Sikar District (Rajasthan State): These are 'vein' type, deep underground deposits in depth of 200-250 meters, similar to the ones in the Singhbhum District, with average uranium in the range of 0.07% U_3O_8. Some 2,000 tons have been proven in this area so far.

3.0 Core Loading Concepts For Utilisation Of Depleted Uranium

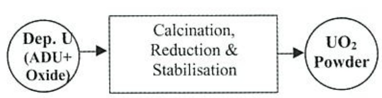
In recent years, a programme has been initiated at NPCIL for utilization of depleted uranium in the initial and equilibrium cores of PHWR 220 and PHWR 540 units. Accordingly, several core loading concepts have been proposed. Figures 5 and 6 show the loading pattern of depleted uranium oxide bundles in the initial cores of PHWR 220 and PHWR 540 units respectively. Some 1516 depleted UO_2 bundles were loaded in the initial core of MAPS 2, in the second quarter of 2003 after retubing. In the same way, some 2208 depleted uranium oxide bundles are proposed to be manufactured for the forthcoming PHWR 540 unit at Tarapur (TAPP 4 to start with). In equilibrium cores of operating PHWR 220 units, some 78 depleted uranium oxide bundles could be utilized in the inner channels as shown in figure 7.

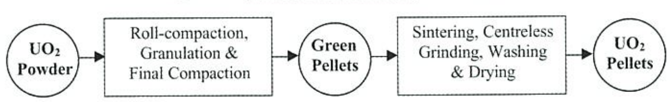
4.0 Fabrication Of Natural & Depleted Uranium Oxide Fuel

The process flowsheet and quality control plan, followed at NFC, for manufacturing natural uranium oxide fuel pellets and fuel bundles for the PHWR 220 units in India have been described in the proceedings of earlier CANDU fuel conferences (1,2). The magnesium di-uranate (MDU) supplied by UCIL and uranium oxide scrap are processed through a series of chemical operations to obtain pure ammonium di-uranate (ADU), which is then subjected to air-calcination followed by hydrogen reduction and stablisation to obtain sinterable grade UO_2 powder. Nearly, 100% of the UO2 powder lot qualified the sinterability test. Modifications in process equipment, steps and parameters led to significant reduction in the consumption of major chemicals like nitric acid, caustic lye, tributyl phosphate, etc. and reduced the uranyl nitrate raffinate cake (UNRC) formation per ton of the UO2 powder. A pilot plant has been set up for recovery of uranium from uranyl nitrate raffinate and UNRC.

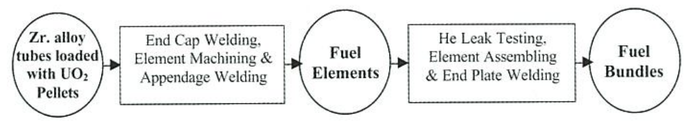


In the spent PHWR fuel reprocessing plant of BARC, plutonium and depleted uranium (Dep. U) are separated from the fission products by the PUREX process. Thereafter, from the pure plutonium nitrate solution, PuO_2 powder is produced by air-calcination of plutonium oxalate. Likewise, the pure depleted uranium nitrate solution is converted to ammonium di-uranate (ADU), which is then calcined at around 500°C and sent to NFC. The as-received depleted uranium from BARC, met all the chemical specifications and was in the form of yellowish powder consisting of a mixture of ammonium di-uranate (ADU) and the higher oxides of uranium as shown in the X-ray diffraction pattern (Figure 8a). Since the powder was chemically pure, additional process steps like nitric acid dissolution and solvent extraction were not required. Instead, the powder was directly subjected to air-calcination at around 650°C in rotary calciner, followed by hydrogen reduction and stabilization. X-ray diffraction pattern of depleted uranium oxide powder thus produced, as shown in figure 8b, show single phase UO_2, which more or less matched with the natural uranium oxide powder (figure 8c).





The powder lots of natural and depleted uranium oxide were processed in the same way to obtain free-flowing press-feed uranium oxide granules. Roll-compaction followed by granulation was adapted for this purpose. Next, zinc stearate lubricant was admixed with the granules. A new organic lubricant has been developed in place of zinc stearate, thereby avoiding deposition of metallic zinc in the interior parts of the sintering furnaces. The granules were subjected to cold-pelletisation in hydraulic press using multi-cavity die punch set. Twelve pellets were compacted at a time. For powder compaction, use of tungsten carbide dies and cryogenic-treated die steel punches improved the tool life by a factor of 40. The green pellets were loaded in molybdenum charge carriers and subjected to high temperature sintering in cracked ammonia in pusher type molybdenum resistance furnace. The microstructure of as-sintered natural uranium oxide and depleted uranium oxide were more or less identical as shown in figure 9. It consisted of single phase equiaxed grains of uranium oxide and uniformly distributed porosity. The high density sintered pellets thus obtained were centreless ground to the specified diameter, washed, dried and loaded in zirconium alloy cladding tubes. Resistance welding is being followed for encapsulation of UO_2 pellet stack, welding bearing and spacer pad appendages on the fuel elements and projection welding of fuel elements with the end plates on both sides of the fuel bundles.



The annual production of natural uranium oxide fuel bundles has increased progressively during the last 5 years, as shown in figure 10. In the year 2002-03, the annual fuel bundle production crossed the 30,000 mark for the first time since the inception of NFC in early 1970s.

Quality control and quality assurance (QC/QA) activities have been geared up to meet the enhanced production target. The synergy among production & QC/QA groups and the customer (NPCIL) has paved the way for uninterrupted and timely delivery of PHWR fuel bundles of high quality to the different operating reactor sites. This contributed to a great extent to improving the PLF of operating PHWRs during the last 5 years and reach international levels of ≥85%.

5. Utilisation Of Mox Bundles For Irradiation In Kaps

As a first step to recycling of plutonium in PHWR, recently, NPCIL has proposed a core loading pattern of MOX fuel bundle in PHWR 220, as shown in figure 11. Accordingly, some 50 numbers of zirconium alloy clad 19-element MOX 7 fuel bundles have been manufactured for irradiation-testing in one of the units of KAPS. The MOX 7 bundles consist of 7 inner fuel elements containing mixed uranium plutonium oxide pellets with 0.4% plutonium and 12 outer fuel elements of natural uranium oxide.

The MOX fuel pellets were manufactured by BARC in collaboration with NFC. The ex-ADU derived UO_2 powder and ex-oxalate PuO_2 powder were subjected to co-milling in an attritor, followed by granulation, admixing lubricant, pelletisation and high temperature sintering in hydrogen atmosphere. Next, the sintered pellets were ground to the desired diameter, washed, dried, inspected, loaded in zirconium alloy cladding tubes and encapsulated by TIG welding of the zirconium alloy end-plugs.

6.0 PHWR Fuel Performance

So far, more than 250,000 fuel bundles have been irradiated in the 12 operating PHWR 220 units to discharge burnup in the range of 6,500 – 7,250 MWd/TeU. The in-core performance of PHWR fuel has progressively improved over the years and presently the iodine activities in coolant circuit in most of the reactors are being maintained at levels less than 5 µCi/l. For the first time, the failure rate has been below 0.1% (actual number: 0.096%). Most of the fuel failures have been mainly due to fuel handling events particularly the ones, which failed at relatively low burn-up. Accordingly the frequency of checking the alignment of tubes in fresh fuel transfer system has been increased. In fabrication side, 100% ultrasonic testing of end plug welds and helium leak testing of finished fuel bundles have brought down significantly the fuel failure rate due to manufacturing defects. The improved fuel management practices now followed at all reactors have also contributed to minimization of fuel failure.

Table 1: Water-Cooled Nuclear Power Reactors in Operation, Under Construction and At Planning Stage In India (Total: 18,420 MWe by the year 2020)

Plant (Site)	Reactor Type	Capacity
IN OPERATION : 2,720 MWe [2 BWRs + 12 PHWRs (220 MWe type)]		
TAPS 1&2 (Tarapur)	BWR-220	2x160 MWe (rerated)
RAPS-1 (Rawatbhatta)	PHWR-220	1x100 Mwe (rerated)
RAPS-2 (Rawatbhatta)	PHWR-220	1x200 MWe (rerated)
MAPS-1&2 (Kalpakkam)	PHWR-220	2x170 MWe (rerated)
NAPS-1&2 (Narora)	PHWR-220	2x220 MWe
KAPS-1&2 (Kakrapar)	PHWR-220	2x220 MWe
Kaiga-2&1 (Kaiga)	PHWR-220	2x220 MWe
RAPS-3&4 (Rawatbhatta)	PHWR-220	2x220 MWe
UNDER CONSTRUCTION: 3,960 MWe [4xPHWR 220 + 2xPHWR 540 +2xVVER 1000]		
TAPP-3&4 (Tarapur)	PHWR-500	2x500 MWe
Kaiga 3&4	PHWR-220	2x220 MWe
RAPP 5&6	PHWR-220	2x220 MWe
KK-1&2 (Kudankulam)	LWR-1000 (VVER type)	2x1000 MWe
PLANNING STAGE: 11,740 MWe [8xPHWR 680 + 1 AHWR 300 + LWR 1000]		
AHWR	AHWR 300	1x300 MWe
PHWR	PHWR 500	8x680 MWe
LWR (including VVER)	LWR 1000	6x1000 MWe

7.0 Concluding Remarks

PHWR and its fuel cycle technology has reached a stage of maturity in India. During the last 3 decades, all operations in front- and back-end of the fuel cycle have been streamlined and are now being carried out on an industrial scale. 12 units of PHWR 220 are in operation and 6 PHWRs are under construction, of which 2 are of the PHWR 540 type. The Department of Atomic Energy has planned 8 more PHWR 680 MWe units in coming years. One of the major challenges being faced is the high cost of natural uranium from indigenous sources because of low-grade (0.04 – 0.06%) uranium ores and the high cost of deep underground mining. In recent years, uranium exploration activities have been significantly augmented in order to intercept concealed, deep underground, high-grade uranium deposits based on indicative 'unconformity'. A few more mines are likely to be opened in coming years. However, till these mines become operational, several alternatives, including large scale utilization of depleted uranium in initial and equilibrium cores, recycling plutonium as MOX and utilization of thorium are being pursued for conserving natural uranium.

Nuclear energy is an inevitable option for India for meeting the ever-increasing demand of electricity. Such non-carbon based primary source of energy should be encouraged through international collaboration for generation of clean or 'green' electricity, particularly in large developing countries like India. Thus, emission of greenhouse gas and in turn global warming would be minimized. For this, the existing international treaties and guidelines of Nuclear Suppliers Group for trading of uranium should be made more flexible in order to facilitate peaceful use of nuclear energy (like generation of electricity) while ensuring non-proliferation of such material for military activities. Countries with expanding nuclear electricity programme, but having modest uranium resources like India, should have easy access to natural uranium from international market for their nuclear electricity programme. Thus, new uranium market would open-up and countries having large uranium resources with no or small nuclear power programme would be encouraged to expand their uranium mining and milling activities.

Acknowledgements

The author is grateful to all his colleagues from different units of DAE associated with PHWR fuel cycle programme. The present paper is a summary of the natural and depleted uranium oxide fuel fabrication activities of NFC during the last 2 years and in-core performance of PHWR fuel. The other papers in this proceedings from NFC and NPCIL gives details of the PHWR fuel cycle activities in India. The author would like to thank Mr. R.M.Sinha, Director, AMD, Mr. A.K.Bagchi, Additional Director, AMD, Mr. S.A.Bhardwaj, Executive Director, NPCIL, Mr. H.S.Kamath, Director, Nuclear Fuels Group, BARC and Mr. R.N.Jayaraj and Mr. Komal Kapoor his colleagues from NFC for preparation of this paper. The author is grateful to the International Atomic Energy Agency (IAEA) Vienna for sponsoring his participation in the 8th International CANDU Fuel Conference and presenting the paper.

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	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
A								7	7	7	7	7	7							
B						7	7	6	6	6	6	6	6	7	7					
C				7	7	6	6	6	6	6	6	6	6	6	6	7	7			
D			7	6	6	6	6	6	5	5	5	5	6	6	6	6	6	7		
E			6	6	6	6	5	5	5	5	5	5	5	5	6	6	6	6		
F		6	6	6	6	5	5	5	8	8	8	8	5	5	5	6	6	6	6	
G		6	6	6	6	5	5	8	8	8	8	8	8	5	5	6	6	6	6	
H	6	6	6	5	5	5	5	8	8	8	8	8	8	5	5	5	5	6	6	6
J	6	6	6	5	5	8	8	8	8	8	8	8	8	8	8	5	5	6	6	6
K	6	6	6	5	5	8	8	8	8	8	8	8	8	8	8	5	5	6	6	6
L	6	6	6	5	5	8	8	8	8	8	8	8	8	8	8	5	5	6	6	6
M	6	6	6	5	5	8	8	8	8	8	8	8	8	8	8	5	5	6	6	6
N		6	6	5	5	5	5	8	8	8	8	8	8	5	5	5	5	6	6	
O		6	6	6	6	5	5	8	8	8	8	8	8	5	5	6	6	6	6	
P		6	6	6	6	5	5	5	8	8	8	8	5	5	5	6	6	6	6	
Q			6	6	6	6	5	5	5	5	5	5	5	5	6	6	6	6		
R			7	6	Q	6	6	5	5	5	5	5	5	6	6	6	6	7		
S					SP	6	6	6	6	6	6	6	6	6	6	SP				
T						7	7	6	6	6	SP	6	6	7	7					

5	**DU Bundles in 11 and 12**		7	**DU Bundles in 1 to 12**
6	**DU Bundles in 1 to 2 and 11 to 12**		8	**DU Bundles in 5 to 12**

Figure 5: Initial Core Loading Maps-2 (PHWR 220) With 1516 Depleted Uranium Bundles

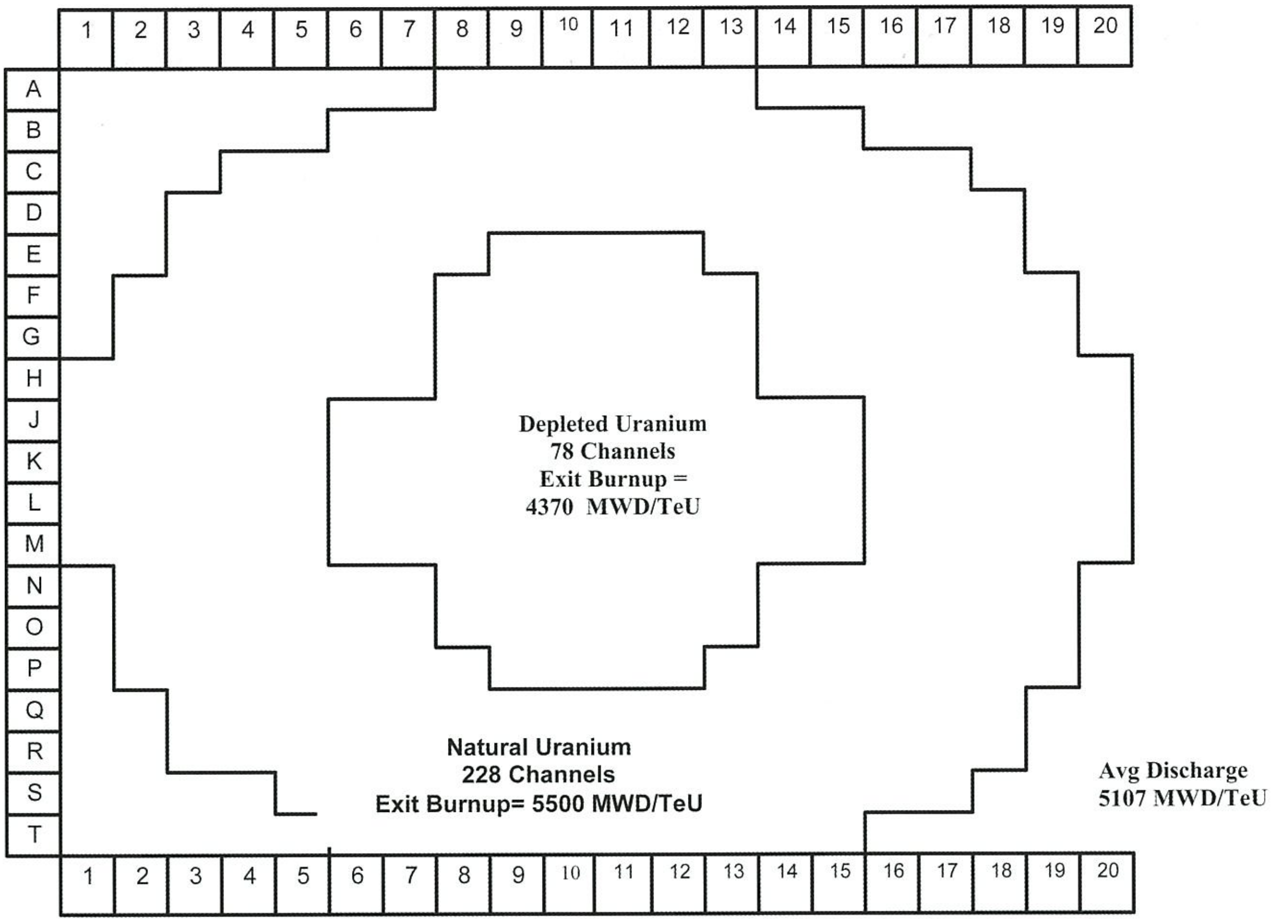
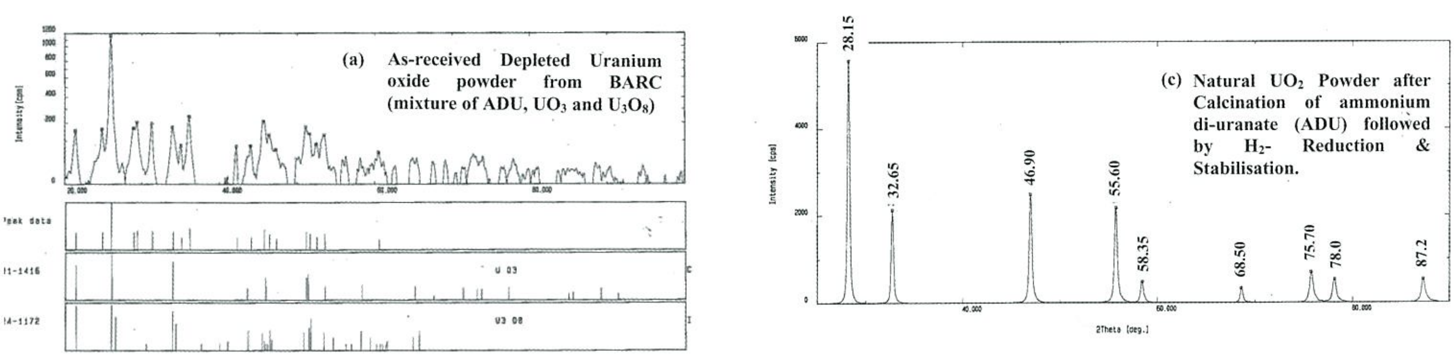


Figure 7: Equilibrium Core Burnup Optimisation For 220 MWe PHWR Depleted and Natural Uranium Fuel.



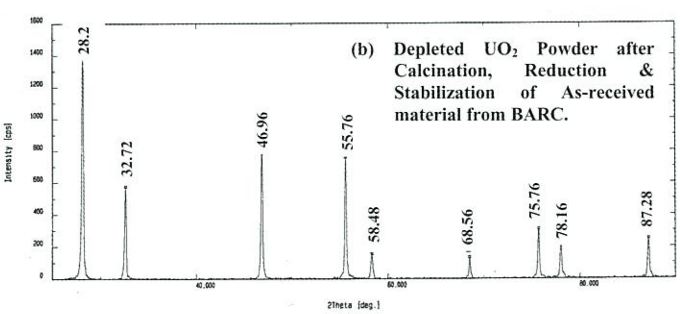
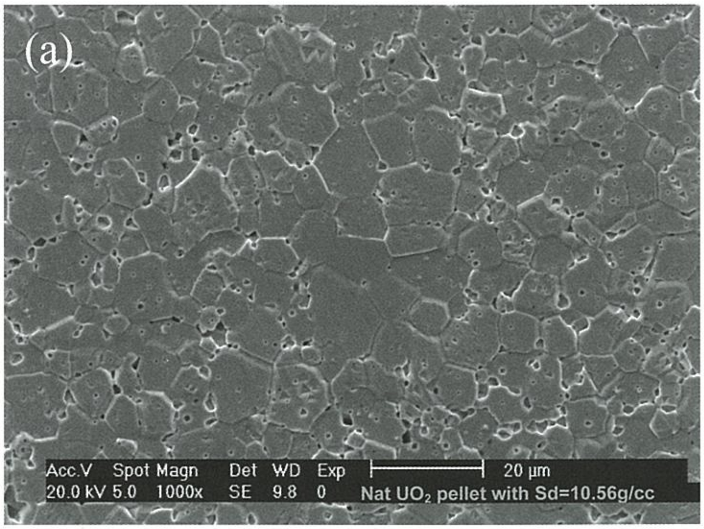


Figure 8: X-ray Diffraction Pattern of (a) As-Received Depleted Uranium Oxide Powder Showing Mixture of ADU, UO_3 & $U3O_8$ (b) After Calcinations, Reduction & Stabilization of As-Received Powder Shwoing Peaks of Pure UO_2 (c) Natural O_2



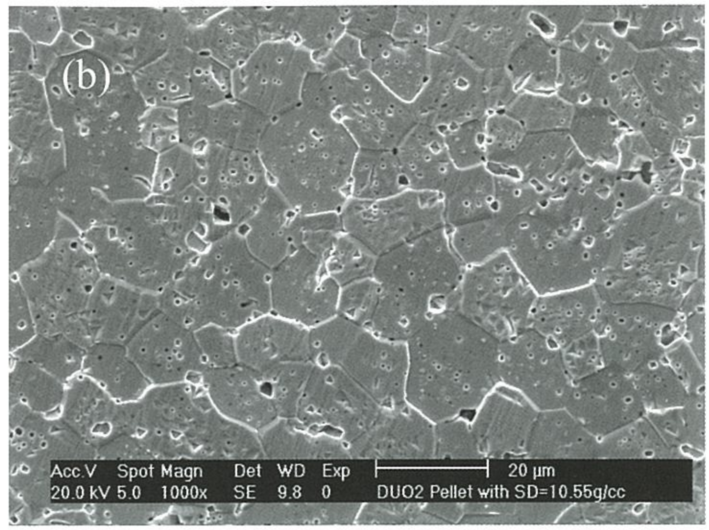


Figure 9: SEM Microstructure of As-Sintered (a) Natural Uranium Oxide and (b) Depleted Uranium Oxide Showing Single Phase, Equiaxed Grains & Residual Porosity in Both Cases.

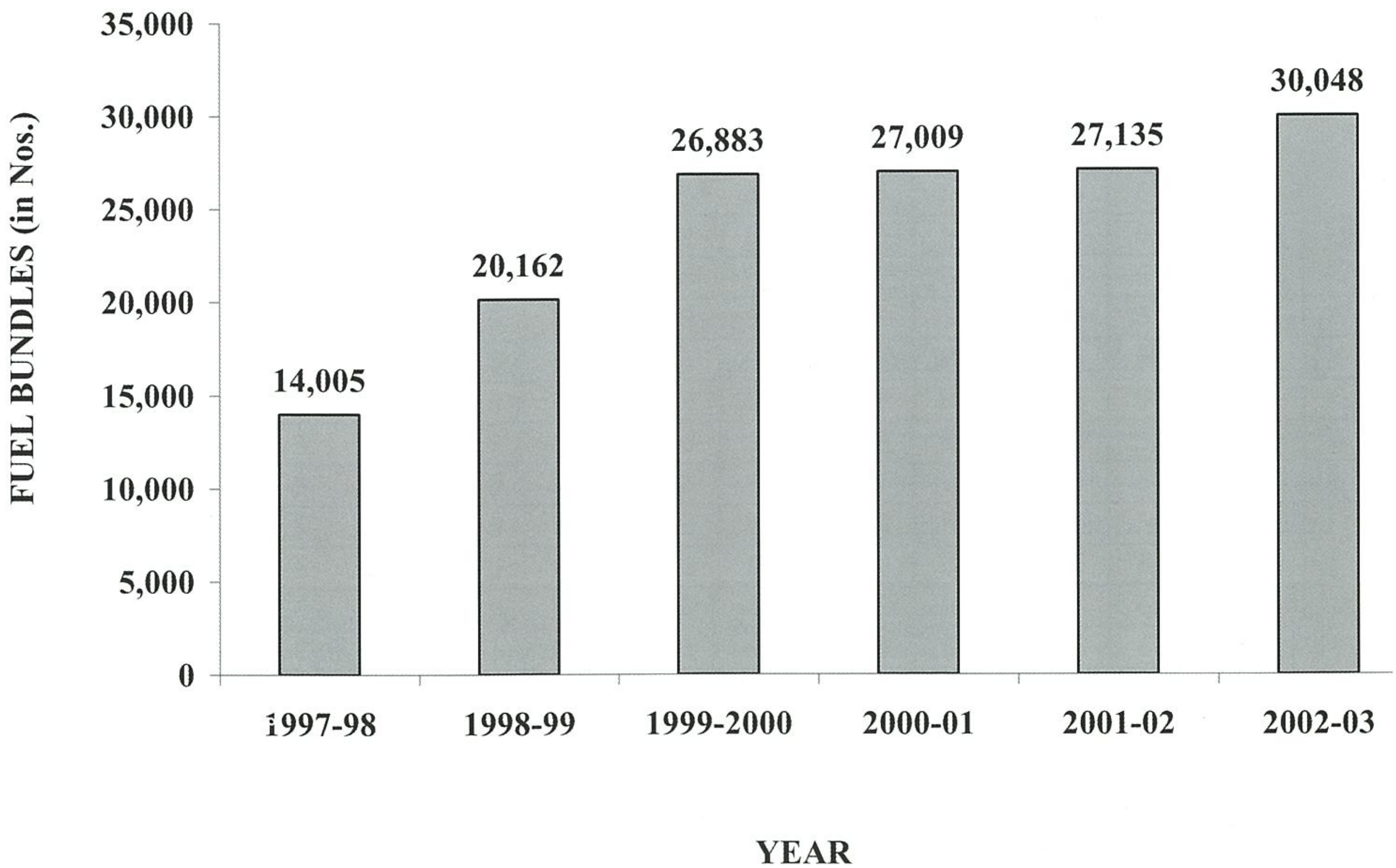


Figure 10: Production of Zirconium Alloy Clad 19-Element Natural uranium Oxide Fuel bundles at NFC for PHWR 220 Units in India.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
A								5	5	5	5	5	5							
B						5	5	5	5	5	5	5	5	5	5					
C				5	5	5	5	5	5	5	5	5	5	5	5	5	5			
D			5	5	5	4	4	4	3	3	3	3	4	4	4	5	5	5		
E			5	5	5	4	4	4	2	2	2	2	4	4	4	5	5	5		
F		5	5	5	5	4	4	2	2	2	2	2	2	4	4	5	5	5	5	
G		5	4	4	4	4	4	2	2	2	2	2	2	4	4	4	4	4	5	
H	5	5	4	3	3	3	3	2	2	2	2	2	2	3	3	3	3	4	5	5
J	5	4	4	3	3	2	2	2	2	2	2	2	2	2	2	3	3	4	4	5
K	5	4	4	3	3	2	2	2	2	2	2	2	2	2	2	3	3	3	4	5
L	5	4	4	3	3	2	2	2	2	2	2	2	2	2	2	3	3	3	4	5
M	5	4	4	3	3	2	2	2	2	2	2	2	2	2	2	3	3	4	4	5
N		5	4	3	3	3	3	2	2	2	2	2	2	3	3	3	3	4	5	
O		5	4	4	4	4	4	2	2	2	2	2	2	4	4	4	4	4	5	
P		5	5	5	5	4	4	3	2	2	2	2	3	4	4	5	5	5	5	
O			5	5	5	5	5	4	3	3	3	3	4	5	5	5	5	5		
R			5	5	5	5	5		4		3	4	4	5	5	5		5		
S					5	5	5	5	5	5	5	5	5	5	5	5				
T						5		5	5	5	5	5			5					

2	NATURAL U (10000 MWD/Teu)	3	NATURAL U (5800 MWD/Teu)
4	MOX-7 (9700 MWD/TeU) 4BSS	5	MOX-7 (10350 MWD/TeU) 8BSS

Figure 11: MOX-7 Bundles in Equilibrium Core of PHWR 220 MWe.

PBNC14

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The Pacific Basin Nuclear Conferences take place every two years. They bring together representatives of the nuclear programs in countries around the Pacific Rim, the countries with the most active nuclear programs in the world.

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CANDU Type Fuel Activities in Argentina

by L. Alvarez, J.A. Casario[1] and C. Moreno[2]

Ed. Note: The following paper was one of the plenary presentations at the 8th International Conference on CANDU Fuel held at Honey Harbour, Ontario. September 21-24, 2003.

Abstract

Domestic fuel performance in Embalse NPP during the last two years has been excellent without a significant occurrence of fuel failures. The defect rate level was reasonably low with a lowest value of 0.02 % in 2002.

The implementation of fuel design optimizations to increase uranium content was fully completed by the end of year 2000. The in-reactor performance was not affected and shows the high degree of maturity reached for both the design and the manufacturing procedures and capabilities.

A feasibility study for the utilization of SEU in Embalse NPP mainly conducted by NA-SA and AECL is almost completed. Some fuel related activities are still in progress. As part of them fuel behavior simulations using simplified power histories were performed to assess the influence of SEU fuel burnup extension.

1 General Overview

This report presents the main activities performed in Argentina related with the type of nuclear fuel used in the Embalse Power Station. Embalse has a CANDU-6 type reactor that is operating since 1983. The organizations involved in nuclear fuel activities in Argentina are CNEA (fuel design and engineering issues), CONUAR (domestic fuel manufacturer) and NA-SA (operator of the nuclear power stations).

This presentation covers the following areas: fuel performance, fuel design and engineering activities, fuel fabrication and advanced programs.

2 Fuel Performance

During the last two years, 2001 and 2002, more than 10000 bundles have been irradiated in Embalse NPP. Almost all of them were fabricated by CONUAR in Argentina. Table 1 provides the relevant information for this period. During both years the average discharge burnup was around 7400 MWd/tU and the failure rate dropped to a 0.02 % value during 2002. Only three FA in 2001 and 1 FA in 2002 were declared failed by the power station. The overall failure rate for the last 5 years is 0.04 %. This is a remarkable improvement respect past failure rates reported in previous International Conferences on CANDU Fuel (1).

Table 2 presents information related with the fuel failures occurred during 2001 and 2002. In only one case out of four the defected fuel assembly was detected during the post irradiation visual inspection. In another case the failed fuel assembly was detected during the wet sipping test and in the other two cases the failed fuels were not identified. The failed fuel rod detected during the visual inspection showed hydrides in both end cap-sheath welding areas and also an opening in one of them. The reason of the failure was not precisely determined.

3 Fuel Design and Fuel Fabrication Issues

The design of the fuel for Embalse has reached a high maturity state. The design of the main components of the fuel assembly remains very stable and only minor changes were introduced to cover manufacturing requirements, to reduce costs or to improve fuel reliability

Despite the excellent performance of the fuel a real need to reduce costs still remains in order to keep the competitiveness of the nuclear energy.

Fuel design contributions developed by CNEA toward this objective were presented during the 1997 CANDU Conference [1]. To increment the U content, several design optimizations were proposed and developed. In a first stage, these modifications affected only the dishing volume and the diameter of the fuel pellets and the length of the fuel stack. Three ways have been considered for the last one: a reduction of the nominal axial gap between fuel pellets stack and endcaps, a reduction of the fuel stack length tolerance and a small increment of the length of the fuel bundle. Table 3 shows the dates when each one of the design changes was

1 Comisión Nacional de Energia Atómica, Argentina

2 NA-SA Embalse Power Station, Argentina

fully implemented in the fabrication line. All the modifications were completely implemented by the end of 2000. Since then all the domestic fuels assemblies irradiated in Embalse had those changes and the overall performance of the fuel assemblies showed no detrimental effects.

The benefits of these design optimizations in terms of U content increment are reported in Table 4 (2).

The increment of the density of the fuel pellets also produces good results but its application depends on another factors like powder quality and pellet fabrication technology. During the last three years the average pellets density remained higher than 12.61 g/cm3.

Design optimizations to reduce fuel-manufacturing cost are developed in a close agreement with the fuel manufacturer.

4 Fuel Fabrication capabilities

During the last 4 years the manufacturing line of CONUAR was improved considerably with new equipment (3). The main modifications were oriented to increase the reliability of the fuel, to reduce intermediate stocks and also to reduce manufacturing costs. These modifications included the replacement of the end cap welding machine and the complete automation of the fuel rod fabrication line.

The advantages of these improvements are summarized in Table 5. Figures 1, 2 and 3 present pictures showing the new facilities for fuel rod fabrication. The most important results are related with the significant reduction of the rejection rate, another advantages are shown in Figure 4.

5 Embalse SEU Fuel Program

NA-SA and AECL has been analyzing under the framework of a co-operative program the feasibility of using Slightly Enriched Uranium (with 0.9 w% 235U) fuel in the Embalse nuclear power reactor. Using SEU fuel would produce a significant increase in the fuel discharge burnup, from 7.35 MWd/kgU currently achieved with natural-uranium (NU) fuel to about 14 MWd/kgU. This would lead to a reduced fuel-cycle cost and a large reduction in spent-fuel volume per full-power-year of operation.

NA-SA and CNEA have already implemented the conversion of the PHWR Atucha I NPP from NU to SEU (0.85 % 235U) during the 90´s with excellent results.

Some activities related with the assessment of the Embalse fuel performance up to SEU typical burnups are still in progress. Among them CNEA has recently performed preliminary calculations to evaluate the typical behaviour of the fuel in nominal design conditions.

To perform these studies NA-SA provided typical instantaneous fuel power distributions at different fuel burnups (3). Simplified power histories were built from those distributions.

Main parameters analyzed were fuel center temperature,

Table 1: Relevant data related with the irradiation and performance of the fuel elements in Embalse NPP.

Concept	YEAR	
	2001	2002
Fuel bundles irradiated	5416	4770
EFPD	353,12	304,82
Loading factor	97,54%	83,57%
Average Discharge Burnup [MWD/t.U]	7423,6	7315,4
Failed Fuel Assemblies	3	1
Annual Failure Rate	0,06%	0,02%
Fuel Failure rate during the last 5 years 1998-2002	0,04%	

Table 2: Information regarding fuel failures occurred during 2001 and 2002

Year	Channel	Failed FA identified by	Position in the channel	Discharge Burnup [MWd/tU]
2001	N17	Visual Inspection	11	5575,9
	K08	Wet Sipping	9	1943,9
	M15	Not identified		4752.9 (*)
2002	K13	Not identified		7829,0(*)

(*) Average burnup of the 8 FA replaced.

Table 3: Dates when each one of the design changes to increase U content was finally implemented for fabrication

Dishing depth reduction	July 97
First increment of FP diameter	October 97
Stack length modifications	May 98
Second increment of FP diameter	November 2000

Table 4: Evolution of the U content in the Embalse fuel bundles during the last three years

Year	Average U content [kg]	Minimum U content [kg]
2001	18.978	18.941
2002	19.008	18.952
2003	19.000	18.949

Table 5: Main advantages of the new fuel rod fabrication line

Failure rate < 2 ppm (2 failed FE/15000 FE means more than 1.500.000 weldings without failures)
Self paid after less than 5 years
Productivity improvement: 22%.

internal gas pressure and cladding strains. Fuel calculations were performed using ELESIM mod.9 and mod.10 codes. These preliminary results are encouraging and allow predicting a very limited impact of the higher burnup on the fuel performance for the current design conditions.

6 Final Remarks

Domestic fuel performance in Embalse NPP during the last two years has been excellent without a significant occurrence of fuel failures. The failures level was reasonably low with a Fuel Failure rate of 0.04 % for the last 5 years (1998-2002).

The implementation of fuel design optimizations to increase U content was fully completed by the end of year 2000. The in-reactor performance was not affected and shows the high degree of maturity reached for both, the design and the manufacturing procedures and capabilities.

During the last 4 years a significant improvement of the manufacturing process was achieved, mainly with the complete automation of the fuel rod fabrication line.

A feasibility study for the utilization of SEU in Embalse NPP conducted by NA-SA and AECL is almost completed. Fuel behavior simulations using envelope power histories were performed by CNEA as part of the assessment of the influence of SEU fuel burnup extension. This work is still in progress.

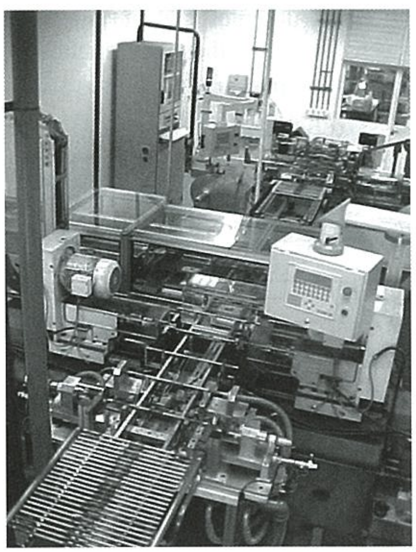
7 References

(1) L. Alvarez, J. Casario, R. Olezza, 5th International Conference on CANDU Fuel, Toronto, CANADA, September 1997.

(2) R. Lamuedra (CONUAR), personal communication, May 2003.

(3) J. Fink (NA-SA), personal communication, August 2003.

Figures 1 and 2: New facilities for fuel rod fabrication – Cladding Machine and Endcap Welding.



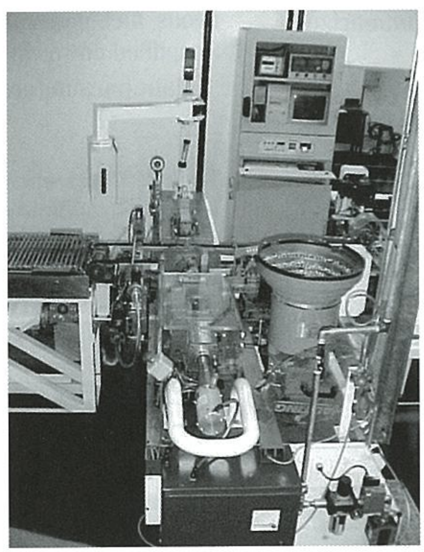


Figure 3: New facilities for fuel rod fabrication. General View.

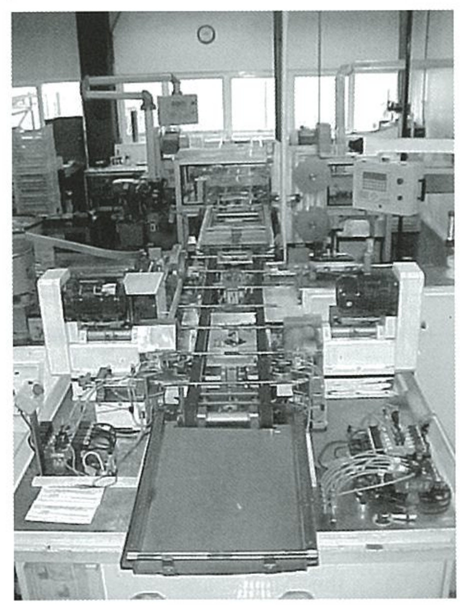
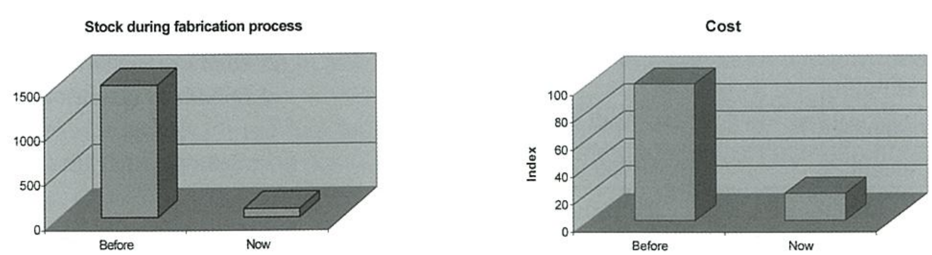


Figure 4: Main advantages of the new fuel rod fabrication equipment at CONUAR.



Advanced Fuel Development in AECL

by Peter Boczar

Ed. Note: *Peter Boczar gave two presentations at the 8th International Conference on* CANDU Fuel, *September 21-24, 2003, at Honey Harbour, Ontario, one titled* Advanced Fuel Development in AECL, *the other* ACR Fuel Design. *They were based on the following paper he and co-authors prepared for a subsequent international meeting. The authors of that paper are:* Peter G. Boczar, Jim D. Sullivan, Mukesh Tayal, Al M. Manzer, Ki-Seob Sim, Steve J. Palleck, Krishnan Chakraborty, *all with Atomic Energy of Canada Limited*

Abstract

The Advanced CANDU Reactor (ACR) offers a significant reduction in capital cost compared to current operating designs, and improvements in inherent safety characteristics and performance. It is built on the traditional CANDU fuel-channel design with on-power refueling, and features light-water coolant, heavy-water moderator and reflector, enriched uranium CANFLEX fuel, and a reduced lattice pitch (distance between fuel channels). This core design retains important benefits of the well-established CANDU 6 reactor, and introduces additional improvements. Fuel is a key enabling technology in the ACR. This paper describes the ACR fuel, the experience on which the design is based, and the fuel qualification program.

1. Introduction

1.1 The Advanced CANDU Reactor (ACR)

The Advanced CANDU Reactor (ACR) is AECL's "Next Generation" CANDU reactor, aimed at producing electrical power at a capital cost significantly less than that of current reactor designs, and competing with natural gas combined-cycle electricity production[1] . The ACR incorporates major improvements in economics, inherent safety and performance, while retaining the proven benefits of the CANDU design. The ACR-700 produces 731 MWe, about the same as the CANDU 6 reactor.

The ACR achieves a substantial reduction in capital cost by using light water coolant with heavy water moderator, in a tight lattice pitch. This more compact core reduces heavy water inventory to about one-quarter of that in the CANDU 6 reactor, and also reduces the size of the reactor shield tank assembly. The tight neutronic coupling in a small core results in exceptional stability. The use of enrichment in the CANFLEX fuel bundle enables a reduction in the number of fuel channels, from 380 in the CANDU 6, to 284 in the ACR-700. Each channel runs at close to the same power, so the radial power distribution across the core is very flat. Other design simplifications include the use of two steam generators instead of four in CANDU 6, and a simpler interface between the Emergency Core Cooling and the Heat Transport System (HTS) since they both contain light water. The reactor building is smaller than for the CANDU 6 reactor. Substantial economic benefits are also achieved by operating at higher coolant and steam turbine supply pressure and temperature, resulting in higher thermal/electric conversion efficiency. The licensing case is enhanced by reactivity coefficients that in general are small in magnitude and negative. In particular, void reactivity is negative.

1.2 ACR Fuel

Fuel is a key enabling technology in the ACR. Fuel burnup was chosen to be about 21 MWd/kg, nearly 3 times that of natural uranium fuel. This burnup is sufficiently high to reduce fuel cycle costs, but low enough to ensure good fuel performance. The quantity of spent fuel will be reduced by about a factor of 3 compared to natural uranium fuel. The CANFLEX fuel bundle was chosen as the bundle design, because the increased subdivision, with its 43 fuel elements compared to the standard 37-element bundle, reduces fuel linear element ratings, facilitating an increase in burnup. The critical heat flux (CHF)-enhancement buttons on the elements also increase thermalhydraulic margins.

Enriched uranium is necessary for the ACR. Enrichment offers greater flexibility in reactor design: it allows the elimination of three-quarters of the heavy water in the CANDU 6 reactor; it allows a slightly thicker pressure tube, to enable higher PHTS pressures and temperatures, leading to higher thermal efficiency and hence improved resource utilization and lower electricity production costs; enrichment also

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allows the judicious use of a neutron absorber (dysprosium, Dy) in the central element of the CANFLEX bundle to tailor the void reactivity.

Optimizing the fuel design is a complex process, since no single fuel design simultaneously meets all the competing objectives, including minimizing

- fueling costs,
- fuel complexity (from a manufacturing perspective),
- residual fissile content in spent fuel (from a proliferation resistance perspective),
- fuel linear element ratings (from a safety and operational performance objective), and maximizing
- fuel burnup, and.
- uranium utilization.

Several options were considered in the conceptual fuel design. The ACR reference design utilizes uniform enrichment in the outer 3 rings of the CANFLEX bundle (2.1% U-235), with ~7.5 wt% Dy mixed with natural uranium in the central element. The enrichment and Dy content are slightly higher than in the original conceptual design, to ensure that coolant void reactivity is negative under all applicable design and operating conditions, accounting for calculation bias and uncertainties. The ACR utilizes the extensive CANDU experience with collapsible fuel cladding, which improves the heat transfer between the fuel, cladding and coolant. The cladding thickness for ACR fuel has been optimized to address the higher PHTS coolant temperatures and pressures. The endcap weld geometry has been designed to minimize cladding stresses. The internal void within the fuel stack has been increased (compared to the natural uranium CANDU fuel design) by modifications to the fuel pellet design to accommodate the additional fission-gas release with extended burnup, but gas plenums are not required. The ACR fuel bundle will have longer bearing pads to provide support during passage over the sealing groove in the endfitting during refueling operations. The sealing groove is a new feature associated with the improved fuel channel closure plug in the ACR. The ACR fuel bundle bearing pads are also slightly higher than those on the CANFLEX Mark IV design, which has been qualified for use in the CANDU 6 reactor.

The higher bearing pads raise and center the bundle in the pressure tube, reducing the coolant flow bypass over the top of the bundle, and increasing thermalhydraulic margins.

1.3 Interaction between Physics, Thermalhydraulics, and Fuel Performance

The ACR design takes full advantage of the interaction between reactor physics, thermalhydraulics, and fuel design. Simply substituting light water coolant for heavy water would result in very large and positive coolant void reactivity during postulated loss of coolant accidents in the CANDU 6 reactor. In the ACR, the tight lattice pitch and larger gap between the pressure tube and calandria tube significantly reduce the coolant void reactivity. The addition of a small amount of neutron absorber to the center of the fuel bundle drives the void reactivity negative[2] . Moreover, there is flexibility in further tuning the design to achieve slight changes in the value of void reactivity, as uncertainties in the physics parameters are reduced through the physics analysis code validation program. As experience is gained with extended burnup fuel performance in the ACR, fuel enrichments and burnups could be progressively increased to reduce fueling costs.

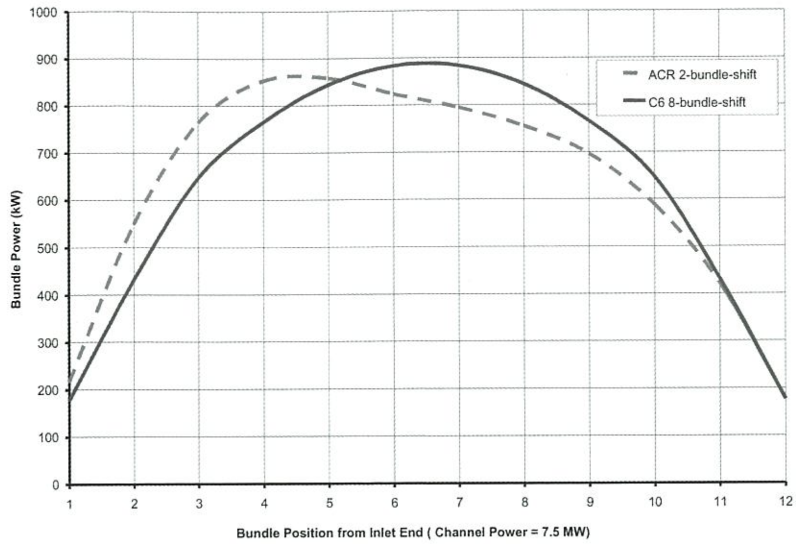

Fig. 1 Bundle-Power Profile in ACR-700 vs. CANDU 6

Fuel management in the ACR is based on decades of experience with on-power refueling in CANDU reactors. A simple bi-directional (adjacent channels are refuelled in the opposite direction), fueling-with-flow, 2-bundle-shift refueling scheme results in an excellent axial power profile (Fig. 1). The power peaks at the inlet end of the channel near bundle position 4, and decreases along the length of the channel to the outlet end. Sustained power boosting during refueling only occurs with relatively low burnup fuel, which is resilient to power boosts; after the initial peak, the fuel sees a declining power history, which ensures good fuel performance margins; the axial power shape provides better thermalhydraulic margins (in terms of critical heat flux, or critical channel power) than a cosine- or outlet-skewed axial power distribution. For a given bundle power limit, the ACR axial power profile allows slightly higher channel powers relative to the traditional cosine-shape CANDU 6 axial power profile.

The CANFLEX geometry allows an increase in average bundle and channel powers relative to the CANDU 6, while still having lower linear element ratings than for 37-element natural uranium fuel in the CANDU 6 reactor.

2. Experience Base For ACR Fuel

ACR fuel is based on three well established CANDU fuel technologies: the CANFLEX fuel bundle, enriched uranium to achieve extended burnup, and Low Void Reactivity Fuel (LVRF).

2.1 Canflex

The CANFLEX bundle is the latest advance in the evolution of CANDU fuel[3] . It has a total of 43 elements, arranged in concentric rings of 1, 7, 14 and 21 elements. The 35 elements in the outer two fuel rings are smaller in diameter (11.5 mm) than the inner 8 elements (13.5 mm). The combination of increased number of elements and 2 element sizes reduces the peak linear element rating for a given bundle power by 15-20% compared to that in the standard 37-element fuel bundle. The lower ratings have both safety and operational benefits, and ensure good fuel performance margins at higher burnup. The bundle also features non-load-bearing appendages, attached to the cladding of all elements at strategic locations to promote turbulence and coolant mixing between subchannels. This increases the thermalhydraulic margins (higher CHF and critical channel power).

The CANFLEX bundle has undergone an extensive qualification program, consisting of in-reactor tests in the NRU reactor at the Chalk River Laboratories (CRL), out-reactor mechanical tests, thermalhydraulic measurements, reactor physics measurements, mechanical analysis, and a demonstration irradiation in a power reactor[4]. A Design Verification Plan (DVP) identified the performance requirements and the tests or analysis required to confirm that requirements were met. All testing and analysis conformed to the quality standard CAN/CSA-N286.2 or equivalent[5]. The DVP called for preparation of a Test Specification, Test Procedure and Acceptance Criteria, and identified the required documentation. Results of the qualification program were documented in the Fuel Design Manual.

The out-reactor qualification tests simulated CANDU 6 reactor conditions and confirmed compatibility of the design with the fuel channel, fueling machine, and other reactor hardware[6].

- Strength tests showed that the fuel withstands the axial loads imposed during refueling.
- Impact tests showed that the CANFLEX bundle withstands impact during refueling.
- Cross-flow tests demonstrated that the bundle could successfully withstand the flow-induced vibration if it were inadvertently stuck in the liner hole region of the channel for more than 4 hours.
- Compatibility tests showed that the bundle is dimensionally compatible with the fueling machines.
- A flow endurance test demonstrated that the bundle maintains its structural integrity during operation, and that fretting wear on the bearing pads, inter-element spacers and pressure tube remain within design limits.

These mechanical tests were supplemented by analysis for cladding strains, fission-gas pressure, endplate loading, thermal behaviour, mechanical fretting, sliding wear, element bow, end-flux peaking, and a range of other mechanical characteristics. Acceptance criteria were based on previous criteria originally developed for CANDU 6 fuel.

Thermalhydraulic measurements were conducted in both Freon-134a and in water. Measurements were made of the single- and two-phase pressure-drop characteristics of the bundle string, and the CHF and post-dryout behaviour[7]. Water-CHF measurements were made in a full-length, 6-m-long 43-element bundle simulator, under a wide range of steam-water flow conditions[8]. Sliding thermocouples scanned the elements both axially and circumferentially, to detect the very first occurrence of dryout. This "onset of intermittent dryout" happens when there is a slight increase in cladding temperature at the dryout location. The rest of the cladding, and the other elements, remain wet. This dryout phenomenon occurs under different thermalhdyraulic conditions than those in a.PWR, and is quite benign, as there is no drastic deterioration of heat transfer. Measurements were also made in freon of the post-dryout cladding temperatures, up to powers 60% greater than the dryout power.

In-reactor tests on CANFLEX bundles were conducted under CANDU pressure/ temperature conditions in the fuel testing loops in the NRU reactor at CRL. These tests subjected the bundles to powers over 25% greater than would be experienced in a CANDU 6 reactor. Power ramp tests confirmed performance for the power changes occurring during refueling in a power reactor. A demonstration irradiation of 24 CANFLEX Mark IV bundles, in a high-power, and a lower-power instrumented channel, was conducted in the Point Lepreau Generating Station in New Brunswick, between 1998 September and 2000 August. In-bay inspections of all bundles after refueling, and PIE of 2 bundles in the hot-cells at CRL confirmed that fuel bundles were defect-free and in good condition; performance indicators (such as residual cladding strain, fission-gas release and grain growth) were within the range expected for CANDU fuel operating under similar conditions[9]. A similar demonstration irradiation is underway in the Wolsong 1 reactor in South Korea.

The ZED-2 facility at CRL was used to measure the fine structure, reaction rates, and reactivity coefficients for CANFLEX natural-uranium bundles, for validation of the reactor physics lattice code WIMS-IST. Good agreement was obtained between measurements and code predictions, with the bias and uncertainty in the key reactor physics parameters similar to that with 37-element fuel[10].

The CANFLEX Mark IV bundle has been fully qualified with natural uranium fuel, for a CANDU 6 reactor. The conversion to a full core of CANFLEX natural uranium fuel in a reactor in Canada will require regulatory approval. The proponent will have to demonstrate that the change in fuel type does not compromise the safe operation of the reactor, based on the existing safety report and supporting docu-

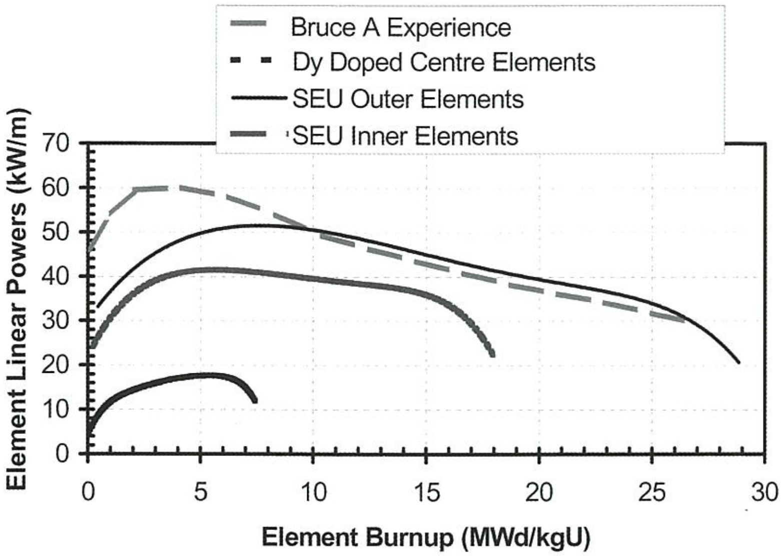

Fig. 2 ACR High Power Envelope vs Bruce A Envelope

mentation. In addition, the assessment must consider the transition between an all-37-element-bundle core, and an all-CANFLEX core, since this process takes place over an extended period of operation at power (about two years). AECL is working with CANDU utilities in Canada to establish the licensing program requirements and the various roles and responsibilities.

ACR fuel uses the CANFLEX Mark V version, which has slightly higher bearing pads than the Mark IV version described above. The higher bearing pads reduce the bundle eccentricity in the channel. By elevating the bundle, the flow by-pass over the top of the bundle in a crept pressure tube is reduced, resulting in further increases in the CHF and critical channel power over the Mark IV design[11]. The ACR fuel design also has longer bearing pads than the CANFLEX Mark IV bundle.

2.2 Enriched Uranium for Extended Burnup

The second key enabling technology in the ACR is enriched uranium to achieve higher fuel burnup. ACR average fuel burnup (21 MWd/kg) is nearly three times higher than for natural uranium fuel. The fuel cycle economics of the ACR are different from those of the CANDU 6 reactor, and favor higher burnup. The burnup chosen was high enough to achieve economic benefit, but within CANDU experience. In the Bruce A power reactor, more than 230, 37-element bundles have achieved burnups in excess of 17 MWd/kg. This is illustrated in Fig.2, in which the Bruce A power/burnup envelope is compared to the ACR high power envelope for the center element, the inner ring of 7 elements, and the outer ring of 21 elements (the ratings for the intermediate ring of 14 elements in the CANFLEX bundle are always lower than for the outer ring). The ACR power envelope is generally below the CANDU high power envelope for which we have considerable operating experience.

This is supplemented by irradiations of experimental bundles in the NRU reactor, where there have been more than 24 bundle and element irradiations with burnup in excess of 17 MWd/kg, and 15 irradiations with burnups greater than 21 MWd/kg. Ten of 24 irradiations also experienced power ramps. The power histories of these irradiations fully bound the operating range of the ACR fuel. This experience is summarized in Table 1. It is noted that all of the NRU irradiations use enriched fuel (in light water coolant) to achieve the power/burnup histories required. Fuel performance parameters germane to extended burnup are summarized in Reference 12. Also, while CANDU power reactor irradiations have used the standard 37-element fuel design, the ACR design features optimized pellet and weld geometries..

In addition to the CANDU-specific experience, of course there is exhaustive experience with LWR fuel with much higher burnups than in the ACR.

2.3 Low Void Reactivity Fuel

Finally, the ACR fuel design is actually a variant of LVRF[13]. This concept, which can be embodied in either the 37-element or CANFLEX bundle, was conceived by AECL in the early 90's as an option for reducing the positive void reactivity in the.CANDU 6 reactor, in those jurisdictions where this might have been perceived as an impediment to licensing. This fuel has a neutron absorber in the center of the bundle (in the central element, and possibly the next ring), mixed with either natural or depleted uranium. Dysprosium was chosen for CANDU applications because of its burnup characteristics. The rest of the fuel elements in the bundle contain enriched uranium.

Both burnup and void reactivity can be varied by selecting the Dy-content and U-235 enrichment. The mechanism for void reactivity reduction is the flattening of the thermal flux across the lattice cell in a postulated loss-of-coolant accident, which increases parasitic absorption in the center of the bundle and contributes a negative component to void reactivity. This flexibility was exploited in the ACR fuel design to independently tailor the values of void reactivity and burnup.

AECL undertook a series of qualification tests for both 37-element and CANFLEX embodiments of the LVRF bundle, to demonstrate technical feasibility. Both variants would have negative void reactivity in the CANDU 6 reactor. The 37-element LVRF design had a burnup of ~10 MWd/kg, while the CANFLEX design had a burnup of ~21 MWd/kg (in the CANDU 6 reactor). The tests included the following:

- Dy_2O_3 -UO_2 pellet fabrication, and prototype bundle fabrication for both physics and fuel performance tests
- irradiation tests in NRU and post-irradiation examination of prototype LVRF bundles, and demountable elements containing Dy-doped fuel

Table 1: Summary of Extended Burnup UO2 Fuel Experience in NRU Loop Irradiations

Bundle ID	Maximum element power (kW/m)	Maximum element burnup (MWd/kg U)	Failure (Yes, No)	Power history mode
GC	43	18.1	No	Declining
GE	43	17.5	No	Constant
GF	37	37.6	No	Constant
XY	33	23.7	No	Constant
ZR	34	23.2	No	Constant
ZS	35	28.5	No	Constant
AAH	37	28.7	No	Declining
AEX	34	19.6	No	Declining
JC	63	26.8	No	Declining
HR	55	18.0	No	Declining
AJN	70	23.1	No	Declining
DME 190	53	23.5	No	Declining
DME 191	57	29.3	No	Ramped
DME 216	66	18.9	No	Declining
DME 217	65	29.3 (Note 1)	No	Declining
ZX	70	21.8	No	Ramped
ZY	67	20.0	No	Ramped
AAW	70	26.8	No (Note 2)	Ramped
AJM	69	22.4	No	Ramped
AKV	49	18.3 (Note 3)	No	Ramped
AKW	45	18.6	No	Ramped
DME 195	66	22.2	No	Ramped
DME 210	64	26.5	Yes	Ramped
DME 215	54	7.8 (Note 4)	Yes	Ramped

Notes:
1. *DME-217 irradiation continues to 41.7 MWd/kg.*
2. *Suspected to be failed based on loop coolant fission product levels; no failures were confirmed in the hot cells.*
3. *AKV irradiation continues to 29.2 MWd/kg.*
4. *DME-215 irradiation continues to 25.0 MWd/kg.*

- substitution experiments in AECL's ZED-2 critical facility for measurements of reactivity, reactivity coefficients and fine structure, and subsequent validation of physics codes
- thermalhydraulics measurements in freon, and modeling
- preliminary safety experiments with Dy-doped fuel, looking at high-temperature interactions with Zircaloy and the grain-boundary inventory of fission products.

This testing provided generic qualification of the LVRF concept. Further qualification is of course required for a specific design and reactor application. However, this testing has provided a solid foundation for application either in existing reactors, or in the ACR.

2.4 Generic Advanced Fuel Development Supporting ACR Fuel

AECL has an ongoing program on advanced fuel technologies that support the ACR, including the reference ACR fuel design and advanced fuels and fuel cycles that can be effectively utilized in the ACR[14]. This includes continued irradiations in the NRU reactor of a variety of advanced fuels (enriched uranium, MOX, DUPIC, thorium), as well as separate-effect irradiations (Dy-doped fuel, optimized internal element design, advanced CANLUB coatings that protect against stress-corrosion cracking) under a variety of power envelopes (high power, extended burnup, power ramps). An important part of our fundamental studies is the refinement of the stress-corrosion power-ramp failure criteria, for multiple power ramps and extended-burnup fuel. A particularly fruitful area of study is the use of Imaging X-ray Photoelectron Spectroscopy (XPS) to study the microchemistry at the fuel-cladding interface, which is yielding new insights into the migration of fission products, and their effect on stress-corrosion cracking. Very accurate measurements are being made of the oxygen-to-uranium ratio axially and radially in pellets in both intact and defected fuel, to determine the effect of oxidation on fuel performance. We have international collaboration to measure the intrinsic diffusion coefficients of important fission products. AECL is also participating in the IAEA FUMEX II project on inter-code comparisons of fuel performance for high-burnup fuel. International collaboration continues on the use of slightly enriched or recovered uranium in CANDU. All of these programs provide ongoing support to the ACR, and are described in more detail below.

2.4.1 Extended Burnup Enriched Uranium

Experiments are being performed on a fuel bundle containing elements having various internal design features to improve high burnup performance. The features include changes to pellet geometry and the inclusion of plena to provide internal void space to accommodate released fission gas. This bundle has now reached a burnup of 30 MWd/kg with the goal to exceed 40 MWd/kg. The bundle has demountable elements that can be removed during the irradiation, and examination of some the elements discharged earlier in this irradiation has provided valuable information that has contributed to the current design of ACR fuel.

2.4.2 MOX

MOX fuel could be burned very effectively in the ACR. AECL maintains a MOX fabrication facility and has performed many irradiations of MOX fuel bundles. Currently irradiations are being performed to assess the criteria used in the specification of CANDU MOX fuel. A fuel bundle was fabricated containing MOX elements with the plutonium dispersed in the uranium matrix using a variety of techniques, resulting in fuel pellets with a range of controlled homogeneity. By irradiating these elements together in the same fuel bundle and under the same conditions, it will be possible to compare their performance directly and correlate performance with plutonium homogeneity, providing valuable support to the rational for the MOX fuel specification[15].

2.4.3 DUPIC

DUPIC (direct use of spent PWR or BWR fuel in CANDU) is a dry-recycle fuel cycle in which spent fuel discharged from a PWR or BWR is reconfigured to make a fuel that can be used directly in a CANDU or ACR[16] . Because there is no wet chemistry involved in the process, there is no possibility to divert special fissional materials, making the process inherently proliferation resistant. By extracting additional energy from the spent PWR or BWR fuel prior to final disposal, the combined fuel cycle produces less waste per kWh of electricity generated, effectively reducing the volume of spent fuel generated by the system. The dry process is also inherently simpler than wet recycling processes (Purex or Urex) and, as a result, should also be less expensive.

AECL successfully fabricated three DUPIC elements making use of spent PWR fuel that was discharged at a burnup of 28 MWd/kg. The DUPIC elements were irradiated in the fuel test loops in NRU to burnups of 10, 16 and 21 MWd/kg respectively. PIE has been completed on the first two elements[17]. The performance of this fuel is not significantly different from fresh fuel that has seen the same power-burnup history, except for slightly higher fission-gas release and microstructural features (primarily metal-particle fission products) typical of high burnup fuel (reflecting the accumulated burnup in both the PWR and NRU of up to 44 MWd/kg).

2.4.4 Thorium

Thorium is a fuel cycle option for ACR that may be attractive to countries that have a domestic supply of thorium but limited resources of uranium[18]. The fuel cycle flexibility of the ACR allows several different thorium fuel cycles to be considered. Thorium has a number of properties that make it superior to UO_2 as a fuel type, including higher thermal conductivity (the fuel runs cooler), higher melting temperature (3300°C vs. 2800°C for UO_2), lower fission gas release (which aids in achieving high burnups), resistance to oxidation (thorium has only one oxidation state) and chemical inertness (should be a more stable waste form that is expected to be resistant to leaching).

There are irradiations underway in the NRU reactor to demonstrate the correlation between ThO_2 microstructure and fuel performance. The fuel types are pure ThO_2 and $(Th,U)O_2$. The ThO_2 has achieved a burnup of ~12 MWd/kg and the (Th,U)O2 ~21 MWd/kg. Current plans are to extend some of these fuel elements to a burnup of about 40 MWd/kg.

2.4.5 CANLUB

Segregation of fission products that can cause stress-corrosion cracking (SCC) of the Zircaloy fuel cladding increases progressively with burnup and power. The CANLUB graphite coating on the inside of the fuel cladding inhibits SCC by acting as a fission-product chemical barrier. The investigation of the power-ramp performance of both the fuel and of CANLUB coatings is advancing on three fronts: microchemical investigations at the fuel-cladding interface, advanced fundamental modeling of the power-ramp thresholds, and in-reactor tests of fuel at the limits of the power-ramp thresholds.

An Imaging-XPS technique has been developed to study fuel microchemistry at the fuel-cladding interface of irradiated fuel that has undergone power ramps. These investigations are providing valuable insights into the mechanisms affecting fuel performance.

To assure successful fuel operation at extended burnup, it is necessary to establish defect thresholds for SCC-induced power ramp failures. At present, SCC defect thresholds are obtained by using an empirical approach developed in mid-70s, using the following parameters: maximum power, power-ramp, burnup, hold time, and CANLUB.

A new formulation has been developed that considers the work density of the fuel cladding and the generation of corrosive fission products, and correlates these features with the irradiation history of the element prior to the power ramp and the change in power experienced during the ramp. This methodology provides a good fit to the existing power ramp data, and because it is mechanistic, it can be more reliably extrapolated to higher burnups[20].

In addition to these investigations, two bundles containing elements with various coating formulations and application procedures are undergoing irradiation in the NRU fuel test loops. These irradiations are to investigate the performance of these coatings under power ramp conditions and will provide insight into the mechanisms affecting power ramp performance.

2.4.6 Fundamental Properties

Development of the coulometric-titration method for assessing oxidation of both irradiated and unirradiated fuels is reaching a mature state[21]. Measurements of the extent and distribution of oxidation in several defected fuels (with holes in the cladding that allowed coolant ingress) have been done. These have shown that the size of the defect and the in-reactor residence time both influence oxidation. Recent studies of $(U,Dy)O_2$ fuels indicate that they have more resistance than UO_2 fuels to oxidation. This work is continuing at higher temperatures (400-1100°C).

Capabilities are being developed to extend the coulometric-titration method to even higher temperatures (from ~1100°C to ~1500°C) and more oxidizing environments. Comparisons are also being made of the oxidation of SIMFUEL against UO_2 and will be extended to irradiated fuels in the future.

Work continues on a novel method of inferring the migration behaviour of fission products in irradiated fuel, which allows for the effects of thermal diffusion, radiation damage and local segregation to be independently assessed[22].

3. ACR Fuel Qualification

3.1 Design Requirements

ACR designers have specified the core-average discharge burnup of ACR fuel (21 MWd/kg) and negative void reactivity during loss-of-coolant accidents. The coolant temperature and pressure in the fuel channel are slightly higher than in the CANDU 6 reactor. The fuel is required to stay intact during its life; be able to navigate available passages and openings during refueling; and interface with available supports in the fuel channel and in the fuel handling equipment.

3.2 Objective and Approach

The objective of fuel design qualification is to confirm that no configuration permitted by design and operational tolerances in fuel and reactor can lead to unacceptable damage to the fuel or to an interfacing component. To achieve this, the ACR fuel qualification plan includes in-reactor tests, out-reactor tests, and analyses enveloping all permitted operational and design configurations. All credible damage mechanisms are being systematically evaluated, individually and in combination as appropriate. The qualification assessments are grouped into four broad areas:

- Thermal integrity, to confirm that various parts of the fuel bundle have comfortable margins against overheating. Some illustrative examples of related damage scenarios include end-temperature peaking and CHF.
- Structural integrity of the fuel element, to confirm that all parts of the fuel element have comfortable margins against cracking or breakage. Some illustrative examples of related damage scenarios include stress-corrosion cracking, internal gas pressure, longitudinal ridging, and oxidation.
- Structural integrity of the fuel bundle, to confirm that all parts of the fuel bundle operate with comfortable margins against cracking or breakage. Some illustrative examples of related damage scenarios include refueling loads and fatigue.
- Dimensional compatibility, to confirm that during various stages of life—loading, irradiation, and unloading—all parts of the fuel bundle stay within available clearances and do not damage each other or interfacing systems. Some illustrative examples of related damage scenarios include bowing, droop, and dimensional compatibility with fuel-channel and fuel-handling systems.

Qualification of the fuel will include sufficient margins for burnup, peak element ratings, coolant temperature and flow.

3.3 Design Criteria

For each credible damage mechanism, design criteria are being established to cover the operating conditions of ACR fuel. As an illustrative example, a design criterion may be defined for internal gas pressure, specifying that under normal operating conditions the internal pressure stays below coolant pressure throughout in-reactor residence of the fuel bundle. To confirm that, we would first calibrate the fuel performance code ELESTRES[23] with all available pertinent data for fission-gas release in fuel with ratings and burnups that cover the ACR range. Then the code would be used in parametric studies to simulate all permitted combinations of power histories and fuel design tolerances to confirm that they all meet the design criterion specified above. Similar combinations of tests and analyses as appropriate are planned for all credible fuel damage mechanisms.

3.4 In-Reactor Tests

All-effects irradiation tests in the NRU reactor at CRL will confirm the overall fitness of the fuel design for ACR operation. Irradiation tests will simulate operation at powers and burnups that bound the ACR operating envelope for both poisoned and non-poisoned elements. Another important objective is to generate data for validating computer codes such as ELESTRES and BOW[24,25] that will be used to qualify many aspects of ACR fuel, as discussed in Section 3.6. Important data towards this end include fission-gas release, cladding ridge strain, fuel-element bowing, and UO_2 grain growth. Other irradiation tests in NRU will confirm the ability of ACR fuel to survive power ramps expected in the ACR reactor.

3.5 Out-Reactor Tests

Sixteen out-reactor tests will confirm the adequacy of design details in a number of diverse areas such as thermal performance, strength, and the dimensional compatibility of the bundle with the fuel-handing system: CHF, pressure-drop, and post-dryout temperatures; bundle strength; endurance; cross-flow; fuel-handling compatibility; spacer interlocking; bent-tube gauge (to ensure the passage of the bundle through a sagged fuel channel); sliding wear; refueling impact; cladding ridging; cladding corrosion; fuel channel component optimization; stress-corrosion cracking; and coolant pressure pulsation.

Some of the tests will provide data that directly qualify a specific aspect of fuel bundle design. Many of the tests will provide data that will be used in subsequent assessments that will scale the data from test conditions to in-reactor conditions and will also account for design and operational tolerances.

3.6 Analyses

The objective of analytical assessments is to systematically confirm that ACR fuel meets all design criteria for all damage scenarios. Some of the assessments will use as input measurements described above from in-reactor and out-reactor tests, and where required, scale them to in-reactor, on-power conditions. The following are some illustrative effects of irradiation that require scaling: reduction in the ductility of Zircaloy with fluence; stiffening of the fuel element due to thermal expansion of the pellet; dimensional changes in fuel due to creep, etc. Likewise, post-irradiation measurements from in-reactor tests require scaling to cover on-power effects. For example, post-irradiation measurements of fission-gas release need to be converted via analyses to peak on-power internal pressure. As well, analyses take data measured from a given set of dimensions and test conditions in a limited number of tests, and convert them to all combinations permitted by design and operational tolerances.

A large number of such assessments are envisioned. The following are some specific illustrative examples: assessment of internal gas pressure; assessment of peak pellet temperature, taking into account the high-power envelope and end-flux peaking during refueling; cladding temperature, taking into account braze voids between bearing pads and cladding, oxidation and crud; on-power cladding deformations; assessment of static and impact strength of the bundle during refueling, taking into account operating temperature and irradiation embrittlement; analysis of bundle fatigue, taking into account flow-induced vibrations and pellet expansion; axial expansion of fuel bundle string; etc.

The above assessments will require a suite of specialized computer codes such as ELESTRES, FEAT[26], FEAST[27], BOW, INTEGRITY[28], BEAM[29], etc. Most of our computer codes already contain the required capabilities. Additional capabilities are being added to our codes where required for ACR application. All computer codes will be fully qualified to ISO 9001-2000 and CSA N286.7.

3.7 Summary: Fuel Qualification

ACR fuel is being qualified by confirming that all parts of the fuel bundle will meet a set of pre-defined design criteria that will ensure thermal integrity, structural integrity, and compatibility with interfacing components. These assessments are being done via a comprehensive integrated set of in-reactor tests, out-reactor tests and analyses, using qualified facilities, computer codes and staff. This program is designed to confirm that no configuration of ACR fuel that is permitted by design and operational tolerances in fuel and in reactor can lead to unacceptable damage to the fuel or to an interfacing component.

4. Conclusions

ACR fuel is based on extensive experience with the CANFLEX fuel bundle, extended burnup experience using enriched uranium, and low-void-reactivity fuel. This provides excellent confidence in the qualification of ACR fuel.

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CNS Quebec and the Institut de Recherche sur l'Hydrogène (IRH) held a joint conference in Trois-Rivières, October 29, 2003. Shown are left to right: Dr. T. K. bose, Director of IRH and President of the Canadian Hydrogen; Dr. D. Rozen, directeur du département de génie physique et titulaire de la chaire Hydro-Québec en génie nucléaire; and Michel Rheaume, HQ, who organized the meeting. (See "Branch News".)

Bruce Power New Fuel Project

Design and Implementation of Bruce CANFLEX-L VRF

by M. Liska and D. McArthur1

***Ed. Note:** The following paper was presented at the 8th International Conference on CANDU Fuel, held September 21-24, 2003, at Honey Harbour, Ontario.*

Abstract

Bruce Power has initiated a plan to refuel the four reactors at its Bruce B facility with low void reactivity fuel (LVRF), beginning in 2006. The LVRF will provide the necessary safety margins to allow the Bruce B reactors to operate at their design capacity to beyond 2015. The project, called "New Fuel Project" (NFP), includes the design of the fuel, out-reactor testing and a demonstration irradiation, and addresses system and process changes pertaining to the receipt, use, storage and management of fresh and used LVRF.

New Fuel Project Overview

Background

At the time Bruce Power entered a lease to operate and maintain the Bruce nuclear facility, the Bruce B units were limited to 90 per cent full reactor power (90%FP) owing to reactor physics issues pertaining to a postulated large loss of coolant accident (LLOCA). In addition, further deratings could be experienced in the future owing to reactor aging resulting in a reduction of margin to fuel dryout during a neutron overpower event. Bruce Power initiated the New Fuel Project to enhance existing safety margins, thereby providing a basis for restoring the reactor units to full output and sustaining that output to beyond 2015.

In addition, reactor operation is restricted to below full power owing to feedwater limitations and increasing reactor coolant inlet temperature owing to heat transport system aging. Additional projects are being implemented concurrent with the NFP to return the reactor units to full power. The suite of projects, including the NFP, is being managed under the Improve Output Program for the Bruce B units. This paper describes the NFP. The scope of the Project is depicted in Figure 1. The Project includes designing and implementing any changes required to support receiving and storing fresh LVRF, operating a reactor with LVRF, and wet storing used LVRF. The Project also includes assessing the impact of LVRF on the transfer of used fuel to dry storage.

Project Management

Bruce Power is managing the overall project while Atomic Energy of Canada Limited (AECL) is managing the design and qualification testing of the fuel. A consortium formed by Nuclear Safety Solutions (NSS) and AECL, under contract to Bruce Power, performs analysis of accidents, system performance, and the potential for out-reactor criticality of the fuel. Various other contracts will be in place for specialized services such as assessing impacts on the fuel handling system, developing an in-house criticality prevention program, designing and testing transportation containers, and enhancing the performance of a limited number of existing support systems where required.

The Bruce Power NFP is progressing concurrently with separate initiatives by Cameco and Zircatec Precision Industries (ZPI) to develop the necessary manufactur-

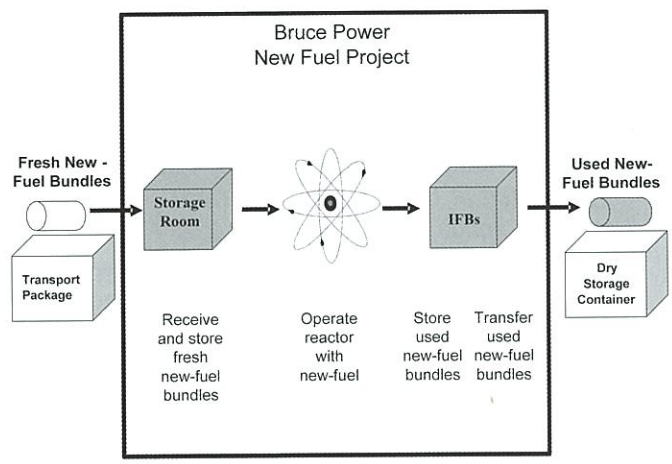


Figure 1: Overview of Bruce Power New Fuel Project
Note: new-fuel = Bruce CANFLEX-L VRF

1 Both authors are with Bruce Power, Kincardine, Ontario

Table 1: Comparison of Bruce CANFLEX-LVRF and Existing 37-Element fuel Bundles

	CANFLEX Bundle	Current Fuel Bundle
Number of elements per bundle	43 - Concentric rings of 21,14, & 7 elements around a central element	37 - Concentric rings of 18,12, & 6 elements around a central element
Diameter of fuel bundle	102.77 mm (maximum)	102.77 mm (maximum)
Length of fuel bundle	495.3 mm	495.3 mm and 508.0 mm
Weight of fuel bundle	23.1 kg	24.2 kg
Diameter of fuel elements	13.9 mm (maximum) for centre and inner elements. 11.9 mm (maximum) for intermediate and outer elements	13.10 mm (maximum)
Bearing pad height	>1.4 mm	1.2 mm
Location of bearing pads	inboard and outboard planes at both ends and at mid-length	inboard and outboard planes at both ends and at mid-length
CHF Enhancement buttons	Yes	No
Concentration of U-235	~ 1.0 %	0.71% (natural uranium)
Weight of uranium per bundle	18.5 kg U (21.03 kg UO_2)	19.2 kg U (21.8 kg UO_2)
Dysprosium in central element	Yes	No

ing industry infrastructure in Canada for CANFLEX and CANFLEX-LVRF designs. In anticipation of future markets for these fuel designs, Cameco is modifying their powder processing facility to accommodate fuel designs with enrichments somewhat higher than that required for the Bruce CANFLEX-LVRF design.

The Bruce Power NFP team consists of a project management team, project technical team, representatives from Cameco, ZPI, and AECL, and various support staff within Bruce Power. Functional roles, specific responsibilities, authorities and accountabilities for each team position are specified in detail in the Project Execution Plan. Representatives from Cameco, Zircatec, and AECL attend all Bruce Power project meetings to ensure there is effective communication at the scope and schedule interfaces. The project team consists of about 15 stakeholder representatives (i.e. representatives of those most affected by the project), most of whom coordinate various sub-teams having specific areas of expertise.

Design

The New Fuel Project will replace the current 37-element fuel bundle design (37NU), used in all four Bruce B CANDU® units, with a 43-element CANFLEX-LVRF® fuel bundle designed by Atomic Energy of Canada Limited (AECL). Each Bruce B unit is a Pressurized Heavy Water Reactor (PHWR) with a design electrical output of approximately 900 MWe at 100% FP. Each reactor vessel contains 480 horizontal fuel channels and each fuel channel contains thirteen 37-element fuel bundles stacked end-to-end and centred across the core. The 37-element fuel bundle design is a cylindrical array of 36 fuel elements in 3 rings surrounding a central element. All elements are of the same diameter and each element contains uranium oxide ceramic pellets formed from natural uranium. See also Figure 2 and Table 1.

The original design of the Bruce B units was for "fuelling against the flow" (FAF). The units are in the process of being converted (Core Conversion) to "fuelling with the flow" (FWF), which partially addresses the reactor physics issues associated with a postulated LLOCA. Currently, a mix of normal (495.3 mm) and long (508.0 mm) fuel bundles are used to reduce the power pulse associated with fuel string relocation during a postulated LLOCA using FAF. As a last step in the Core Conversion Project, the inlet fuel bundle will be removed such that each channel will have only 12 fuel bundles. The reduced fuel string length eliminates damage to the fuel channel at the inlet resulting from fuel bundle vibration. In addition, the reduced string length eliminates stresses on the fuel that could potentially arise during a LLOCA if the fuel string during thermal expansion became constrained by the fixed components at the end of the fuel channel.

Following Core Conversion, including the removal of the thirteenth bundle, the remaining 37-element fuel bundles will be replaced during normal FWF refuelling with the new 43-element Bruce CANFLEX-LVRF design. The length of the new bundles will be normal length only. The CANFLEX-LVRF design comprises 2 distinctly different design concepts: CANFLEX and LVRF.

Similar to the existing 37-element design, the 43-element CANFLEX fuel bundle is a cylindrical array of 42 fuel elements in 3 rings around a central element. The elements in the outer 2 rings, though, are smaller in diameter than the remaining elements. The element power ratings for a CANFLEX bundle are lower than those for a 37-element bundle for the same bundle power, making CANFLEX an ideal carrier for enriched fuel and for extended burnup. The CANFLEX fuel bundle was also designed to compensate for reductions in fuel dryout safety margins in aging reactors. The critical heat flux (CHF) enhancement "buttons" offset the reduction in thermal hydraulic safety margins caused by reactor aging, particularly fuel channel diametral creep. As a result of neutron absorption during normal reactor operation, the fuel channel diameter expands, allowing coolant to be redirected over the top of the bundle positioned eccentrically at the bottom of the fuel channel. This phenomenon is known as flow bypassing. The CHF enhancement buttons promote turbulent mixing of the coolant flow, thereby improving heat removal capability. Owing to its design features, the CANFLEX design has been selected as a carrier, or platform, for the LVRF concept.

The LVRF design includes a neutron absorber in the central element to reduce the positive void reactivity effect associated with a postulated LLOCA. During a LLOCA, the reduction in coolant pressure results in steaming and the formation of voids in the fuel channels. The presence of voids increases neutron reactivity until the shutdown systems terminate the rise in reactor power. During voiding, the neutron flux peaks at the centre of the bundle. Thus, the presence of an absorber in the centre of a bundle reduces the positive void reactivity effect. Since the neutron absorber is also present during normal reactor operation, all remaining fuel elements in the fuel bundle must contain slightly enriched uranium (SEU) to offset the reduction in bundle burnup that would result during normal operation if SEU were not used.

Dysprosium, the neutron absorber selected for use, is a non-radioactive rare-earth metal used in magnets, halogen lamps and in the electronics industry. A form of dysprosium oxide will be mixed with uranium oxide powder and formed into ceramic pellets for use in LVRF. The fuel composition (i.e. the amount of dysprosium and SEU used) is determined primarily by the magnitude of void reactivity reduction (VRR) and burnup required.

The concentration of Uranium-235 (U-235) used in Bruce B fuel will be increased slightly from the naturally occurring level 0.7 per cent U-235 to approximately 1.0 per cent. In comparison, reactors around the globe use fuel with a U-235 content anywhere from 3 to 5 per cent. The target burnup of the new fuel design is the same as that for existing 37-element fuel. Higher burnup fuel, which would require slightly higher enrichment to about 1.2 per cent U-235, may be considered in the future to reduce annual fuel costs.

Implementation

A Demonstration Irradiation (DI) of the Bruce CANFLEX-LVRF design is planned for the Fall of 2004 at the prevailing reactor power level. The DI is considered to be a confirmatory step in the design process given that:

- a DI for the CANFLEX natural uranium design has already been performed in a CANDU 6 reactor.
- some out-reactor testing and analysis has already been done for the CANFLEX-LVRF design
- only slight modifications and refinements (e.g. bearing pads, end caps, specific fuel composition) were made to the CANFLEX-LVRF design to develop the Bruce CANFLEX-LVRF design.
- all planned out-reactor testing and relevant analysis specific to the Bruce CANFLEX-LVRF, and a formal design review will have been completed by that time.

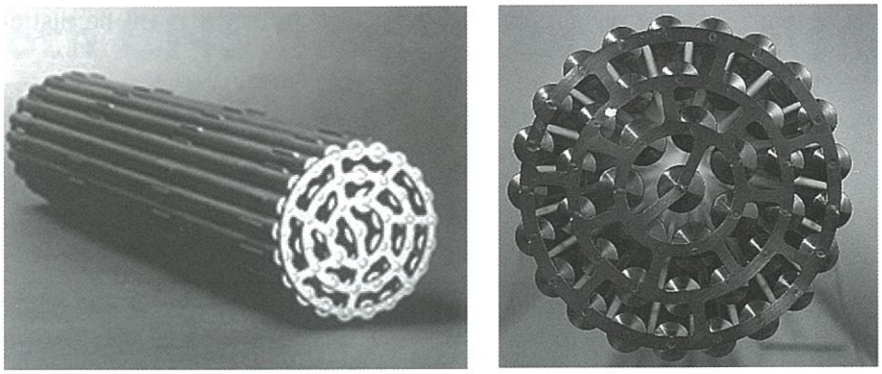
Twenty-four CANFLEX-LVRF bundles will be loaded into two relatively high power channels (twelve bundles in each channel) in one of the Bruce B reactor units. The bundles will be loaded in accordance with the normal refuelling schedule. At least one channel will be in the high power four-bundle shift refuelling region. Complete refuelling of the selected channel will take three refuellings of four bundles each over approximately a nine-month period. Discharge of the last bundles will occur after about twelve months. Some of the discharged bundles will undergo detailed post-irradiation examination. To minimize any differences or concerns between the DI bundles and production fuel, the DI bundles will be manufactured using equipment and conditions as near as practically achievable to the equipment and conditions that will be used for production fuel manufacturing.

Following receipt of a sufficient quantity of production fuel bundles, one of the Bruce B reactor units, the target unit, will begin normal refuelling with Bruce CANFLEX-LVRF bundles. Approximately four months later, the first channel will have to be refuelled. If after that refuelling there are no problems with the new fuel in the target unit, the remaining units will begin normal refuelling with Bruce CANFLEX-LVRF bundles. Approximately forty months are required to complete the replacement of all existing fuel bundles in each reactor unit. However, each unit is expected to be returned to full power after about twenty months and only 95 per cent of the existing bundles have been replaced. The last 5 per cent of bundles to be replaced have a negligible effect on restoring safety margins. The management of the transition core and obtaining regulatory approvals for subsequent raising of reactor power will be executed under a separate project.

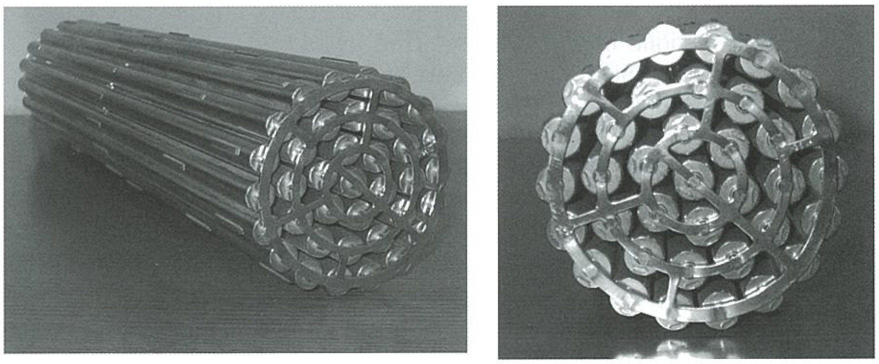
Considerations For Design And Implementation

Normal Operation with CANFLEX-LVRF

The new fuel design is similar to the existing fuel design in terms of dimensions, flow resistance, burnup, and overall



CANFLEX Fuel Bundle[2]



Bruce B Current-Fuel Bundle

Figure 2: CANFLEX and Current Fuel Bundles[3]

bundle power output. Thus, the impact on Heat Transport System (HTS) thermal hydraulic performance is expected to be small. There may be changes in the gamma and thermal neutron fluxes at the in-core detector locations, although the changes are expected to be small. Given the impacts on the HTS and flux detectors are expected to be small, the impact on Reactor Regulating System (RRS) operation is expected to be small. The localized increase in reactivity experienced during refuelling (i.e. refuelling ripple) is expected to be slightly higher with the LVRF, but no significant impact is expected for fuel management or safety analysis. The compatibility of dysprosium with materials in the HTS and other possible effects of dysprosium are being investigated under an out-reactor dysprosium behaviour program including testing and assessments.

Fuel Handling

Fuel handling within Bruce B includes receipt and storage of fresh LVRF bundles, reactor refuelling, wet storage of used LVRF bundles, and transfer of used bundles to Dry Storage Containers (DSCs). It is expected that the fuel handling process for LVRF bundles will be essentially identical to the existing process except for additional physical or procedural barriers implemented to prevent out-reactor criticality owing to the presence of SEU. International experience is being reviewed to determine the appropriate overall approach to managing the risk of criticality.

Fresh LVRF bundles will travel from the manufacturer to site in transportation containers specially designed to meet transportation requirements of the Canadian Nuclear Safety Commission (CNSC) Packaging and Transportation of Nuclear Substances Regulations (which adopt and reference International Atomic Energy Agency (IAEA) regulations) for fuel assemblies containing enriched uranium. Each transportation container is expected to consist of inner packaging holding the bundles and robust outer packaging (overpacks) designed to withstand normal conditions of transport and accidents. The types of tests include resisting water ingress, fire, and being dropped. The number of bundles transported per shipment will decrease and the number of shipments increase as a result of increased volume of packaging required to transport the bundles.

While the fuel handling process has not yet been finalized, it is expected that the process will be essentially identical to current practice. The transportation containers will be removed from the truck, overpacks will be removed and returned to the fuel manufacturer, and the inner containers stored on existing storage racks in the New Fuel Storage Room. Preliminary assessment indicates adequate margin to criticality exists for all credible fuel handling scenarios in the New Fuel Storage Area. Criticality analyses for severe conditions such as fire are being done to determine the likelihood of criticality and what mitigating provisions may be required (e.g. improvements in fire detection, or barriers, or protection systems, or additional storage space to increase separation).

When required for reactor refuelling, pallets of LVRF bundles will be moved to the New Fuel Loading Transfer Room where the bundles will be loaded into the New Fuel Transfer Mechanisms, which load the Fuelling Machines. No change in the existing process is expected except for the possibility of limiting the number of bundles in the New Fuel Storage Room to prevent criticality. Analysis shows that there is no possibility of criticality occurring in either the New Fuel Transfer Mechanisms or Fuelling Machines.

As per existing practice, used LVRF bundles will be discharged and stored in light water in the Primary Irradiated Fuel Bay (PIFB) prior to being relocated to the Secondary IFB to await transfer to the DSCs. Preliminary assessments indicate there is adequate margin to criticality during normal fuel handling and storage of used LVRF bundles in the IFBs. The potential for criticality as a result of an unplanned fuel handling event is being investigated.

2 Not actual bearing pad configuration for Bruce B.

3 See Table 1 for dimensions of fuel bundles.

No change to the dry fuel storage process is expected for transferring discharged LVRF bundles to DSCs. Similar to current practice, used LVRF bundles will be transferred to dry storage after typically 10 years of wet storage. Changes to the design of the DSCs are not expected. Modifications to DSCs, if required, would be pursued under a separate project.

Radiation Protection

Radiation protection for handling existing fresh 37NU fuel includes regular work coveralls, cotton gloves, and personal dosimetry. No additional radiation protection equipment is required for handling fresh LVRF bundles. Although the dose rate increases with use of SEU, the total dose associated with handling LVRF bundles remains an insignificant contribution to overall station dose. Few changes in radiation protection procedures are expected.

Failed Fuel Detection and Location

The Bruce B reactor units employ a Gaseous Fission Product (GFP) monitoring system to detect the presence of a defective element and a Delayed Neutron (DN) monitoring system to locate the specific fuel channel containing the fuel defect. At this time, the design concepts for the existing systems appear to be adequate for detecting and locating a defective LVRF bundle, although further assessment is continuing.

Accident Analysis

The CANFLEX-LVRF design is being implemented to substantially improve safety margins. As mentioned above, the LVRF feature specifically addresses the reactor physics issues associated with a postulated LLOCA. The CANFLEX feature improves safety margins associated with fuel cooling as a result of improved critical heat flux. Hence, Neutron Overpower Trip setpoints for the shutdown systems and maximum reactor power allowable by RRS will be increased. The Bruce CANFLEX-LVRF design is similar to the existing 37NU design in many respects. Hence, although all accident analyses are being reviewed for impacts, no significant adverse impact is expected.

Quality Assurance and Quality Control

The New Fuel Project will be carried out following Bruce Power internal procedures and practices, which comply with the Canadian Standards Association N286 standards. AECL will conduct its own internal design review of the Bruce CANFLEX-LVRF design. Bruce Power will conduct a design review of all aspects pertaining to fuel design, qualification, and system and procedure modifications prior to performing the Demonstration Irradiation. In addition, the project will conduct self-assessments and be subjected to a Bruce Power internal audit by the Audit, Inspection, and Investigation Department.

As with existing fuel, the Bruce CANFLEX-LVRF bundles will be manufactured to CSA Z299.1 standards and fuel powders will be processed to CSA Z299.2 standards. Dysprosium and SEU pellets and elements will be distinguishable to prevent mix-up.

Environmental

A Project Description has been formally submitted to the CNSC in accordance with the requirements specified under the Canadian Environmental Assessment Act (CEAA). Implementation of the CANFLEX-LVRF design does not require any new construction of a structure on site, nor modification to any existing major structure (e.g. containment) nor modification to a major system (e.g. RRS, HTS, or moderator system). As such, only minor modifications to support systems or components (e.g. changes to fuel trays or stacks in irradiated fuel bays) may be required, none of which is expected to introduce a new pathway to the environment. There are no new or increased gaseous or liquid releases anticipated from the use of CANFLEX-LVRF. Dysprosium and its irradiation products will be minor additions, both radiologically and in terms of conventional toxicity. Consequently, environmental impacts are considered to be insignificant. Notwithstanding the expectation that environmental impacts will be negligible, the assessment of environmental impacts is an integral part of the Engineering Change Control process. Bruce Power is qualified to ISO 14001 international standards.

Regulatory

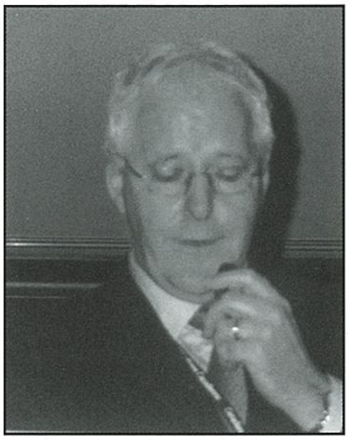
In accordance with the Reactor Operating License for Bruce B, Bruce Power will request the CNSC to approve the fuel design and, where required, to approve system or component modifications prior to the Demonstration Irradiation and production loading. Bruce Power has prepared and presented to the CNSC the outline of a formal Safety Case and is providing the CNSC on a regular basis updates on the development of the Safety Case.

The Safety Case is intended to include mutually agreed upon acceptance criteria, scope of reviews, and documents and analyses to be submitted. Although the action of raising reactor power and the approvals necessary to do so are not part of the New Fuel Project (they will be handled separately under another project for business reasons), such approvals and acceptance criteria are being discussed with the CNSC as an integral part of the Safety Case.

In Canada, the CNSC represents the International Atomic Energy Agency's (IAEA) interest in safeguarding new and used nuclear fuel. Information received by the IAEA to date confirms that IAEA operations and the use of surveillance monitoring equipment will remain the same as is currently used.

Postscript: In a letter to the Canadian Nuclear Safety Commission, November 2003, Bruce Power informed that it intends to conduct a demonstration irradiation of the new fuel in two channels in 2004 with production loading to begin in 2006. Ed.

6th CNS International Conference on CANDU Maintenance



Martin Reid

Phillip Klintworth

Duncan Hawthorne's words, "Good maintenance is essential to keep our existing plants running", set the tone of the **6th CNS International Conference on CANDU Maintenance,** held in Toronto, November 16- 18, 2003. About 270 people wound their way to the Holiday Inn on King for the conference whose theme was *Maintenance for Life.*

Similar to the previous conferences on this topic the technical sessions were augmented by a set of interesting and informative exhibits by companies engaged in the maintenance of CANDU plants. *(See list below.)*

The conference was concentrated into two days. Following the half morning opening plenary session the balance of the meeting was structured with four technical sessions running in parallel. Nevertheless, the organizers allowed ample break periods when delegates gathered in the exhibit area to share experiences while enjoying the ample refreshments.

A reception in the exhibit area on the Sunday evening provided an opportunity for delegates to meet each other and observe some of the latest tools to aid in maintenance and inspection.

Following the official opening of the conference by **Martin Reid** on the Monday morning, **Duncan Hawthorne,** president and CEO of Bruce Power, gave a short talk on his perspective of the electricity situation in Ontario and the role of nuclear. More generation is needed, he said, and refurbished nuclear is the cheapest route. The price cap on electricity prices that the former Ontario government had established needs to be reviewed, he contended. Referring to the "blackout" in August, he commented, "Good maintenance could have prevented it". All our plants are at mid-life, he observed, "like most of us", and need more check-ups. "If the plants could speak they would ask for a day off", he quipped. The decision to build new nuclear plants should be a clear one, he stated, but there are perception problems However, even if a decision were made now to build new plants they would not be available until 2011, he observed. Until then we must keep the current plants running. Referring to a "distorted" article in a major Toronto newspaper the previous Saturday, he called on delegates to be ambassadors and tell the good story of nuclear.

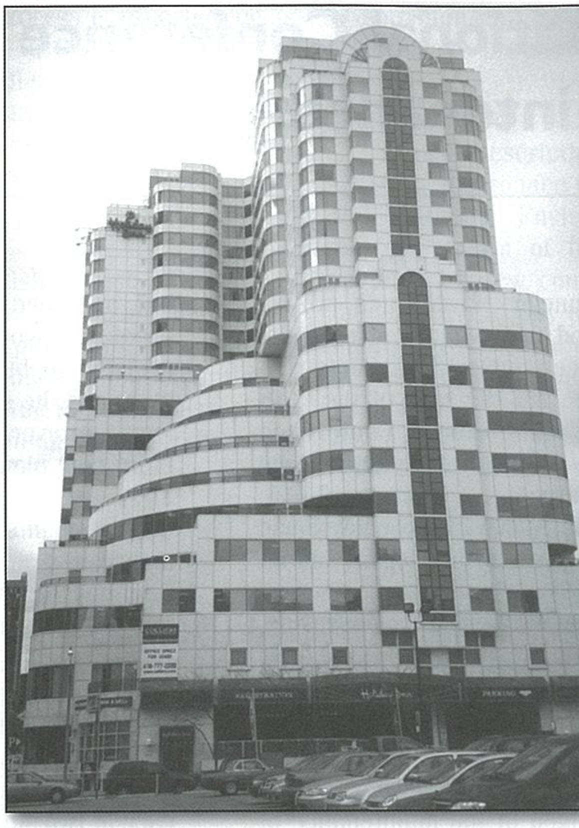
The first plenary speaker was **Jarmo Tanhua,** Manager of Electrical and Automation Maintenance with the TVO utility in Finland, who began by providing an overview of the nuclear power situation in his country. Finland has four nuclear power units, two each at two sites with separate owners. A new nuclear plant has been approved by the Finnish parliament and a waste disposal facility is being constructed. TVO is a non-profit company owned by a number of power-consuming companies to which it sells electricity at cost. TVO's operating policy is, he said, "disturbance free operation", especially when the price is high. They have had a capacity factor of about 96% for years, he noted. Emphasis is put on predictive, efficient, advanced maintenance, he stated. The practice is to exchange full packages, he commented, not to make repairs. Their units have been uprated over the years, and they now propose to increase the capacity of the original 660 MWe plants to 840 MWe. Despite this focus on high capacity factor he emphasized that a high safety culture was very important.

David Scott, General Manager, Commercial products and Field Services at AECL, spoke on CANDU Maintenance Challenges, noting specifically: pressure tubes, steam generators and feeders. Showing a photograph of the end face of a CANDU unit he commented that each feeder is unique and access for repair is difficult. He noted that feeders were subject to degradation by erosion, corrosion and cracking. There is a need for the development of specialized tools, he stated, and showed samples of advanced simulations that enabled better planning of maintenance. Good site organization is essential, he emphasized.

With the third proposed plenary speaker, from Argentina, unable to attend, **Duncan Hawthorne** stepped in to provide some updates on steam generator problems at Bruce B unit 8 and plans for Bruce A units 1 and 2.

Erosion of tubes in a steam generator at Bruce 8 appeared to be caused by flow, Hawthorne commented. This has led to a fitness for service examination of all units. It shows the importance of life-cycle management, Hawthorne stated, and of changing data into information.

Bruce Power is studying the possible rehabilitation of units 1 and 2 of Bruce A, Hawthorne said, and the company is willing to spend up to $500 million next year. A factor in the decision will be what credit will be given for replacing coal generation. That credit will also affect a decision on new nuclear, although he noted the continuing political and investment uncertainty. We must produce well, he said, to convince investors.



Venue of 6th CANDU Maintenance Conference.

Following the mid-morning break, the meeting split into four parallel sessions, a pattern that continued for the remainder of the conference. Papers were grouped under the following headings:

- Fuel Channels & End Fittings - Assessments
- Fuel Channels & End fittings - Inspections
- Fuel Channels & End Fittings - Maintenance
- Fuel channels & End fittings - Universal Delivery Machine
- Water Upgrading
- Performance & Plant Life Improvement
- Steam Generator Life Management
- Steam Generator Modifications
- Steam Generators - Inspections
- Steam Generators - Assessments
- Maintenance Programs
- Feeder Inspections
- Feeder Assessment & Mitigation
- Valve Maintenance
- Instrumentation and Control
- Inspection Technology
- Fuel Handling

An unscheduled talk on the response of Ontario Power Generation to the "blackout" of August 14 was given by **Shane Ryder**, Assistant Operations Manager, Pickering B, during the first period on the Tuesday morning. He began by identifying the status of the Darlington and Pickering units prior to the shock to the system. All four Darlington units and Pickering 5,6, and 8, were operating at full power. Pickering 7 was in a planned maintenance outage. Pickering 4 was at 12% F.P. and in a turbine run up.

Following the system event Darlington 1, 2, and 4, initially step-backed to 60% and then had a setback to 50% on high flux tilt. The breakers opened, the reactors were run down to 5% and the turbines tripped. Shutdown System 1 tripped and there was a loss of class 4 power. Reactor power at Darlington 3 was decreased to 13% then raised to 59% to prevent poison out. The generator was loaded to 52 MWe to supply unit loads and the unit was re-synchronized to the grid later that day.

At Pickering the three operating units experienced a generator trip followed by a reactor setback and actuation of Shutdown System 1 followed by firing of Shutdown System 2.

Ryder outlined the many challenges and issues facing the operators, including difficulty communicating with the Independent Market Operator. There were a number of technical problems such as the primary system being in a non-standard state and control room panels for the emergency injection system not functioning because of power fluctuations. At Pickering operators faced an extended outage of class 4 power. He emphasized that all of the operators at both plants responded professionally, following procedures. Conservative decisions were made with safety having the highest priority.

Darlington unit 2 was reconnected to the grid on August 17 and units 1 and 4 on August 18. Pickering 4 was reconnected on August 21, unit 8 on August 22, unit 5 August 23, unit 6 August 25 and unit 7 following its planned outage) August 29.

There were speakers at both of the luncheons. On the Monday it was **Mike Lees,** Director of Business Development and Marketing, for Babcock and Wilcox Canada, who spoke on the market for nuclear services. In the USA, he noted, there is a much smaller number of owners than a few years ago, while in Canada we have a new one [Bruce Power]. There is a wide spread concern about financial risk. This has led to a move towards partnerships between suppliers and plant owners.

At the second luncheon, **Ken Hedges,** vice-president, ACR

Business Unit, at AECL, reviewed the potential market for new nuclear. There is a growing demand for electricity and a need for security of supply, he observed. Ontario alone will need an additional 10,000 MWe in 20 years, There is also a need for diversification of supply and protection of the environment. The goal for ACR (Advanced CANDU Reactor) is to optimize maintenance and operability, with a target for a plant to be in-service in Canada in 2011. Qinshan showed, he said, that plants can be built on budget and on schedule. Targets for ACR are: 93% C.F.; forced outage 1 to 2 %; planned outage period 3 years; outage duration 20 -25 days. A 60 year life is proposed, which will require life management from the beginning.

A lighter but still relevant topic was the subject of the dinner speaker, **Phillip Klintworth,** a retired US nuclear submarine commander. Klintworth was commander of the submarine base in San Diego when the US Navy permitted it to be used as a set for the movie *The Hunt for Red October.* He titled his talk "Salt Water and Movie Studios" or "Unlikely Adventures of a Nuclear Submariner". With a series of slides he provided an amusing tale of the making of the 1989 movie. One scene of a submarine doing an emergency surfacing (with it emerging at an apparent 30°) took three tries, he said, because of the difficulty of determining exactly where it would surface. He rounded out his talk with comments on an exercise in the Indian Ocean, the new method of launching nuclear submarines and several humorous anecdotes..

There were 14 exhibitors: Nova Machine Products; Weed Instruments; GE Canada; RCM Technologies; Farris Engineering; Summit Controls; AECL; R/D Tech Probes; Kinetrics Inc.; Intech International; Babcock & Wilcox Canada; Justram Equipment Inc.; Schultz electric Company; Entertech.

The conference was organized by a large committee chaired by Martin Reid of Ontario Power Generation. Marc Paiment, Bob Tapping and Greg Shikaze looked after the technical program, while Mike Schneider, Ima Kalos and Amira Nour coordinated the exhibits. Other involved were: Ken Belfall; Ben Rouben; Steve Solomon; Denise Rouben; Micheline Verney; Kathy Davies; Tim McLaughlin; Bob Gunn; Marc Leger; Ed Price; Jerry Cuttler; Carl Daniel; Eric Williams; Narinder Bains; Aman Usmani; Heather Smith; Malcolm Lightfoot; John Slade.

A book of abstracts was available for the conference and the proceedings will be published on a CD, which will be available from the CNS office.

Universal Delivery Machine
– Design of the Bruce and Darlington Heads

by Michal G. Gray[1] and Roy Brown[2]

***Ed. Note:** The following paper was presented at the 6th International Conference on CANDU Maintenance held in Toronto, Ontario, November 16 - 18, 2003 under the title* Design of the Bruce and Darlington Universal Delivery Machine Heads.

Abstract

The Universal Delivery Machine (UDM) was designed and supplied to reduce the time required to perform channel inspection services. The Bruce UDM was the first to be completed followed by Pickering and Darlington. The Bruce and Darlington machines are nearly identical. Design concepts applied include a rotating, multiple tool station magazine, a rigid chain driving telescoping rams, a common drive package, and an external support frame to meet seismic qualification requirements.

Introduction

Following an initial concept study, GE Canada was contracted by Ontario Power Generation's Fuel Channel Inspection and Maintenance Department to design and manufacture the Universal Delivery Machines (UDM) to significantly reduce the time required to deliver various channel inspection and maintenance services. The first machine was completed for the Bruce B station, followed by one for Pickering B and one for Darlington. The Darlington design was to be identical to Bruce except for specific, negotiated differences made necessary by station or channel configuration, or requests by the customer. Design and supply included both mechanical and control equipment and software. This paper deals with aspects of the mechanical design of the UDM head.

General Arrangement

The Bruce and Darlington Universal Delivery Machines were designed to be hung on the reactor area bridges and operated on the reactor remotely, similar to the fuelling machines. Attachment to the bridge carriage is the same as the fuelling machines at each respective station. Drives in the UDM suspension and sensors in the snout locking mechanism allow for fine homing prior to lockup on an endfitting.

Experience with previous inspection machines such as CIGAR and SLAR indicated that the UDM head should have the capability to work independently of the fuelling machine. It would need to be able to home and lock to the target channels and remove, store and replace the closures and shield plugs. The Bruce UDM would need the ability to remove fuel from the channel by pushing it into a fuelling machine on the opposite end. Pushing fuel was not an initial requirement for the Darlington UDM head but the ability to add this feature later has been incorporated in the design.

Some of the concepts for UDM grew out of the previous experience with the Bruce A MiniSLAR machine, where a snout locking mechanism was mounted onto a fixed plate while a closure mechanism, shield plug mechanism and SLAR Tool Mechanism were mounted onto a sliding plate. UDM also attaches a snout locking mechanism onto a fixed plate, but in order to accommodate up to three tool mechanisms and improve sealing, the moveable plate is round and rotates about the center of the fixed plate. The individual mechanisms are mounted on a center support tube and operate when magazine rotation brings them to the top site in line with the snout.

Magazine Mechanism

One of the first decisions required centered on the number of magazine stations. As it was desirable to carry a spare closure, the concept of storing the spare in a location within the fixed plate of the magazine eliminated the need to have a second closure mechanism mounted on the rotating portion. This left room to have up to three stations available for the channel inspection and maintenance tools. The Bruce UDM head magazine has a closure mechanism, a shield plug mech-

1 GE Canada Nuclear Products, Peterborough, Ontario.

2 OPG Inspection Services Division and Bruce Nucelear Power Development

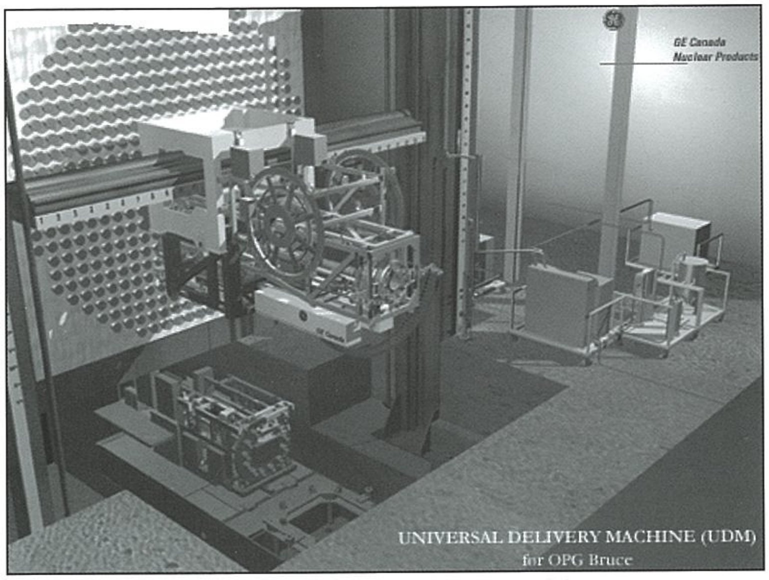


Figure 1: Artist's rendering of UDS System at reactor face

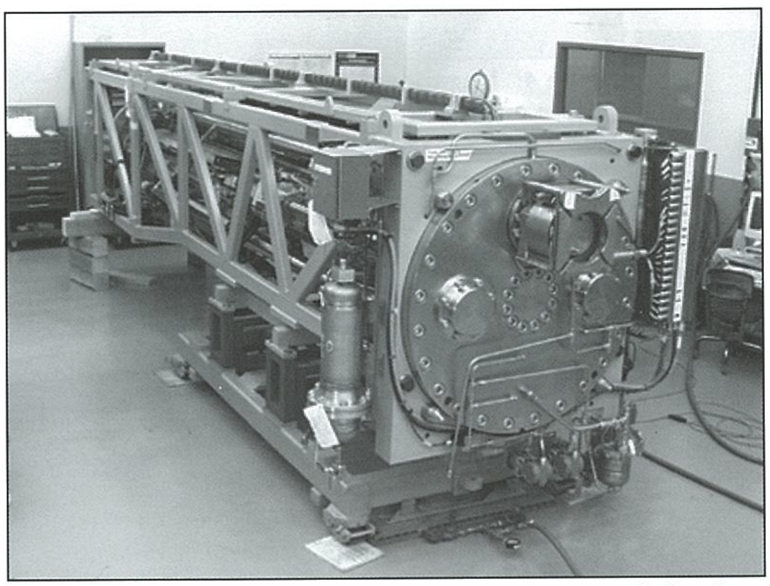


Figure 2: Bruce UDM Head Assembly nearing completion in Clean Room

anism and three tool mechanisms, one of which is used for the fuel push. The Darlington UDM has a closure mechanism and a shield plug mechanism but only two tool mechanisms, and a space where the fuel push mechanism can be added later.

The main magazine components consist of a turret housing, a fixed plate, a rotating plate and the turret tube. The turret housing provides structural support for the front portions of the magazine and attaches directly to the external support frame, which mounts the head into the suspension. The turret housing holds a large diameter, thin section ball bearing, which allows the rotating plate to rotate freely.

The fixed plate is bolted directly onto the turret housing and provides the front portion of the pressure retaining boundary. Penetrations are provided for mounting of the snout, where components pass through, and for fittings for head filling and draining. Two closure caps are mounted onto the fixed plate, either side of center. One cap is available to store the active closure while the other one normally houses a spare. A center retaining plate is bolted to the inside of the fixed plate and helps retain the rotating plate through a large capacity thrust ball bearing.

As its name suggests, the rotating plate rotates with respect to the fixed plate and turret housing on the two ball bearings. The rotating plate forms the back side of the magazine pressure retaining boundary, and is sealed to the fixed plate by two pairs of large diameter lip seals. The seals ride on two spigots extending from the fixed plate. A wiper ring rubbing on the end of each of these spigot extensions helps prevent particulate from contacting the seals. An interspace bleed off passage between the individual seals of each pair allows the seal leakage to be monitored when the head is pressurized.

A large spur gear is attached to the back side of the rotating plate. The magazine drive package mounts to the back of the turret housing and provides the motive power to drive the gear. 292 degrees of motion are provided to the rotating plate to ensure that any of the five magazine sites can be aligned to the snout.

The turret tube extends rearward from the rotating plate and provides support to each of the five mechanisms. The shield plug mechanism and each of the three tool mechanisms are mounted on ball bearing linear slides. The rail for each of the linear slides is aligned and firmly bolted onto the turret tube. As the turret tube is made from a large, hollow pipe, a stack of lead counterweight discs was conveniently located inside the tube at the front end. The back flange of the turret tube rides inside segmented bushings mounted in the back end of the external head support frame. This allows the turret tube to rotate freely with the rotating plate to which it is fastened.

Snout Assembly

The snout assembly consists of the snout, clamping jaw sub-assembly, jaw drive, guard plate and the mounting collar along with the various sensors for homing and jaw position indication. In concept the design is similar to MiniSLAR, but has been considerably strengthened to meet the pressure and seismic requirements of UDM.

The snout forms the forward most portion of the UDM head pressure boundary. It mounts onto the fixed plate around the penetration provided for it at the top centerline of the head and is sealed to the fixed plate with a single o-ring. The bore of the snout is large enough to allow passage of the closure and is hard chrome plated to prevent scoring or other damage. Proximity sensors are mounted in the front face of the snout to measure head alignment during the homing process.

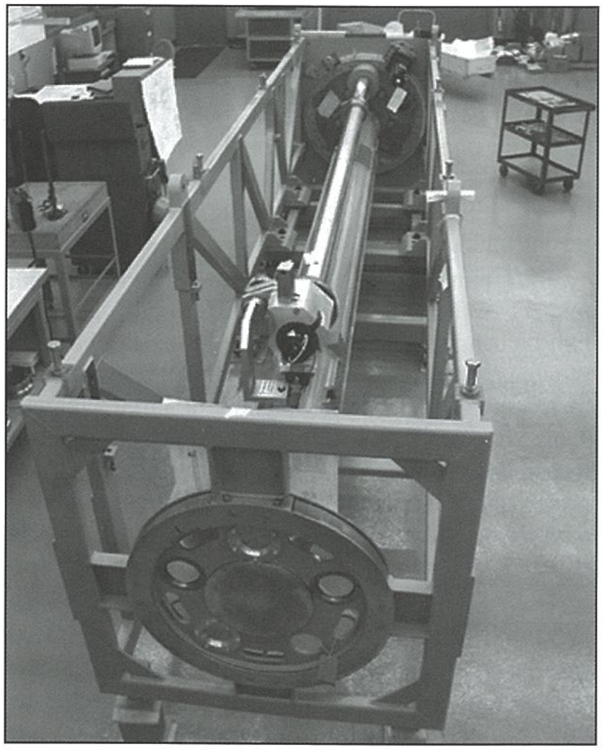


Figure 3: Magazine and Turret Tube mounted in Head Support Frame

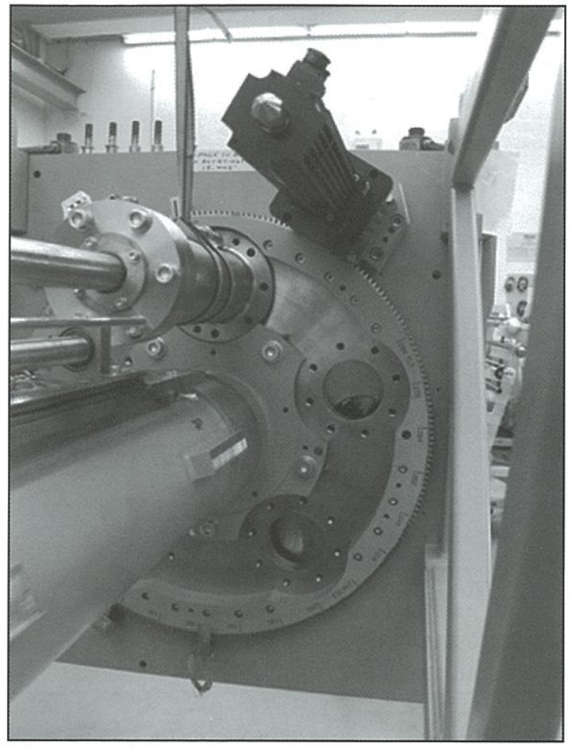


Figure 4: Magazine and drive assembly with closure mechanism mounted.

The locking jaws are a robust pair of clamp halves attached at the front end of the snout. Both halves are supported by a pivot pin mounted directly under the snout. When open, there is sufficient clearance to allow passage over the flanged hub of the endfitting. When closed, the jaws clamp simultaneously against the back side of the flanged hubs on both the snout and the endfitting. A Grayloc style metal seal ring, with an elastomeric o-ring on each side, is squeezed between the snout and the endfitting to provide the seal. This seal ring is identical to MiniSLAR.

The locking jaws are driven by a screw having both a left and a right hand threaded portion. Each portion mates with a matching nut pinned between the lugs extending up from its respective jaw half. The concept is similar to the fuelling machine except that only two jaw segments are used, as was the case on MiniSLAR.

The guard plate is mounted at the front of the UDM head. It has a hole through it large enough to pass the endfitting hub, and has four spring loaded pads equipped with sensors in the event that it contacts an endfitting during homing. If coarse alignment is off, the endfitting will cause a pad to depress, triggering a limit switch to stop the forward motion of the head. The control system then makes the necessary position corrections and the homing sequence continues.

Closure Mechanism

Of the five mechanisms mounted on the rotating magazine, the closure mechanism is unique in that it has its own dedicated drives. This was made necessary by the fact that the closure mechanism needed to be able to drop off or pick up a closure at one of the closure storage locations on the fixed plate, that is, it had to operate away from the top magazine site. The closure mechanism is also unique in that it is completely mounted on a separate baseplate, which in turn fastens onto the magazine turret tube.

The business end of the closure mechanism ram is identical to the equivalent fuelling machine part, and the sequence of operations used to remove or replace a closure has also been kept identical. Unlike MiniSLAR, which had a hydraulically driven closure ram, the UDM closure mechanism is driven by electric motors, as are all the other UDM drives.

The closure mechanism ram head is housed within a hollow support sleeve attached to the back of the magazine rotating plate at the penetration provided for it. Rotary motion is provided by a shaft, which penetrates lip seals in the pressure boundary, and couples to the output shaft of the rotary drive gearhead. The rotary drive is mounted to a housing, which, in turn, is mounted on ball bearing linear slides attached to the baseplate. Axial motion is provided to the housing by the action of a ballscrew and nut driven by the axial drive motor and gearhead.

Shield Plug Mechanism

The shield plug mechanism design used on MiniSLAR was often problematic. The fundamental problem was the lack of rotary stiffness in the long mechanism. The drive and position feedback encoder were located at the back end of the machine, whereas the ram head had to interface with

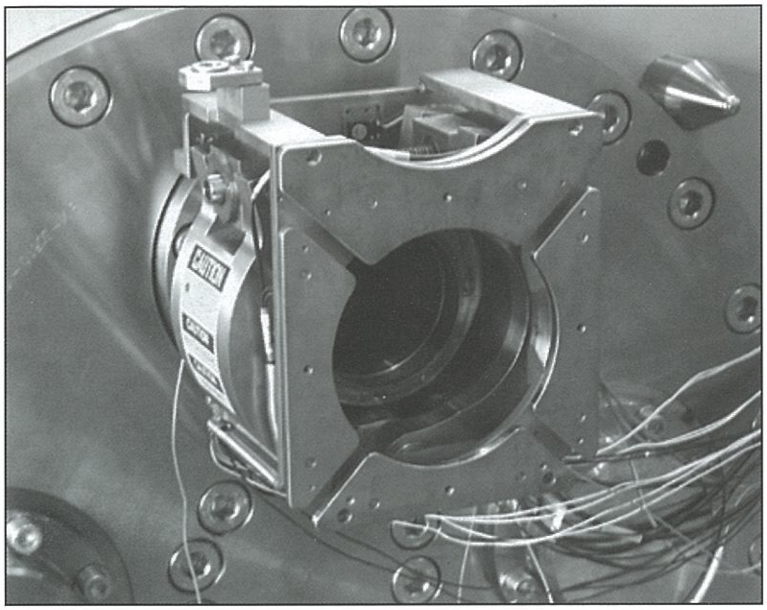


Figure 5: Snout assembly prior to drive motor installation.

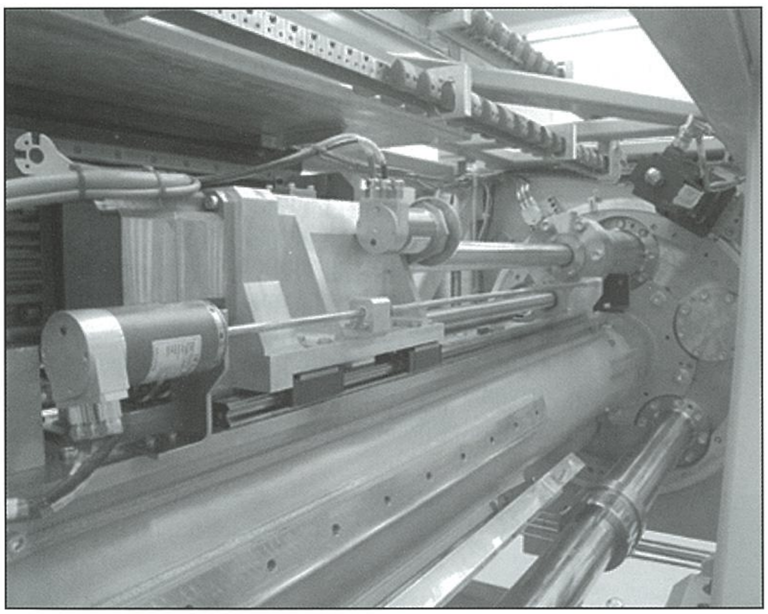


Figure 6: Closure mechanism mounted on turret tube.

the shield plug inside the endfitting. All too often the shield plug removal sequence was completed, apparently successfully, only to discover that the shield plug was still locked in the endfitting.

The problem was addressed on UDM by designing the shield plug mechanism ram as a long, thick walled and robust liner tube. The front end is fitted with fingers used to rotate and grip the shield plug. The back end of the liner is supported in ball bearings in a housing, which, in turn is mounted on a linear bearing. The single rail of the linear bearing is mounted on the magazine turret tube. Axial motion is provided by the UDM common drive assembly carriage, which pushes and pulls on the liner support housing.

The liner penetrates through lip seals in a seal carrier at the rear of the support sleeve, which is part of the head pressure boundary. The support sleeve is long enough to completely contain an inlet shield plug when the shield plug mechanism liner is fully retracted after shield plug removal. A spur gear is keyed to the liner near the back end to provide the rotary motion necessary for shield plug locking and unlocking. The meshing pinion gear is mounted on the front of a short shaft passing through the liner support housing. On the back end of this shaft is mounted a rotary coupling half. The mating half coupling is provided by the common drive assembly and connected directly to the output shaft of the common rotary drive gearhead.

A long shaft passes through the center of the liner and connects a linear actuator at the back end to a shield plug locking lug in the liner head at the front end. Following the CCW rotation of the liner assembly to unlock a shield plug and get the head fingers fully hooked inside the shield plug, the locking lug is driven forward to prevent any relative rotation which may cause unhooking of the fingers. The lug remains in this extended position until the shield plug is once again returned to the channel.

Tool Mechanism

The UDM tool mechanism design follows the concept used on MiniSLAR. It consists of a long, thick walled liner tube, which contains a set of nested, telescopic ram tubes. UDM uses four ram tubes whereas MiniSLAR had three. Each ram tube has three full length keyways on its outside diameter, which engage with mating keys on the ram guide at the front end of the next larger ram. The keyways on Ram #4, the largest ram, engage with integral keys on an insert fastened deep inside the bore of the liner tube. The insert also acts as the forward end of travel stop for Ram #4. Since the keys and keyways remain engaged at all times, rotary motion applied to the liner tube is passed through to the smallest ram, Ram #1. This ram has the channel inspection tool attached at its front end while its back end is driven by the ram drive chain. Each ram, in turn, is driven forward when the smaller ram inside it contacts its ram guide at the front. The inspection tool umbilical cable passes through the center of Ram #1 and exits at the back through the special ball and socket arrangement used to connect the drive chain to the ram.

Each ram has a pair of elastomeric piston seals at its back end, which form a seal with the bore of the next larger ram. The travel of each ram is limited such that its piston seals always remain within the bore of the next larger ram. The seals on Ram #4 travel inside, and seal to, the smooth bore inside the back part of the liner tube.

The liner tube carries the inspection tool and nested ram assembly into the endfitting and delivers it up to the channel latch or spacer, ready to enter the pressure tube. The liner penetrates through a pair of lip seals inside a seal carrier at the back of a cylindrical support sleeve. The support sleeve is bolted onto the magazine rotating plate at one of the penetrations provided for tool mechanisms. As mentioned earlier, the Bruce UDM head has three tool mechanisms, whereas Darlington only has two and a space where a third can be added. To cap the hole left in the Darlington magazine rotating plate, a special flanged, pressure boundary plug was designed and fitted.

A thin, cylindrical nosepiece is attached to the front end of the liner tube. The nosepiece supports the liner as it travels in and out of the endfitting. The forward end is reduced in diameter to pass by the endfitting liner locking lug and enter the channel latch, where it has the ability to open the latch fingers and hold them open to allow free passage of inspection tools into the channel. The nosepiece also contains a special calibration sleeve, which is made from a short piece of pressure tube. The calibration sleeve is sealed inside the nosepiece such that it is surrounded by a thin, annular air gap. The sleeve contains multiple man made defects, or artifacts, used in the calibration of the various sensors on the inspection tools.

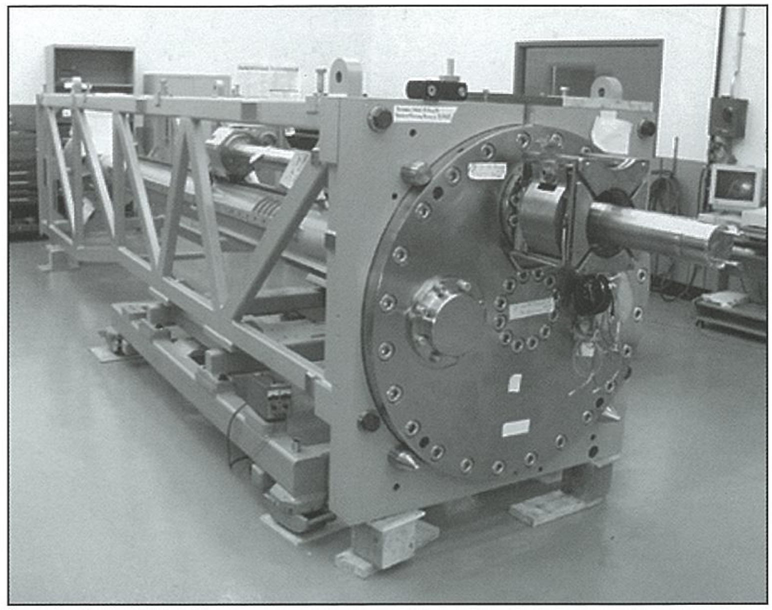


Figure 7: Shield plug mechanism liner head protruding from snout.

The back end of the liner tube is similar to that of the shield plug mechanism. It is supported in ball bearings inside a housing, which in turn, is mounted on a ball bearing linear way. A spur gear keyed to the liner passes rotary motion coming from the common rotary drive. Axial motion is provided by the common axial drive through its carriage, which can push or pull on the tool mechanism liner support housing.

Support Frame

The head support frame is constructed of hollow structural steel. Its main function is to provide rigidity to the UDM head in order to maintain mechanism alignment and meet seismic requirements. The frame has mounting pads on the bottom where the head is bolted to the suspension Y-frame. The support frame bolts to the turret housing at the front and holds the segmented bushings at the back where the turret tube flange rotates.

The support frame top frame provides a convenient place to mount the storage track for holding the chain drive rigid chain. At the back end, the support frame holds cable guides used to bend the inspection tool cables in a controlled manner and guide them towards their respective cable accumulator. The support frame also provides mounting points for the common drive assembly guide track.

Common Drive Assembly

Early in the UDM concept definition phase, it was decided that a common approach to providing motions to the various mechanisms would make the most sense. There were to be three tool mechanisms plus one shield plug mechanism, all requiring both axial and rotary motion. A common drive approach meant that all required liner motions could be provided by a single axial drive motor and a single rotary drive motor.

The concept followed was to equip UDM with a drive carriage, which could travel back and forth in line with the top magazine site and provide axial motion through its contact with the individual liner support housings. The carriage is supported from two ball bearing linear ways on a guide track mounted on the inside of the head support frame. The carriage is driven by a ballscrew, which is also mounted on the guide track, and rotated by the common axial drive motor and gearhead.

The common drive carriage provides forward axial motion to a mechanism via direct contact with its liner support housing. In order to retract a mechanism, a liner retract finger assembly is provided on the carriage. The finger is driven by an actuator down into a groove on the mechanism's support housing such that when the carriage is driven back, it pulls the housing via the retract finger.

A torque limiting coupling is installed between the axial drive motor gearhead and the input end of the common axial ballscrew. The torque limiter protects both the UDM equipment and channel components from overload in the event of an impact during common axial motion. It can also limit the maximum force applied to the fuel string during a fuel push operation using a tool mechanism.

The common rotary drive is also mounted on the common drive carriage. It consists of a drive motor and gearhead with an Oldham style half coupling mounted directly on the gearhead output shaft. When the carriage is in contact with a mechanism, this half coupling mates with a similar half

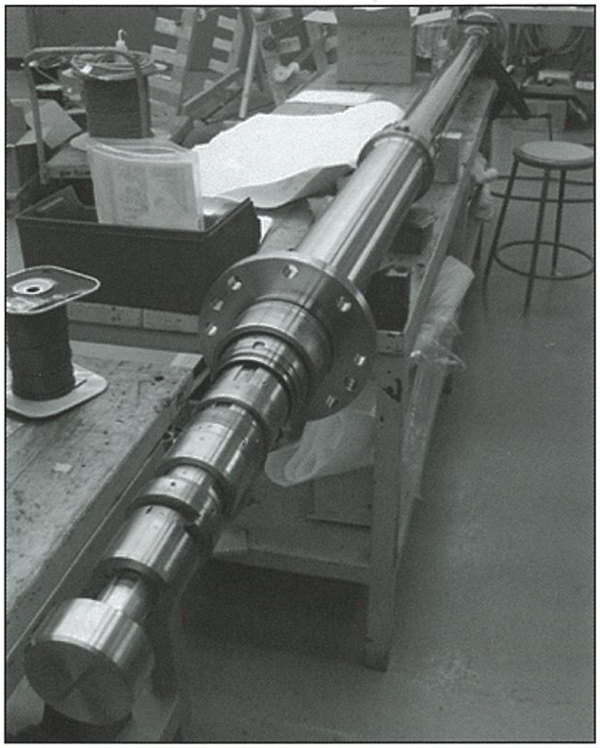


Figure 8: Tool mechanism liner, support sleeve and ram assembly.

coupling on the mechanism liner support housing. Driving the motor then rotates the spur gear mounted near the back of the mechanism liner tube thereby causing the liner to rotate. The common rotary drive is capable of providing continuous rotation of a liner, however, for a tool mechanism the rotary stroke available is limited by control software to prevent excessive twist of the inspection tool umbilical cable. Rotation of the shield plug mechanism is controlled to that needed to remove, store and reinstall a shield plug.

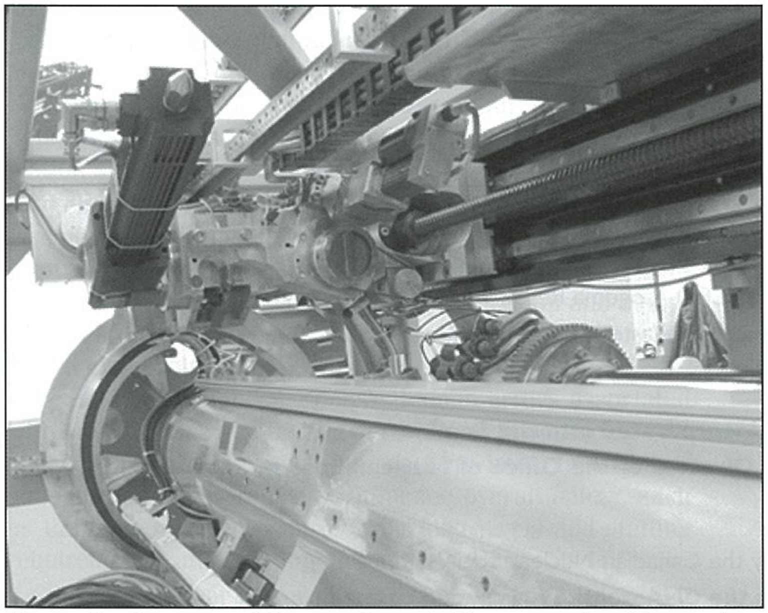


Figure 9: Common drive carriage, ballscrew and guide track.

Ram Drive

One of the challenges in the design of UDM was how to deliver an inspection tool 300 inches into a fuel channel from a delivery machine that can't exceed about 22 feet in length. Several concepts were considered, but the solution implemented was the one originally used for the SLAR ram drive on MiniSLAR. The main component is a Serapid rigid chain, a chain that can fold in one direction but not the other. With proper guiding, this chain is capable of transmitting both push and pull forces.

The chain is guided in two ways. First, the chain drive housing contains two horizontal and parallel guide plates, one just above and the other just below the chain pins. These serve to maintain the chain horizontal while it passes by the drive sprocket. The second guides are sets of polyethylene spacers attached to some of the side links of the chain. These keep the chain centered in the rams, which prevents buckling, and by being plastic, avoid scratching the inside bores of the rams as the chain slides along.

The chain drive housing contains the drive sprocket and the aforementioned guide plates. The drive motor and right angle gearhead are mounted on one side of the housing such that the gearhead output shaft penetrates into, and is keyed directly to the sprocket shaft. The chain drive housing is bolted to the common drive carriage. The chain drive is part of the common drive philosophy, in that the single chain and drive provides the axial motion for any one of the three tool mechanism ram assemblies. Of course, only the tool mechanism at the top magazine site can be driven.

A storage area has to be provided to store the long length of chain when a ram assembly is retracted. On MiniSLAR this came in the form of a relatively heavy, steel chain storage box mounted above the tail end of the SLAR ram mechanism. UDM did not use this design for two reasons: the box would be too high, causing interference with the ceiling of the fuelling machine duct; and the box would add undesired weight at the back end of the machine. Instead, a simple storage track was designed to mount onto the support frame top frame with the track extending the full length above the frame and doubling back on the underside. This is sufficient to store the entire chain when both the chain and the common drive carriage are parked in their fully retracted positions.

Unlike MiniSLAR, the need to service up to three tool mechanisms meant that the rigid chain on UDM could not be permanently attached to any of the ram assemblies. Instead, a reliable method of connecting to and disconnecting from the tool mechanism ram at the top magazine site had to be devised. The solution was the design of a special ball and socket arrangement.

A spherical ball connector is fitted to the first pair of links on the drive chain. Stops in the chain drive housing ensure that this front portion of the chain will never interface with the sprocket. The lower part of the ball is machined away to allow room for the inspection tool umbilical cable to pass out the end of the ram. There is also room between the chain side links for the cable.

A socket and collet assembly is attached to the tail end of each Ram #1 (the smallest, innermost ram). When the chain is driven forward into the ram, the chain ball enters the collet by forcing the collet fingers to open, and comes to rest against the spherical contact surface on the socket inside. The collet fingers close around the ball and a sleeve surrounding the collet is moved back into position around the fingers. Since the fingers are now prevented from opening, the ball is trapped and is now able to push or pull on the ram. Another feature of the ball and socket arrangement is that it allows relative rotation between the chain and the ram. This reduces the twisting loads on the chain as the tool mechanism rotates, and also reduces bending loads as the ram passes through a sagged channel.

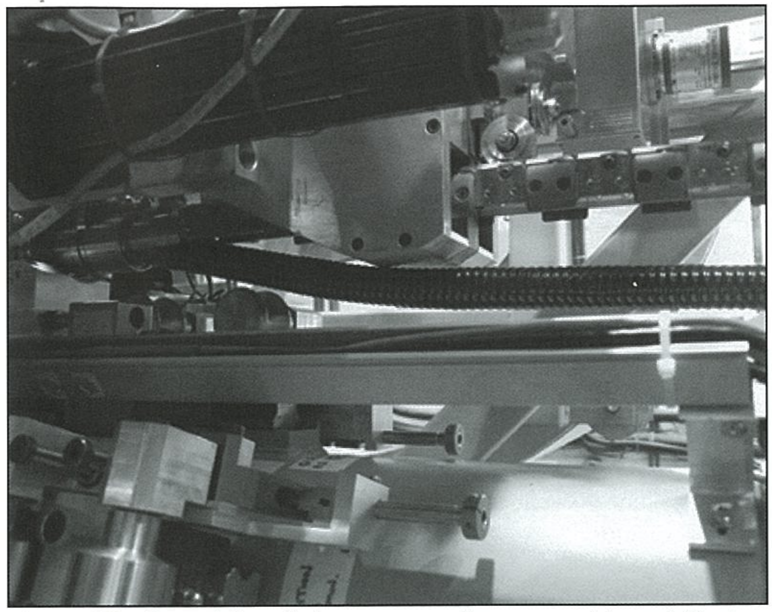


Figure 10: Ram drive chain connected to ram with umbilical cable protruding.

Canadian Nuclear Association
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Nuclear Industry Seminar 2004

The Canadian Nuclear Association's *Nuclear Industry Seminar* for 2004 will be held February 18 and 19 in the Fairmont Chateau Laurier hotel in Ottawa, Canada

The theme will be:

Nuclear Energy - a hard look at the future

An *Atlantic Canada Buffet Reception* begins the Seminar on the evening of February 18

Speakers include:

- **John Ritch** — Director General, World Nuclear Association
- **Linda Keen** — President and CEO, Canadian Nuclear Safety Commission
- **Duncan Hawthorne** — President and CEO.Bruce Power
- **Robert Van Adel** — President and CEO, Atomic Energy of Canada Limited
- **Tim Gitzel** — President and CEO, Cogema Resources
- **Elizabeth Dowdeswell** — President, Nuclear Waste Management Organization
- **Dwight Duncan** — Minister of Energy, Ontario

For the first time the W. B. Lewis Lecture will be incorporated into the Seminar
It will be given after lunch by William Magwood, Director of the Office of Nuclear Energy, Science and Technology, U.S. Department of Energy

The event is co-hosted by the Canadian Nuclear Society
For further information go to the CNA website <www.cna.ca>

Incorporating Human Factors into Design Change Processes – A Regulator's Perspective

L. Staples and H. McRobbie[1]

Ed. Note: *The following paper was presented at the24th CNS Annual Conference held in Toronto, Ontario, June 2003*

Abstract

Nuclear power plants in Canada must receive written approval from the Canadian Nuclear Safety Commission (CNSC) when making certain changes that are defined in their licenses. The CNSC expects the design change process to include a method for ensuring that the human-machine interface and workplace design support the safe and reliable performance of required tasks. When reviewing design changes for approval, the CNSC looks for evidence of analysis work, use of appropriate human factors design guide-lines, and verification and validation testing of the design. In addition to reviewing significant design changes, evaluations are conducted to ensure design change processes adequately address human performance. Findings from reviews and evaluations highlight the need to integrate human factors into the design change process, provide human factors training and support to engineering staff, establish processes to ensure coordination between the various groups with a vested interest in human factors, and develop more rigorous methods to validate changes to maintenance, field operations and testing interfaces.

1. Introduction

The Canadian Nuclear Safety Commission (CNSC) regulates the nuclear industry in Canada. Regulatory Policy P119 [1] requires CNSC staff to verify that licensees and license applicants minimize the potential for human error.

The regulatory regime in Canada is non-prescriptive. In accordance with this philosophy, CNSC reviews focus on the processes established by the applicant or licensee to minimize human error potential. These include the processes for the training and qualification of staff, the development and maintenance of procedures, and the design of systems and equipment. For nuclear power plant licensees, an important issue is the control of changes to the design of station systems and equipment.

To ensure that a design change delivers a net safety benefit, human performance issues must be considered with the same degree of rigor as more traditional engineering considerations, such as environmental qualification and equipment reliability. For example, the seemingly straightforward addition of a valve to improve system redundancy may introduce new failure modes, such as human errors associated with maintenance and testing of the valve. In addition, engineering changes can have far reaching impacts on other elements of the system that influence overall safety, such as the content of procedures, workload levels and staffing requirements, and training and qualification.

Canadian nuclear power plants date from 1971 (Pickering A) to 1989 (Darlington). Design changes to existing facilities, such as these, pose particular challenges. Challenges include balancing the benefits of introducing new technology against compatibility with existing technology, dealing with constraints such as layout and space issues, and maximizing user involvement and acceptance of the changes [2]. For major refurbishment projects, the cumulative effects of the design changes on human performance must be considered.

The remainder of this paper elaborates on CNSC expectations for integrating human factors into the design change process and regulatory review methods. Observed strengths and areas for further improvement are also discussed.

2. Expectations

The CNSC expects licensees to consider human performance when making design changes that affect operations, testing, and maintenance. In order to consistently address human performance, licensees need a design change process that includes a method for ensuring that the human machine interface and workplace design support the safe and reliable performance of required tasks. Furthermore, the process must address the impact of a change on supporting mechanisms, such as training and procedures. The first column of Table 1 shows steps in a design change process that would allow for systematic consideration of factors which may impact on human performance.

1 Canadian Nuclear Safety Commission, Ottawa, Ontario, Canada

For any design changes, the CNSC expects licensees to have a process in place that defines the appropriate level of human factors effort required. The level of human factors effort should be based on human performance and safety, and consider factors such as the safety consequences of human error, probability of human error, and the impact of the change on tasks performed by operators and maintainers.

When the screening process indicates the change will be significant, the CNSC expects a Human Factors Engineering Program Plan (HFEPP) to be submitted. The CNSC has developed guides on HFEPPs and Human Factors Verification and Validation Plans to explain regulatory expectations to licensees [4],[5]. The guides are based on international guidelines and standards, such as NUREG 0711 [3] and IEEE 1023 [6]. The HFEPP is a way for licensees to ensure that human factors activities are adequately developed, executed, managed and documented when a significant design change is undertaken. The Human Factors Verification and Validation plan and results are a way of demonstrating that the design meets specifications and facilitates safe and reliable task performance [7].

Table I: Steps in a design change process to address human factors and examples of outputs (based on NUREG 0711) [3]

Process	Examples
Planning	-Screening process defines level of human factors effort required
Analysis	-Operating exerpeince review -Task analysis -List of functions and allocation to humans and automation -Strengths and weaknesses of possible design options from a human performance perspective -HMI and workplace design requirements
HMI and Workplace Design Application of human factors guidelines	-Drawings and mock-ups -Design specifications / descriptions
Human factors verification and validation	-Verification and validation plan results
HMI Issue Resolution	-Design change requests -Training / procedural modifications

Human Factors staff evaluate the effectiveness of licensee processes through document reviews of individual design changes and evaluations of licensee design change processes. Most nuclear power plants in Canada have processes in place that meet the expectations defined in CNSC guides [4], [5]. Promotion of regulatory expectations continues with all plants to improve processes and to improve adherence to processes.

3. Desktop Reviews

Nuclear power plants in Canada must receive written approval from the CNSC when making the following changes:

- any changes to special safety systems that would render the descriptions and analyses in the Safety Report inaccurate or
- any change to equipment or procedures that could result in hazards or risks different in nature or greater in probability or magnitude than those stated in the design and analysis documents.

Human factors staff at the CNSC review design changes with significant human performance implications. This is accomplished through desktop review of design change packages, which contain a description of the proposed change along with supporting analyses.

When reviewing a significant design change for approval, CNSC staff look for evidence that a process for incorporating human factors into the design change has been followed. CNSC Human Factors staff confirm that human performance issues have been identified and analyzed and that appropriate human factors design guidelines have been followed. CNSC staff also confirm that verification and validation testing of the design has been completed and that any significant issues have been resolved.

4. Evaluation Methods

In addition to reviewing significant design changes, evaluations are conducted to ensure design change processes adequately address human performance. The CNSC's Quality Management Section evaluates design change processes to confirm compliance with Quality Assurance Program Requirements for Nuclear Power Plants [8]. Human Factors Specialists have participated in these evaluations, focusing on incorporation of human factors into the design change process.

The starting point of an evaluation is ensuring the licensee's documented process adequately incorporates human factors. Without a process, it is unlikely that human factors will be dealt with consistently between designers and between projects.

It is expected that human factors work done on a project is documented in sufficient detail that an evaluator could follow the work through from the screening process to resolution of HMI and workplace design issues. During an evaluation, CNSC Human Factors staff select several design changes that appear to impact on human performance. Design documentation, such as HFEPPs, design requirements, design descriptions, and project execution plans, are reviewed. In the design documentation, evidence is sought of appropriate screening, human factors analysis, use of human factors guidelines and station-specific design guidelines, and human factors verification and validation testing of the design. Any questions arising from the document reviews are pursued during interviews with licensee staff, such as engineering and design staff, managers, operations and maintenance staff, human factors specialists.

In addition to reviewing documents and interviewing design staff, meetings related to design are attended during the time on-site. For example, some licensees perform Constructability, Operability, Maintainability and Safety (COMS) reviews at various stages of the design. These reviews draw together various specialists and stakeholders in a brainstorming session, which includes a systematic review of human performance issues. When observing design meetings, human factors staff focus on how issues related to human performance are addressed and resolved.

In 2002, the CNSC began rating programs and implementation of programs separately in the re-licensing process. Therefore, the process for incorporating human factors into the design change process may be acceptable, but if the process is not followed, implementation of the program may be unacceptable.

5. Strengths And Areas For Improvement

Over the past ten years, the CNSC has observed an improvement of the incorporation of human factors into design change processes at nuclear power plants (NPPs). There is a greater awareness about human factors among NPPs which has led to increased hiring of Human Factors Specialists. More station specific design guides are available to ensure consistency when design changes are made. A number of major licensees and design agents have established a systematic process for incorporating human factors into the design change process [9].

The CNSC has also observed an increasing number of HFEPPs submitted at the early stages of design projects. By keeping the regulator informed of progress and by addressing areas of concern throughout the design process, there is less chance of unanticipated issues arising late in the project, when corrections may be more difficult. Although the CNSC has seen several improvements, areas requiring further improvement have also been observed.

In an evaluation of a plant with an adequate process for addressing human factors in design changes, it was found that the process was not being followed. An apparent causal factor was the screening question used to determine the level of human factors effort required was answered incorrectly. The screening question indicated the change would not impact on human performance; however, when design documentation was re-viewed by a CNSC Human Factors Specialist, it was apparent that the change would impact on human performance. This finding emphasizes the need for plants to do internal evaluations to ensure the processes are working as expected.

The screening process is important for determining the level of human factors effort to exert in a design project. Due to the importance of screening questions, they should be tested with those who will be completing them. In addition, those who will complete the screening questions should receive an appropriate level of human factors training and support, so the level of human factors effort required will be defined correctly.

For human factors to have the greatest impact, it should be a consideration when the design option is selected. For example, if a decision is made to go with an "off-the-shelf" option, reducing the potential for human error in the design should be part of the acceptance criteria. If human factors is only considered after the design option is selected, the chosen option may not be the best one for reducing human error potential.

Findings from reviews and evaluations also highlight the need to integrate human factors activities into the design plan. It is important that time frames for deliverables committed to in a HFEPP are included in the master time line for the project. This is especially critical when design packages are contracted out or when human factors work is performed by an external consultant. If all design team members are not aware of human factors deliverables, such as verification and validation testing results, the final design may not take full advantage of the human factors work performed.

Several groups have a vested interest in human factors analysis results, including training, procedure-writing, and simulator staff. Often, procedure validation is done separately from human factors validation testing of the design change. In reality, human factors validation testing of the design and procedures could be integrated. Furthermore, operators may identify problems with the design during training in the simulator. Therefore, simulator staff, who train operators about upcoming design changes, are an important source of information on usability problems. It is therefore important to establish processes to ensure coordination between groups with a vested interest in human factors.

When the human-machine interface changes, whether inside or outside the main control room, there is a potential to impact on human error. Although most NPPs systematically consider human factors for changes affecting Main Control Room operations tasks, less emphasis tends to be placed on changes to maintenance, field operations and testing tasks. In particular, the techniques for validating changes to maintenance, field operations, and testing interfaces require further development.

6. Conclusions

The CNSC has a mandate to evaluate measures implemented by licensees to address human factors and to determine whether these measures provide for protection of the environment and the health and safety of persons. The CNSC expects the design change process of licensees to include a method for ensuring that the human-machine interface and workplace design support the safe and reliable performance of required tasks.

A familiar argument is that human factors is common sense. Those who truly understand the field of human factors know that designing a system to reduce the potential for human error requires a systematic design process. Findings from regulatory reviews indicate that facilities without a process are unlikely to adequately incorporate human factors into design changes.

CNSC reviews of design changes and evaluations of design change processes highlight several areas for development. These include the need to integrate human factors into the design change process, to provide human factors training and support to engineering staff, to establish processes to ensure coordination between the various groups with a vested interest in human factors, and to develop more rigorous methods to validate changes to maintenance, field operations and testing interfaces.

7. Acknowledgement

The authors gratefully acknowledge the contributions of T. Taylor and F. Harrison for developing this paper.

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Calls for Papers

Following are some planned meetings in the USA for which papers are still being invited.

2004 Annual Meeting and 2004 International Congress on Advances in Nuclear Power Plants (ICAPP 04)
50th Anniversary of ANS: A Golden Anniversary – A Golden Opportunity,
June 13-17, 2004, OMNI William Penn Hotel, Pittsburgh, PA
Call for Papers available at: http://www.ans.org/goto/nad.cgi?id=1068271200-15

Spectrum 2004
August 22-26, 2004, Hyatt Regency Hotel, Atlanta, GA
Call for Papers available at: http://www.ans.org/goto/nad.cgi?id=1068271200-16

4th International Topical Meeting on Nuclear Plant Instrumentation, Control and Human Machine Interface Technology (NPIC & HMIT 2004),
September 19-22, 2004,
Hyatt Capital Square, Columbus, OH
Call for Papers available at: http://www.ans.org/goto/nad.cgi?id=1068271200-19

Royal Society of Canada Symposium on Energy, Environment and Society

Making Choices

– Little reference to nuclear at symposium

The theme of the one-day symposium on *Energy, Environment and Society,* held in Ottawa, November 25, 2003, was Making Choices but it is unlikely that any of those attending went away with a clear idea of what should be the appropriate choice of energy source. Certainly there was little acknowledgement of the role that nuclear power can play in reducing green-house gas (GHG) emissions.

The symposium, which was held in the Museum of Science and Technology, was organized by the Royal Society of Canada in collaboration with the National Research Council. Several other organizations including Atomic Energy of Canada Limited, Canadian Nuclear Society, BIOCAP, Natural Science and Engineering Research Council, were co-sponsors.

There were good presentations on the historical use of energy, on the carbon cycle, the Kyoto accord and on the consequences of Kyoto on the manufacturing sector, aboriginal peoples, and municipalities but little on the energy choices that must be made.

Donald Johnston, secretary-general of the Organization for Economic Cooperation and Development (OECD) was slated to speak about nuclear energy but appeared to be constrained by the divergent views of the member countries of OECD, despite his statement that he was not representing the organization. Nuclear power is needed to reduce GHG emissions, he said, but commented that even if 700 new nuclear plants were built by 2030 nuclear would still only supply 9% of the world's total energy. Referring to the optimism at the time of Eisenhower's "Atoms for Peace" speech 50 years ago he noted that past problems of the nuclear industry had led to public scepticism. Given the environmental benefits of nuclear power he said he is puzzled by the opposition of European "Green" parties.

The strongest advocate for nuclear power was **David Sanborn Scott,** International Association for Hydrogen Energy. Although focussing on hydrogen as an energy "currency" he spoke strongly about the role of nuclear as a primary energy source to produce hydrogen. Today, he noted, nuclear is used primarily to produce electricity. In the future he predicted that half of nuclear power would be used to produce hydrogen, which could displace petroleum in transportation, the largest source of GHG. Hydrogen could also support distributed electricity generation, which, he said, would make blackouts such as August 14 impossible.

Bill Rees, UBC, the lead speaker, gave a fascinating overview of the historical use of energy. Just in the last 200 years fossil fuels had displaced human and animal energy, and this had led to a population explosion. Earlier the energy source for food production was solar, now it is 90% fossil, he commented. Noting that since 1980 the world has consumed more oil than has been discovered, he predicted that the global production of oil would peak by 2017, which could lead to serious societal problems.

The carbon cycle was the subject of the presentation by **David Layzell,** CEO of BIOCAP Canada. Carbon is stored in (in descending order) rocks, the oceans, fossil fuels, land and the atmosphere. Only in the last 20 years has the emission from fossil fuels exceeded the net effect of cutting forests. Currently, he stated, the biosphere is a net sink for carbon but there is a question how long this will continue.

Heather Smith, author, spoke specifically on the Kyoto Protocol. It is only nominally about the environment, she contended, primarily about politics and competitiveness. She asserted that Canada followed US interests in Kyoto discussions. Kyoto is only a beginning, she noted, but even so Canada and most countries have missed [GHG reduction] targets to date.

Allan Amey, CEO of Climate Change Central, an Alberta public-private partnership, spoke about reducing GHG emission in the oil and gas industry. We can move to a "carbon constrained" future, he said, through a combination of policy, technology and market forces. He noted that the "carbon intensity" of our economy (relationship of carbon emissions to GDP) had decreased over the past few decades. A price for carbon emissions combined with research and development on clean technologies could help break the link between GDP growth and emissions.

The morning session concluded with comments by **Richard Gilbert,** Centre for Sustainable Transportation, followed by a panel discussion. He remarked that he accepted Johnston's claim that nuclear could provide unlimited energy but questioned if this was a good thing.

In the afternoon the speakers focussed on specific areas. **Vasudha Seth,** Dofasco, outlined how her industry is addressing the key issues arising from the Kyoto Protocol by pursuing technologies to reduce CO_2 emissions. **Fred Wilson,** CEP Union, spoke on implications of the Accord for labour. **Eva Ligeti,** Clean Air Partnership, talked about action at the municipal level and **Carol Crowe,** Indigenous Visions, emphasized the need to consult aboriginals. **Jayson Myers,** Canadian Manufactures and Exporters, complained that the federal plan for Kyoto only dealt with part of the issue, that there was no integration of policies, rules were in constant flux, and there was no mitigation plan.

The afternoon closed with a panel discussion followed by a small reception.

Further information on the symposium and the Royal Society of Canada can be obtained at its website <www.rsc.ca>.

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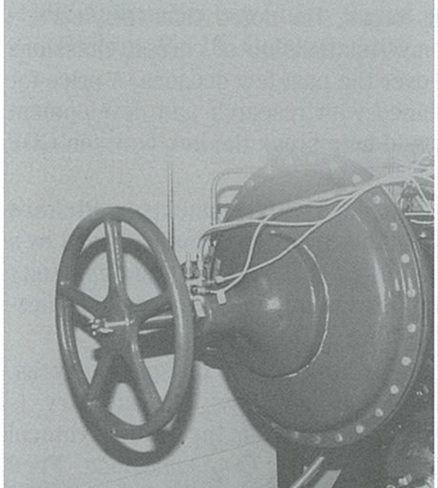
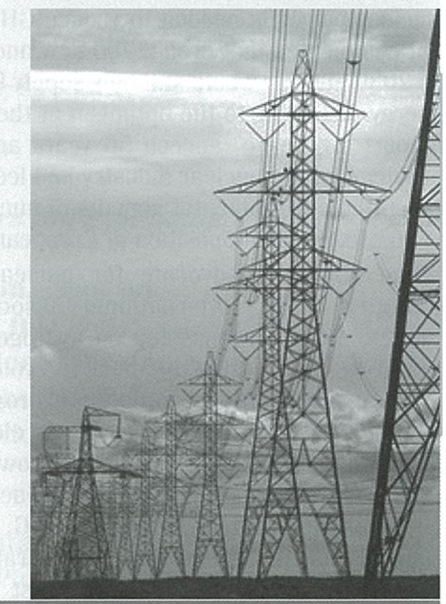
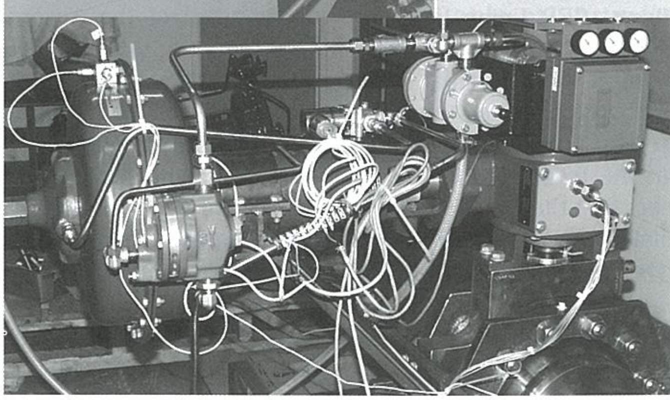
- Pre-commissioning and System Turnover
- Utility Information System Update







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GENERAL **news**

New president for CNA

In late November Bill Clarke, president and CEO of the Canadian Nuclear Association, announced that the CNA Executive Committee had selected **Murray J. Elston** to be his successor. Clarke had informed the CNA Board in the summer that he wished to retire by the end of the year. Following an extensive and professionally managed search campaign, Elston, who is currently President of the national industry association *Canada's Research-Based Pharmaceutical Companies,* was chosen. He has extensive advocacy experience in Ottawa, Ministerial responsibility and authority in the Ontario government, and a sound knowledge and understanding of nuclear energy through his representation of the Bruce constituency in the Ontario legislature. He is a graduate of the University of Western Ontario and practised law before being elected to the Ontario legislature in 1981. During his 13 years as an MPP he held several posts, including Minister of Health. Elston will be assuming his new post in January 2004.

NB Electricity Act to be proclaimed

On November 27, 2003, Bruce Fitch, Minister of Energy in the New Brunswick government, announced that the new *Electricity Act* will be proclaimed on April 1, 2004. This will result in:

- a competitive electricity market for 39 large industrial customers and 3 municipal utilities,
- reorganization of NB Power into five distinct units (a Holding company and 4 subsidiaries including generation, transmission, distribution and customer Service, and nuclear),
- empowering the Public Utilities Board to play a greater role in the electricity market and protecting ratepayers, and
- creation of a System Operator as an independent body to manage the market rules and the electricity transmission system.

Two weeks earlier, on November 13, NB Power's acting President and CEO **Stewart MacPherson** announced his intention to retire effective March 31, 2004. MacPherson joined NB Power in 1966 and has held senior positions in the areas of rates, load forecasting, materials management, business information systems, customer service and corporate planning. He is the architect of NB Power's business development plan that will see the refurbishment of major generating facilities and expansion of the transmission infrastructure over the next several years.

The NB Power Board of Directors has begun the search for a new President and Chief Executive Officer. Under the corporate structure that will be created with the implementation of the *Electricity Act,* the position will be both the President and CEO of the Holding Company, as well as, for a period of time, the CEO of each operating company.

NWMO invites comments

On November 28, 2003 the Nuclear Waste Management Organization (NWMO) announced the publication of its first discussion document, ***"Asking the Right Questions?*** *Future Management of Canada's Used Nuclear Fuel."* The document describes several possible technical methods, including three the NWMO is required to study, and it poses ten key questions that might be used to analyze them. The NWMO is inviting comments on this report and on how it proposes to assess nuclear fuel waste management options.

The NWMO will utilize a number of engagement techniques, including its website, over the course of its study. Comments, questions and concerns that are raised will help shape the detailed assessment of options and will be reflected in another discussion paper expected in mid-2004. The organization will then present its recommendations for scrutiny in a draft final report, which will be issued before an implementation plan is delivered to the Minister of Natural Resources Canada in November 2005.

"Asking the Right Questions? *Future Management of Canada's Used Nuclear Fuel"* is available for download at: www.nwmo.ca.

China: AECL Agreement To Expand Technological Co-operation

Atomic Energy of Canada (AECL) and the China National Nuclear Corporation (CNNC) have signed a memorandum of understanding that sets out their plans for the 'extension and expansion of co-operation' on nuclear power technology. The agreement was signed in Beijing on October 21, 2003 by AECL president and chief executive **Robert Van Adel** and CNNC president **Kang Rixin.** The agreement is expected to lead to a strengthening of business co-operation including:

- Technical support to the third Qinshan nuclear power plant;
- Application of "advanced project management and construction techniques" to China's nuclear power projects;

- Assessment of the "potential for Candu-type reactors to use recycled spent fuel from light water reactors and thorium resources";
- Co-operation in the use of uranium resources for nuclear power;
- Co-operation in the "commercialisation of Candu technology outside of China";
- Improvement of China's nuclear power plant equipment design capability.

Two days later, October 23, a ceremony was held at Hangzhou, with Chinese Vice-Premier **Zeng Peiyan** and Canadian Prime Minister **Jean Chretien** attending, to mark the completion of the third phrase of the Qinshan Nuclear Power Plant, the biggest cooperation project between China and Canada. Zeng described the project as an "important milestone" in the history of economic cooperation between China and Canada. Before the ceremony, the two leaders met and discussed investment and economic cooperation. After the meeting, they attended a ceremony unveiling a monument marking the completion of the project.

AECL's Corporate Plan tabled in Parliament

The Summary of Atomic Energy of Canada Limited's Corporate Plan for 2003 - 2008 was tabled in Parliament in early September. The plan contains AECL's high level strategy required to achieve corporate-wide objectives with clearly defined outcomes and related measures to monitor performance.

The Corporate Plan identifies three major objectives and several strategies to strengthen AECL's leadership role as a commercial Crown corporation:

- Achieve $1 billion in annual revenue by 2007-08;
- Achieve recognition as a leader in health and safety and nuclear as a clean air solution; and
- Achieve progress in managing the Canadian nuclear platform obligations and effectively supporting the CANDU asset life cycle through innovative solutions to increase performance to customers.

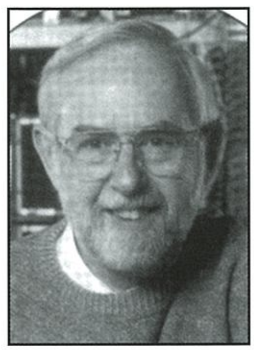
To achieve its revenue target, AECL will focus on growth in several major business lines, including reactor services; waste management, environmental and decommissioning; refurbishment; and continued development of the Advanced CANDU Reactor (ACR).

During the Plan period, AECL will accelerate its objective of being recognized as a global leader in environmental and health related technologies and site operations. To achieve this goal, AECL will communicate the benefits of nuclear as a safe and environmentally friendly energy solution to increase public recognition and acceptance of nuclear as well as AECL's contributions. Significant investment will be made in staff development and training and application of staff resources to key commercial and technology development activities. The Plan projects AECL's net income to steadily increase from a breakeven position in the first year to $98 million in the last year.

MAPLE saga continues

At the meeting of the Canadian Nuclear Safety Commission November 27, 2003, CNSC staff reported that they were still waiting for a further report from Atomic Energy of Canada Limited on the positive power coefficient observed earlier this year during commissioning of the MAPLE 1 reactor. (MAPLE 1 is one of two small reactors, designed for isotope production, being built at the Chalk River Laboratories of AECL for MDS Nordion.) Although the unit has been in a secured shutdown state since early summer CNSC staff have authorized periodic operation up to 2 kW.

Fuel was loaded into MAPLE 2 and the reactor achieved first criticality on October 9. Commissioning continues with a hold point at 2 kW.



SNO Director awarded Herzberg Medal

Dr. Arthur McDonald, director of the Sudbury Neutrino Observatory (SNO) and Queen's University professor, was awarded the Gerhard Herzberg Canada Gold Medal for Science and Engineering and the Award of Excellence of the Natural Science and Engineering Council (NSERC) at an awards banquet in Ottawa, November 25, 2003. These are the highest honours in Canada for science and engineering and are accompanied by a large research grant from NSERC. Following is an excerpt from the citation for the award.

"To build SNO, Dr. McDonald managed the creation of the most sensitive neutrino detector to date. It was a massive engineering project that involved the construction of an ultra-clean, ten-storey-high neutrino detector including 1,000 tonnes of heavy water (worth $300 million, on loan from AECL), two kilometres underground in INCO Ltd.'s Creighton nickel mine in Sudbury. SNO would be the first neutrino detector with equal sensitivity to all three kinds of neutrinos (electron, muon, and tau) and to be able to distinguish electron neutrinos from the others."

SNO went into operation in 1996 and has already disproved the "Standard Model" which assumes that the three kinds of neutrinos do not change from one type to another and that they do not have a mass greater than zero.

McArthur River update

On November 3, 2003 Cameco Corporation submitted a report to the Canadian Nuclear Safety Commission on the water inflow incident of last April. (See Vol. 24, No. 2 May 2003 issue of the CNS Bulletin.)

The primary conclusion identified by Cameco is stated as:

"A fall of ground occurred in the 7320 E freeze drift on April 6, 2003. The ground fall was the result of inadequate ground support for the size of the drift. The ground fall breached the grout cover above the opening allowing a significant inflow of water to enter the mine."

The report acknowledges a number of factors contributed to the incident.

- rock mechanics information was not communicated effectively
- there was no formal risk assessment
- the complexity of the geology was not fully appreciated
- water handling, control and treatment processes were inadequate for the emergency

The report also informed about discharges in October which exceeded the radium concentration limit of the mine's licence.

The incident report and the radium discharges are still being reviewed by CNSC staff.

Algae shut down plant

The severe weather and strong winds resulting from hurricane Isabel on Friday, September 19, 2003, resulted in an algae run in the Pickering B screenhouse. As a proactive measure Unit 7 was shut down to reduce the cooling water load and avoid tripping multiple units. By the Sunday afternoon the threat of an algae run was reduced sufficiently to allow Unit 7 to return to power.

MDS Nordion gets positive review

At the meeting of the Canadian Nuclear Safety Commission, November 27, 2003, CNSC staff submitted a mid-term report on the operation of MDS Nordion's facility in Kanata, now part of the amalgamated City of Ottawa. Nordion operates under a Class 1B Nuclear Substance Processing Facility licence issued in November 2000 and running to October 2005. CNSC staff concluded "that the overall performance of MDS Nordion over the past three years meets expectations..... [and] the doses to workers and radioactive emissions from the facility have been well below regulatory limits".

DoE's Magwood to give W. B. Lewis lecture

William Magwood, Director of the Office of Nuclear Energy, Science and Technology in the U.S. Department of Energy, will be the 2004 W. B. Lewis lecturer.

For the first time the 2004 W. B. Lewis lecture will be held during the Annual Seminar of the Canadian Nuclear Association to be held in Ottawa on February 19, 2004. The series of lectures honours the memory of Dr. W. B. Lewis, director of the Chalk River Laboratories from 1946 to 1952 and vice-president of Atomic Energy of Canada Limited from 1952 to 1973, who is considered the "father" of the Canadian nuclear power program.

Mr. Magwood is the senior nuclear technology official in the United States Government and the senior manager for all of the programs of his Office. Mr. Magwood is leading the Department's *Nuclear Power 2010* initiative, aimed at building new nuclear plants in the U.S. by 2010 as a key to long-term energy security. He is also leading the United States' *Generation IV* initiative, working closely with the *Generation IV International Forum* – an international collective of 10 leading nations and the European Union's Euratom–dedicated to development of next generation advanced nuclear energy technologies.

Interim report on "blackout"

On November 19 the Interim Report of the Canada–U.S. Power System Outage Task Force was released. This report identifies the causes of the August 14, 2003 blackout that affected Ontario and northeastern USA. The Interim Report assesses conditions on the electric transmission grid that contributed to the blackout; outlines the actual physical causes of the outage; and discusses events and conditions that allowed the blackout to spread. The report is available on the website of Natural Resources Canada <www.nrcan.gc.ca/media>.

The Interim Report sets the stage for the next phase of the process — the development of recommendations aimed at reducing the possibility of future outages. As one part of this process the public and stakeholders in both countries will be asked to give their comments on the report and their views on enhancing the reliability of the electricity system. The Final Report of the Task Force will focus on identifying specific recommendations to reduce the likelihood of future blackouts and bolster the reliability of the electricity infrastructure.

Comments invited: The Task Force will be holding a public forum in Canada for stakeholders and the public to comment on the Interim Report. Interested parties who wish to provide written submissions in English or in French are invited to do so through NRCan's website by December 12, 2003. Submissions can also be sent by mail or fax to Dr. Nawal Kamel, Special Advisor to the Deputy Minister, Natural Resources Canada, 21st floor, 580 Booth Street, Ottawa, Ontario K1A 0E4; Fax: (613) 995-0087. All public feedback will be posted on the Web site in the language in which it was submitted.

Bruce 8 outage extended

Bruce Power is extending the maintenance outage on Unit 8 until later in the fourth quarter of 2003 after inspections identified some erosion on support plates in three of the unit's eight steam generators.

A detailed investigation and analysis is ongoing regarding the condition of the support plates, which separate the boiler tubes as they run through the steam generators. No damage to the boiler tubes has been identified.

Due to the nature of the repairs, any modifications will require review and approval by the Canadian Nuclear Safety Commission before the unit returns to service.Inspections from previous outages show no evidence of a similar condition in the other steam generators at either the Bruce B or Bruce A generating stations.

RedR seeks volunteers

RedR is an international organization working to save lives and reduce suffering. Its members provide technical, managerial and logistical support to front-line humanitarian relief agencies like Oxfam, CARE and Médecins sans frontières. RedR provides assistance by placing its members for short-term assignments in overseas disaster areas, that typically involving one or more of the following:

- re-establishing fresh water and food supplies
- rebuilding roads and bridges
- setting up relief camps and ensuring adequate sanitation and hygiene in them
- restoring communications

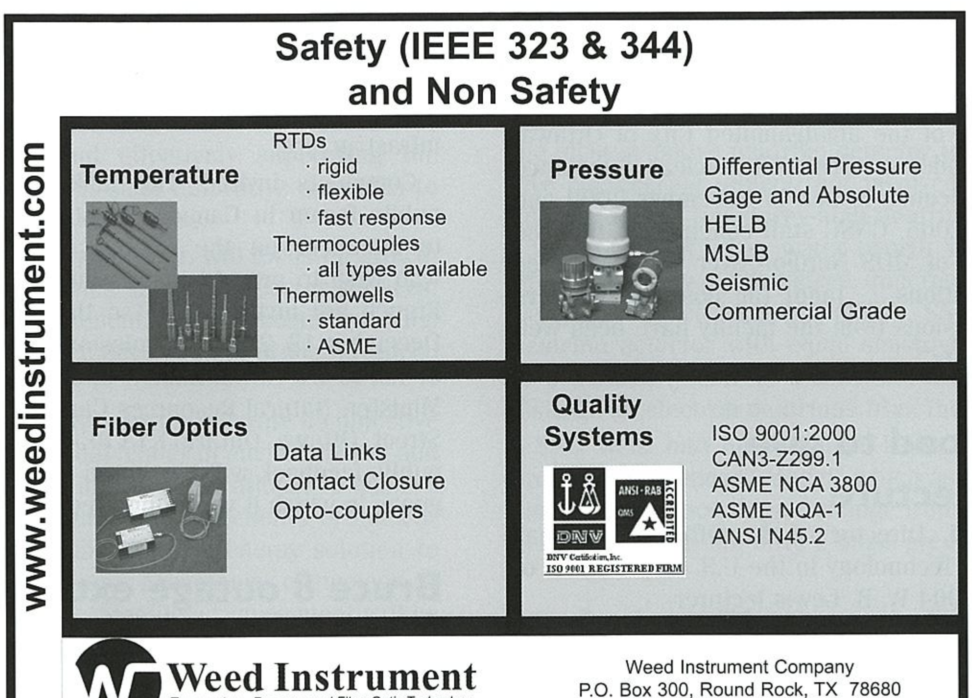
and many other types of relief assistance.

RedR Canada was established in 2002 and is beginning to make placements of members with clients in Canada and the USA. It needs part-time volunteers to help in human resource management, and administration. Volunteers must be mature, well-organized self-starters with a successful track record. These positions would appeal to retired persons who wish to keep busy – but not full-time –with a charitable organization. Needed are: a Placements Manager; Placement Professionals and an Administration Manager. The tasks require a commitment of 10 to 15 hours a week on average. Most of the work in Placements could be done from home. Office space is available at RedR Canada's offices in Ottawa. For further information, please contact Kirk Thompson, Executive Director, RedR Canada. E-mail: kthompson@redr.ca, or telephone 613-232-9999.

Principles for a Carbon Market Agreed

On October 23, 2003 the Canadian Working Group on the Carbon Market (CWGCM) of the International Emissions Trading Association (IETA) and the Government of Canada agreed to a set of principles for the design and functioning of a market for greenhouse gas emission (GHG) compliance instruments in Canada.

The domestic emissions trading system will be an important tools for large final emitters, such as power, oil and gas, and certain mining and manufacturing industries, in meeting the reduction of 55 million tonnes of GHG emissions under the Government of Canada's Climate Change Plan for Canada. Canada's target under the Kyoto Protocol is to reduce GHG emissions by six percent from 1990 levels by the period 2008 to 2012. Industry considers that an effective and efficient domestic emissions trading system is an important element in efforts to balance climate change commitments and a healthy economy.



Gordon Brooks

The following TRIBUTE TO GORD BROOKS was prepared by George Pon, a former vice-president at AECL and a long time close associate and friend.

Gordon Leonard Brooks was born in Edmonton, Alberta on November 4, 1930 and passed away in Cambridge, Ontario on September 26, 2003.

Gord graduated in Chemical Engineering from the University of Alberta in 1952 and joined the Chalk River Nuclear Laboratories of Atomic Energy of Canada Limited shortly afterwards. He spent most of his 39 year career with AECL but was employed by Canadian Westinghouse for a short period

In the period 1959-61 he was seconded to the UK Atomic Energy Authority. Upon returning to Chalk River he joined the Advanced Reactor Engineering Division where he led a team on the design of advanced reactors. In 1965 he transferred to Toronto to design the Gentilly -1 reactor. From that beginning he was to take a leading role over the next 25 years on all of AECL's nuclear power projects: the eight Pickering reactors, the eight Bruce reactors, the four Darlington reactors, and the reactors in New Brunswick, Quebec, Argentina, Korea and Romania. He retired from AECL in 1991 with the position vice President and Chief Engineer.

Gord was one of Canada's pioneers in nuclear power engineering. In recognition of his extraordinary achievements the Canadian Nuclear Association presented him with the W. B. Lewis Award. AECL recognized Gord's magnificent contribution by establishing a $1000 award to be presented annually to a young AECL engineer who had the potential to become the next Gord Brooks.

His approach to engineering was profound knowledge, deep understanding, horse sense, and lots of laughs,

When I'm asked what I did in my working career I will respond with humility and great pride that I worked with Gord Brooks on the Canadian nuclear power program.

Over and above his exceptional technical accomplishments Gord was very much an exceptional human being.

He was a man of many talents and interests. Suffice it to say he was a photographer of professional calibre, an auto mechanic who's car never broke down, a skilled and careful cabinet maker who never cut off a finger, a star of the Deep River Drama Club where his portrayal of General Bull Moose in the Lil Abner Musical will be long remembered. He sang loudly and laughed louder. I always felt if Gord did not laugh at least a hundred times a day there was something awfully wrong.

During his long four year battle with the incurable Lou Gehrigs disease Gord's remarkable character shone through this terribly wasting illness. On first entering the long care home he was confined to a wheel chair. Gord used his feet to paddle around as he did not have the strength in his arms to move the chair but he did have the strength of character to maintain a remarkably cheerful disposition and outlook on life.

At the beginning of his illness we had many telephone conversations and in later stages of his adversity I and many friends were frequent visitors to his long care home in Kitchener and later in Cambridge. We talked, joked, bantered and told outrageous stories just as we had done for the last forty years. We hid our sorrow but pretended all was well - and he was mentally very well. His mind was as sharp as ever. His comprehension of current world affairs was astounding. He continued to fill his knowledge bank through daily visits to the Internet via his voice-actuated computer. At first he was able to type his e-mails with his fingers and later when he couldn't bring his hands to the keyboard he would peck out very short messages using a straw in his mouth. He played the hand that fate had cruelly dealt him with heroic skill.

Of course he appreciated immensely our telephone calls and visits from his many friends. Gord especially enjoyed the visits of friends from Japan and Korea and the best wishes of associates from across Canada.

His good spirits and vibrant conversation was infectious. Those who visited him came away refreshed. Our lives arc nobler and richer. His spirit of optimism and selflessness was a profound example for all who are blessed to have known and loved him.

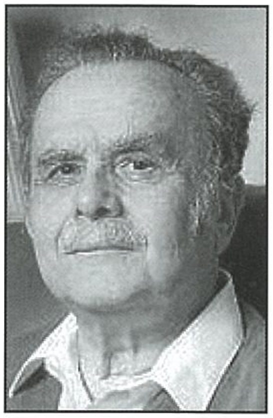
He showed us in spite of his illness that life was to be enjoyed to the fullest. He was the most positive person that I have ever known. He projected such a positive light - never a complaint - a person of extraordinary strength and faith - he felt that life was a gift to be treasured.

In my last visit with Gord my eyes saw a withered, motionless, friend but my heart saw him standing tall on an island of greatness with his unforgettable laugh booming loudly

In over 40 years o£ a wonderful and vibrant friendship Gord showed me how to live and in four years of Lou Gehrig's he taught me how to die.

A memorial service and scattering of ashes was held at Pine Point near Deep River on Saturday, November 1, 2003, with many of Gord's colleagues and friends attending.

Following his retirement Gord wrote "Short History of the CANDUJ Nuclear Power System" and, together with John Foster, a set of papers on "CANDU Origins and Evolution". These can be accessed through the CNS website <www.cns-snc.ca>.



Bertram Brockhouse

Bertram Neville Brockhouse, winner of the 1994 Nobel Prize in Physics, died in Hamilton, October 13, 2003.

Bertram Brockhouse was born in Lethbridge, Alberta, on July 15, 1918. He obtained a B.A. in physics from the University of British Columbia in 1947 and a Ph.D. from University of Toronto in 1950. The following year he joined the Chalk River Laboratories of Atomic Energy of Canada Limited to work in a newly created group on neutron spectroscopy. It was there that he developed the triple-axis neutron spectroscope and used it to study condensed matter, the work for which he was awarded the Nobel Prize. He shared his physics prize with American physicist, Clifford Shull,

He moved to McMaster University in Hamilton in 1962 and stayed there until his retirement in 1984. He was chairman of the Physics Department from 1967 to 1970.

Brockhouse received many honours over the years, including the Tory Medal of the Royal Society of Canada, the Buckley Prize of the American Physical Society, the Duddell Medal and Prize of the (British) Institute of Physics and Physical Society "for excellence in experimental physics", and the Centennial Medal of Canada. He was a Companion of the Order of Canada, a Fellow of the Royal Societies of Canada and London, and a Foreign member of the Royal Swedish Academy of Sciences. He received honorary D.Sc. degrees from the University of Waterloo and McMaster University. He was also a member of the Philosophy of Science Association

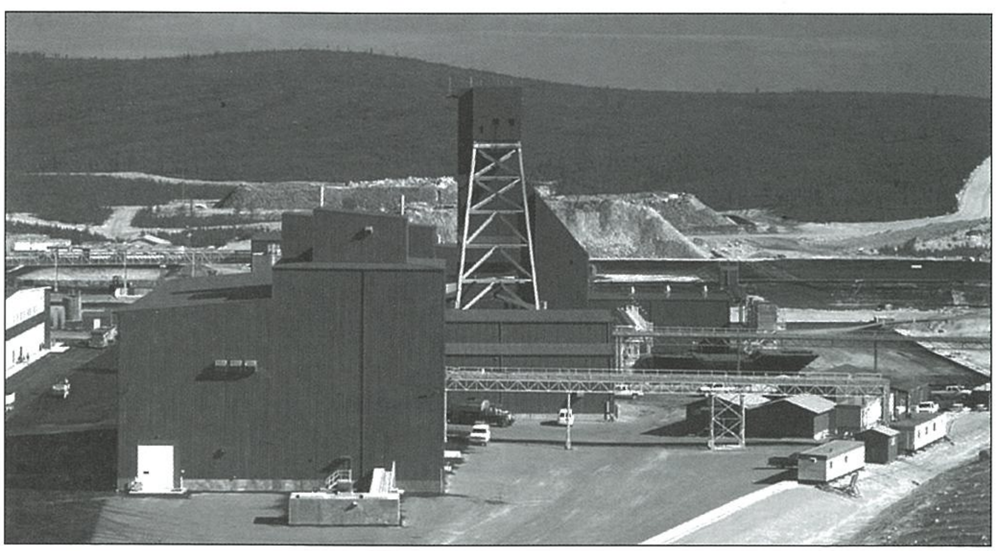
William Campbell

William Munro Campbell, one time Director of Chemistry and Metallurgy at the Chalk River Laboratories of Atomic Energy of Canada Limited, died September 14, 2003.

Bill Campbell was born January 21, 1915 on a farm at Kilmartin in south western Ontario. After high school, he spent a couple years at home working on the farm before going to the University of Toronto where he obtained a degree in chemical engineering. He then went to the Case Institute of Technology in Ohio for his Master of Science before returning to work at the Ontario Research Foundation in Toronto, where he met and married Mary Jeffrey, in 1941. After a period working for Shawinigan Chemicals and a year teaching at Queen's University, he attended the University of Illinois where he obtained his doctorate.

He joined the Chalk River Laboratories of AECL in 1950 and became head of the Chemistry and Metallurgy Division in 1956. In 1965 he moved to become Director of Research at the Ontario Research Foundation then returned to AECL as Manager of Industrial Applications at Sheridan Park. He finished his professional career as Project Director for Wardrop London Power Consultants in Winnipeg, retiring in 1978.

In 1957 he was named first editor of the newly established Canadian Journal of Chemical Engineering, and Chairman of the Board of Directors of the journal from 1958-61. In 1970-71, he was President of the Canadian Society for Chemical Engineering. In 1976 he was awarded a Queen's Silver Jubilee medal for his contributions to his profession.



A view of the surface works of the McArthur River mine.

CNS news

President visits McArthur River

In September 2003 CNS President Jeremy Whitlock visited Cameco Corporation's McArthur River mine in Northern Saskatchewan with other members of the Board of the Canadian Nuclear Association. Following is his account of that visit.

A Visit to the McArthur River Uranium Mine

by Jeremy Whitlock

Our journey begins in Saskatoon at 6:30 AM, where a twin-prop aircraft whisks us off on a 620 km trip due north to the McArthur River mine. The single-digit temperature and misty rain of the Saskatoon morning would turn out to be the best it would ever get up north that day.

As we descend over the Athabasca Basin a couple of hours later, the view from the plane window reveals a sparse landscape: stunted black spruce, pine, and everywhere glacial debris. The area around McArthur River seems to be littered with moraines, drumlins, and eskers. There was much action here 10,000 years ago as the last of the great glaciers advanced and retreated, scarring and shaping the terrain as it passed.

The Athabasca Basin itself goes back much further than that of course. Almost 2 billion years, about half the age of the earth. At that time the Precambrian continental shelf began to fill with sediment from the erosion of Himalayan-sized mountains that today are completely vanished. The resulting sandstone basin goes down 1400 metres at its deepest point.

Today our journey will take us only 600 metres down, somewhere along the lip of the basin, to the boundary between sandstone and granitic basement rock. It is at the boundary, or "unconformity", where interesting things tend to happen, including the mineralization and deposition of uranium.

About 20% of the world's uranium, two-thirds of Canada's entire production, comes from this one mine in northern Saskatchewan. Put another way, fully 2% of the world's electricity supply is fueled from this one 50-hectare site.

The plane puts down on a gravel runway amongst scrub spruce trees and pebbly sand. The short trees bespeak both a recent fire and a painfully slow re-growth period. The sand is a more familiar sight to someone from Petawawa or Deep River, Ontario: in both cases a raging glacial meltwater stream dumped its sediment at the confluence with a greater waterway, many millennia ago.

But enough of the surface.

After a brief welcome by our host, Gerald Grandey, Cameco President and CEO, we are headed underground. Also accompanying us is COGEMA VP Bob Pollock. COGEMA owns 30% of McArthur River, while Cameco owns the remaining 70% and operates the site.

Our first stop is the 530-metre level, the upper of two main levels used at McArthur River. Dressed in coveralls, boots and hardhats, we feel well prepared for the drizzling water and occasional muck. Everywhere there is evidence of the accidental flooding this past Spring. Temporary pumps and hoses are commonplace, and here and there grouting operations are in full swing next door to resumed uranium mining. We hear about the heroic team effort that kept the interruption of production to a mere three months.

Production, simply put, is awesome to behold. Earlier this year Cameco won a CNS/CNA Innovative Achievement award for its solution to the problem of extracting high-grade uranium ore from saturated sandstone, with ground-water pressures not unlike that found at the bottom of an ocean.

Their solution to the water problem was to create a frozen wall that isolates the saturated sandstone from the production zone. We saw the tops of the pipes circulating –30 degree Celsius brine through the sandstone; somehow the thought of all that water being held back was more provocative that the thought of half a kilometre of Canadian Shield above us.

The second problem, that of mining ore with significant radiation fields, was solved with a raise-bore operation and a combination of remote handling and underground milling. Rather than conventional drilling and blasting, the ore is first drilled from above with a "pilot hole", then reamed out from below.

At the 530-metre level we can see the raise-bore rigs drilling down through the ore body below us. Later, at the 640-metre level, we see the other end of the operation: 3-metre reaming heads waiting to be used in the next raise.

A worker demonstrates his skill at manipulating a remotely controlled front-end loader. We hear the echoing roar before we see anything, and then the metallic creature advances from a side tunnel, does a three-point turn, and poses in front of us. It is the loader's job to remove rock waste and uranium ore falling down from each raise. Each scoop of ore is held up to an overhead gamma meter that tells the operator the grade, and therefore the next step to take.

The higher-grade ore is blended down to 15% at this stage, so that it can be received by the mill at nearby Key Lake. The entire concept of BLENDING DOWN ore to a mere 100 times the world average concentration, must be a bit bizarre to the rest of the mining community.

Down a few more metres and we see the huge grinding mill which turns the ore into sand-like consistency. It is then mixed with water, thickened, and the resulting slurry pumped to the surface. This unique setup was designed, like much else down here, to reduce worker exposure. Amazingly, we are told that every massive piece of machinery we see came, piece by piece, down the same elevator that brought us to these depths.

Worker safety appears to be a guiding principle, which is comforting. Everywhere we see radon monitors hooked to "traffic lights" that are brilliant in their simplicity: red for no-entry, green for safe, amber for probable danger requiring investigation.

Once back at the surface we see the trucks that carry the slurry 80 km by road to the Key Lake mill. A dozen or so of these trucks leave McArthur River each day, with 5000 pounds of U308 slurry per truck. At about 18 million pounds per year, with a current price approaching $12 per pound, and 25 more years of reserves in the ground, one certainly begins to see the point of it all.

McArthur River is a phenomenon, with people at its heart. We meet senior geologist Brian McGill, who tells us a fascinating story of the legendary day in August 1988 when the very last borehole of the prospecting season yielded the mother lode. We sense the pride in the voice of Bob Pollock, COGEMA VP of Environment, Health and Safety, as he speaks of the state-of-the-art waste water treatment. At lunchtime we share the cafeteria with the day shift, which easily reflects the claim of 50% aboriginal employment. The workforce lives at the mine site on a one-week on, one week off basis.

The plane deposits us back in Saskatoon, but we'll not soon forget our journey beneath the Canadian Shield. Despite the scale of the operation we are left with a strong impression of the efficiency of uranium as a fuel. We'll also not look at a tiny generator at a nuclear plant the same way again, after seeing first hand the front end of the vast infrastructure that keeps those magnets spinning.

New Members

We would like to welcome the following new members, who have joined the CNS recently.

Nous aimerions accueillir chaudement les nouveaux membres suivants, qui ont fait adhésion à la SNC récemment.

Mohamed Dahmani, École Polytechnique de Montréal

Roy McGillivray, Babcock & Wilcox Canada

Peter King, Babcock & Wilcox Canada

Ken Hill

Al Benton

Alan Ripley, Bruce Power

Gregory Jackson, McMaster University

Amandeep Bedi, McMaster University

David Harris, Kinectrics Inc.

Brian Coulas, Bruce Power, Plant Design Engineering

Aamir Husain, Kinectrics

Faramarz Akbari, Royal Military College of Canada

Barbara Sawicka, AECL

Scott MacDonald, Comstock Canada Ltd.

Yong Shao, University of New Brunswick

Carol A. Gregoris, Ontario Power Generation

Macit Cobanoglu, Atomic Energy of Canada

Daniel Schneider, McMaster University

Jocelyne Martin, CNSC

Gerald Crawford

Kevin T. Routledge, Nuclear Safety Solutions Limited

James Rippon, Nuclear Safety Solutions Limited

Charlie B. Boyle, Kinectrics Inc.

PetreJifcu, Nuclear Safety Solutions Ltd.

Paul Shipp, Babcock & Wilcox Canada

Jeffrey Thomson, AECL

Jeffrey Millman, Babcock & Wilcox Canada

Greg M.Shikaze, Babcock & Wilcox Canada

John R. MacQuarrie, Babcock & Wilcox Canada

Gord Fountain, Wardrop Engineering

Sixth International Conference on Simulation Methods in Nuclear Engineering

2004 October 13-15
Montréal, Québec, Canada

Call for Papers

The Canadian Nuclear Society is organizing its Sixth International Conference on Simulation Methods in Nuclear Engineering, to be held in Montréal, Québec, Canada, 2004 October 13-15. Come enjoy the culture, sophistication, and cuisine of Canada's francophone metropolis!

The objective of the Conference is to provide a forum for discussion and exchange of information, results and views amongst scientists, engineers and academics working in various fields of nuclear engineering.

The scope of the Conference covers all aspects of nuclear modelling and simulation, including, but not limited to:

- Reactor Physics
- Thermalhydraulics
- Safety Analysis
- Fuel and Fuel Channels
- Computer Codes and Modelling

Deadlines

- Receipt of summaries: 2004 March 14.
- Notification of acceptance: 2004 April 20.
- Receipt of full papers: 2004 August 15.

Guidelines for Summaries

Summaries should be approximately 750-1200 words in length (tables and figures counted as 150 words each). They should present facts that are new and significant or represent a state-of-the-art review. Proper references should be included for all closely related published information. Summaries should contain:

- an introductory statement indicating the purpose of the work
- a description of the work performed
- the results achieved

NOTE

For a paper to appear in the Conference Proceedings, at least one of the authors must register for the Conference by the deadline for receipt of full papers (2004 August 15).

Submission Procedure (Summaries and Full Papers)

Submissions should be made electronically, preferably in MS Word format, through the Conference web page at:

http://www.cns-snc.ca/simulation2004.html

Technical Program Chair

Prof. Jean Koclas
Directeur, Institut de Génie Nucléaire
Ecole Polytechnique de Montréal
e-mail: jean.koclas@polymtl.ca
Tel.: (514) 340-4711 ext. 4263

General questions regarding the Conference

Denise Rouben, CNS Office Manager
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Tel.: 416-977-7620

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BRANCH ACTIVITIES

Chalk River *(Morgan Brown)*

The Branch held an Annual General Meeting on October 9, 2003 and elected the following executive:

- Chair: Morgan Brown
- Vice-Chair: vacant
- Treasurer: Bryan White (until his retirement)/ Marcel Heming (thereafter)
- Program Coordinator: Michael Stephens o Members-at-Large: Bill Bourns, Blair Bromley, Uditha Senaratne
- Past Chair: Michael Stephens

Following the AGM, Dr Romney Duffy, Principal Scientist at AECL, spoke on the subject of his recent book co-authored with Dr John Walton Saul, entitled: "Know the Risk: Learning from Errors and Accidents: Safety and Risk in Today's Technology".

Blair Bromley is heading up the essay contest (see the link from the CR branch web page or from the News page on the web - when it's up and running again).

On November 27 Dan Meneley and Malcolm Lightfoot, spoke about the CANTEACH program.

Darlington *(Jacques Plourde)*

Because of workload at the station the Darlington Branch has been relatively inactive. However, Jacques Plourde, Branch chair, assisted by John Luxat is developing plans for the re-activation of the Branch following the November planned outage.

Together with Dr. George Bereznai, of the University of Ontario Institute of Technology a joint CNS/UOIT activity is tentatively scheduled for March 4, 2004 during Engineering Week.

A membership drive will begin shortly and a Branch General Meeting is planned for mid-January. Tentative plans for the first part of 2004 include:

- Jan: General Meeting - Gregory Smith tentative speaker
- Feb: New Executive in Place
- Mar: UOIT Eng Week 'Nuclear' Day - Jeremy Whitlock tentative speaker
- Apr: Luncheon Session at Darlington - speaker TBD
- May: Evening Session & Darlington Tour - speaker TBD
- Jun: Golf Tournament

New Brunswick *(Mark McIntyre)*

On Tuesday, November 25, 2003, the New Brunswick Branch organized a talk at the Saint John Regional Library by Debbie Boudreau, past principal of the Nuclear Medicine program at the New Brunswick Community College on Practical Radiation Dose Responses.

The NB Branch Executive numbers have decreased due to retirements and that leaves a couple of openings. The NB Branch will be posting a call for volunteers among members soon.

The NB Branch is also planning a membership drive for December 2003.

Ottawa *(Bob Dixon)*

On October 16, Murray Stewart gave an updated presentation on Fusion energy and ITER: An Exciting Opportunity for Canada. Sadly Murray commented that without a commitment from the federal government the bid to have the ITER project located in Canada is in jeopardy.

On November 27, Fred Boyd gave a talk about the Early Days of the Canadian Nuclear Program, which was based on one he presented in Deep River last January.

Branch members participated in the one day Royal Society of Canada symposium on Energy, Environment and Society, November 25.

Quebec *(Michel Rhéaume)*

Successful joint CNS and IRH conference was held in Trois-Rivières on October 29th. About 30 attendees coming from the Université du Québec à Trois-Rivières and from the Québec Branch Members. For the first one in the Trois-Rivières area this is a good start.

Toronto *(Bob Hemmings)*

The Toronto Branch has been rejuvenated with a new executive:

Chairman:	R L Bob Hemmings (Canatom)
Vice-Chair	Nima Safaian (NSS)
Treasurer	Andrew Lee (ret'd)
Web Guru	Dan Quach (OPG)

The Branch has organized two programs for the fall of 2003.

On November 18th Dr Jeremy Whitlock, President of the CNS, gave a dual presentation on: What is the CNS to You? and The New Maple Reactors at Chalk River

On December 12th Dr Ken Petrunik, AECL, is scheduled to talk on: The International Success of the CANDU 6 - CANDU New Build (CANDU 6 and the evolutionary ACR)--a Potential Answer to Ontario's Power Needs

The Branch has agreed to provide support to the 2004 Sci-Tech Fair in Toronto, with $100 in prizes and $300 in support of the fair organization, and, and is well on the way to finalizing the Spring schedule of presentations and speakers.

Canadian Nuclear Society Twenty-Fifth Annual Conference

2004 June 6-9

Toronto Marriott Eaton Centre, Toronto, Ontario, Canada

Call for Papers

The Canadian Nuclear Society's 25th Annual Conference will be held in Toronto, Ontario, Canada, 2004 June 6-9, at the Marriott Eaton Centre in downtown Toronto.

The main objective of the Conference is to provide a forum for discussion and exchange of views on the technical aspects, challenges and opportunities for nuclear technology in what appears to be a renaissance for nuclear power. As usual, papers are solicited on technical developments in all subjects relating to nuclear technology.

Conference Web Page **http://www.cns-snc.ca/conf2004.html**

Deadlines

- **Receipt of summaries: 2004 January 18.**
- **Notification of acceptance: 2004 February 15.**
- **Receipt of full papers: 2004 April 18.**

The full paper may also be submitted by the January 18 deadline, in which case no summary is required. This one-step process can shorten the time required for the internal review of papers by the authors' companies.

Guidelines for Submissions

Summaries and full papers should present facts that are new and significant or represent a state-of-the-art review. Proper reference should be made to all closely related published information.

Summaries should be approximately **750-1200** words in length (tables and figures counted as 150 words each). They should include:

- an introductory statement indicating the purpose of the work
- a description of the work performed
- the results achieved

Full papers should include enough information for a clear presentation of the topic. Usually this can be achieved in 8-12 pages, including figures and tables. The use of 12-point Times New Roman font is suggested. The name(s), affiliation(s), and contact information of the author(s) should appear below the title of the paper. **An abstract of 50-100 words should be placed at the beginning of the full paper, after the title and author names.** Abstracts will be collected in an Abstract Book as a guide to the contents of the presentations.

Copyright in papers or written submissions to CNS events such as conferences, workshops, seminars, or courses remains with the author but the CNS may freely reproduce it in print, electronic or other forms. The CNS retains a royalty-free right to charge fees for such material as it sees fit.

NOTE: For a paper to appear in the Conference Proceedings, at least one of the authors must register for the Conference by the "early" registration date (2004 May 1).

Submission Procedure (Summaries and Full Papers): The required format of submission is electronic (Word or pdf). Submissions should be made through the Conference web page.

Questions regarding papers and the technical program: e-mail: cns2004@aecl.ca

General questions regarding the Conference **Denise Rouben, CNS Office Manager**
e-mail: cns-snc@on.aibn.com
Tel: 416-977-7620

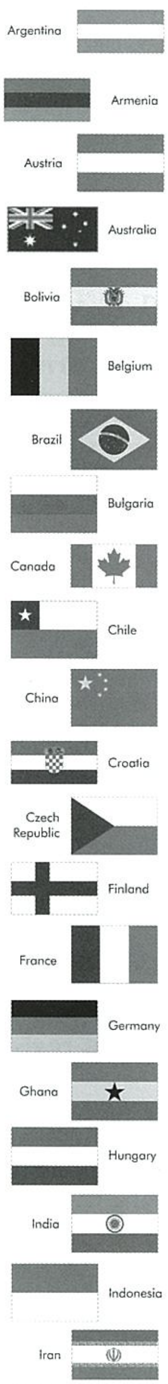
International Youth Nuclear Congress

YOUTH, FUTURE, NUCLEAR

May 9-13, 2004

in

Toronto *Canada*



What is IYNC?

We, as representatives of the new generation of professionals in the nuclear field agree to seize the opportunity of gathering together by creating an international network, the "International Youth Nuclear Congress", to:

Develop new approaches to communicate benefits of nuclear power, as part of a balanced energy mix.

Promote further peaceful uses of nuclear science and technology for the welfare of mankind.

Transfer knowledge from the current generation of leading scientists to the next generation and across international boundaries.

IYNC Past and Future

Bi-Annual Event:

2000 - Bratislava, Slovakia
2002 - Daejeon, South Korea

Each Enjoyed:

250-350 attendees from 40 countries
15-20 major international sponsors
20-30 prominent keynote speakers
75-100 technical papers and posters

The 3rd IYNC will be held May 9-13, 2004, hosted jointly by the CNS and NA-YGN, and promises to continue the proud tradition and ideals of IYNC.

Important Dates for IYNC 2004

Summary submission deadline	September 30, 2003
Notification for oral or poster presentations	December 20, 2003
Early registration deadline	March 1, 2004
Paper submission deadline	March 15, 2004
Early rate hotel booking deadline	April 9, 2004

For More Information

International Youth Nuclear Congress	**www.iync.org**
Alexandre Tsiboulia, IYNC Network Chair	**alexts@iync.org**
Adam McLean, IYNC 2004 Local Chair	**adam.mclean@utoronto.ca**

Once Upon A Time And Again

by Jeremy Whitlock

Physicists are a hard lot, it is well known. Rough and bawdy, loyal to the core and fiercely fraternal.

Here in Unit 2 of Canada's newest reactors, that old MAPLE mojo is spinning its magic like there's no tomorrow – or maybe it already is tomorrow. The light and space have put a zap on the weird scenes inside this gold mine.

All the greats are in attendance, for this is hallowed ground and going first-critical is a hallowed ceremony. In the corners, by the cabinets, laughing it up by the I-beam, the ghosts of Reactors Past have come to the party. It was here, on this very spot, over half a century ago, that Canada split its first self-sustaining chain-reacted atom.

ZEEP is somewhere else now. Cold, broken, cast aside. The old timers remember, and through wreaths of cigar smoke the stories are retold. Over there, by Shutdown System Two, three of them sit playing cards and shooting the bull, under a gaudy mural of Champlain and his astrolabe. It's happy hour at the Byeways but they wouldn't have missed this moment for the world.

Across the room a grizzled pair relax like they live in their chairs, sharing a single malt and singing songs.

In days of old,
When men were bold,
And neutrons weren't invented,
We stoked our fires,
With lone desires,
And felt ourselves contented.

Here is W.B. Lewis drawing an approach to criticality on the blackboard. George Laurence goes by with a sack of graphite. Gib James is fixing something behind the control console. Harry Thode shares a joke with Bertrand Goldschmidt, and Allan Nunn May looks like he has a secret.

Rutherford bleat,
That as for heat,
These atoms were a Bohr,
Then Meitner's muse,
Lit Fermi's fuse,
And the beggars won a war.

The nuclear giant is coming unshackled, bursting forth its milliwatts of power. You can barely see in the control room for the cigar smoke. The card players call a pause and look over, whispering their own suppositions as to why it's taking so long.

Behind a pair of size 10 faded yellow shoes propped on a bank of scalars, a tuneful lament emanates from one not far from unconsciousness.

Millie Kay, Millie Kay,
Dance for me,
You're hardly worth a dollar,
But I'll give you three.

A fight now between two bespectacled lads over the best way to calculate reactivity. Lew Kowarski wades in and breaks it up. Kowarski is here looking for his own reactor, asking how you spell Quonset.

It is especially fitting that Kowarski be present. In his honour a reverse-Polish calculator sits on the main console, untouched.

The moment of truth arrives, passes, and is gone. A trio of old NRX hands by the door are still laughing giddily over a joke about MAPLE's size, and miss the occasion.

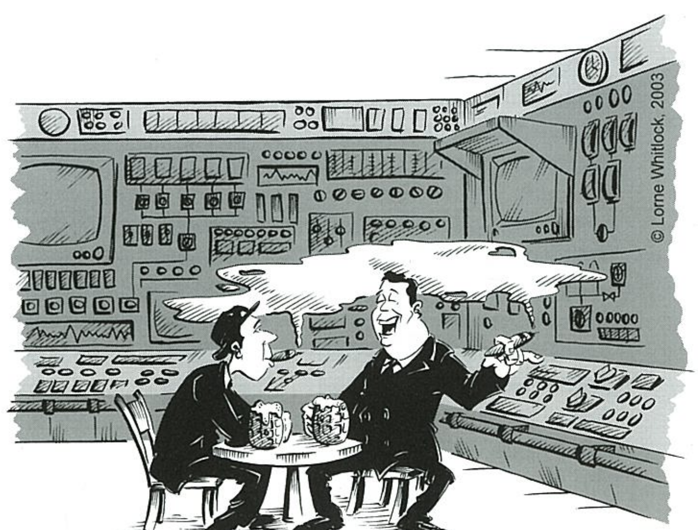
Yip hey, yip ho,
Keep them neutrons a-go,
Keep them rolling and twisting,
And send that flux high.

Some linking of arms, some spilling of beer. A flashbulb goes off and the moment is frozen in time.

Yip hey, yip ho,
Out the header below,
Where the Ottawa River flows by.

The milestone put to bed, the room empties like a QA course at coffee break. They disappear on skis, on bicycles, and some go looking for their old bus. They're surprised to find it still in use.

But the Old Guard won't be gone for long. For old physicists never die; they just keep redefining the problem.



CALENDAR

2004

Feb. 18, 19	**CNA Nuclear Industry Seminar** Ottawa, Ontario website: www.cna.ca
Mar. 28 - 31	**10th International Topical Meeting on Robotics and Remote Systems** Gainesville, Florida website: www.ans.org/meetings/robotics
Mar. 21 - 25	**PBNC 14** **14th Pacific Basin Nuclear Conference** Honolulu, Hawaii website: www.ans.org/meetings/pbnc
Apr. 25 - 29	**PHYSOR 2004** Chicago, Illinois website: www.td.anl.gov/PHYSOR2004
May 2 - 6	**International Topical Meeting on Advanced Nuclear Installation Safety** San Francisco, California website: www.ans.org/meetings
May 9 - 14	**10th International Conference on Radiation Shielding** Maderia, Portugal website: www.itn.mces.pt/ICRS
May 9 - 13	**IYNC3** **3rd International Youth Nuclear Congress** Toronto, Ontario Contact: Adam McLean e-mail: adam.mclean@utoronto.ca
June 6 - 9	**25th CNS Annual Conference** Toronto, Ontario Contact: Denise Reuben Canadian Nuclear Society Tel: 416-977-7620 e-mail: cns-snc@on.aibn.com
June 13 - 17	**ANS Annual Meeting** Pittsburgh, Pennsylvania Contact: American Nuclear Society website: www.ans.org
June 13 - 17	**International Congress on Advances in Nuclear Power Plants** Pittsburgh, Pennsylvania website: www.ans.org/goto/icapp04
Aug. 22 - 26	**SPECTRUM 2004** Atlanta, George website: www.ans.org/spectrum
Sept. 22 - 26	**4th International Topical Meeting on Nuclear Plant Instrumentation, Control and Human Machine Interface Technology (NPIC & HMIT 2004)** Columbus, Ohio website: www.ans.org
Oct. 13 - 15	**6th International Conference on Simulation Methods in Nuclear Engineering** Montreal, Québec website: www.cns-snc.ca/simulation2004
Nov. 14 - 18	**ANS Winter Meeting** Washington, D.C. website: www.ans.org

CNS Website

CNS members and others accessing the Society's website (www.cns-snc.ca) will probably have noticed the improvements and expansion over recent months.

Some technical changes have also taken place. The CNS server has now been moved to the COG offices in Toronto (where the Society sub-leases its office space) from Chalk River and a contract made with COG to have their IT staff service it. The move required shutting off service, re-aligning domain name servers with new address, moving computer, re-starting and loading software. Through the efforts of the CNS Webmaster, Morgan Brown, with assistance from CNS president, Jeremy Whitlock, and others, the site was down for only five days.

With the new set-up Morgan is looking into better access for CNS Branches and other units of the Society to look after their own pages and provisions to allow payments, such as membership dues, to be made securely by credit card.

In the last six months the site has averaged 44 hits per day. Servers in Canada and from around the world are logging on to the CNS site.

Over the past few months a number of new pages have been added. These include several in French with the help of Jaro Franta. The history section has been expanded, including a page on Gord Brooks, former chief engineer at AECL, who died in September. Morgan, who is also the CNS' historian, has placed a remarkable amount of Canadian nuclear history on the CNS website, making it the best source for such information.

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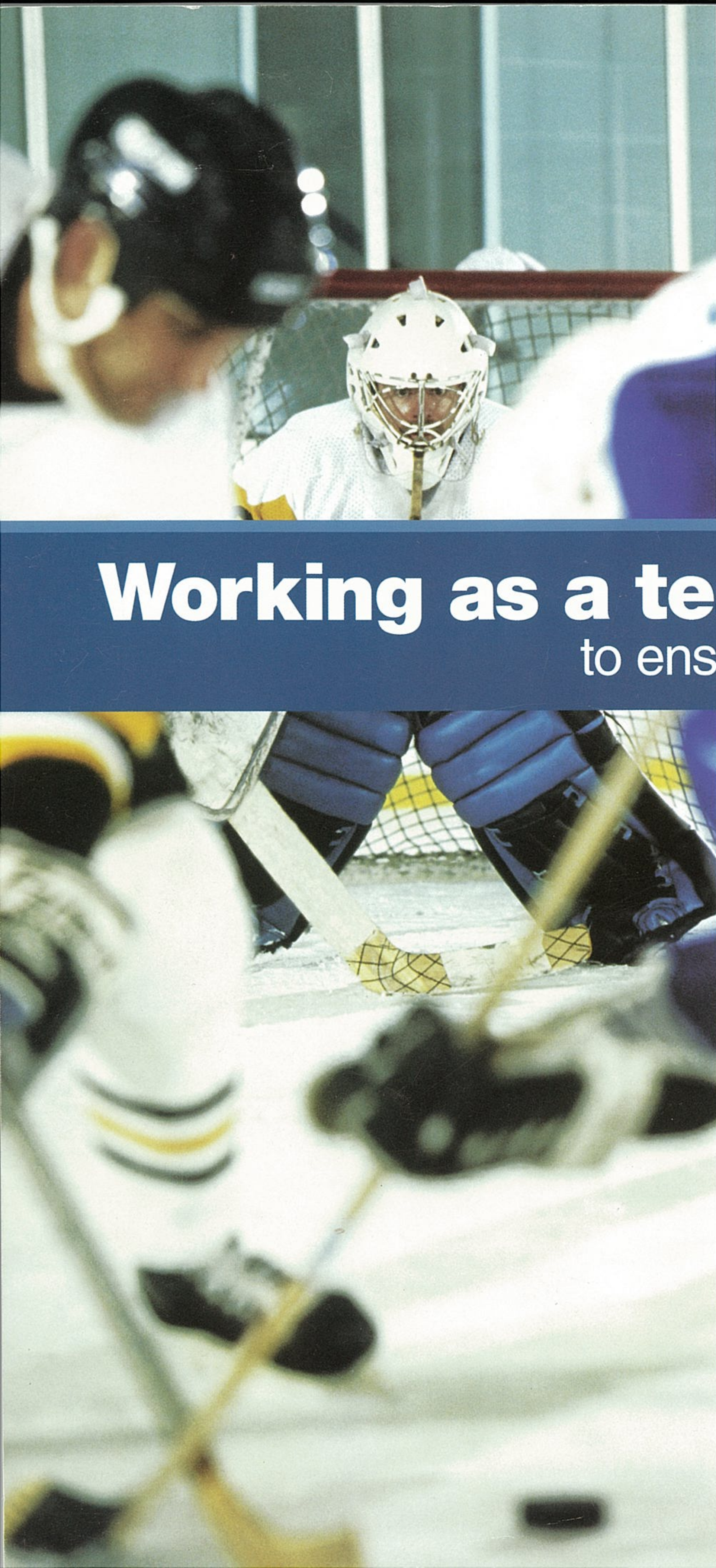
CNS WEB Page

For information on CNS activities and other links

http://www.cns-snc.ca

adriaan buijs



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