Canadian Nuclear Society

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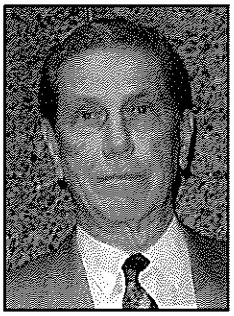
- Simulation Conference
- Point Lepreau Refurbishment
- Proposed Licensing Basis
- New Waste Repository
- Canadian Light Source
- Generation IV

EDITORIAL

Encouraging news, but...



The past couple of months have brought some positive developments to the Canadian nuclear scene.

Among those was the choice of members of the new Board of Directors of Ontario Power Generation. Four of the new members have senior level backgrounds with the nuclear industry and some of the others have experience in the energy field or related industries.

What a marked difference to the ineffective Boards of Ontario Hydro in the 1990s whose behaviour almost killed the organization and, with that, Canada's nuclear program. Hopefully, this new Board will provide real oversight of OPG and, in particular, its nuclear operations.

Another welcome event was the announcement that the new Advance CANDU Reactor will be one of two designs to be supported by the U.S. Department of Energy in its new pre-licensing program.

Then, there is the apparent agreement for a waste repository at the Bruce site. Although this is just for low and intermediate level radioactive wastes it is a marked step forward as an example of how it is possible to obtain public support in this contentious area.

However, not all news is positive. The Maple project continues to drag on, now, reportedly, focussed on the question of whether or not the reactor has a positive power coefficient. The fact that the miniscule size of the coefficient, whether positive or negative, has no real bearing on the safety of the reactor appears to have escaped the regulator and the proponent.

In the area of radiation protection it appears, from the Forum reported in this issue, that not only are we in for more of the same, with continued emphasis on the linear no threshold concept, but are likely to be faced with specific regulations for "non-human" biota (plants and animals). We see the day when our uranium mining companies will be spending much of their time trapping and fishing.

And, there have been unofficial reports that the rehabilitation of Pickering 1 is already over budget. Let us hope that is not the case and that OPG has finally learned how to manage these difficult jobs.

But, in perspective, it is good to see some things going right.

Fred Boyd

IN THIS ISSUE

Much of this issue is drawn from the successful **6th International Simulation Conference** held in Montreal in October, beginning with a short report on the event and followed by the opening plenary paper **Refurbishment of Point Lepreau.**

Two other papers also originated at that conference. One was another plenary presentation **Generation IV Power for the Future.** The other is one of the technical papers, **Bruce A Restart Phase B Commissioning Physics Tests.**

The final borrowing is the thoughts of Dan Meneley in his talk at the conference dinner which he titled, **Now that we have arrived, where do we go?**

Then, turning to another theme, there is a paper on **Proposals for a New Canadian Licensing Basis,** which is basically the executive summary of a report from consultants Allan Brown et al to the Canadian Nuclear Safety Commission.

The next article reports on a significant development in the on-going challenge of waste management, **Waste repository planned for Bruce site.**

There is a short report on a gathering in Ottawa in early November dealing with radiation protection, **Forum discusses ICRP draft recommendations.**

The last article is a report on the official opening of the **Canadian Light Source,** the impressive new synchrotron facility on the campus of the University of Saskatchewan.

There is the typical eclectic selection of items in **General News,** followed by what has unfortunately become a usual item, an **Obituary,** this time for **Frank Stern.**

The **CNS News** section, again prepared by Bryan White, provides a good overview of a very active Society.

Finally, there is the special view of our world in **Endpoint** by Jeremy Whitlock

Again this issue has been assembled against a background of some personal strains. Please overlook any omissions or errors. Nevertheless, we hope you find something of interest and invite your comments and submissions.

Contents

Cover Photo

The cover photograph is a view looking north of the Bruce site, the location of the proposed Deep Geologic Repository for low and intermediate radioactive waste.

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6th International Simulation Conference

Hong Huynh, Conference Chair

The **6th International Conference of the Canadian Nuclear Society on Simulation Methods in Nuclear Engineering** drew over 100 specialists from 11 countries to Montreal, October 12 to 15, 2004 to share approaches and techniques in the use of computer simulations to analyse nuclear reactors. Representatives from a 12th country were unable to attend because of visa problems. Countries represented were: Argentina, Japan, Korea, Lithuania, Mexico, Spain, Sweden, Turkey, UK, USA, (and Canada).

As suggested by the title, most of the papers and presentations focussed on the analysis of difficult engineering problems associated with nuclear reactors with the aid of complex computer programs. Many included comparison of calculations with observations from experiments or actual plant operations. A measure of the advances that have been made in simulation methods was the good agreement between calculation and measurement, a fact that has led to increasing reliance on simulation methods.

On the Tuesday evening, October 12, a reception offered the opportunity for delegates to meet, some for the first time after long periods of sharing information by e-mail.

The conference proper opened on the Wednesday morning, October 13 with a plenary session of six presentations related to, but not directly on, the conference theme. The first plenary paper was given by Paul Thompson, on *Refurbishment of Point Lepreau Generating Station.* He reviewed the history of the station, the considerable work done to date and the planned work for a major refurbishment of the plant in 2008. *(His paper is reprinted in this issue of the* CNS Bulletin.)

The other plenary papers were:

- *Constitutive Model for Reinforced Concrete Applied in the Analysis of the Gentilly-2 Reactor Building*
  by V. Gocevski, Hydro Québec
- *LWR Reactivity Accident analysis and its Significance in the U.S. Regulatory Process*
  by D.J.Diamond et all,
  Brookhaven National Laboratory
- *Approach and Methods to Evaluate the Uncertainty in Systems Thermalhydraulic Calculations*
  by F. D'Auria, University of Pisa
- *Implementation of Low Void Reactivity Fuel in Bruce B*
  by R. M. Chun, F. C. Iglesia, et al, Bruce Power
- *Generation IV Power for the Future: Status of the SCWR*
  by R.B.Duffy, AECL

There were two luncheons and a banquet included in the conference program, with a speaker at each.

At the luncheon on the first day, Michel Beaudet, Chef Sûreté nucléaire, Gentilly 2, Hydro-Québec, spoke about *Energy Supply and Demand in the Québec Scene.* HQ is the largest generator of hydro electricity in the world, he stated. The corporation is now divided into three distinct operating organizations: Production, TransEnergie (trans-

Venue of 6th Simulation Conference in Montreal.

mission), and Distribution. There is a steady growth of demand, which is expected to reach 191 TWHr by 2011. Plans are underway for the refurbishment of the Gentilly 2 station in 2010 / 2011, however there are many steps still to go before final government authorization is given. Public hearings will begin in 2005.

Dan Meneley, former chief engineer at Atomic Energy of Canada Limited and now (semi-retired) director of the CANTEACH program, was the invited speaker at the banquet. His topic was: *"Now That We Have Arrived, where Shall We Go?"* A slightly edited version of his remarks is included in this issue of the CNS Bulletin.

CNS President Bill Schneider addressed the group at the luncheon on the final day of the conference. He reviewed briefly the history of the Canadian Nuclear Society since its creation 25 years ago as the technical society of the Canadian Nuclear Association, its incorporation as a legally separate entity six years ago and its current activities. He noted, in particular, three new courses being offered by the Society in the late fall of 2004 and early part of 2005.

The titles of the various sessions give some indication of the scope of topics addressed:

- Codes and Modelling
- Safety Analysis
- FuelChannels
- Simulator
- Operations Support
- Reactor Physics
- Thermalhydraulics
- Neutronics Methods
- Containment

This successful conference was organized by a large committee chaired by Hong Man Huynh of Hydro Québec (and a former president of the CNS). The technical program was put together by Jean Koclas, Laurence Leung and Eleodor Nichita. Others involved were: John Tong, John Luxat, Ajit Muzundar, Marv Gold, Ron Aboud, René Girard, Ben Rouben, Siamak Kaveh, Ovidiu Nainer, and Monique Ip. Denise Rouben, Isabelle Beaulieu and Melissa Boyd handled registration and other local matters.

A CD with the full text of most papers is available from the CNS office.

This series of conferences has typically been held every two years. The venue and date for the next one are still being discussed.

Delegates mingle prior to the conference banquet

Refurbishment of Point Lepreau Generating Station

by P.D. Thompson[1], M.A. Petrilli[2], Raj Jaitly and N. Ichiyen[3]

***Ed. Note:** The following paper was the first presentation of the opening plenary session of the 6th International Conference on Simulation Methods in Nuclear Engineering in Montreal, Quebec, October 13, 2004.*

ABSTRACT

NB Power is planning to conduct an 18-month maintenance outage of the Point Lepreau Generating Station (PLGS) beginning in April 2008 (Reference-1). The major activity would be the replacement of all 380 Fuel Channel & Calandria Tube Assemblies and the connecting feeder pipes. This activity is referred to as Retube (Reference-2). NB Power would also take advantage of this outage to conduct a number of repairs, replacements, inspections & upgrades (such as rewinding or replacing the generator, replacement of shutdown system trip computers, replacement of certain valves & expansion joints, inspection of systems not normally accessible, etc). These collective activities are referred to as Refurbishment. This would allow the station to operate for an additional 25 to 30 years.

The scope of the project was determined from the outcome of a two-year study involving a detailed condition assessment of the station that examined issues relating to ageing and obsolescence (Reference-3).The majority of the plant components were found to be capable of supporting extended operation without needing replacement or changes. In addition to the condition assessment, a detailed review of Safety & Licensing issues associated with extended operation was performed. This included a review of known regulatory and safety issues, comparison of the station against current codes and standards, and comparison of the station against safety related modifications made to more recent CANDU 6 units.

Benefit cost analyses (BCA) (Reference-4) were performed to assist the utility in determining which changes were appropriate to include in the project scope. As a Probabilistic Safety Assessment (PSA)for PLGS did not exist at the time, a risk baseline for the station had to be determined (Reference-5) for use in the BCA. Extensive dialogue with the Canadian Nuclear Safety Commission staff was also undertaken during this phase. A comprehensive Licensing Framework was produced upon which the CNSC provided feedback to NB Power. This feedback was important in terms of achieving clarity of the regulatory position and thus to minimize the financial risk associated with regulatory uncertainty.

The Refurbishment outage is preceded by a detailed Engineering Project Phase that includes:

- Finalizing details of the Retube process including modeling, tooling development, site facilities and training of personnel
- Perform Engineering activities related to design modifications, safety analysis and level II PSA
- Construction of new waste storage structures to house Retube Waste and other additional waste storage structures for the extended life of the station[4]
- Setup necessary temporary construction facilities (offices, storage areas, change rooms, decontamination and maintenance areas)
- Perform detailed outage planning
- Initiate development of detailed lay-up, commissioning and return to service procedures
- Procure equipment & components

Although final project approval is still pending[5], NB Power has been carrying on a limited scope of activities that are important in reducing overall project financial risk. A number of these up-front activities relate to safety analysis and licensing issues related to life extension. In particular, a level II PSA along with additional safety analyses are being performed to complement that which currently support the existing Operating Licence for the station.

This paper discusses the Safety & Licensing activities that were involved in defining the project scope and outlines the safety analysis related activities that will be performed in support of the Refurbishment project and extended operation.

Background:

The Point Lepreau Generating Station is a 680 MW(e) CANDU-6 reactor located in the province of New Brunswick on the Atlantic coast of Canada. It is owned and operated by NB Power, which is the provincial utility[6]. PLGS was constructed between May 1975 and the summer of 1982. The initial Operating Licence was granted in July of 1982, with

1 NB Power, Point Lepreau Generating Station
2 MAPSAN
3 Atomic Energy of Canada Limited
4 An environmental assessment for these additional structures was conducted and approved as part of the projects early start activities.
5 The final decision on project approval is expected in late 2004.
6 On Oct. 1, 2004 NBP was restructured into a holding Company (NB Power Holding) and four subsidiary companies, one of which is NB Power Nuclear company

commercial operation beginning on February 1,1983. The Station has proven to be an economic and environmentally sound source of electricity generation and has achieved a lifetime capacity factor of 82.9%[7]. The station provides about a third of the power consumed in the province of New Brunswick and has a significant positive economic impact in the southern part of the province, employing over 600 people and having an annual operating budget of over 100 Million dollars. In addition the station is an important element in achieving provincial environmental emission targets.

Although the station continues to perform well, key reactor components (the pressure tubes and feeders) are nearing the point in time in which they will need to be replaced. Although pressure tubes and feeders can be replaced on an individual basis, the number of tubes requiring replacement increases significantly starting about 2008 –2010, making the economics of continued operation during this time less and less favorable. For this reason the refurbishment outage is planned to start in April 2008.

The original design of the station anticipated an operating period of about 30 years. This set the basis for the assumed number of transient cycles and the number of full power running days that equipment could either experience or be exposed to, and thus had to be designed to withstand[8]. This implies that PLGS would reach the end of the planed period of operation around 2013. Because of the significant cost associated with retubing the reactor, there was a need for the refurbishment outage to extend the period of operation out another 25 to 30 years. In addition, a number of new regulatory requirements have been issued since the station was granted its first operating licence. These issues drove the need to perform the comprehensive condition assessment and the safety & licensing reviews to determine the scope and cost of refurbishment, to ensure the station could be operated over the extended period, and to understand the regulatory aspects.

Regulatory Climate

The station Operating Licence is renewed on a periodic basis[9] . This drives an on-going process at the station related to updating the design, safety analysis and station programs. As a result, in response to emerging issues, a number of safety related design changes have been introduced over the years. Similarly, a number of station programs and processes have likewise been introduced. There is also an on-going Safety Analysis program. Safety Analysis is performed on an as needed basis in response to proposed design changes, internal and external operating experience (such as response to plant ageing, unanticipated plant response to a transient, unexpected research and development or analysis findings, plant events, improving the definition of safe operating envelope), and safety related issues raised by the regulator under an "Action Item" process. When new analysis is performed, it is conducted with current methods. The analysis is summarized in the station Safety Report. This report is reviewed and updated on a 3-year basis. Thus over time, much of the analysis in the Safety Report has been redone and updated with modern methods.

The periodic licence renewal and continuous safety improvement process used in Canada easily accommodates extending the planned period of operations (sometimes referred to as "life extension" in other jurisdictions). The central safety and licensing items performed in support of refurbishment were as follows:

- Identify additional safety related design changes through review[10] of:
  - Comparison of station against current codes & standards
  - Safety related changes introduced at newer CANDU 6 stations
  - Outstanding safety issues where the optimum solution would be a design change that is not likely to be introduced unless there was an extended station outage followed by prolonged period of station operation
  - Results of Level II PSA against set of established goals and targets

- Identify additional Safety Analysis to be conducted through consideration of:
  - Events not included in the Safety Report but required for stations licensed to more recent regulatory documents
  - Conditions expected with refurbished and fresh core
  - Design changes to be introduced during refurbishment
  - Support to the level II PSA

- Assess plant condition via a comprehensive Plant Condition Assessment
- Assessment of equipment and cable qualification over the extended operating period
- Determine the changes to Operating Policies & Principles to cater to the defuelled core state

7 Capacity Factor for in-service since March 1983 up to the end of 2003.

8 Based on the condition assessment review, it was identified that the number of actual plant transients and cycles experienced were far less than the number assumed in the original design and in most cases sufficient to allow an additional 25 to 30 year operation.

9 Historically this was based on a nominal 2-year renewal period. Recently longer licence periods have been introduced with the period related to demonstrated safety performance.

10 Safety significant issues raised during the comparison reviews then under went a benefit cost analysis to determine whether or not a change was warranted.

- Development of a benefit cost analysis process to assist in decision making (back fit issues)
- Conduct a level II PSA to replace the earlier Safety Design Matrix studies (Reference 6)
- Perform Environmental Assessment for additional structures to be built and operated at the on-site Solid Radioactive waste facility to support the refurbishment and extended operations (References 7 and 8)

Guiding principles related to safety & licensing were established at the onset of the project and presented early to the regulatory authority (the Canadian Nuclear Safety Commission - CNSC). These guiding principles evolved into a detailed licensing framework for the project. To achieve regulatory clarity, the regulator was asked to provide comments on the licensing framework. This then allowed for focussed discussions. The central principles are provided in Appendix I.

Key steps in the regulatory interaction were as follows:

- Understanding of the importance of achieving regulatory clarity related to the project and extended operations
- Establishment of primary contacts
- Communication of guiding principles
- Outlining overview of project execution plan (scope, timeframe, interfaces etc. relating to project management, QA, design, assessments, outage activities, fuel loading, commissioning and restart)
- Discussions on safety reviews to determine design changes and analysis
- Discussion on key design and safety analysis issues such as shutdown system modifications, fuel channel design, PSA methods goals and targets, etc.

Although the station condition and plant processes and programs are subject to on-going reviews and inspections from the CNSC as well as from WANO and the insurance brokers, CNSC staff determined that a review along the lines of the IAEA Periodic Safety Review should also be performed. Because a good portion of such a review had already been completed through the condition assessment and the comparison of the plant to current codes & Standards, NB Power choose to perform an Integrated Safety Review (References 9 and 10).

The remaining portion of the paper outlines the PSA and Safety Analysis scope related to the refurbishment:

PSA

The plan for producing the PSA's calls for the production of a level II PSA for internal events as well as for External events involving internal fires and internal floods. In addition, the shutdown state PSA for internal events and a seismic margin assessment will also be produced.

The main objective of the PSA is to provide insights into plant design and performance, including the identification of dominant risk contributors and the comparison of options for reducing risk to verify that the Point Lepreau refurbished station will meet the currently internationally accepted safety goals.

The main tasks associated with the level II PSA are as follows:

Level I PSA – Internal Events:

The internal events PSA starts of by systematically identifying a number of initiating events that cause a plant disturbance and may potentially lead to core damage. By considering the similar plant response the initiating events are grouped and for each group an event tree is drawn indicating the required mitigating systems and operator actions to bring the plant to a safe stable state, or which otherwise lead to core damage. The probability of mitigating systems being unavailable or failing during the mission period is estimated by fault tree analysis. Fault trees are developed to identify combinations of individual component failures that can cause the system failures modelled in the event trees. All of this information is then synthesized to perform a quantification of the frequencies of the accident sequences shown in the event trees. Main elements of the Level I internal events PSA are noted below:

- Initiating event analysis to establish a comprehensive listing of internal initiating events for on power as well as the shutdown state.
- Develop a plant-specific dependency matrix to gain a full understanding of the dependencies, which exist between plant systems, as well as initiating events and plant systems. Two dependency matrices will be developed, one system-system matrix, and one initiator-system matrix
- intermediate size event tree / fault tree linking approach to defining accident sequences)
- Develop fault trees for the plant design.
- Incorporate Common Cause Failures (CCFs) in the fault tree analysis using the Unified Partial Method (UPM).
- Perform Human Reliability Analysis (HRA) related to pre accident as well as post accident operator actions. This analysis methodology is consistent with ASEP, which is a simplified version of the more analysis-intensive THERP method, developed by the US NRC.
- Perform Accident Sequence Quantification (ASQ) to evaluate frequency of the core damage related end states in each event tree.
- Perform uncertainty analyses on parameters such as: failure rates, component unavailabilities, initiating event frequencies, and human error probabilities. The uncertainties for each of these quantities will be expressed in terms of probability distributions about their mean or best-estimate values.
- Perform sensitivity analysis to test the impact of certain changes in key input values (different maintenance practices, testing intervals, mission times, etc.) to PSA results.

Level I PSA – External Events and Seismic Margin Assessment:

Based on previous CANDU experience with PSA for external events, the events selected for Lepreau site are seismic, internal fires and internal floods due to failure of the large piping/expansion joints in the RSW and/or CCW systems. For the Lepreau site, this list of external events is judged to be sufficient. The external events PSA will rely heavily on the Level I PSA models (event trees and fault trees) for the internal events summarized above. Main elements of the Level I external events PSA are noted below:

- Plant Walkdowns: Plant walkdowns are used to both verify and supplement the information contained in the fire and flooding database. The walkdowns also provide a greater understanding of the failure modes of structures, equipment due to spatial interaction during a seismic event.
- Develop a fire database to define "fire zones" and "fire areas", typically based upon the location of barriers to fire propagation and develop hazard scenarios for the various fire zones. Only fires occurring at-power will be considered. Scenarios may be qualitatively screened from further analysis, based on a variety of reasons such as: fire in an area does not cause a demand for plant shutdown, the area does not contain any safety-related equipment, small volumes of combustible materials in an area. The intention of the screening is to focus analysis efforts on the critical areas of the plant. For those scenarios not screened out, fire initiating event frequencies will be established based on the amount and type of fire sources in the area. The generic / CANDU-specific component-based and location-based data developed by AECL will be used for this task, in which the total fire frequency for each component type is apportioned to the various fire zones.
- Fire progression modelling will be performed using the computer code COMPBRN IIIe. This code has been used in various nuclear plant PSA applications and is currently maintained by EPRI and the USNRC.
- Accident Sequence Quantification (ASQ) and Recovery Analysis for core damage sequences resulting from fire.
- Flooding PSA: Develop i) Database to define flood areas typically based upon the location of flood sources, ii)Flood hazard scenario development for various flood zones and iii) Flood Progression Modelling including calculations of probability of water spray interactions and time to submergence calculations.
- Accident Sequence Quantification & Recovery Analysis for core damage sequence frequencies resulting from flooding.

A PSA based Seismic Margin Assessment will be performed as the US NRC recommends this approach in SECY 93-87, to avoid the problems encountered with seismic PSA's where the results of severe core damage frequency (SCDF) may be dominated by the uncertainties in the hazard curve. The Seismic Margin Assessment involves essentially performing all the steps of a seismic PSA (fragility analysis, event tress & fault trees) except convolution of fragilities with the hazard input. It thus provides all the design insights expected of a seismic PSA without making the results vulnerable to the large uncertainties typically encountered in site hazard input. Main tasks involved in performance of the Seismic Margin Assessment include:

- Establish seismic safety target defined in terms of plant HCLPF (High Confidence of Low Probability of Failure)
- Seismic fragility evaluation for structures and equipment which affect consequences or mitigation of the seismic event
- Perform Failure Mode and Effect Analysis for Seismic Failures
- Develop Plant Models
- Generate Minimal Cutsets for Seismic Core Damage Sequences
- Calculate the HCLPF value for each seismic core damage sequences
- The plant HCLPF is the lowest sequence HCLPF
- Uncertainty & sensitivity analyses for external events

Level II PSA

The Level II PSA will be performed to calculate the frequency and timing of various modes of containment failure that may cause releases of radioactive material to outside of the containment boundary. The Level II PSA takes as an input the core damage sequences from the Level I PSA, and calculates the frequency and timing of various modes of containment failure. As such, the Level II PSA will incorporate both probabilistic and deterministic aspects. Main tasks for the Level II PSA are summarized below:

- Severe Accident Plant Damage State Definition of SCDF sequences from Level I PSA results.
- Develop Plant Parameter File for severe accident progression and containment performance analysis: The consequence analysis will be performed with the use of the MAAP4-CANDU that is the industry standard codes for CANDU Level II PSA accident progression analysis. The code requires a large volume of input data on the plant systems and structures, such as geometries, instrumentation and control equipment set points, solid and fluid inventories and operating pressures and temperatures.
- Perform Sequence Analysis for representative plant damage state sequences by the MAAP4-CANDU code to model the accident progression and establish the timing and nature of any radionuclide releases.
- Containment Event Tree Development and Quantification: Containment event trees will be developed for each of the various severe accident plant damage states or for a group of plant damage states if their characteristics are sufficiently similar. These event trees model the systems

and physical phenomena, which have an impact on the containment response following a core damage accident. Information is taken from the MAAP4-CANDU sequence analysis, and plant damage state analysis to identify the final release category for the plant end-states.

Safety Analysis

The Point Lepreau Refurbishment safety analysis plan was developed based on:

- A systematic review of the initiating events, event combinations and event sequences that are described in the C6 rev. 1 (Reference-11). The events that are not covered by accident analyses currently in the Safety Report and which have a frequency higher than 1x10 -6 /year have been included in the safety analysis plan. Moreover, the common cause events relevant to the Point Lepreau site will be documented. The likelihood of various relevant common cause events will be specified, the inherent design features that provide protection against these events, and the contingency procedures to mitigate their consequences will be identified.
- A systematic review of the Safety Report to identify the accident scenarios whose consequence assessments could be affected by the design changes that are planned to be implemented during the refurbishment outage or by the plant conditions that will prevail after the refurbishment outage and which are not currently covered in the Safety Report, namely the use of fresh fuel and new uncrept fuel channels.
- A provision to perform the deterministic safety analyses required to support the PSA is included in the plan.

A summary of the main analyses included in the Safety Analysis Plan is provided below.

Large LOCA Analysis

The large LOCA analysis that will be performed in support of the Project aims at confirming the adequacy of the reference fuel channel design to accommodate fuel string expansion in the conditions that will prevail after refurbishment. The impact of pre-equilibrium core conditions on the large LOCA power pulse will also be assessed.

Shutdown System depth analysis for Pressure Tube/Calandria Tube rupture with loss of ECC with fresh fuel and Pressure Tube Rupture Trip Coverage

The objective of the pressure tube rupture analysis that will be performed in support of the Project is mainly to assess shutdown system number 1 reactivity depth for a pre-equilibrium core. The calculation will be performed for both fresh fuel and equilibrium core cases. Options to improve the reactivity depth, such as various initial fuel loadings of depleted Uranium fuel, will be assessed. The trip coverage analysis for pressure tube rupture will also be revised to take into account the high moderator level trip that will be implemented during refurbishment. This trip coverage analysis will be performed for various power levels and will account for the maximum amount of poison in the moderator during a restart following a long outage.

An assessment of the adequacy of the SDS #1 reactivity depth following a flow blockage occurring at full power will also be performed using a methodology similar to the one referred above for the analysis of the SDS # 1 reactivity depth following a pressure tube rupture. The analysis will assume that the maximum amount of poison that could be present in the moderator is equal to the amount that would be present during steady-state operation with the reactor at the plutonium peak. The maximum amount of poison that could be present in the moderator during a start-up following a long outage is larger than the value during steady-state operation and therefore would be more limiting. However, since flow verification tests are performed during reactor start-ups, it is considered not credible that a complete flow blockage could occur at high power during a reactor start-up.

Loss of Heat Transport System flow events

The objectives of the loss of forced circulation analysis that will be performed in support of the PLR Project are to assess the trip coverage improvement provided by the additional SDS2 high pressure trip instrumentation on PHT outlet headers 3 and 7, and to finalize the redesign of the trip setpoints and conditioning power levels for the low flow, low pressure differential and high reactor outlet pressure trips. The analyses will cover both pre-equilibrium and equilibrium fuel core configurations. The analysis will cover all the loss of forced circulation events currently included in Point Lepreau Safety Report. Coupled thermalhydraulic and 3-D physics simulations will be performed for the limiting cases.

Regional Over Power Trip analysis with new HTS conditions (to restore set-points)

The analysis of slow loss of reactivity control will be updated to determine the ROPT detector trip setpoints that will apply following the refurbishment outage. The critical channel power distributions for the various flux shapes will be recalculated using the thermalhydraulic conditions that are predicted to prevail after refurbishment. The channel and bundle power distributions used to calculate the setpoints applicable during the pre-equilibrium phase will be based on RFSP simulations. The setpoints for the equilibrium core will be based on current ripples.

Moderator Events

The review of C6 has identified that in addition to the slow moderator drain event currently included in Point Lepreau Safety Report, a number of other failures that can affect the moderator system have to be analysed. Among these other events, the most significant ones are the Loss of Service Water to the Moderator Heat Exchangers and the Loss of Moderator Circulation. The analyses of these accident scenarios will be performed. The slow moderator drain analysis, already included in the Safety Report, will be revised to account for the new trip on low moderator level. The analyses will cover both fresh and equilibrium fuel. Assessments of the consequences of the moderator temperature control, moderator heat exchanger tube failure and moderator cover gas system failure will also be performed.

Loss of End Shield Coolant, Flow and Heat Sink

The review of C6 has identified that more detailed analyses of the failures of the shield cooling system have to be included in the Point Lepreau Safety Report. An assessment of the consequences of the various following failures will be performed:

- Loss of end shield cooling inventory, including shield cooling single heat exchanger tube failure,
- Loss of end shield coolant flow, and
- Loss of end shield cooling heat sink.

The response of the end shield cooling system following each postulated accident will be assessed to determine if the potential exists for differential tubesheet deformation, which can damage fuel channels and/or shutoff rod assemblies. Based on the results of the thermalhydraulic calculations, stresses on the calandria assembly will be assessed.

Shutdown cooling events (including Failure of LRV during entrance in shutdown cooling mode)

The review of C6 revision 1 has identified that a number of events affecting the shutdown cooling system have to be analysed. More specifically the analysis of the various following events will be performed:

- Loss of primary coolant inventory during operation in the shutdown cooling mode, including shutdown cooling single heat exchanger tube failure,
- Loss of primary coolant flow during operation in the shutdown cooling mode,
- Loss of heat sink during operation in the shutdown cooling mode,
- Shutdown cooling isolation valves failure,
- Pressure relief valve failure during cooldown of the plant

Multiple boiler tube failure

The review of C6 has identified that the consequences of the failure of a large number of boiler tubes is currently not in the Point Lepreau Safety Report. Consistent with the methodology developed for Darlington and used for recent offshore projects, a simultaneous guillotine break of 10 steam generator tubes will be analysed. The analysis will include an assessment of:

- The trip coverage effectiveness,
- The thermalhydraulic response of the primary and secondary heat transport circuits,
- The radionuclide releases to the secondary side and from it to the environment, and
- Population dose.

All the analyses realized in support of the Point Lepreau Refurbishment will follow rigorous quality assurance requirements. They will be performed using the most up-to-date methodologies and computer codes that comply with current standards. The methodologies for performing those analyses are currently being finalized and discussions with the CNSC have been engaged on the analyses performed to support the considered design changes. The station's safe operating limits will be updated to account for the results of these new analyses. The Safety report will also be revised to reflect the new analysis.

Summary

NB Power is expecting approval by late 2004 of a project to extend the operating life of the Point

Lepreau Generating Station by 25 to 30 years. If approved, there would be an 18 month maintenance outage starting in the spring of 2008 to replace the fuel channel assemblies and feeder tubes as well as make certain safety improvements and equipment refurbishments. The project also includes the production of a level II PSA and upgrading of the deterministic Safety Analysis of a number of specific events not currently required to be assessed, to support the intended design modifications and the non equilibrium conditions that will be experienced immediately following refurbishment. This work complements the on-going safety analysis program that has been in place at the station since the station went into service. Work in this area is currently on-going in anticipation of project approval.

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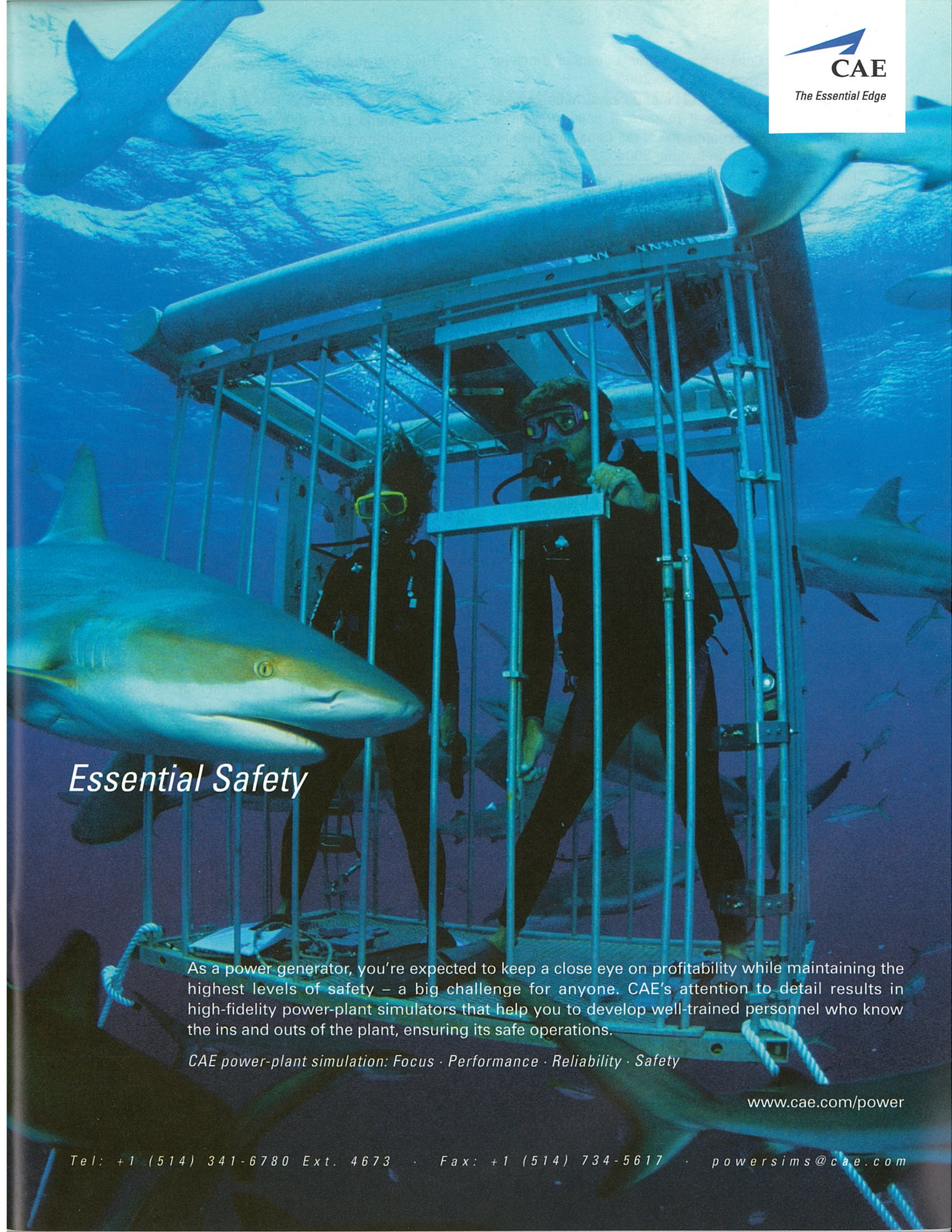
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APPENDIX I

Safety & Licensing guiding principles

There should be no conceptual impediment for extended operation (life extension)

- regulatory changes should be evolutionary not revolutionary
- will utilize a risk informed approach on a going forward basis
- The condition Assessment and on-going system health monitoring program at the station ensure long term equipment safety performance

The Licensing basis of the plant should remain essentially unchanged[11]; however

- will perform a level II PSA[12]
- will conduct a review of Safety Report against AECB Consultative Document C6 Rev 01 and perform assessments for events either not covered off or bounded[13]
- will conduct a review of the station against current codes & standards
- will use a risk informed approach (BCA) to assist with the resolution of complex safety & licensing issues

There should be no additional design changes or requirements associated with the outage and return to power other than those included in the project scope[14] and those that might arise from the PSA.

The outage is considered to be, and should be treated as, a maintenance outage. This includes aspects relating to:

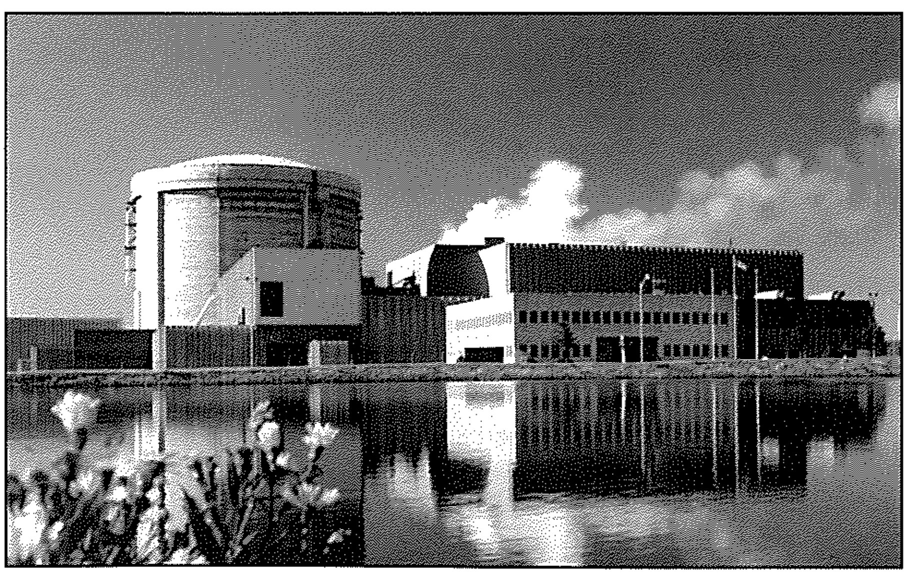
- the Operating Licence
- commissioning and return to service principles
- operator training and certification

On-going issues are covered by existing processes and should be kept separate from project scope, and restart of reactor, except in such instances where the only practical solution is a design change that lends itself to the refurbishment outage window.

Human Factors will be considered in the design process for any given change that will be made, however changes will not be made solely on a Human Factors perspective as this is covered by existing processes.

There needs to be early agreement on key design items such as the modification to the shutdown systems (both improvements to trip coverage as well as trip computer hardware and software design issues), and fuel channel design, etc.

The guidelines for the Environmental Assessment need to be agreed upon early on in the project in recognition of the need to construct the additional structures prior to the start of the Refurbishment outage.



Generation IV Power for the Future: Status of the SCWR

Romney B Duffey[1]

Ed. Note: The following paper was presented at the opening plenary session of the 6th International Conference on Simulation Methods in Nuclear Engineering, Montreal, October 2004.

Abstract

This paper summarizes the approach, history and most recent developments in Supercritical Water Reactor (SCWR) design and technology, both nationally and the planned international collaborations.

Atomic Energy of Canada Limited is advancing every aspect of CANDU technology by employing an evolutionary development strategy, applied to both the key technologies used in the reactor and the reactor's applications [1]. Our evolutionary development strategy ensures that AECL's innovations are based firmly on current experience and keeps our development programs focused on one reactor concept. This focus reduces risks, development costs, and product development cycle times. Central to this strategy is the discipline of basing next generation technology on the current technology. This approach ensures that CANDU investments today will result in technology that is relevant now and for the foreseeable future.

The development path for the next several decades is shown in Figure 1.

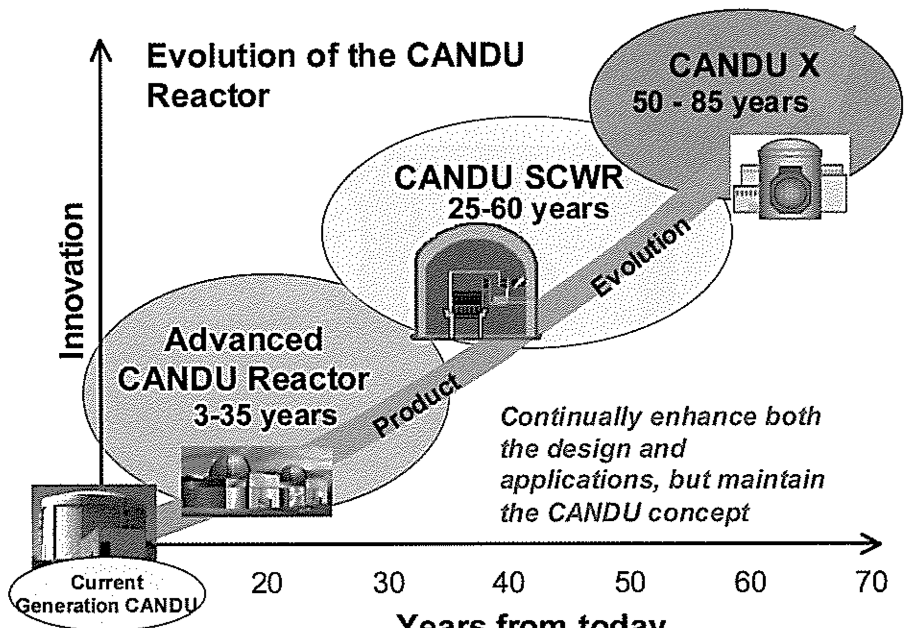


Fig. 1. Evolution of the CANDU Reactor

The "Generation" labels in Figure 1 refer to a classification scheme developed by the U.S. Department of Energy. Following present Generation III commercial designs, Generation IV reactors are in the conceptual development phase, and we have defined Generation V as the ultimate vision for a particular technology based on an extrapolation from current knowledge. The Generation III+ 2 ACR-700 is based on the existing CANDU 6, the Gen IV SCWR is based on the Advanced CANDU Reactor™ (ACR™), and the CANDU X is based on extending even further the super-critical concept, all based on the characteristics summarized in the introduction to this paper.

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The Supercritical Water Reactor

One of the key technical questions asked at the outset of our thinking on CANDU long-term evolution was the potential performance, safety, and cost benefits of using a different coolant to allow operation at higher temperatures. We are publishing a review of SCWR systems, which is summarized here [2].

Supercritical boilers have been operating for some time in coal-fired power plants at 500° C or more, primarily to raise the thermal cycle (Carnot) efficiency to greater than 40%. Thus, prior technical experience at industrial scale already exists. Moreover, supercritical water has no phase change, so the fuel dryout limit on heat flux and power vanishes. The high density still provides excellent neutron moderation capability, and requires several times less mass flow for the same heat removal due to the large heat capacity, hence reducing the required pumping power. So, theoretically, up to 30% higher channel powers are possible, with less pumping, enhanced safety margins, and increased thermal/electrical output than with existing technology. Simplification and cost reduction using a direct cycle is also possible. In addition, the plant has the potential to produce large quantities of low cost heat, which can be used for other industrial processes, while at the same time producing power at total cycle efficiencies close to 50%. We concluded that significant cost, safety, and performance advantages would result from the SCWR design, plus the flexibility of a range of plant sizes suitable for both small and large electric grids.

Hence, the next major phase of CANDU development will be the SCWR, using the same evolutionary concepts as the ACR-700, with the exception that there will be further improvements to materials to allow operation at higher temperatures, and therefore higher efficiencies and lower costs. Aggressive targets have been set for enhanced safety margins, cost reduction, resource sustainability, and economical and efficient plant operation for a wide range of plant sizes.

Supercritical fluids: the history

The use of supercritical (SC) fluids in different processes is not new and, actually, is not a human invention. Nature has been processing minerals in aqueous solutions at near or above the critical point of water for billions of years [3]. The first works devoted to the problem of heat transfer at supercritical pressures started as early as the 1930s [4] and [5]. Investigations of free convection heat transfer of fluids at the near-critical point with the application to a new effective cooling system for turbine blades in jet engines. They found [6] and [7] that the free convection heat transfer coefficient at the near-critical state was quite high and decided to use this advantage in single-phase thermosyphons with an intermediate working fluid at the near-critical point [4].

In the 1950s, the idea of using supercritical water (SCW) appeared to be rather attractive for steam generators. At supercritical pressures there is no liquid-vapour phase transition; therefore, there is no such phenomenon as critical heat flux (CHF) or dryout. Only within a limited range of parameters would any deterioration of heat transfer occur. The objective of operating steam generators at supercritical pressures was to increase the total thermal efficiency of a coal-fired power plant, with the combustion gases heating the fireside of the steam generator. The supercritical water was heated and circulated by forced convection on the secondary side, and used to drive a SC turbine. Work in this area was mainly done in the former USSR and in the USA in the 1950s – 1980s [8] and also in Germany and Japan.

In general, the total thermal efficiency of a modern power plant with subcritical parameters steam generators is about 36 – 38%, with supercritical parameters, i.e., water pressure 24 – 26 MPa, is about 45% and with ultra supercritical parameters, i.e., water pressure of 30 MPa and higher, is about 50%. The highest total thermal efficiency achieved in the power industry is about 56 – 58% with the combined thermal cycle, i.e., gas turbine – steam turbine.

As early as the end of the 1950s and the beginning of he 1960s, some studies were conducted to investigate the possibility of using supercritical water in nuclear reactors (see the review [9, 10, 11, 12, 13, 14]). Several conceptual designs of nuclear reactors using water as the coolant at supercritical pressures were developed in the USA, Great Britain, France and the USSR. However, with the emergence and dominance of Light Water Reactors (LWRs), this idea was abandoned for almost 30 years and did not regain support until the 1990s.

Use of supercritical water in power-plant steam generators is the largest application of a fluid at supercritical pressures in industry. However, many other areas exist where supercritical fluids are used or will be implemented in the near future.

The design of SCW nuclear reactors is seen as the *natural and ultimate evolution* of today's conventional modern reactors from a number of factors. Some designs of the modern Pressurized Water Reactors (PWRs) already work at quite high pressures of about 16 MPa; and Boiling Water Reactors (BWRs) are a once-through or a direct-cycle design, where steam from nuclear reactor is forwarded directly into the turbine. Some experimental reactors use nuclear steam superheaters with outlet steam temperatures well beyond the critical temperature but at pressures below the critical pressure. Finally, and most key, modern supercritical turbines, have operating pressures about 25 MPa and inlet temperatures of up to about 600°C, and have operated successfully at thermal power plants for many years.

The SCWR concepts therefore follow two main types [15, 16]: the use of either (a) a large reactor pressure vessel with wall thickness of about 0.5 m to contain the reactor core (fuelled) heat source, analogous to conventional PWRs and BWRs, or (b) distributed pressure tubes or channels analogous to conventional CANDU and RBMK reactors. The latter is used to avoid a thick-wall vessel. The coolant for both concepts is usually water, although carbon dioxide has also been considered. Using a thermal neutron spectrum, light water is usually used in the core flow, plus either solid graphite or zirconium hydride, or a liquid heavy water as a neutron moderator.

To reduce the severe axial flux tilt due to the large density decrease as the coolant is heated, (the density may reduce by 50% or so) the core flow can be a re-entrant in the vessel option, coming down unheated first and then turning into an up-flow; or interlaced or re-entrant in channels with flow in opposite directions. Both options also allow the reduction of pressure boundary temperatures, by partly insulating the pressure-retaining vessel or the channel wall using the first pass of the unheated flow. Typical outlet temperatures are near 600°C after the core pass. There is also the option of a re-entrant or return superheat pass to further raise the outlet temperatures if needed.

The limit on SCW outlet temperature is effectively set by the fuel cladding, since the peak clad temperature will be some 20% higher than the average, and the potential corrosion rates much higher. Estimates of the peak values of temperatures and wall thinning have been made to establish the margins and clad lifetime expected before refuelling.

Moreover, one of the unique features of the once-through SCW reactors is the very low- coolant mass-flow rates that are required through the reactor core because of the high specific thermal capacity. Preliminary calculations showed that the rate can be about eight times less than in modern PWRs, significantly reducing the pumping power and costs. This improvement is due to the considerable increase in enthalpy at supercritical conditions. Therefore, tight fuel bundles are more acceptable in supercritical pressure reactors than in other types of reactors. These tight bundles have a significant pressure drop, which in turn can enhance the hydraulic stability of the flow. Since the SCW is a single-phase "gas", then the cladding surfaces can and should be finned or ridged to enhance turbulence levels and the channel flows can be adjusted by orificing. This is done for Advanced Gas-Cooled Reactors (AGRs) today, and hence will flatten the outlet temperature distribution to avoid hot spots, increase the heat transfer, and reduce peak cladding temperatures in normal operation.

To optimize thermal efficiency and capital cost, there are also options for the thermal cycles [17, 18], being either direct cycle into a SCW turbine, or indirect using a heat exchanger.

However, the most significant technical problem seems to be with the materials reliability at high temperatures, pressures and neutron fluxes within a highly aggressive medium such as SCW.

Recent R&D directions

As part of the SCWR concept development and feasibility, AECL and others have examined and reported on a number of SCWR thermal cycles and configurations, using available methods benchmarked against current designs [19, 20, 18].

The corrosion behavior of candidate materials for pressure tubes, piping insulators and fuel clad, was studied in autoclaves, initially up to 450°C, which confirmed that a number of steels existed that could withstand the aggressive thermo-chemical conditions. These materials were derived from successful operation of SC boilers.

To maintain low-pressure tube temperatures to avoid excessive strain at the operating pressures of 25 MPa, two approaches are possible. One is to use an insulating material inside the channel to separate the hot-coolant from the pressure boundary. The other approach would be a re-entrant or double flow path, with the colder water-cooling the tube before being heated by the fuel. R&D has started on examining replaceable insulating liners that would ensure channel life and would provide reactor decay heat removal directly to the moderator heat sink without forced cooling.

The conceptual design for an insulated channel is shown in Figure 2, where the insulator must be both resistant to corrosion and also have a relatively low neutron absorption cross-section. Thus, the creep rate is negligible and the tube life itself is effectively infinite. Both zirconium-based and steel alloys are candidate materials for the pressure tube.

As a result of the R&D to date, the preferred concept is a direct cycle reactor using supercritical water with an outlet temperature close to 650°C, a thermal cycle efficiency of ~45%, and a reactor electrical output of ~400 MW(e), suitable for introduction in staged additions or extensions of power grids.

Extension of the design to even higher temperatures is solely a matter of materials choice and compatibility, and will be pursued in conjunction with the Generation IV International Forum (GIF) efforts to develop the VHTR.

International Developments: the GIF Program

The GIF is a consortium of nations, with proposed government-to-government Agreements to cover the R&D needed for the development of reactor concepts.

The development of the Generation IV SCWR (including the CANDU concept outlined above) is of high international interest. An evaluation performed under the auspices of the Generation IV International Forum and the US Department of Energy has resulted in a Roadmap for R&D for Generation IV systems. Over 100 different systems were evaluated, but only 6 were selected for further development, including the SCWR [21].

The R&D planning in Canada is integrated with the Roadmap plans, and only diverges when design-specific differences between the vessel and pressure tube options preclude total integration. Using input from experts in the field, the following are identified as the areas for SCWR R&D:

- Fuel cladding materials;
- Corrosion/fission product transport;
- Optimized core neutronics and fuel cycles;
- Safety and accident analysis; and
- Validation of neutronic, thermal hydraulic and fuel codes.

In the interests of sustainability, hydrogen production by an SCWR will also be included as part of the system requirements, where the methods for hydrogen production will depend on the outlet temperature of the reactor. For temperatures <625°C, electrolysis will be the only viable choice.

For temperature >850°C, either direct methods such as transforming biomass, or high temperature electrolysis, can be considered. AECL's approach is to establish the technology at the more modest temperatures, and then evolve the SCWR to higher temperatures as the operating experience and knowledge base both advance.

With the resurgence in interest in nuclear energy as a greenhouse–gas and pollution free energy source; there is a growing market potential for nuclear energy in the 21 st century. Our analyses show that it is possible to stabilise global emissions, and meet the future transportation needs using hydrogen fuels [22], and the SCWR and other Generation IV concepts are uniquely positioned to bring about this major transformation.

Concluding Remarks

The SCWR is a natural evolution of today's LWR. In summary, SCWR has the promise to, according to the US DOE Generation IV Nuclear Energy Systems Roadmap [23] and the work discussed here:

- Significantly increase thermal efficiency up to 40 – 45%, competing with alternate thermal cycles;
- Eliminate steam dryers, steam separators, re-circulation pumps and steam generators, and reduce balance-of-plant systems, thus saving cost and further simplifying;
- Decrease reactor coolant pumping power thus reducing cost and increasing plant net efficiency;
- Potentially lower containment loadings during Loss-Of-Coolant Accident due to low specific energy; and
- Allow the production of hydrogen at SCWR due to high-coolant outlet temperatures, either directly or indirectly.

These are all performance enhancements compared to existing designs, and hence encourage further long-term development. These also change considerably the analytical simulation challenges compared to present designs. Appropriate development of simulation tools and their benchmarking will continue to be an active area for R&D for such Generation IV concepts. In fact, synergisms exist with other renewable energy sources and with the hydrogen economy that represent an irresistible opportunity.

Acknowledgements

The author thanks his numerous technical colleagues in the Generation IV International Forum and at AECL for their original contributions to the SCWR concept, and the CNS for stimulating this paper.

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Bruce A Restart Phase B Commissioning Physics Tests – A Comparison Between Measurements and Calculations

by C. Ngo-Trong, D.A. Jenkins, W. Shen, A. Mao, P. Schwanke[1], M. Gold[2]

Abstract

The computer codes RFSP-IST, WIMS-IST and DRAGON-IST were used to pre-simulate the Bruce A Nuclear Generating Station Unit 4 Restart Phase B commissioning physics tests. Comparisons between calculations and measurements have validated all Bruce A Restart accident analyses, previously done with the same computer codes and modelling methodologies.

Post-simulations of Phase B commissioning physics tests were also performed, which differ from the pre-simulations mainly in the use of an improved methodology, the side-step method, for calculating the incremental cross-suctions for reactivity devices. The post-simulation results showed better agreement with measurements than those of the pre-simulations, particularly for the reactivity worth of the liquid zone controllers.

1. Introduction

After a five-year shutdown and a two-year intensive preparation, Bruce A Nuclear Generating Station (NGS) Unit 4 was restarted on 2003 October 7, and Unit 3 was restarted on 2004 January 8. The restart of both units was preceded by a series of on-power tests of the units' safety and operating systems, including a complete program of Phase B commissioning physics tests, typical of start-ups of new CANDU[1]" reactors.

The reasons for this complete Phase B program are the following:

Both Bruce A Units 3 and 4 were restarted with cores containing all fresh fuel. Major safety studies for both the equilibrium core and the initial core have been carried out with the industry standard toolset (1ST) reactor physics code suite RFSP/WIMS/DRAGON [1-3]. Physics tests at low power, so-called Phase B commissioning tests, provided station-specific data to validate the use of these computer codes and models for Bruce A restart accident analyses, to supplement the generic validation of these codes.

Reactor physics tests at low power would have also helped to detect severe abnormalities in the initial core loading or major degradation in the worth of reactivity devices, if any.

2. A Short Description of The Bruce A Reactors

Bruce A NGS has 480 fuel channels per reactor. The core length of each fuel channel is equivalent to 12 standard 37-element CANDU fuel bundle lengths. Each fuel channel contains 13 such bundles, with half a bundle being positioned outside the active reactor core length at each channel end.

The reactor power is 92.5% of the original design power; the thermal power to coolant is now 2492 MW.

Figure 1 shows the locations of reactivity control units. There are 30 shutoff rods (SORs), four control absorbers (CAs) and six zone control units, which are divided into a total of 14 liquid compartments or zone controllers (ZCRs).

Figure 2 shows the initial core loading. The Bruce A restart initial core has more depleted fuel (fuel with 0.4% U-235 by weight) bundles than that of the original start-up (714 bundles versus 432 bundles), and the depleted-fuel bundles are distributed in a more complex loading scheme. This has been necessary due mainly to two factors: the need to substantially reduce the critical moderator poison concentration at the plutonium peak, as compared to that of the original loading scheme; and the need for more stringent limits on the maximum channel and bundle powers.

Figure 3 shows the locations of the flux monitor units used during Phase B flux scan activities.

3. Pre-simulations and Post-Simulations of Phase B Physics Tests

Pre-simulations of the Phase-B commissioning physics tests were performed with the same 1ST code suite RFSP/WIMS/DRAGON and the same core model as that used in the accident analyses [4].

Post-simulations of Phase-B commissioning physics tests were also performed, with further work done to improve on the general simulation methodology.

The post- and pre-simulations of the tests differ in the following areas:

- The core conditions (moderator and coolant purities and temperatures, average ZCR fills) used in the test pre-simulations are not exactly the same as the actual core conditions at the time of the tests. In the post-sim-

1 Atomic Energy of Canada Limited

2 Candesco Research Corporation

ulations of the tests, the actual core conditions during the tests were used instead. Table 1 shows the core conditions used in the pre- and post-simulations of the Phase B commissioning physics tests.

- Incremental cross-sections of the ZCRs (full and empty), SORs, CAs and their guide tubes were calculated in the pre-simulations with the "all-DRAGON" method. They were re-calculated in the post-simulations with the WIMS/DRAGON side-step modelling method [5], which is more suitable for the clustered nature of various zirconium tubes (feeders and scavengers) inside the ZCR assemblies. In the side-step method, W1MS-IST, rather than DRAGON-IST, was used to generate the 89-group macroscopic cross-sections, and homogenisation of the cluster geometry of the ZCR in 2-D transport calculations was performed. Additional changes in the post-simulation calculations of the above reactivity device incremental cross-sections include the use of a newer version of DRAGON, version 3-04L (instead of version 3-04Bb in the pre-simulation calculations), the use of the ENDF/B-VI library instead of the ENDF/B-V library and the use of finer meshes in DRAGON 3-D flux calculations.
- Under cold coolant conditions, it was estimated that, because of core aging resulting in the axial elongation of pressure tubes and the subsequent change in the fuel channel fixed ends, the fuel bundle latches on the west side of the reactor moved further west by an average of about 0.7 cm, and the latches on the east side moved further east by an average of about 4.3 cm. The resulting net bundle latch displacement is about 5 cm in an average channel. Since the initial core fuel loading includes one to three depleted-uranium fuel bundles, placed near the downstream end in the central fuel channels at bundle positions 8, 9 and/or 10, this bundle latch displacement introduces an east-west flux tilt and needs to be modelled. RFSP-IST had to be modified to model this phenomenon, through the use of fictitious "mini-bundles" whose lengths are equal to an integral fraction of the real fuel bundle length. In the pre-simu-

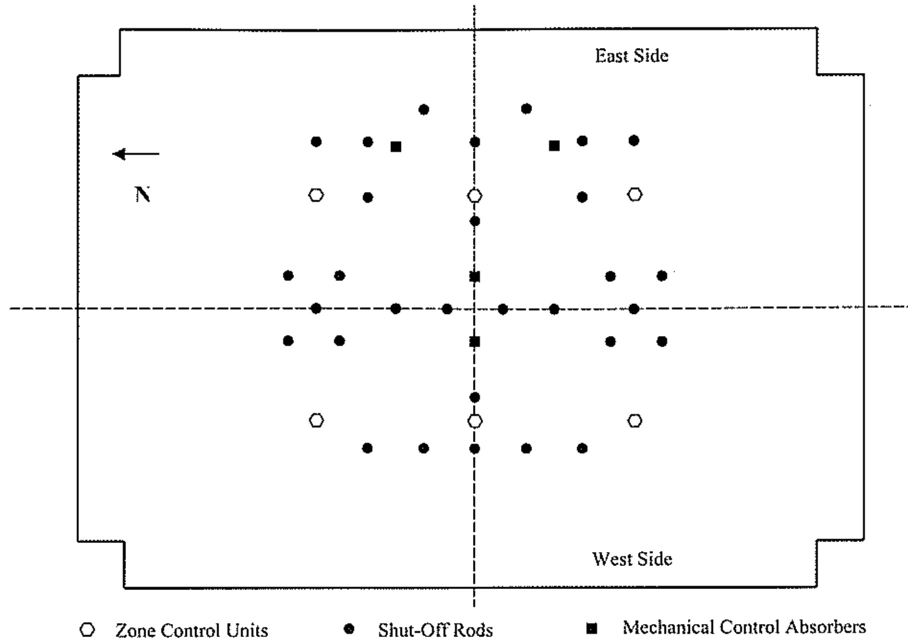

Figure 1: Bruce A Core Reactivity Device Locations (viewed from the Top of the Calandria)

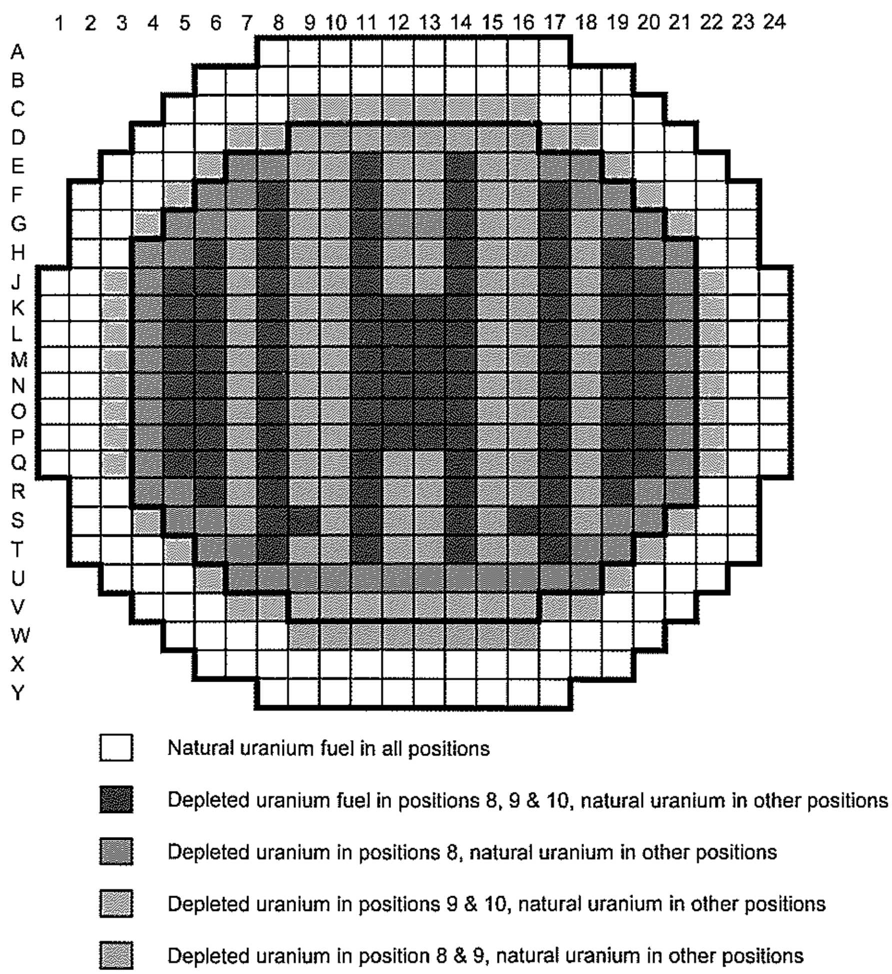

Figure 2: Bruce A Restart Initial Core Loading

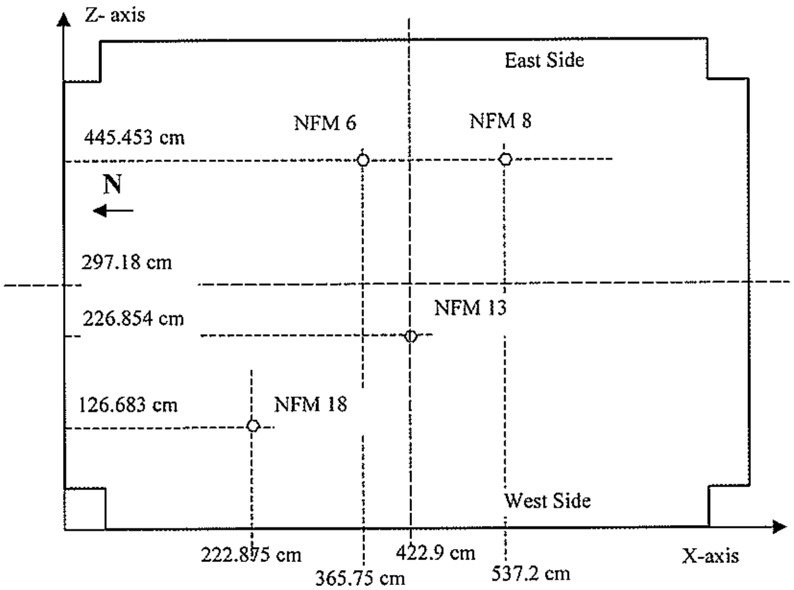

Figure 3: Locations of Flux Monitoring Units

lations, with more limited changes to the code (because of time constraints), the net latch displacement was assumed to be about 6.2 cm instead of 5 cm, allowing the use of 6.2-cm mini-bundles (i.e., 1/8 the length of real bundles) in the model. In the post-simulation of the test, more changes to RFSP-IST were made, allowing the more accurate 5-cm bundle-latch average displacement to be modelled, through the use of 5-cm mini-bundles (i.e., 1/10 the length of real bundles),

It should be noted here that a successful comparison between pre-simulation results and measurements, after adjustments for differences in core conditions, is necessary to permit the reactor to proceed to Phase C commissioning tests (at-power tests); the pre-simulations of the tests used the same computer codes and the same core modelling methodologies as those used in the accident analyses. Post-simulations of the test are only meant to show the possibility of better modelling methodologies for future use,

The existence should be also noted here of other post-simulations of Bruce A restart Phase B commissioning tests. These other simulations are more limited in scope and are not discussed in this paper. They are described in Reference [6],

The following sections provide short summaries of the Phase-B commissioning physics tests and the corresponding acceptance criteria, measurement values, pre-simulation results and post-simulation results.

4. Approach To Critical

Gadolinium was used as the moderator poison in the core-shutdown state and during Phase B commissioning. First criticality was attained through gadolinium withdrawal by ion exchange columns. The predicted moderator gadolinium concentration at critical was 1.80 ppm, with an acceptance criterion of ± 0.4 ppm. Adjusted to the actual core conditions, the predicted value was 1.82 ppm. The measured critical moderator gadolinium concentration was 1.85 ppm. Post-simulation calculations showed a critical moderator gadolinium concentration of 1.875 ppm. Note that core criticality corresponds to a calculated keff = 1.00000.

5. Zone Controller Reactivity Worth Calibration Test

The change in ZCR reactivity worth as a function of average zone fill was measured by adding successive pre-measured gadolinium packets to the moderator and recording the average zone level decreases, from an initial level of 77.7% down to 13.2%.

Table 2 shows the measured reactivity worth changes and the corresponding pre-simulation and post-simulation results.

The pre-simulation change in core reactivity was 3.58 mk, with an acceptance criterion that the measured core reactivity change be within ± 0,4 mk of the calculated value. The measured core reactivity change for this 64% drop in ZCR fill was 3.01 mk, or a difference from prediction of 0.57 mk Including all intermediate measurements within the above range, the root-mean-square (RMS) difference between prediction and measurement was 16.4%. The non-compliance of tins test with &e acceptance criterion was discussed in the Phase B commissioning test report [7], It was shown there that the lighter-than-predicted ZCR worth has no negative effect on the results of the existing accident analyses and on reactor operability.

In the post-simulation, the RMS difference between calculations and measurements was reduced to 5.0%. In particular, the post-simulation calculation of core reactivity change due to a decrease in average ZCR fill from 77.7% to 13.2% was 3.19 mk, or a 0.18-mk difference from the measurement. The improved agreement is mainly a result of the finer mesh representation of the supercell. The use of side-step method in calculating the ZCR incremental cross sections is also an important contributor to reducing the observed discrepancies.

6. Control Absorber and Shutoff Rod Reactivity Worth Measurements

The four CAs and 30 SORs were successively inserted and withdrawn, and the consequent changes in average zone levels were recorded. The device-measured reactivity worth was then extracted from a third-degree polynomial that was fitted to the curve of measured core reactivity change vs. zone level of the preceding test. The acceptance criterion for the test is that the measured reactivity worth of each device be within ± 15% of the pre-simulated value. Control absorbers and SORs are identical in construction. Their reactivity worths differ because of the differences in flux levels at their locations.

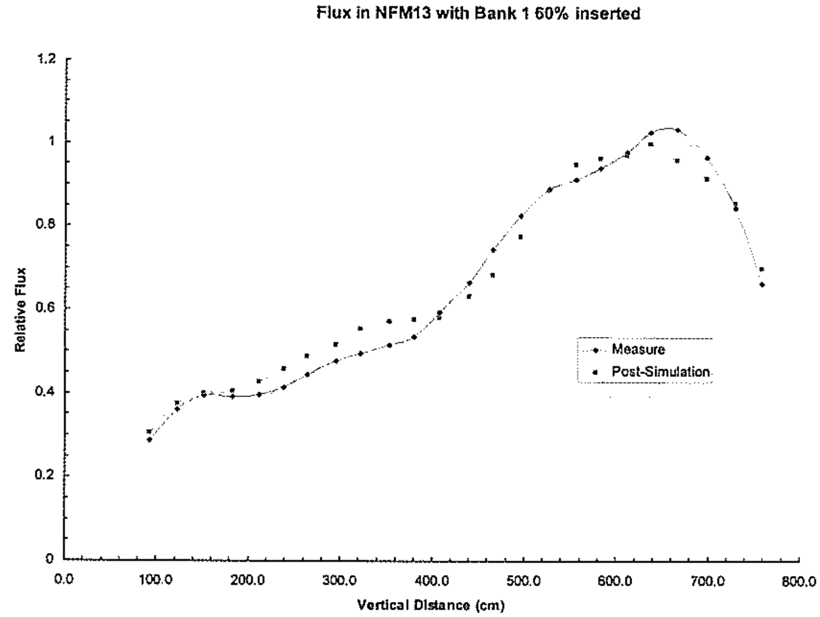

Figure 4: Comparison between Measured and Calculated Fluxes for Flux Scan Number 7

Table 3 shows the measured, pre-simulated and post-simulated CA and SOR reactivity worths.

The measured worths for a nominal core configuration range from a low of 0.72 mk (SOR03) to a high of 1.94 mk (SOR29), with an average of 1.36 mk. The pre-simulated rod worths range from 0.84 mk (SOR03) to 1.90 mk (SOR18), with an average of 1.38 mk. Relative differences between predictions and measurements are within the range from -17.4% to +13.6%, with three rods outside the ± 15% range. The RMS value of the differences is 9.0%.

The measured discrepancies in CA and SOR worths will have little effect on accident analysis results. While there is an apparent spatial discrepancy in the device worths, the total worths of the reactivity devices are only slightly decreased (roughly a 4.3% decrease in the combined CA worth and a 1.3% decrease in the combined SOR worth). Therefore, the test results confirm that the CAs and SORs will fulfill their safety function.

In the post-simulation, the average rod worth is 1.35 mk; the RMS value of the differences from measurements is 8.7%, but all differences are within the ± 15% range.

7. Control Absorber Bank Reactivity Worth Measurements And Flux Scans

Fission chamber vertical scans were conducted in central wells in four vertical flux-detector assemblies, for five successive CA bank configurations: all CAs out-of-core; bank 1 inserted 60%; bank 1 fully inserted; bank 1 fully inserted and bank 2 inserted 60%; and all banks fully inserted. Also, for each configuration change, the resulting ZCR level change was measured. The measured CA bank worths were then extracted, using the same ZCR worth vs. level curve that was mentioned in the preceding section.

Table 4 shows the measured, pre-simulation and post-simulation CA bank worths.

The acceptance criterion for the reactivity worth component of the test is that the measured reactivity worth of each bank be within ± 15% of the calculated value. This criterion was essentially met by both pre-simulation and post-simulation calculations. The RMS values of the differences between measurements and calculations are 7.1 and 8.3% for pre-simulation and post-simulation, respectively.

Table 5 shows a summary comparison of calculated and measured thermal neutron fluxes for the 20 flux scans (five CA bank insertion configurations at four scan locations). Each scan has up to 24 assessment points, located at or near the midpoints between fuel-channel rows. Calculated (measured) fluxes were interpolated (recorded) at every 1/5 of

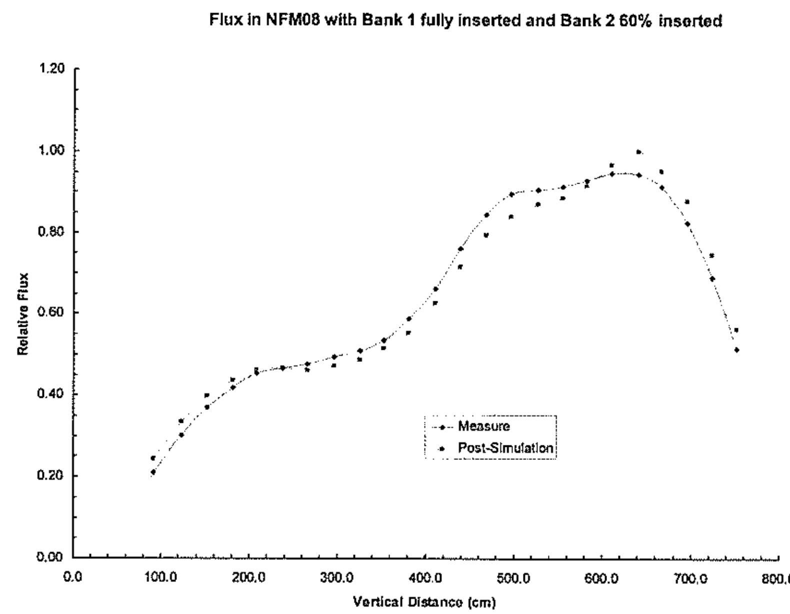

Figure 5: Comparison between Measured and Calculated Fluxes for Flux Scan Number 14

Parameter	Value	
	Pre-Simulation	Post-Simulation
Moderator temperature (degrees C)	25	30
Coolant temperature (degrees C)	25	29
Moderator purity (wt% D_2O)	99.95	99.95
Coolant purity (wt% D_2O)	99.10	99.15
Natural uranium fuel density (g/cc)	10.610	10.610
Depleted uranium fuel density (g/cc)	10.627	10.627
Pressure tube diametrical creep (%)	1.59	1.59

Table 1: Phase B Physics Test Core Conditions

Poison addition (mk)	Cumul. poison addition (mk)	Measured Average Zone Level (%full)	ZCR Worth Change (mk)		Relative Error (%)	
			Pre-Simulation	Post-Simulation	Pre-Simulation	Post-Simulation
0.000	0.000	77.7	0.000	0.000	0.00	0.00
0.354	0.354	68.8	0.404	0.359	-14.24	-1.41
0.354	0.708	60.5	0.821	0.740	-15.93	-4.52
0.354	1.062	53.0	1.224	1.114	-15.24	-4.90
0.354	1.416	45.8	1.630	1.487	-15.11	-5.01
0.354	1.770	38.7	2.044	1.859	-15.50	-5.03
0.354	2.124	31.6	2.469	2.233	-16.22	-5.13
0.354	2.478	24.3	2.910	2.611	-17.44	-5.37
0.177	2.655	20.4	3.147	2.805	-18.53	-5.65
0.354	3.009	13.2	3.582	3.194	-19.06	-6.15
AVERAGE					-16.36	-4.80
RMS					16.44	4.96

Table 2: Comparison between Measured and Calculated ZCR Fill Worths

a lattice pitch along the y-axis of the RFSP-1ST core mesh array, which is also the vertical direction of the flux monitoring units, starting from the first point at y = 80 cm, to the last point at y = 765.8 cm, for a total of 120 points along each of the four assemblies. The origin of they-axis is at the top of the calandria main shell inner surface. The top edge of the first row of lattice cells is at y = 80 cm. The calandria midplane is at y = 422.90 cm. The x- and z-coordinates of the central wells in the flux monitor units NFM 6, 8, 13 and 18 are shown in Figure 3. The calculated (measured) flux at each of the 24 assessment points is the average of the five nearest interpolated (recorded) fluxes.

Figures 4 and 5 show the comparisons between measurements and post-simulation results for two typical scans, namely scan numbers 7 and 14 of Table 5.

The acceptance criterion for flux scans is that each measured curve agree with the pre-simulated curve to within ±15%, once both are consistently normalised. A least-squares-fit normalisation scheme, applied to each curve, was used to give the results shown in Table 5 and Figures 4 and 5. The measured flux scans are in generally good agreement with the pre-simulated flux scans, and meet the acceptance criterion for almost all data points—the only locations where measured and pre-simulated fluxes differ by more than ± 15% are in the bottom one or two rows and the very top row (all relatively low-flux regions) in six of the 20 scans. The RMS value of all data point differences is 7.3%. In the post-simulation, the agreement is slightly better, with the RMS value of all data point differences reduced to 7.1 %. The number of data points with differences outside the acceptable range of ±15% also decreases from 14 (out of a total of 471 data points) to 7.

8. Low-power Shutdown Systems 1 and 2 Rundown Tests

The fission chambers employed in the flux scans were inserted in four different vertical positions (in wells) in the same four detector assemblies as those used in the flux scan test, to record the power rundowns upon actuation of shutdown systems SDS1 and SDS2. The acceptance criterion for SDS1 rundown is to demonstrate that the rundown is at least as effective as that assumed in the safety analysis, using such assumptions as OP&P (operating principles and procedures) limits for SOR insertion. The acceptance criterion for SDS2 rundown is to demonstrate that SDS2 is at least as effective as SDS1. Comparisons between the recorded and pre-simulation calculated times from trip, for the flux to reduce to 95, 90 and

Device	Measured worth (mk)	Calculated worth (mk)		Relative error (%)	
		Pre-Simulation	Post-Simulation	Pre-Simulation	Post-Simulation
CA01	0.97	1.11	1.10	-14.09	-13.44
CA02	1.02	1.12	1.10	-9.37	-8.15
CA03	1.58	1.63	1.56	-3.05	1.68
CA04	1.77	1.72	1.63	2.84	7.65
SOR01	0.83	0.95	0.95	-14.52	-14.13
SOR02	0.87	0.96	0.96	-10.66	-10.16
SOR03	0.72	0.84	0.84	-17.38	-17.70
SOR04	0.88	1.01	1.01	-15.07	-14.17
SOR05	0.99	1.09	1.09	-10.28	-10.41
SOR06	0.95	1.01	1.01	-7.24	-6.94
SOR07	0.78	0.84	0.85	-8.11	-8.36
SOR08	0.92	1.07	1.05	-16.55	-14.45
SOR09	0.98	1.07	1.05	-9.29	-6.74
SOR10	0.97	1.07	1.05	-10.57	-7.73
SOR11	1.16	1.27	1.23	-9.99	-6.33
SOR12	1.36	1.48	1.41	-8.75	-3.26
SOR13	1.45	1.55	1.48	-6.45	-1.92
SOR14	1.43	1.47	1.41	-3.23	1.21
SOR15	1.23	1.29	1.24	-4.83	-0.60
SOR16	1.79	1.83	1.74	-2.22	3.17
SOR17	1.91	1.90	1.80	0.46	5.76
SOR18	1.91	1.90	1.80	0.61	6.06
SOR19	1.84	1.84	1.74	0.33	5.46
SOR20	1.28	1.35	1.30	-5.31	-1.25
SOR21	1.51	1.55	1.49	-2.39	1.64
SOR22	1.53	1.56	1.49	-1.82	2.23
SOR23	1.58	1.57	1.50	1.08	5.41
SOR24	1.34	1.36	1.30	-1.15	2.81
SOR25	1.48	1.37	1.34	7.61	9.76
SOR26	1.75	1.60	1.59	8.92	9.41
SOR27	1.93	1.73	1.72	10.19	10.82
SOR28	1.90	1.64	1.65	13.64	13.47
SOR29	1.94	1.72	1.71	11.11	11.54
SOR30	1.76	1.60	1.59	9.30	9.95
			RMS	9.05	8.73
Average of all CAs and SORs	1.36	1.38	1.35	-1.63	1.23

Table 3: Comparison between Measured and Calculated CA and SOR Reactivity Worths

50% of the initial values, given in Table 6, show that SDS1 was clearly inserted faster than assumed in the safety analysis. Indeed, the pre-simulation calculated times are all higher than measured, for the three flux reductions, at four different flux monitoring points.

There was no post-simulation of the SDS1 rundown test as it was deemed unnecessary, in view of the slight reduction in the post-simulation calculated average SOR reactivity worth.

The SDS2 power rundown test was not simulated, as the test involved only comparing recorded transient fluxes. It suffices to mention here that the acceptance criterion for SDS2 rundown was also met.

9. Conclusion

The above successful simulations of Bruce A Unit 4 Phase B commissioning physics tests represent a major achievement for the 1ST code suite RFSP/WIMS/DRAGON, considering the fact that the Bruce A Unit 4 fresh-core configuration is a very complex core to model, with uneven bundle latch displacements, coupled with a particularly complex initial-core fuel loading.

The calculation of incremental cross sections of reactivity devices with an annular structure like a CA or SOR could be very satisfactorily performed with either the existing all-DRAGON methodology, or the newly conceived side-step methodology. However, the side-step method has some advantage for devices with a cluster structure like ZCRs.

Acknowledgements

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The authors wish to thank Mr. Robert Chun of Bruce Power for permission to use the Phase B pre-simulation results and measured data in this paper.

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Control absorber bank position	Measured worth (mk)	Calculated Worth (mk)		Relative Error (%)	
		Pre-Simulation	Post-Simulation	Pre-Simulation	Post-Simulation
Bank 1 half in	1.421	1.228	1.204	13.6	15.3
Bank 1 fully in	2.801	2.698	2.653	3.7	5.3
Bank 1 fully in & bank 2 half in	3.710	3.643	3.580	1.8	3.5
Bank 1 and bank 2 fully in	4.286	4.307	4.255	-0.5	0.7
AVERAGE				4.6	6.2
RMS				7.1	8.3

Table 4: Comparison between Measured and Calculated CA Bank Worths

Scan Number	Control absorber bank configuration	Scan location	Number of assessed points	Pre-Simulation RMS error (%)	Pre-Simulation Number of points outside +/- 15% error band	Post-Simulation RMS error (%)	Post-Simulation Number of points outside +/- 15% error band
1	All banks out	NFM6	24	6.71	0	6.54	0
2	All banks out	NFM8	24	9.05	3	8.59	2
3	All banks out	NFM13	24	6.41	0	6.36	0
4	All banks out	NFM18	23	7.36	0	7.38	0
5	Bank 1 60% in	NFM6	24	11.18	4	10.02	0
6	Bank 1 60% in	NFM8	24	11.29	3	10.19	2
7	Bank 1 60% in	NFM13	24	5.48	0	6.42	0
8	Bank 1 60% in	NFM18	23	8.59	2	8.80	2
9	Bank 1 fully in	NFM6	24	7.53	0	7.16	0
10	Bank 1 fully in	NFM8	24	7.70	0	7.27	0
11	Bank 1 fully in	NFM13	24	4.82	0	5.31	0
12	Bank 1 fully in	NFM18	22	6.82	0	6.67	0
13	Bank 1 fully in, bank 2 60% in	NFM6	24	7.19	1	6.95	1
14	Bank 1 fully in, bank 2 60% in	NFM8	24	6.48	0	6.24	0
15	Bank 1 fully in, bank 2 60% in	NFM13	24	5.13	0	5.55	0
16	Bank 1 fully in, bank 2 60% in	NFM18	21	6.90	0	6.81	0
17	Both banks fully in	NFM6	24	5.79	1	5.49	0
18	Both banks fully in	NFM8	24	5.99	0	5.78	0
19	Both banks fully in	NFM13	24	4.97	0	5.40	0
20	Both banks fully in	NFM18	22	6.67	0	6.60	0
All 20 scans			471	7.32	14	7.11	7

Table 5: Comparison between Measured and Calculated Fluxes in Flux Scans

Flux detector location	Time from trip for flux to reach 95% (s) Measured	Time from trip for flux to reach 95% (s) Pre-Simulation	Time from trip for flux to reach 90% (s) Measured	Time from trip for flux to reach 90% (s) Pre-Simulation	Time from trip for flux to reach 50% (s) Measured	Time from trip for flux to reach 50% (s) Pre-Simulation
319 cm from top, in NFM6	0.486	0.523	0.530	0.567	0.699	0.746
440 cm from top, in NFM8	0.686	0.788	0.776	0.879	1.159	1.236
384 cm from top, in NFM13	0.588	0.660	0.661	0.731	0.957	1.041
337 cm from top, in NFM18	0.540	0.606	0.588	0.665	0.835	0.954

Table 6: Comparison between Measured and Calculated Flux Reduction Times in the SDSI Power Rundown Test

CNA/CNS Nuclear Achievement Awards

Each year the Canadian Nuclear Association and the Canadian Nuclear Society join in honouring individuals and groups who have made significant contributions to the Canadian nuclear program. See the last issue of the CNS Bulletin for those recognized in 2004.

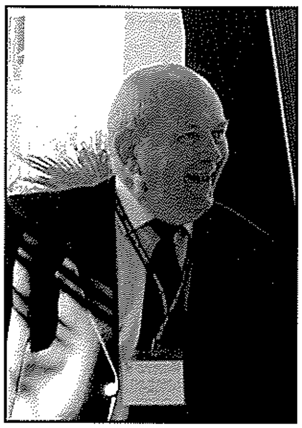
The formal call for nominations will be issued shortly. Readers are urged to begin thinking of colleagues that deserve recognition and start to gather the information needed to support a nomination.

For further information contact Ed Price, chair of the Joint CNA/CNS Honours and Awards Committee. His e-mail address is: edward.price@sympatico.ca

Now That We've Arrived, Where Shall We Go?

by Dan Meneley

Ed. Note: *Dan Meneley was the guest speaker at the banquet of the 6th International Conference on Simulation Methods in Nuclear Engineering held in Montreal, October 12 -15, 2004. He has had a broad background in the Canadian nuclear scene, including: chief engineer at Atomic Energy of Canada Limited , professor at University of New Brunswick, visiting professor in China and other roles. "Semi-retired" he now is director of the CANTEACH program. Following is a slightly edited version of his presentation.*



The title of this talk came to mind more than 40 years after the start-up of NPD marked the start of the development of uranium-fuelled electric power in Canada. These have been busy and productive years. We can ask the question of where to go next at several levels, ranging from the details of simulation models and mathematics up to the broadest of generalities. Specifically, - "Whither CANDU?"

Figure 1 is an old slide that was the result of an earlier speculative look at the nuclear future.

In about 2050-2060 there could be as many as 1000 reactors in world, up to 50 on one site. There could be on-site fuel fabrication, reprocessing, and waste management with some fissile atoms coming from either fast breeder reactors (FBRs) or accelerator breeding.

To give an idea of the assumed scale, about 250 large nuclear units are needed to produce transportation fuels equivalent to today's North American gasoline consumption.

This talk is based on two earlier exercises. The first was an options study done at AECL several years ago - as a precursor to the very serious studies of ACR, SCWR, and CANDU X. The second exercise was conducted at the June 1997 IAEA International Conference on the future of the world nuclear industry.

At the very beginning, during and after the Montreal Project during World War II, we were both hampered and aided by the climate of secrecy surrounding this industry. We were hindered by restricted access to what other groups were doing around the world - especially in the US, UK, and Russia. But these restrictions helped us as well - they fostered the single-minded objective of producing a natural-uranium reactor with excellent neutron economy. This might prove to be the most important legacy of those who founded this enterprise. CANDU is different because we started independently and from a different viewpoint.

A. Mathematical Models

We've moved from a development climate that was primarily experimental with a few theoretical models (some of which became quite sophisticated) to a climate in which many design decisions are based on very complex mathematical models of components and systems in the plant. Neutron behaviour now can be described using stochastic models that are, in some respects, more precise and informative than are the practical experiments. Simpler (cheaper) physics models now can be tested against benchmark Monte Carlo calculations.

The more complex calculations of transient thermal-hydraulics are less fully mastered than those of reactor physics, but very good progress is being made. We are almost at the stage where further refinement of these models will be unnecessary, at least for systems fairly similar to those that are already in service.

B. Plant Simulation

Simulation of whole-plant behaviour using a combination of analytical and empirical models has reached a sufficient level of accuracy that operator training can be carried out in a very realistic way. Design simulation of plant processes is still less well developed, due more to insufficient effort than to lack of capability.

It seems, a bit surprising that design optimization has essentially ceased as a part of the logical process of plant design. More than 30 years ago the computer codes AESOP and CANCAP had reached quite an advanced stage of development. Many of the design parameters now considered "standard" for CANDU were derived from those models. It is less surprising that these methods have not been used recently when one considers that most key parameters of CANDU have remained fairly constant since the mid 1970's. Current redesign efforts may revive the use of these elegant models.

Simulation of plant hardware has reached a very useful stage, in which one can expect that we will soon reach the goal of 'building the plant twice'-- once on the computer and once more in the field. Recent experience, such as the Qinshan project, has shown the great advantages of this procedure in design/construction cost as well as schedule. It is expected that use of these models for configuration control and maintenance planning will also offer solid advantages in operating cost and in reduction of outage time.

C. Structural Materials

The spectacular initial success of CANDU fuel design now has reached what appears to be a practical limit. Extreme demands are placed on a few critical components - e.g. pressure tubes and steam generator tubes. Pressure tubes dictate the major rehabilitation interval for today's power reactors.

It would be interesting to once again look at methods for improving pressure tube performance. One possibility is to introduce the cold pressure tube design - now being considered for the CANDU X concept.

Another interesting possibility that was considered many years ago is carbon fibre reinforced zirconium alloy. The objective would be to relax limitations imposed by axial and radial expansion in-service. Increased strength of calandria tubes could improve channel sag and some aspects of safety following an in-core break.

D. Power Projects

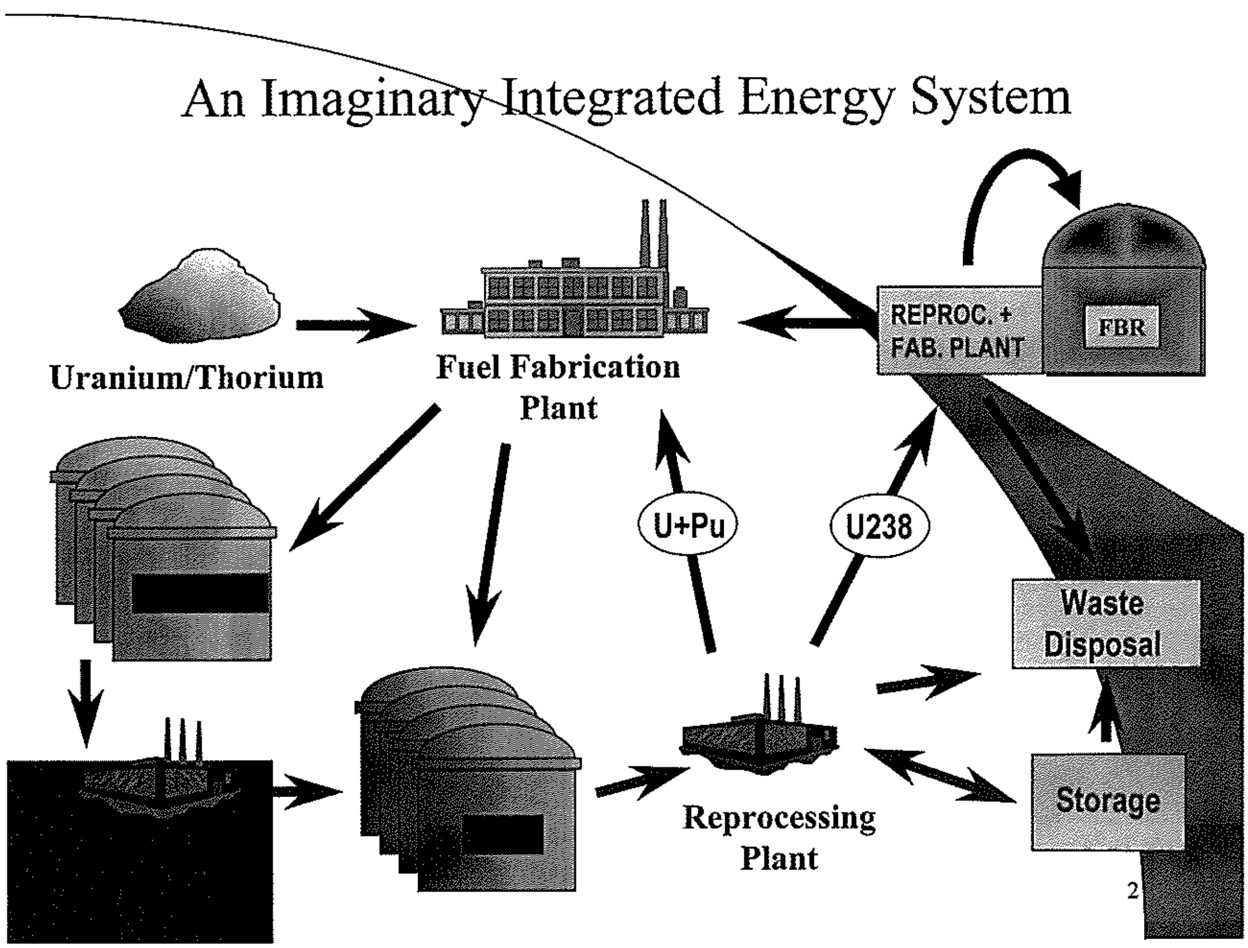
After a rapid start in the early years the progress of nuclear power production in the world was slowed by economic factors, poor performance, and public opposition -and new construction came to a complete halt in the USA. The first oil crisis effectively halted new installations in Ontario in the early 1990's.

Then, poor management and a few bad mistakes in Ontario plants placed a heavy weight on this enterprise and nearly brought it to an end. But we are recovering in spite of present overburden of bureaucracy compounded by a confusion of objectives. But we have good news as well.

During the same period of time when the CANDU program was crippled in Canada, the international program was making steady headway. The CANDU 6 design has emerged as a first-class product that can compete anywhere in the world with other designs available today. This success was the result of painstaking detailed engineering, slow design evolution, feedback from operations, excellent project management, and creative construction methods.

Two examples of design evolution are the use of an interleaved feeder layout in the CANDU 9 design - to reduce the coolant void power pulse, and the option of low-enriched fuel to allow higher output from the same number of fuel channels. That design also essentially eliminates the possibility of gross fuel melting in severe accident sequences. The new AECL design of the Advanced CANDU Reactor is attracting a great deal of attention in the world due to its low capital cost and very short construction schedule.

At the present time the most intense CANDU project activity is major rehabilitation of the second-generation



plants in Ontario and elsewhere. This condition likely will persist for a number of years. The effort will need a large commitment of staff as well as manufacturing facilities.

Furthermore, it is becoming apparent that Ontario, in particular, must increase its total electricity production capacity in the very near future. Most of this increase must be achieved by building new nuclear plants ~ given the cost and availability limits of all other electricity supply options. I expect that the next few decades will be extremely busy ones for all of us.

More generally, the world's nuclear industry should be careful of what it wishes for in the future. A likely scenario is one in which hundreds or thousands of nuclear units will be needed within the next few decades. Achievable unit output, available plant sites, and fuel supplies then will become major concerns.

E. Human Resources

It is not clear that there is an actual shortage of trained staff in our industry. The majority of new staff, at least in the operating side of our industry, can be drawn directly from high schools and community colleges. This strategy offers distinct advantages. First, specific training can be done in a shorter time. Second, training can be more specific to the tasks assigned. Furthermore, the required practical-minded students are drawn to these institutions and job openings. Much of the training can be done in-house, but in the future it might be taken over by community colleges, especially if new regulations arise that increase requirements for formal certification of engineers and technologists in operating stations.

Future support for university programs will be contingent on the availability of funds. The UNENE program of support for university departments offering nuclear engineering will provide much-improved support for such programs, at least for the next few years.

Generally, we now must consider the separate needs of three different nuclear industries - operations and plant engineering, isotope production, and plant design development. The large majority of staff will, from now on, be located in operations. Engineering staff will be located partly in design offices and partly in operations - reflecting the clear need for engineering support throughout plant life. Isotope production is a very different venture than power production. Plant design development also is very much a separate activity. Research activities within that industry will most likely be much more diverse, and will include an increasing fraction of non-power applications.

Future Directions

The future development direction of CANDU power plants will depend on the needs of the marketplace.

Plants ordered at any particular time may not be the best ~ or even the cheapest - plants available on the market. Utility companies are very conservative and cautious of potential downside risk, especially of risk inherent in a new design. They will tend to order plants similar to those that have served their purposes well - in general, plants with high reliability, low maintenance cost, established safety, and those with which their staffs are already familiar. Minimum (and definite) project time from order date to in-service also weighs very heavily in the choice of any particular plant. The capital cost needs to be driven lower and lower. The ACR represents a very good move in this direction.

Public attitudes are very important in the decision on plant type. The basic idea is that the public must TRUST the plant operators day and night for a lifetime. Trust is built best through the public's observation of good behaviour over a long period. Trust is easily lost but must be painfully earned. We should be able to prove that human injury outside the plant boundary is not a real possibility.

The Next Decades

Given the present fragile state of public acceptance, one can judge that excellent operation must be the most important focus of our attention. Trust must be earned, and built up steadily. Unless we are successful at this task the society may not accept the introduction of thousands of new nuclear plants, whatever their apparent merits.

The world will face a massive energy shortage over the next decades - petroleum will not run out but will become more and more expensive in real terms. How to respond? Both short-term and long-term priorities apply.

In the short term we must be prepared to install dozen, if not hundreds of new power plants every year, around the world.

In the medium term we must be prepared to continue installing even larger numbers of installations but must make them cheaper. The ACR-700 development is well suited for this medium term, along with a number of other commercially available plant designs that soon will become available.

In the longer term, two major problems are apparent today. (I do not include waste management because it appears to be an "artificial" problem - one that will be solved over time through rational discussion.) The two obvious problems are available sites for new power plants, and adequate fuel supply for those plants.

In addressing the first problem, developers should look at both unit size and flexibility of plant sites. Large unit size is often judged to equate to better economics. However, this might not always be true. Smaller units fare better in terms of capital cash flow and operating flexibility. Plants that can be located on offshore islands, or even on-board barges, can help to ease the siting problem - and smaller unit sizes are favoured in such cases. If the plant is intended to produce only electricity then smaller unit size plus site flexibility may offer the best combination.

Fuel supply is a much more complex problem. First, it is clear that extensive fuel recycle will be mandatory within the next hundred years in order to control the real price of uranium supply. Second, the fraction of potential energy extracted from each mined ton of uranium must be increased substantially. A useful target minimum is 20 percent of potential energy extracted, though 50 percent would be much better - this to be compare with today's 1 to 2 percent. Twenty percent extraction would allow us to win uranium from phosphate fertilizers, and 50 percent would make economical direct extraction from ocean water.

Nuclear energy supply may become essential in fields far removed from electricity production. The most obvious is production of transportation fuels. Many other non-electric energy systems now depend on petroleum in one form or the other -increasing prices of petroleum will eventually force these industries to seek an alternate supply of primary energy.

Nuclear plants with several optional outputs - electricity, hydrogen transportation fuels, industrial process gases desalination, oil production from tar sands, etc. would appeal to a much broader market base and could be more flexible in response to changed economic conditions.

Can nuclear meet the demand?

This brief talk cannot begin to cover all the bases. Four recommendations do stand out, however:

(1) We must build the trust of the general population - current nuclear plants must be run well.

(2) We need to design for increased safety - so that it can be shown that there will be no need for evacuation.

(3) We need to ensure long-term nuclear fuel supply - by increasing the energy from each ton of uranium.

(4) We need to design nuclear plants that are faster to build, better performing, and cheaper.

Proposals for A New Canadian Licensing Basis

by R.A. Brown[1], P.H. Wigfull[2], G.H. Archinoff[3]

Ed. Note: *When Allan Brown saw the paper Moving Along the Risk Informed Path by S Petrella et al in the September 2004 issue of the CNS Bulletin he contacted me about the work referenced below. After some discussion it was agreed that we would run the Executive Summary of his report to the Canadian Nuclear Safety Commission in the CNS Bulletin with the hope that it will stimulate discussion among those concerned about the licensing of Canadian nuclear power plants. Having been involved with the early "risk based" approaches to reactor safety I am pleased to see this initiative. Fred Boyd*

Abstract

A contract was issued to R.A. Brown & Associates Ltd to assist the CNSC staff develop a new licensing basis document for use in assessing the acceptability of new reactor designs submitted for licensing under the Nuclear Safety and Control Act. The Project had two main deliverables: Licensing Guide: Design and the Basis for Licensing Guide: Design. The approach taken to develop these documents was top-down, systematic and comprehensive. Current regulatory requirements and industry standards and practices for the licensing of a CANDU reactor were examined, and the suitability for application to the ACR assessed. Where necessary, changes were proposed and/or new requirements recommended. The IAEA Safety Standards Series Document NS-R-1 entitled "Safety of Nuclear Power Plants; Design" was used as the template for the Licensing Guide: Design.

The final reports submitted to the CNSC propose modifications intended to make the overall licensing process more risk informed than the current deterministic based approach. It requires a combination of deterministic analysis and Probabilistic Safety Assessments. The reports recommend the adoption of Quantitative Safety Goals, and a new event classification scheme for analysis of accidents is proposed. Recommendations are also made to modify several of the current rules for the design of systems in the areas of reliability, shutdown requirements, trip requirements, sharing of instrumentation and equipment between process and safety systems, safety classification, containment leakage requirements and the introduction of Operating Limits and Conditions. These modifications, if accepted by the CNSC, will bring the Canadian licensing process more into line with accepted international practice; at the same time ensuring plants built to these requirements will provide a high level of safety.

As far as is practicable the proposed requirements can be applied to both future CANDU and future non-CANDU reactors.

Introduction

The Canadian Nuclear Safety Commission (CNSC) is undertaking a review of the Advanced CANDU Reactor (ACR) being designed by Atomic Energy of Canada Limited (AECL). The objective of the review is to provide a statement as to whether there are any fundamental barriers that would prevent the licensing of the new CANDU reactor design in Canada under the Nuclear Safety and Control Act. It is intended that the review should use a new Licensing Basis Document as a guide for the assessment.

A scoping study carried out in 2003 by two of the authors proposed that the most appropriate method of preparing a new Licensing Basis Document would be to conduct a top-down, systematic review based upon the IAEA Safety Standards Series Requirements Document NS-R-1 *"The Safety of Nuclear Power Plants: Design"* (Ref.1), modified to take into account specific Canadian licensing requirements and the unique features of the CANDU reactor. NS-R-1 was recommended as the primary template for the review since it is believed to reflect best international practice for both existing and future nuclear power plants. It has been developed in a systematic manner and is, with a few exceptions, very comprehensive. A proposal based on the scoping study was accepted by CNSC and the findings of the work are summarized in this article.

A systematic review of each section of NS-R1 was under-

1 R.A. Brown & Associated Limited

2 Wigfull Consulting

3 Candesco Research Corporation, resigned from the project in May 2004

taken to determine if its requirements are: adequately covered by existing Canadian regulatory documents, or adequately covered by CSA and other national standards, or adequately covered by AECL/Utility documents. A review of the existing CNSC regulatory documents was carried out to determine if these: are fully applicable to the ACR design, or are inadequate, or impose requirements that are either unnecessary or unrealistic. It was determined by the authors that in some instances the current requirements were overly restrictive and that there were areas covered in NS-R1 that were not encompassed by the current requirements.

The original proposal was directed at the preparation of a Licensing Basis Document for the ACR-700 only, but at the request of CNSC staff the document was developed to apply to any future design of CANDU reactor and, to the greatest extent possible, to other non-CANDU reactor types. In response to this request the project team was successful in making the majority of the requirements technology neutral, but concluded that a relatively small number must remain reactor design specific.

These proposals are under consideration by the CNSC. CNSC staff have advised the authors that a Licensing Basis Document, reflecting the views of CNSC staff, will be issued as a guide for the assessment of the ACR design in December 2004. The new Licensing Basis Document may differ in some aspects from the proposal outlined in this article.

Objectives

The objectives of this project were to:

(1) Develop a Licensing Guide for the design of future CANDU reactors, and as far as practical, for non-CANDU reactors, in Canada,

(2) Develop a Basis for the Licensing Guide that explains the rationale for the proposed new requirements, and

(3) Ensure that the requirements are more risk-informed and more consistent with accepted international practice than existing requirements.

Initial Finding

Work on the project started with an examination of the ACR-700 design to identify key features that might not meet existing regulatory requirements. This examination was conducted as a means of raising potential issues and not for identifying changes to accommodate the ACR-700 itself. The two most important questions raised by this initial work were:

- Is the long-standing requirement for two equally effective shutdown systems still necessary for a reactor, which, unlike existing CANDU reactors, has a negative void coefficient of reactivity, and hence behaves differently under some accident conditions?
- Is the arrangement where certain equipment is shared between systems acceptable, even though this does not conform to current regulatory requirements?

These questions were not resolved until the next phase of the work that involved a systematic review of current Canadian regulatory requirements and practices against those of NS-R-1. It quickly became evident that an overall framework was required to put any proposed new requirements into context.

The existing Canadian approach originated in a document known as the Siting Guide (Ref. 2) which introduced the concept of dual failure accident analysis. Basically this required that the nuclear power plant be designed for a single system failure, such as a pipe break, combined with a coincident failure of a safety system, such as a shutdown system. Over the years this concept of dual failures was developed further by the addition of specific requirements for safety systems, and included requirements for analysis of initiating events with failures of other safety systems.

The approach is unique to Canada and is not widely understood outside the country. It was very relevant at the time it was introduced in the 1960s and 1970s, but has not been critically examined over the years. As a consequence, the authors concluded that the approach is out-dated in that it no longer captures many of the advances that have been made in the international nuclear safety community. These include the use of safety goals, advances in reliability engineering, use of probabilistic safety assessments and the need for severe accident management. While several of these advances have been covered by informal agreements between the CNSC and the licensees, they have not been captured in a formal, integrated and comprehensive manner. The project's initial finding therefore was that comprehensive new regulatory requirements could only be developed by introducing an overall framework based on modern international practice.

Basic Approach

The project concluded that the new framework should be based on the application of the principle of defence-in-depth originally developed by the International Nuclear Safety Advisory Group (INSAG) and subsequently embodied in NS-R-1. The defence-in-depth approach has wide support among member states.

The project also concluded that the defence-in-depth model should be complemented by the adoption of formal safety goals to ensure that the design is optimized in terms of risk and that the overall approach to demonstrating the design adequacy is more risk-informed.

Defence-in-Depth Concept

The application of the concept of defence-in-depth in the design of a plant provides a series of levels of defence (inherent features, equipment and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails.

1. The first level of defence is to prevent deviations from normal operation, and to prevent system failures. This

leads to the requirement that the plant be soundly and conservatively designed, constructed, maintained and operated in accordance with appropriate quality levels and engineering practices, such as the application of redundancy, independence and diversity.

2. The second level of defence is to detect and intercept deviations from normal operational states in order to prevent Anticipated Operational Occurrences (AOOs) from escalating to accident conditions. This level of defence is provided by control systems.
3. The third level of defence assumes that, although very unlikely, the escalation of certain AOOs may not be arrested by a preceding level, and a more serious event may develop. This level of defence is provided by engineered safety features, more generally known as safety systems.
4. The fourth level of defence addresses accidents, including severe accidents, in which the design basis may be exceeded and ensures that radioactive releases are kept as low as practicable. The most important objective of this level is the protection of the confinement function. This may be achieved by complementary measures and procedures to prevent accident progression, and by mitigation of the consequences of selected severe accidents, in addition to accident management procedures.
5. The fifth and final level of defence is aimed at mitigation of the radiological consequences of potential releases of radioactive materials that may result from accident conditions. This requires the provision of an adequately equipped emergency control centre, and plans for the on-site and off-site emergency response.

This concept requires that systems be designed to provide overlapping layers of defence-in-depth. It also requires that these systems be designed using conservative criteria and take into account a wide range of both operating and accident conditions. For the second and third levels of defence-in-depth, i.e., for AOOs and design basis accidents, it must be shown by conservative, deterministic analysis that applicable reference dose limits are not exceeded.

Systematic application of the defence-in-depth concept ensures that the requirements are derived in a consistent manner and they are graded according to importance to safety. The concept remains, however, essentially deterministic in nature and, by itself, does not meet the objective of moving towards a more risk-informed regulatory environment.

Safety Goals

Safety goals were originally introduced to determine if a nuclear power plant designed using traditional deterministic design rules is safe enough. Safety goals consist of numerical goals or targets that are directly linked to potential health effects to people in the neighbourhood of the plant. These targets are expressed in terms of the risk of a fatality caused by the operation of the nuclear power plant being a very small percentage (<1%) of the risk posed by other activities. The use of safety goals represents a risk-informed approach that requires probabilistic techniques, i.e., probabilistic safety assessments (PSAs), to determine the overall safety of the plant.

There are two fundamental safety goals, one relating to early fatalities and the other relating to late or delayed fatalities. Early fatalities are linked to accident rates (e.g. industrial, traffic, etc.) while late fatalities are linked to cancer rates. The actual numerical safety goal limits proposed in this project are conservative surrogates of these two goals to simplify their calculation.

The first of these surrogates, a defence-in-depth measure designed to limit reliance on the containment system, is the severe core damage frequency goal which requires that the frequency of accidents that could lead to severe damage is very low, i.e., less than once every hundred thousand years. The numerical value is based on that suggested by INSAG for new nuclear power plants. It is widely accepted in the international nuclear community.

The second surrogate is the large release frequency goal. This goal refers to the frequency of an off-site release that would result in the need for long-term, or even permanent, evacuation of the surrounding population as a result of extensive ground contamination. This requirement is more restrictive than that needed to meet the fatality goals. A numerical value of once every million years is recommended as a suitable level for such events. Again this value is widely accepted in the international nuclear community.

Overall Framework

The main elements of the defence-in-depth approach complemented by the use of safety goals are shown schematically in Figure 1. The process requires that the station be designed to conservative, deterministic rules and the effectiveness of the design be assessed by a combination of deterministic and probabilistic analyses. The authors believe that this has been applied systematically and consistently in developing the requirements in this report. The process was used to ensure that a rational balance between deterministic and probabilistic requirements has been achieved and that these requirements are self-consistent.

Major Findings

General

This report recommends a large number of new regulatory requirements for the licensing of future nuclear power plants in Canada. It is, however, important to note that many of these new requirements represent the good practices that have already been adopted by the industry. Examples of such requirements include those relating to the design of the core and the reactor coolant system. They have essentially been included here to provide a comprehensive set of requirements for the design of the plant. Some of the new requirements have been added to cover

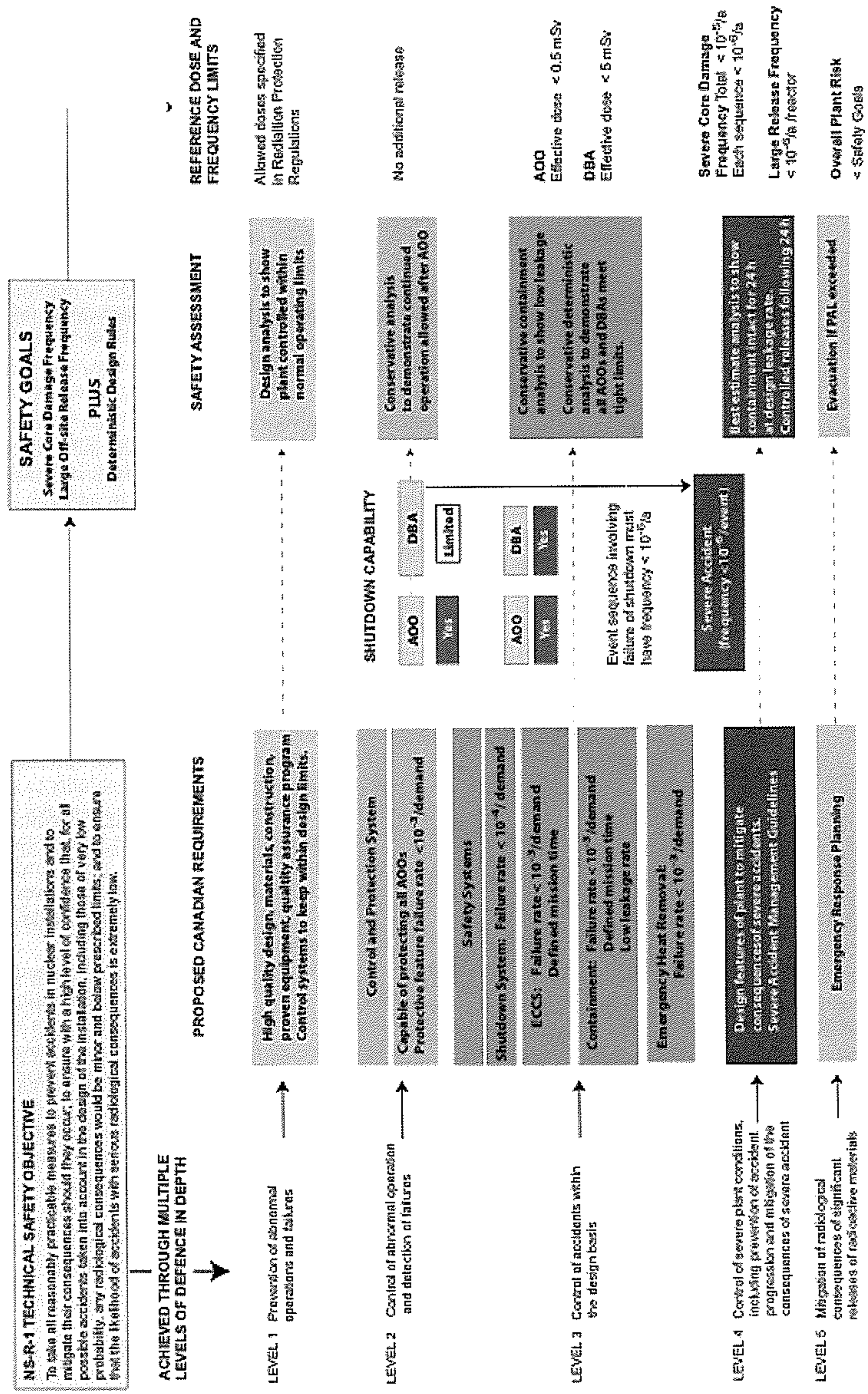


Figure 1: Licensing Basis Framework

topics not explicitly addressed in the current Canadian licensing framework (e.g. severe accidents).

However, the authors have also concluded that existing Canadian regulatory requirements do impose some conditions that appear to be overly restrictive, and are not consistent with modern international practice. It has also been concluded that, although the existing requirements were originally based on a risk model, some have been implemented in a manner that does not clearly reflect risk-informed principles. This is particularly evident when examining the requirement that CANDU reactors should have two equally effective shutdown systems.

In the following sections, the major findings of the study are summarized.

Adoption of Reliability Requirements

Currently safety systems are required to meet unavailability targets during operation. Specifically they are required to demonstrate that they are available for 99.9% of each year. This value was originally chosen as a requirement because it is measurable and does not depend on the availability of sound reliability models, which did not exist at that time.

The unavailability requirements remain unique to the Canadian licensing process and their application has led to a number of problems over the years. The most serious of these problems is the misconception that if the unavailability target is met during operation, there is no need to investigate causes of failure further or to improve the overall reliability of the system. This is also evident when the targets are not being met since it is common practice to increase the testing frequency rather than address the fundamental reliability issue.

It is proposed that formal reliability requirements should replace the current unavailability requirements. This proposal is made in the recognition that reliability engineering in the Canadian nuclear industry has advanced significantly over the years and that sound reliability models for the safety systems can be constructed. It is consistent with the standard approach used in reliability engineering worldwide and more closely reflects the real reliability requirement of a safety system (i.e., failure on demand) rather than the concept of availability per year. It overcomes the apparent weaknesses associated with the current unavailability approach and focuses attention on component failure rates and the trending of those rates.

Furthermore, the change allows for more realistic system reliabilities to be credited. This has a significant impact on overall reactor systems design, particularly that of the shutdown mechanisms.

It should also be noted that the proposal for adopting reliability requirements is fully consistent with Regulatory Document S-98 (Ref. 3).

New Shutdown System Requirements

As the project proceeded, it became clear that aligning the requirements for reactor shutdown with the levels of defence-in-depth recommended in NS-R-1 would be more logical from the risk perspective than the current deterministic approach. In the authors' opinion, this re-alignment does not represent a reduction in safety, but simply requires a re-configuration of the control and safety systems in a more logical and systematic manner.

NS-R-1 requires that control system(s) (level 2 defence-in-depth) shall be capable of dealing with all Anticipated Operational Occurrences (AOOs) and that safety systems (level 3 defence-in-depth) shall be capable of dealing with all AOOs plus Design Basis Accidents (DBAs). This differs from the situation for existing CANDU reactors, which have control systems that are not capable of dealing with the full range of AOOs.

Furthermore, to be consistent with the proposed safety goals, it must also be shown that any event sequence that can lead to severe core damage shall have a frequency of occurrence of less than once in one million years. In existing CANDU reactors this is met by having two independent shutdown systems, but there is an additional requirement that each shutdown system shall be "equally effective". The "equally effective" requirement means that all events are treated in the same manner, irrespective of their frequency of occurrence. There is a further deterministic requirement that there shall be two trip parameters for each event sequence, which is difficult to justify on the basis of cost-benefit and which has never been met fully in practice. Overall, the current requirements for reactor shutdown are not optimal in terms of risk.

It is proposed that there should be two independent and diverse systems which have the capability to shut down the reactor in accordance with the requirements of NS-R-1:

1. A protective component of the control system (level 2 defence-in-depth), which can shut down the reactor from all operational states and in the event of any AOO. It shall be fully buffered from other parts of the reactor control system.
2. A safety system (level 3 defence-in-depth), which can shut down the reactor from all operational states, in the event of an AOO, all design basis accidents and beyond design basis accidents which have an assessed frequency of occurrence of greater than once in one million years.

This proposal does not change the fundamental Canadian requirement of ensuring that severe accidents are of very low frequency. It does, however, simplify the design and reduce both the plant cost and the amount of maintenance and testing required. It removes arbitrary deterministic requirements and is fully consistent with risk-informed approach proposed in this article..

Revised System Classification

Existing CANDU reactor systems are classified as either process systems or special safety systems (i.e., the two shutdown systems, the emergency core cooling system and containment). Other systems have been classified as safety-related, but the practice for designating systems as safety-related has varied over the years and differs among utilities. It is proposed that a revised system classification, based on the NS-R-1 levels of defence-in-depth, should be adopted.

Structures, systems and components shall be classified as:

1. Process Systems (The first level of defence-in-depth)
   Systems that have as a primary objective the production of electrical power and/or steam.
2. Control and Protection Systems (The first and second levels of defence-in-depth)
   Systems that are intended to control process systems and to detect and intercept deviations from normal operating states in order to prevent AOOs from escalating into accident conditions.
3. Safety Support Systems (Support the third level of defence-in-depth)
   Systems that are designed to support the operation of the safety systems and mitigate the consequences of design basis accidents. These include all systems that may be required to supply electricity, cooling water and hydraulic or pneumatic pressure, and means of lubrication.
4. Safety systems (The third level of defence-in-depth)
   The Shutdown system, the Emergency Core Cooling system and the Containment and the Emergency Heat Removal System.

Revised Accident Classification

Existing regulatory requirements call for two types of accidents to be analyzed. Single failures are system failures such as a pipe break or a loss of control, and dual failures are single failures co-incident with a failure of a safety system.

The analysis of single failures is primarily to define the design of the safety and safety support systems. It is carried out using conservative assumptions. Similar analyses are carried out for all reactor types worldwide. The analysis of dual failures is unique to Canada and has several problems. The same standards of analysis as that for single failures is currently required, independent of the frequency of occurrence of the event and which, in some cases, goes well beyond the knowledge base making the results speculative.

It is proposed that accidents be classified in accordance with general international practice as follows:

1. Design basis accidents
   These are single failures and common cause events that are used to set the design requirements of safety and safety related systems. They are analysed using conservative assumptions. It must be shown that these single failures do not result in releases greater than the current single failure reference dose limit.
2. Beyond design basis accidents
   These are all failure sequences (initiating event plus failure of one or more safety and safety support systems) and combinations of events (an initiating event followed by another initiating event during the post-accident period) that have a frequency of greater than about once in ten million years. They are analyzed as part of a PSA using realistic or best estimate calculations. They include a category of events, known as severe accidents, in which there are multiple failures that lead to significant degradation of the core. It must be shown that the total frequency of all severe accidents meets the severe core damage goal of less than once every hundred thousand years.

The above proposals are considered to be more systematic and comprehensive than current requirements. They impose requirements for both PSA and severe accident analysis that are not formally covered at the present time. While the use of realistic or best estimate calculations for beyond design basis accidents may seem a relaxation, this is not really the case because of the inherent uncertainties in the current dual failure analyses.

Design for Severe Accidents

It is proposed that there should be a requirement that severe accidents be considered in the design and that severe accident management guidelines should be put in place. Basically the designer must consider what design features could be incorporated in the design which could deal with a severe accident to the greatest extent reasonable and practicable.

There is one design rule only proposed for severe accidents: the containment must be shown to remain intact for a period of at least 24 hours following such an accident. This rule is required to ensure that there is adequate time to evacuate the surrounding population in the very unlikely event that a severe accident should occur.

Sharing of Safety System Equipment

Current regulatory requirements prohibit the sharing of instrumentation and other equipment between safety systems and between safety systems and process systems. The intent is to ensure that each safety system is separated as far as practicable so that it may be considered as fully independent. This requirement is largely deterministic rather than risk-informed and has led to designs that are complex and require significant additional maintenance. It is a requirement which is unique to Canada; most other jurisdictions allow extensive sharing of equipment, subject to certain conditions

The project has reviewed the technical factors involved with sharing of equipment between safety systems and process systems and concluded that some sharing should be allowed. The sharing should, however, be subject to a number of defined rules. For example, there shall be no sharing of instrumentation between the shutdown safety system and the shutdown function of the control and protection system. Sharing of process and safety functions by a system may be permitted if these functions are not both required or credited at the same time and the system is designed to the standards of the system of higher importance with respect to safety. Where sharing of instrumentation is allowed, adequate isolation between safety and process systems must be demonstrated.

More Restrictive Containment Requirements

Containment leakage rates in existing CANDU reactors are higher than those associated with other designs. For future reactors it is proposed that the containment be designed such that leakage rates are comparable to the best available internationally. Additionally it shall be demonstrated that using a very high source term, the single failure reference dose limit shall not be exceeded. This requirement is a factor of 50 lower than that currently required.

Single Failure Criterion

It is proposed that a single failure criterion shall be applied to each safety system and its safety support systems. The design of these systems must ensure that they perform all safety functions required for a DBA in the presence of any single component failure, all failures caused by that single failure, and all failures and spurious system actions that cause or are caused by the DBA requiring the safety functions.

In Canada the single failure criterion has been required for the design of safety systems only. The proposal logically extends this to cover their support systems, i.e., those systems which supply the cooling water, the electrical power and the compressed air necessary to ensure that the safety systems continue to function.

Introduction of Operating Limits and Conditions (OLCs)

It is proposed that OLCs shall be required to ensure that plants are operated in accordance with design assumptions and intent. OLCs are not currently in place on Canadian reactors, although the industry has made a number of attempts to introduce them in the past. They are considered good practice and are required for most reactors worldwide.

OLCs typically include items such as safety limits, safety system settings, limits and conditions for normal operation and surveillance. The OLCs form a logical system in which these elements are closely interrelated and in which the safety limits constitute the ultimate boundary of the safe conditions.

Conclusions

This document presents the findings of the first comprehensive review of Canadian licensing requirements for many years. It introduces many elements that are consistent with modern international practice, including a formal requirement for safety goals. It attempts to balance deterministic and probabilistic requirements in a comprehensive and systematic manner. The result is a package of overall requirements which is self-consistent and from which individual items should not be selected or rejected. If CNSC staff wish to modify the recommendations great care should be taken to make sure that the approach remains systematic, and the balance between deterministic and probabilistic requirements is maintained. If this is not done, the authors believe that the current opportunity to move Canadian licensing requirements towards a more risk-informed and rational basis may be lost.

Although some current requirements are relaxed, this is proposed only in those areas where, in the authors' opinion, they are either unnecessary or where they cannot be justified in terms of risk. There is no reduction in safety in these cases, only a more appropriate application of risk information.

In many areas new requirements, such as the formal introduction of OLCs and design for severe accidents are recommended. These are not currently regulatory requirements and are intended not only for completeness, but to ensure a higher level of plant safety.

The authors believe that these proposals, if adopted by the CNSC, will result in nuclear power plant designs, which are not only simpler than current designs, but ones which are safer as well. These objectives are in accordance with the recommendations of the IAEA.

References

1 IAEA Safety Standards Series NS-R-1, Safety of Nuclear Power Plants: Design, Requirements, 2000

2 AECB "Reactor Siting and Design Guide" (Boyd and Jennekens) November 1964

3 Regulatory Standard S-98 "Reliability Programs for Nuclear Power Plants"

Waste Repository Planned for Bruce Site

Ed. Note: *At the October 22, 2004 meeting of the Council of the Canadian Nuclear Society, Frank King, of Ontario Power Generation, gave an interesting overview of the proposed repository described below. Subsequently he provided the CNS Bulletin with the material that is the basis for the following article.*

Ontario Power Generation (OPG) and Kincardine, the municipality nearest the Bruce site, have agreed in principal to the construction of a deep geologic repository for low and medium level radioactive waste on the site. The two parties signed the *"Kincardine Hosting Agreement"* on October 13, 2004 to proceed with planning, seek regulatory approval and further public consultation of the proposed project. A Construction Licence is not expected before 2013.

(Although Bruce Power has leased the eight reactors on the site OPG continues to manage the waste from those reactors as well as from its own plants at Pickering and Darlington. OPG operates the Western Waste Management Facility located on the Bruce site.)

The saga began in 2002 when OPG and Kincardine signed a Memorandum of Understanding (MOU) for the development of a plan for the long-term management of low and intermediate level waste at the Western Waste Management Facility (WWMF).

Golder Associates were engaged to carry out an *Independent Assessment Study (IAS)* of alternatives. The study, completed in early 2004, included geotechnical feasibility, safety and environmental analyses, a community attitude survey and interviews with local residents, businesses and tourists, and economic modeling to determine the potential benefits and impacts of three options. *(The study report can be accessed at http://ias.golder.com.)*

The three options studied in the IAS were:

- Enhanced Processing and Storage,
- Covered Above-ground Vault, and
- Deep Geologic Repository.

The IAS concluded that each of the options was feasible, could be constructed to meet international and Canadian safety standards with a considerable margin of safety, would not have significant residual environmental effects, and would not have a negative effect on tourism. The geology of the Bruce site was noted as being ideal for the Deep Geologic Repository option.

In April 2004, Kincardine Council endorsed the project and selected the "Deep Rock Vault" option as the preferred course of study for the management of low and intermediate level radioactive waste because it had the highest margin of safety and is consistent with best international practice. Subsequently the surrounding municipalities of Saugeen Shores, Brockton, Arron-Elderslie, and Huron-Kinloss expressed support for the Deep Geologic Repository proposal.

The Deep Geologic Repository involves the construction of rock vaults within stable, low permeability bedrock using conventional mining techniques. The geology at the Bruce site is ideally suited to isolation and containment of nuclear waste. The reference depth for the proposed repository on the Bruce site is 660 m below ground surface in low permeability limestone, which is overlaid by shale.

The underground repository would initially consist of a number of caverns or vaults arranged in parallel rows on either side of central access tunnels. A concrete floor would be poured to provide a stable base for stacking of the waste packages. The repository would have a modular design that would allow vaults to be added, as required, to meet OPG's low and intermediate level waste disposal needs.

Support buildings would be located on ground surface above the underground workings. Access to the repository would be through a vertical, concrete-lined shaft. A second shaft would be constructed for ventilation and emergency egress purposes.

The estimated expenditures associated with the proposed project amount to $800 million. Sufficient funds have already been deposited in the Ontario Nuclear Fund administered by OPG.

The model for the *Kincardine Hosting Agreement* was the Port Hope agreement, which was negotiated between the federal government and the communities of Port Hope, Welcome, and Clarington. The Port Hope agreement was negotiated for the long-term storage of more than one million cubic metres of historic radioactive waste, currently existing in those communities.

The key terms of the Hosting Agreement are:

- OPG will seek regulatory approvals to construct the proposed Deep Geologic Repository and Kincardine will support OPG's applications
- Kincardine and surrounding communities to receive $35 million (2004 dollars, inflation protected) in lump sum and annual payments over 30 years subject to achieving key milestones:
- Positive Community Consultation in Kincardine 2005

- Environmental Assessment Guidelines 2007
- Environmental Assessment Approval 2010
- Construction Licence 2013
- Operating Licence 2017
- Provision for all low and intermediate level waste produced during reactor operations until 2035, and for waste from decommissioning all 20 OPG reactors; approximately 200,000 m3
- Provision to negotiate repository expansion for additional low and intermediate level waste from any new-build reactors in Ontario
- No used fuel will be placed in the proposed deep geologic repository
- OPG will locate new jobs associated with the facility at the WWMF
- OPG will provide property value protection
- OPG and Kincardine will support the concept of a nuclear centre of excellence, trades and vocational schools, and international tours
- Prior to OPG moving to the regulatory approval stage, Kincardine Council will formally consult with Kincardine residents to determine if they support the Council resolution favouring the Deep Geologic Repository option

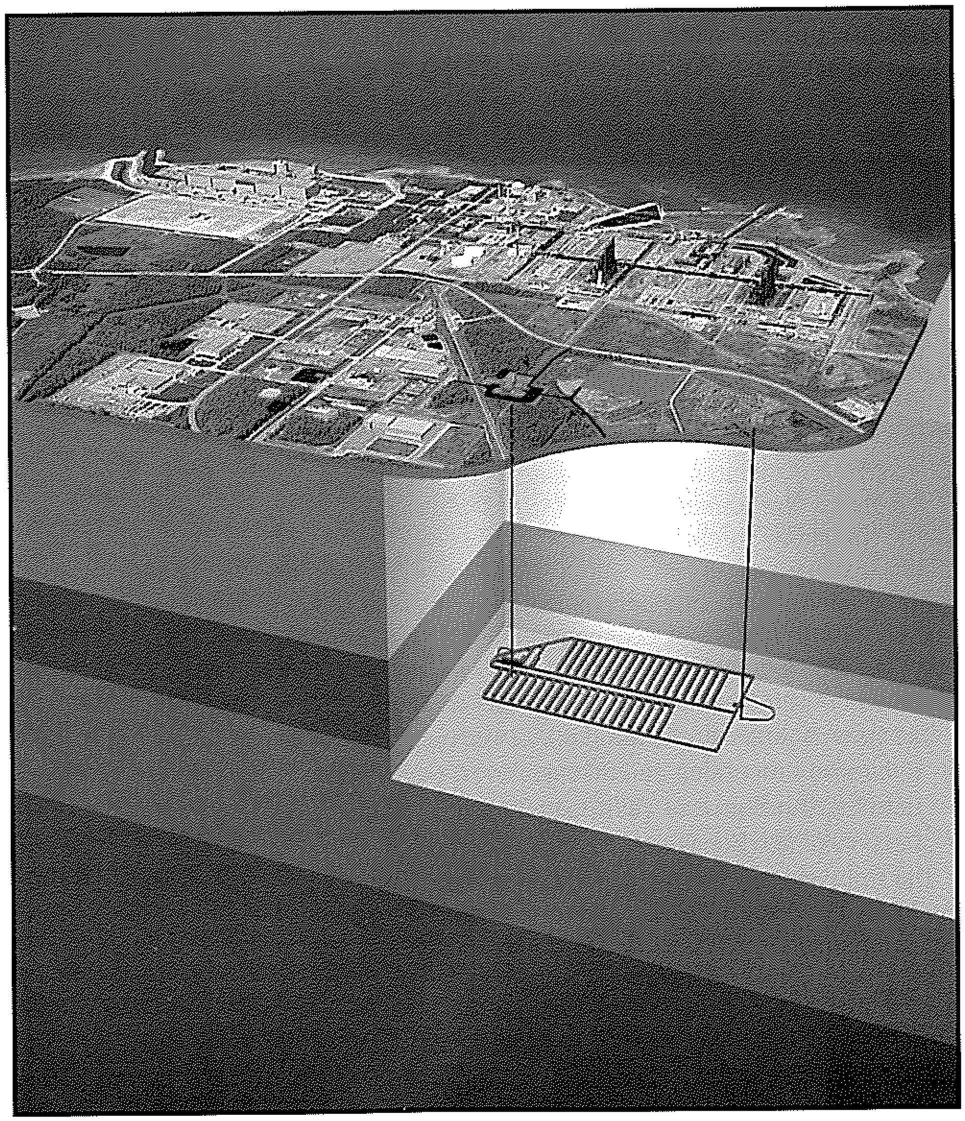
From mid-October 2004 to January 2005, Kincardine, assisted by OPG, will undertake a public dialogue consisting of provision of educational materials to all households in Kincardine. A storefront operation was opened at 759 Queen Street in Kincardine on October 15, 2004 and will remain open until January 22, 2005. It will provide a location where residents can discuss the proposed Deep Geologic Repository and obtain information.

An independent consultant will undertake Community Consultation, consisting of telephoning each residence in Kincardine during the first three weeks of January to determine the level of community support. The telephone calls would be followed up with a mail out if necessary.

The regulatory approvals process will include preparing an environmental assessment, completing site characterization and safety assessment studies, and obtaining a construction licence before construction could begin in 2013. These activities will provide additional opportunities for the public to receive information about the proposed project and to provide feedback on it.

Quintessa Limited, a UK firm with considerable experience in nuclear waste management, conducted a preliminary post-closure safety assessment of the proposed Deep Geologic Repository. They concluded that the proposed project could safely manage all the planned low and intermediate level waste.

Their assessment followed the standard approach recommended by the International Atomic Energy Agency. A reference assessment was devised to illustrate the expected evolution of the repository. This scenario dealt with the potential release of radioactive material from the repository and its subsequent movement. In addition potential future human intrusion into the repository was considered. This safety assessment will be updated with site specific data collected during site characterization studies and during construction.



Artist's rendition of proposed Deep Geologic Repository

Forum discusses ICRP draft 2005 Recommendations

ICRP proposes more of the same, plus rules for non-human biota

About 100 delegates from the nuclear and radiation protection communities gathered in Ottawa, November 1, 2004 to learn about and comment on the Draft 2005 Recommendations from the International Commission on Radiological Protection (ICRP).

The ICRP is in the process of bringing out a new set of basic recommendations to replace those of ICRP 60, which was published in 1990. In a new move, ICRP has issued a "draft" of the proposed 2005 recommendations and seeking comments. The November 1 Forum was convened with an objective of developing a consensus on a Canadian set of comments. It was organized and sponsored by: Canadian Nuclear Safety Commission (CNSC); Canadian Nuclear Association (CNA); Canadian Radiation Protection Association (CRPA) and the Federal/Provincial/Territorial Committee on Radiation Protection (FPTRPC).

Al Shpyth, of Cameco Corporation and chair of the CNA Committee on Regulatory and Environmental Affairs, welcomed delegates and outlined the objectives and structure of the meeting.

The opening speaker was Dr. Lars-Erik Holm of Sweden who is the incoming chairman of ICRP. Since publication of ICRP 60 in 1990 the Commission has issued 10 further publications, he noted, which included nearly 30 different numerical restrictions on dose and has adopted a policy for environmental protection.

The aim of the 2005 Recommendations is to consolidate all of this onto a single coherent set of Recommendations. "ICRP 60 still works but needs to be updated", he stated. That includes continuing the adoption of the LNT (linear no-threshold) hypothesis for dose-effect relationship.

The ICRP 2005 system of protection involves: justification; quantitative recommendations; and optimisation. Justification, Holm commented, is largely a political decision whether or not to allow a particular practice. For "quantitative recommendations" ICRP is recommending limits and constraints. Individual dose limits remain at 20 mSv/yr for workers and 1 mSv/yr for members of the public. Optimisation, he said, is more than ALARA (as low as reasonably achievable), it is a frame of mind, a protection culture.

ICRP's schedule is to adopt the new Recommendation in 2005 but printing may not occur until 2006. Given past experience, Holm said he did not expect any country to implement new regulations until 2009 at the earliest.

The next speaker was Kevin Bundy, acting Director of Radiation Protection at the CNSC. He said the CNSC basically supported the proposed recommendations for protection of workers and the public He noted that some of the "dose constraints" recommended were already being applied by the CNSC. The CNSC also endorses the ICRP sentiment on "safety culture" to engender a "state of thinking in everyone responsible for control of radiation exposure, "Have I done all I can to reduce doses?"

He was followed by Wayne Tiefenbach, co-chair of the Federal / Provincial / Territorial Radiation Protection Committee who began by explaining the role of the Committee to try to harmonize the regulations applying to radiation protection issued by the provinces and territories with those of federal departments and agencies. The Committee is still studying the possible impact of the proposed ICRP Recommendations.

The forum then broke into small groups to discuss the ICRP Recommendations for protection of humans.

After lunch, Dr. Lars-Erik Holm gave a further presentation, this time on ICRP proposals for the protection of non-human species. He said the ICRP decision to develop recommendations for non-human biota was not driven by any particular concern over environmental radiation hazards but by the need to fill a conceptual gap in radiological protection. ICRP 60 had stated that "standards of environmental control needed to protect man ... will ensure that other species are not put at risk". However, he noted, no explicit scientific evidence was quoted. Further attitudes toward the environment had changed. The objective is to: conserve species or habitats; maintain diversity of habitats and species; protect habitats and designated areas.

In recent years there had been four international conferences specifically on radiological protection of the environment: Stockholm 1996; Ottawa 1999; Darwin 2002; Stockholm 2003. ICRP has chosen an approach that uses a reference set of dosimetric models and environmental geometries that will be applied to reference animals and plants. The Commission is still developing its recommendations in this area. A Task Group on Reference Animals and Plants was formed in 2003 and is expected to report in 2005 when a new ICRP Committee 5 will be created.

Dr. Patsy Thompson of the CNSC followed, commenting that the CNSC was preparing regulations for the protection of non-human biota.

Al Shpyth gave a brief report on industry's view of protection of nun-human species from ionising radiation. He said industry welcomes ICRP acknowledgement that the current system has, in practice, provided an appropriate standard of environmental protection but can accept the desire to fill a conceptual gap. Industry urges that radiation protection be kept in perspective. Maintain the focus on humans, he recommended, and, in the environment, protect populations not individuals.

There was another round of small group discussion before the forum reconvened for the closing. Although there appeared to be some consensus on a few points the organizers said they would be reviewing all the comments and preparing a report for further consideration.

Canadian Light Source

Official opening of Saskatchewan synchrotron facility

***Ed. Note:** The following article is based on publications or news releases from Canadian Light Source Inc.*

The **Canadian Light Source** facility at the University of Saskatchewan was officially opened on October 22, 2004 marking the end of a five year construction project

The $174 million synchrotron and associated laboratory was built on time and on budget. It had received an Operating Licence from the Canadian Nuclear Safety Commission in July but the October event marked the real beginning of the huge project.

Up until now Canada was the only industrialized country without a synchrotron. Proposals had been made as early as 1974 but a combination of factors, especially tight government budgets, precluded any progress.

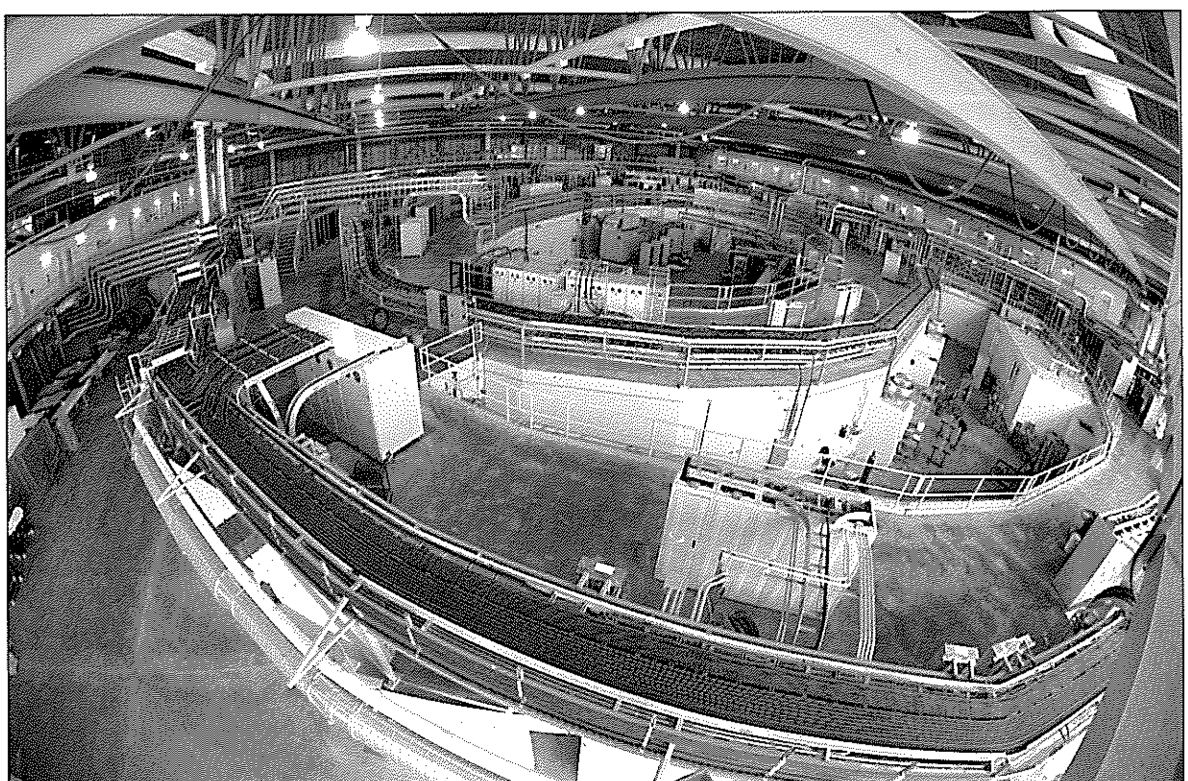
In 1990, the Canadian Institute for Synchrotron Radiation (CISR) was founded to co-ordinate and facilitate Canadian synchrotron-based research and improve access to synchrotron facilities. The national organization began to make Canadians aware of the business, industry, government and university uses of synchrotron radiation.

In 1994, the Committee for Materials Research Facilities of the Natural Science and Engineering Research Council (NSERC) published a document entitled: "Major Materials Research Facilities in Canada's Future." that recommended Canada should make an immediate commitment to develop a fully equipped dedicated national source for synchrotron light research

In early 1996, NSERC set up an international peer review panel to review proposals from institutions interested in building and operating the synchrotron light facility.

The group at the Saskatchewan Accelerator Laboratory (SAL) developed a design which was reviewed at a workshop at the University of Saskatchewan in November 1994. The SAL design -- a third-generation synchrotron with insertion devices, low electron beam emittance, 1.5-2.5 GeV ring (increased to 2.9 GeV in the final proposal), competitive flux and brightness both above and below 10 keV -- received the approval of the Canadian synchrotron user community.

As a result of the workshop, a compact, 12-sided, double-bend achromat (DBA) lattice was agreed upon as affordable



Birdseye view of bosster and storage rings Courtesy of Canadian Light Source Inc.

and suitable for the insertion devices that were planned.

At a May 1995 CISR meeting, it was decided to circulate the design to the Canadian scientific community and solicit bids for review. The solicited bids would be peer reviewed by a panel established by NSERC members. At that meeting, the proposed facility name was also approved as the **Canadian Light Source (CLS).**

NSERC established an Advisory Committee on Site Selection for the Canadian Light Source. Two institutions submitted proposals: the University of Western Ontario and the University of Saskatchewan. In May of 1996, the NSERC committee unanimously chose the Saskatchewan proposal.

The University of Western Ontario and the University of Saskatchewan teams then worked together to produce a combined proposal that was submitted to NSERC. The updated proposal called for an increase in the maximum ring energy to 2.9 GeV and the use of a small gap undulator to increase the availability of hard X-rays.

In March of 1999, the Canada Foundation for Innovation (CFI) awarded the project the entire request of $56.4 million which amounted to 40 per cent of new construction costs of $140.9 million (an existing building and other equipment account for the remaining $32.6 million of the project's $173.5 million total value). This has the largest CFI award to date.

Crucial to the proposal's success was the $25-million commitment from the Government of Saskatchewan. The U of S also agreed to own the facility during the construction phase and for five years after commissioning.

The operation of the facility will be carried out by CLS Inc., a wholly owned subsidiary of the University of Saskatchewan.

Design

The CLS consists of a 200-300 MeV electron linac, a booster to ramp the beam to 2.9 GeV, and the main booster ring which is designed to operate at an energy of 2.9 GeV and at currents up to 500 mA. The ring lattice is based on the double bend achromat (DBA) cell. Twelve straights (9 available for insertion devices), 24 bending magnets, and over 40 possible beamlines are more than enough to satisfy the needs of the Canadian community of SR users for years. The storage ring which circles the booster ring is 54 metres in diameter.

Brightness of the CLS is comparable to other "3rd generation" sources of similar operating energy.

An optimization of the focusing properties has been done to achieve a suitably low emittance while maintaining a relatively small machine circumference. This places non-zero dispersion in the straights reserved for insertion devices which has been shown to enhance the net brightness of the photon beam.

The characteristics of the CLS lattice will permit use of the new small gap undulator technology that is becoming available. A super-conducting wiggler will enhance photon energy.

The peak design energy for the machine will be 2.9 GeV. This will be done in a machine with a circumference of 171 m. The emittance is 18.2 nm-rad. This is a respectable performance for a ring of such small circumference and is comparable to other light sources in the same energy range.

With the 500 mA beam currents anticipated, the CLS will provide brightness and flux to satisfy the Canadian synchrotron research community. The design of the beamlines permits up to 100 experiments to be conducted simultaneously.

Applications

The concentration of the work at CLS will be in support of research in materials, environmental and life sciences. However, about 25& of its capacity will be available for industrial applications.

Materials science

Most of the early synchrotron users in Canada and worldwide used synchrotron radiation to study surfaces and materials and this is still a very major use of synchrotron radiation in Canada and abroad. Techniques such as photoemission or photoelectron spectroscopy (often called XPS), along with XAFS, have been incredibly important for studying metals, alloys, semi-conductors, overlayers, nanomaterials, superconductors, and some non-conductors such as polymers, minerals, and the surface reactions of these materials with gases and liquids, and industrial oils.

Many geoscientists are now using synchrotron radiation to study the chemistry of different minerals, meteorites, and glasses (and very small inclusions in these materials), and the surface reactions of many minerals.

Environmental science

The hard x-ray micro-XAFS line will have a major impact on both academic and industrial research. XAFS can be obtained on virtually any type of environmental sample - gas, liquid, solid of any type The detailed chemistry of almost all elements heavier than Ti at the ppm level can be obtained on very small amounts of sample. The CLS industrial effort is presently concentrating on the determination of the chemistry of As and Se in amorphous materials from mining operations.

Life sciences

A large group of Canadian scientists will be using the protein crystallography (PX) beamline to determine the atomic structures of biological macromolecules such as proteins by single crystal X-ray diffraction. Protein crystallography has shown exponential growth since the late 1970's when only about a dozen protein structures were known, and PX is an essential tool to all the biological scientists working in virtually all fields of biological and medical sciences. Protein crystal structures of functional proteins of viruses are important to understand the mechanisms of virus infections, and to provide targets for virus control. Recent antiviral drugs against AIDS are enzyme inhibitors, and their design took advantage of detailed protein-drug interactions provided by crystal structures of the enzymes with the drugs--so-called structure function relationships.

GENERAL **news**

New OPG Board has Nuclear Expertise

Four of the seven new members of the Board of Directors of Ontario Power Generation appointed in mid October 2004 bring extensive nuclear experience. Among this group is **Gary Kugler** who retired early this year from the position of Senior Vice-President, Nuclear Products and Services at Atomic Energy of Canada Limited. Gary is a charter member of the Canadian Nuclear Society.

The other three are:

James Hankinson

James Hankinson served from 1996 to 2002as president and chief executive officer of New Brunswick Power Corporation, which operates the Point Lepreau nuclear generating station.

Donald Hintz

Donald Hintz is the retired President of Entergy Corporation, where he was responsible for Entergy's 30,000 megawatts of generating assets, including 10 nuclear plants. Prior to his appointment as President he spent seven years as President and CEO of Entergy Operations Inc. where he oversaw the improvement of Entergy's nuclear operations to top quartile performance.

Corbin A. McNeill Jr.

Corbin McNeill is the retired Chairman and Co-Chief Executive Officer of Exelon Corporation, which was formed by the merger of PECO Energy and Unicom Corp. At PECO, he had been Chairman, President and CEO, having joined PECO in 1988 as Executive Vice-President, Nuclear. Prior to PECO, he oversaw nuclear operations at the Public Service Electric and Gas Company and the New York Power Authority.

The other appointees are:

David J. MacMillan

David MacMillan is non-executive director of Killingholme Power, and has extensive international experience in power projects and financing. He is also a former Vice President and Regional Director of Finance for International Generating Company (InterGen).

C.Ian Ross

Ian Ross, a member of the Law Society of Upper Canada, served at the Richard Ivey School of Business at the University of Western Ontario from 1997 to September 2003, most recently as Senior Director, Administration in the Dean's Office. He currently serves as an officer and/or director for a number of corporations including: World Heart Corporation, GrowthWorks WV Canadian Fund Inc., PetValu Canada Inc., Comcare Health Services and Praeda Managements Systems.

Marie C. Rounding

Marie Rounding is the former President and Chief Executive Officer of the Canadian Gas Association (CGA) and served as Chair of the Ontario Energy Board (OEB) from 1992 to 1998. She has extensive background in regulatory and administrative law, and as a leading regulator was involved in the deregulation of the natural gas markets and the early restructuring of the electricity sector in Ontario.

William (Bill) Sheffield

William Sheffield is the former Chief Executive Officer of Sappi Fine Paper plc., and a former Executive Vice President of International Operations and Corporate Development at Abitibi Consolidated. He has experience in operating large international industries. He also spent 17 years with Stelco.

David G. Unruh

David Unruh is a lawyer currently serving as Vice Chairman of Duke Energy Gas Transmission Canada, a Duke Energy company. In this role, he acts as Vice Chairman and as a director of Westcoast Energy Inc. (based in Vancouver and Calgary) and Union Gas Limited (based in Ontario). He is also a director of Pacific Northern Gas Ltd, a director of the Wawanesa Insurance Group of companies, and a director of RAV Project Management Ltd.

Chairman of the Board is **Jake Epp**, a former federal cabinet minister, who was appointed in April 2004. He had served as interim Chairman from December 2003. In early 2003 he led a panel to review the delays and cost overruns at the Pickering A nuclear generating station and was a member of the provincial government's review committee that was created in December 2003 and headed by John Manley, to look at OPG's future role in the province's electricity market; examine its corporate and management structure; and decide whether OPG should go ahead with refurbishing three more nuclear reactors at the Pickering A nuclear power plant.

SNC-Lavalin Takes Over Canatom NPM Inc

SNC-Lavalin Inc. has acquired all the remaining shares of Canatom NPM Inc. held by AECON Construction Group Inc., and will become the sole owner of the Toronto company. SNC-Lavalin was the majority shareholder in Canatom, holding 61.25% of its shares. Aecon held 38.75%.Canatom has served the nuclear industry for 40 years in Canada, on projects associated with the Pickering, Darlington and Bruce facilities in Ontario, Point Lepreau in New Brunswick, and internationally, on the Qinshan Nuclear Power Plants units 1 & 2 in China, and on some of the Wolsong Nuclear Generating Stations in South Korea.Canatom, formed in 1967, is the largest private sector nuclear engineering company in Canada. It offers a complete range of services in project, supply and construction management, design engineering, operating plant support and the management and decommissioning of radioactive materials.The SNC-Lavalin companies form one of the leading groups of engineering and construction companies in the world. They employ nearly 15,000 people in offices across Canada and in 30 other countries around the world and are currently working in some 100 countries.

ACR in US Licensing Demonstration

The Advanced CANDU Reactor design of Atomic Energy of Canada Limited is one of two projects granted awards by the US Department of Energy under its Nuclear Power 2010 program to begin the first phase of DoE's Nuclear Plant Licensing Demonstration program. The projects will demonstrate the untested combined Construction and Operating License (COL) regulatory process.

The application was made by Dominion Energy of Richmond, Virginia in response to a Nuclear Power 2010 program financial assistance solicitation issued by DOE on Nov. 20, 2003. The Dominion project could lead to a licence to build and operate an Advanced CANDU Reactor (ACR-700) at the North Anna site in Louisa County, Virginia. The Dominion-led team includes AECL and its U.S. subsidiary AECL Technologies; Bechtel Power Corporation, both at Frederick, Maryland; and Hitachi America Inc, located in Tarrytown, New York. If a nuclear power plant order results from this work, Dominion potentially could have a new nuclear power plant in operation as early as 2014.

The other award went to NuStart Energy of Chester County, Pennsylvania. The NuStart Energy consortium will evaluate the Westinghouse Advanced Passive Pressurized Water Reactor (AP-1000) and the General Electric Economic Simplified Boiling Water Reactor (ESBWR). The consortium plans to select a final reactor technology and a site by 2007. If a nuclear power plant order results from this work, NuStart Energy could also have a new nuclear power plant in operation as early as 2014. NuStart Energy consists of nine nuclear power companies – Exelon Generation, Entergy Nuclear, Southern Company, Constellation Generation Group, Duke Energy, Tennessee Valley Authority, Florida Power & Light Company, Progress Energy, and EDF International North America, and two reactor vendors - General Electric and Westinghouse Electric Company.

Cooperative agreements for each of these projects are anticipated to be in place by December 2004 and a detailed project planning phase will be completed in FY 2005. A final decision by DoE and the industry consortia whether to proceed to the implementation phase of the projects will be made during the project planning phase.

Vacuum Building Inspection at Bruce B

On October 13, 2004 Bruce Power completed a 25-day inspection of the Bruce B generating station's vacuum building

A unique safety feature of CANDU reactors, the vacuum building is designed to prevent the release of radioactive material to the environment in the event of an accident. A large cylindrical structure, it is connected to the generating station by a pressure relief duct and kept at negative atmospheric pressure so any release of radioactive steam can be sucked into the vacuum building.

The Canadian Nuclear Safety Commission requires a thorough examination of the structure every 12 years. Since the vacuum building is a shared safety system, all four Bruce B units had to be taken off line so crews could check the integrity of the structure and examine any penetrations where pipes or ducts are run to ensure there were no leaks or cracks.

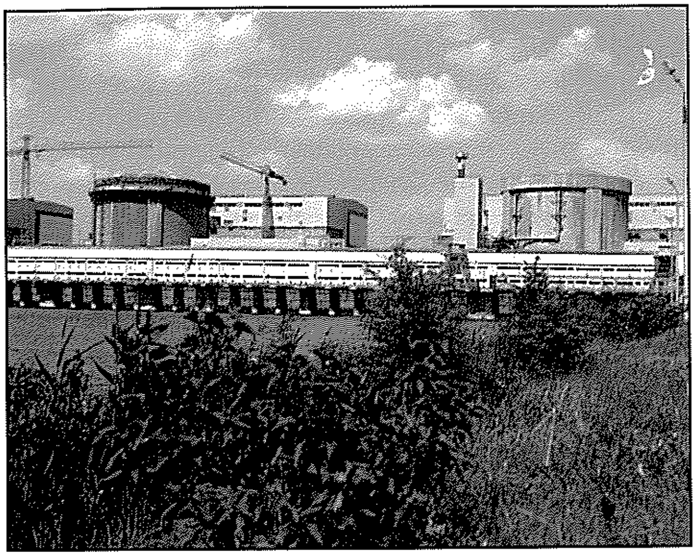
When the station's vacuum building was last inspected, in 1992, it was 36 days before the first unit returned to service. "Completing it in 25 days while meeting our high standards for safety and quality is a credit to the skill and expertise of our staff.", said Bruce CEO Duncan Hawthorne.

The vacuum building program was coordinated to run in concert with the scheduled inspection of Unit 6, which was taken off line on Sept. 11 for approximately three months. Unit 5 will also be kept off line for a short duration after tests during the vacuum building inspection showed a heat transport pump requires additional maintenance.

Throughout the Bruce B vacuum building inspection, from September 18 to October 13, Units 3 and 4 at Bruce A continued to operate at 100 per cent capacity.

Commissioning of Cernavoda 2 begun

In mid October Atomic Energy of Canada Limited (AECL) announced that commissioning has begun of the second CANDU 6 unit at Cernavoda, Romania. Cernavoda is located near the Black Sea, approximately 180 kilometres



from Bucharest. AECL and its partner, ANSALDO Energie of Italy, are managing the engineering, procurement, construction and commissioning processes.

The commissioning milestone was celebrated with the energizing of the main service transformer and associated switchgear. A ceremony, commemorating the event, was attended by Senator Peter Stollery, Chairman of the Senate of Canada's Foreign Affairs committee and Dr. Ken Petrunik, Chief Operating Officer of AECL.

"We are extremely pleased with the work and the progress that has been made in such a short period of time," stated Dr. Ken Petrunik, Chief Operating Officer of AECL. "This is another solid example how AECL is able to consistently manage new-build projects on schedule and within budget."

As of October 1, 2004, Cernavoda II project was 74% complete, with project completion scheduled for March 2007. More than 1,500 workers are employed during the construction of which 110 are AECL experts from Canada and 80 ANSALDO employees from Italy

Over the next two years, construction and commission will be completed on numerous plant systems including fuel loading, initial start up and connection to the grid for an in-service target of early 2007.

The Cernavoda NPP Unit 2 project is the second in a series of five CANDU 6 Power Plants that began construction in the early 1980's. Cernavoda Unit I nuclear power plant has been successfully operating since 1996.

AECL and the Romanian Nuclear Agency signed a Memorandum of Intention to extend a Memorandum of Understanding for cooperation in the research and development of nuclear energy and technology.

AECL signs agreements with Chinese agencies

In September Atomic Energy Canada Limited (AECL) signed a "Memorandum of Understanding" (MOU) on cooperation in nuclear safety with the National Nuclear Safety Administration (NSSA) of China.

This M(U on cooperation is for the pre-application review of AECL's Advanced CANDU Reactor (ACR). The MOU was signed by Dr. Ken Petrunik, Senior Vice President and Chief Operating Officer for AECL and Mr. Li Ganjie Director General of the National Nuclear Safety Administration of China.

AECL has also signed an agreement with the Nuclear Safety Centre (NSC) of the State Environmental Protection Administration of China, which defines the detailed program for the ACR pre-application review.

"Signing of this MOU provides opportunities for meaningful exchanges on nuclear safety culture, ideas on safety design and licensing processes to enhance the safety of nuclear power plants. It also helps to advance the CANDU capability in China for applications now and in the future." said Petrunik.

The work of NNSA/NSC experts will focus on selected areas of the advanced ACR design. The agreement defines in detail the cooperation model, which provides for training sessions in China for NNSA/NSC staff, as well as participation of NSC staff in the ACR review by the Canadian Nuclear Safety Commission.

Conference on Waste Management

The Canadian Nuclear Society will be holding a national conference on **Waste Management, Decommissioning and Environmental Restoration,** for Canadian nuclear activities in Ottawa, May 8 - 11, 2005

This is the first national conference on these topics in eight years. Over 140 summaries of papers have been received by the Technical Committee, and organization of the plenary and technical sessions is now under way. The Conference, which will be held in the Crowne Plaza hotel, will start with an evening reception on Sunday, May 8, followed by the plenary and breakout sessions on May 9-11. Technical visits to four sites will follow on Thursday, May 12.

For more information including an informative "backgrounder", registration form and hotel information, go to the CNS website www.cns-snc.ca.

The deadline for the reduced early conference registration fee is March 31, 2005.

Bruce Power considering investing in Point Lepreau

After considering a number of potential investment partners for the Point Lepreau generating station NB Power has asked Bruce Power for a proposal.

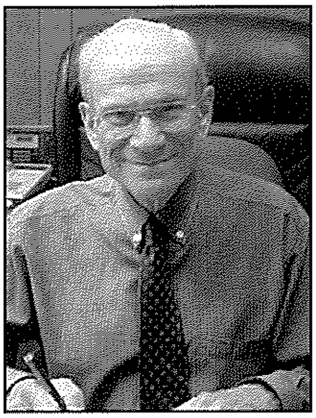
Bruce Power will be sending a team to Point Lepreau starting in December 2004 to conduct a due diligence examination. NB Power president David Hay has stated that the visit of the team does not commit Bruce Power in any way.

NB Power is also holding discussions with Atomic Energy of Canada Limited over the scope and cost of the proposed rehabilitation of the Point Lepreau plant. (See paper by Paul Thompson in this issue of the CNS Bulletin.)

Review of cost estimates

Dr. J. A. L. (Archie) Robertson, a retired senior researcher from the Chalk River Laboratories of Atomic Energy of Canada Limited and a veteran participant of many inquiries into Canada's nuclear program has prepared an insightful review of the cost estimates made over the years by Ontario Hydro and Ontario Power Generation. It can be accessed at his website: www.maga.ca/~jalrober.

Obituary



Frank Stern

Frank Stern, a pioneer of the Canadian nuclear program and one of its few successful entrepreneurs died in Hamilton, November 29, 2004, in his 81st year

Born in Breslau, Germany, Frank immigrated to England in 1939. He graduated in 1950 with a B.Sc. in Engineering from the University of London, England, and married his wife Jane in 1951. A year later they moved to Canada, where Frank joined the Motor Division of Canadian Westinghouse in 1955, soon transferring to the newly formed Atomic Power Division. He was attached to the Chalk River Nuclear Laboratories of Atomic Energy of Canada Limited for three years in the late 1950s, working in the reactor development group.

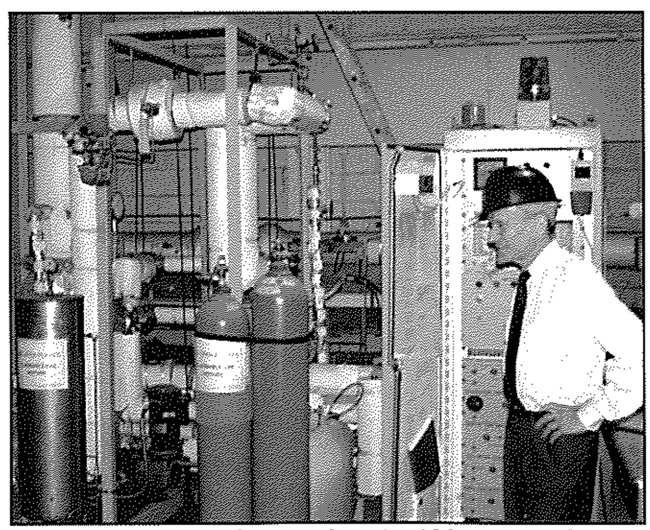
Canadian Westinghouse decided to build a testing laboratory as a means of entering the Canadian nuclear program, since its major competitor, Canadian General Electric Company had earlier obtained the design role for the first nuclear power plant, NPD. Frank returned to Hamilton in 1959 and secured contracts that lead to the formation of the Westinghouse Systems Test Laboratory in 1962. He was appointed manager of that small laboratory and continued in that role until 1987 when Westinghouse decided to get out of the nuclear business in Canada and sold the laboratory and its fuel manufacturing plant in Port Hope.

The purchasers did not wish the laboratory and offered to sell it to Frank. Although by then in his 60s, Frank, together with many of his associates at the laboratory, accepted the challenge. Renamed as the Stern Laboratories the organization has provided essential testing and research capabilities, primarily in thermalhydraulics, and has served customers in the USA and overseas as well as Canada. Frank was president of the company until the mid 1990s when he passed that role to Gordon Hadaller and became chairman. Nevertheless he was at the laboratory almost every day and took an active role in its continuing work.

In 2003, the Canadian Nuclear Society and Canadian Nuclear Association honoured Frank with the Outstanding Contribution Award as an outstanding provider of innovative thermalhydraulic research and development services. The citation also noted his contributions as a mentor to many engineers who were first introduced to nuclear power through their employment in his company.

Frank was an avid athlete, who until this fall followed his daily mile-long swim with a walk of at least three miles. He always preferred climbing stairs to riding elevators and at age 75 went high-altitude trekking in the Andes.

A gathering to remember Frank was held in Stoney Creek on Saturday, December 4th.



A photo of Frank Stern taken in 1997 in his favorite haunt - his laboratory.

CNS news

CNS Council Changes

There have been several changes to the 2004-2005 CNS Council since the Annual General Meeting. Walter Thomspon resigned his position as 1st Vice-President, as of August 26. Walter's work commitments were making it difficult to ensure that he would be able to meet the expanding obligations that would necessarily accompany the 1st VP and President roles. Rather than persevere, with the likelihood of being unable to make a satisfactory contribution to the activities of the CNS, Walter decided that it would be best to resign, giving a replacement candidate the opportunity to learn for almost a full year, before undertaking the President's commitments.

John Luxat accepted the 1st VP position. Dan Meneley was appointed by Council to fill the 2nd Vice-President position. In addition to his other duties, John Luxat now chairs the Program Committee.

Additionally, David Malcolm resigned his position as a Member at Large of Council.

The CNS Council has adopted a **Privacy Policy** to comply with the ***Personal Information Protection and Electronic Documents Act*** (PIPEDA), the Canadian federal privacy law, which came into effect on 2004 January 1. For further information on the Privacy Policy, see the CNS website.

The CNS Officers' Seminar was held August 26-27 at Hilton Garden Inn in Cambridge. The Bruce, Darlington, Golden Horseshoe, New Brunswick, Ottawa, Sheridan Park, and Toronto Branches were represented as were most Divisions and Committees.

Most of the attendees of the CNS Council's "Officers' Seminar" which was held August 26, 27 at Cambridge, Ontario, took advantage of CNS President Bill Schneider's invitation to visit the plant of Babcock & Wilcox Canada. Shown is the group on the steps to the entrance of the B & W offices.

Education and Communications Committee

CNS Council has approved a new scholarship. The proposal was developed by Elisabeth Varin of Ecole Polytechnique, Eleodor Nichita of University of Ontario Institute of Technology, Blair Bromley of AECL, and David Jackson of McMaster University. The scholarship is designed to support summer work projects by undergraduate students at universities related to nuclear science or technology, or that are industry related. Two awards will be made, each of $1500. Check the CNS website for further information.

Bryan White, ECC Co-Chair attended the first annual Science & Technology Awareness Network held in Toronto, November 10. The CNS joined STAN earlier this year to improve our knowledge of other organizations that promote science and technology education and public awareness in Canada. (www.scienceandtechnologynetwork.ca)

In 2005 Canada, together with many other nations around the world, will be celebrating the World Year of Physics (WYP2005) under the auspices of the IUPAP. Each country is arranging their own events to mark this year, which was chosen to celebrate the 100th Anniversary of Albert Einstein's three famous publications in physics on the theory of relativity, quantum theory, and the theory of Brownian motion.

The ECC is preparing a poster to display a timeline of the development of nuclear science and technology in Canada as a WYP2005 activity. Member's suggestions for events to include in this poster will be appreciated.

Inter-Society Affairs Committee

The CNS is a member of the Engineering Institute of Canada, and is participating in the organizing committee for the EIC Conference on Climate Change Technology:

Engineering Challenges and Solutions in the 21st Century, May 9-12, 2006 - Ottawa Congress Centre, Ottawa, Ontario. The program will examine engineering solutions that either mitigate or adapt to climate change. This three-day conference will interest engineering and environmental technology practitioners of all disciplines; delegates from industry, manufacturing, academia, government agencies and regulators; consulting engineers, and special interest groups; economists, financial, and legal experts and other specialists working in the climate change field.

Design and Materials Division

As CANDU units age, there is an increasing need to understand the behaviour of systems, equipment and components and to adopt sound technical practices to manage the challenges and maximize station performance. As such, the Design and Material Division of the CNS is proposing a series of CANDU Life Cycle Management workshops.

The general objective of this series of workshops is to heighten awareness amongst:

- Plant/Utility staff of component aging issues, system/component interactions and experience at other plants
- Design & service support organizations of how utility engineers address aging in their plant programs and to explore existing and emerging strategies.

The first workshop is dedicated to the Heat Transport System. The workshop is being hosted by OPG, Darlington and it will be held February 21- 22, 2005.

For further information, check out the CNS website, or contact

James Nickerson AECL, Mississauga. Tel: 905-823-9060 nickersonj@aecl.ca	Dan Meneley Tel: 705-657-9453 mmeneley@sympatico.ca
Prabhu Kundurpi Consultant Tel: (416) 292 2380 kundurpi@sympatico.ca	Jacques Plourde OPG - Darlington Tel: 905-623-6670 x7348 jacques.plourde@opg.com

Environment and Waste Management Division

The Environment and Waste Management Division of the CNS is organizing a National Conference on **Waste Management, Decommissioning and Environmental Restoration For Canada's Nuclear Activities: "Current Practices and Future Needs"** to be held at the Crown Plaza Hotel in Ottawa, Ontario on 2005 May 8-11. Information on the conference and a call for papers are available on the CNS website.

The main objective of the 2005 conference is to provide a forum for discussion and exchange of views on the technical, regulatory and social challenges and opportunities in radioactive waste management, nuclear facility decommissioning and environmental restoration activities in Canada. The conference is organized into several plenary sessions and eight technical tracks:

1. Low-and intermediate-level wastes
2. Uranium mining and milling wastes
3. Used nuclear fuel
4. Decommissioning
5. Environmental restoration
6. Policy, economics and social issues
7. Licensing and regulatory issues
8. Radioactive materials transportation

This three-day conference will interest waste management, decommissioning and environmental technology practitioners; delegates from industry, academia, and government agencies and regulators; consulting engineers; financial and legal experts; and other specialists working in the field. While the conference is focused on the Canadian scene, about 10% of the submissions received come from foreign and international organizations, which will provide insights into how other countries are dealing with similar issues. An equipment and services exhibition will be held in conjunction with the Conference.

A guest program will take advantage of various attractions in the Ottawa area. Four technical visits are being organized to several nuclear facilities: AECL's Chalk River Laboratories, the Low-Level Radioactive Waste Management Office activities at Port Hope, Elliot Lake uranium mines, and Hydro-Québec's Gentilly 2 nuclear generating station and AECL's shutdown Gentilly 1 prototype reactor.

Deadlines

Receipt of paper summaries:	2004 September 30
Notification of paper acceptance:	2004 October 30
Receipt of draft full papers:	2005 January 15
Receipt of final full papers:	2005 February 28

For further information, contact the Conference Chair Michael Stephens, Quality Assurance Manager for the Decommissioning & Waste Management Unit of AECL, Tel: (613) 584-8811, email: stephensm@aecl.ca.

Fuel Technologies Division

The Fuel Technologies Division organized a CANDU Fuel Technology Course held at the Holiday Inn Mississauga, Ontario, October 18-20.

Nuclear Science and Engineering Division

The CNS **6th International Conference on Simulation Methods in Nuclear Engineering** was held in Montréal, Québec, October 12-15. The Conference Executive Chair, Hong M. Huynh prepared the following summary.

The conference started with the Plenary Session which was followed by parallel sessions.

In total, there were 15 sessions which covered all aspects of nuclear modelling and simulation, including, Reactor Physics, Thermalhydraulics, Safety Analysis, Fuel and Fuel Channels, Containment, and Numerical Methods. The Organizing Committee was very pleased to witness strong interest and support from international experts in addition to Canadian colleagues. Indeed, there were 82 presentations including the 6 plenary papers, from more than 10 countries, such as Argentina, China, Canada, Germany, India, Italy, Korea, Lithuania, Mexico, Sweden, United Kingdom, and the United States. There were more than 120 attendees with 117 paid registrations to the conference.

Two conference lunches were served respectively with guest speakers Michel Beaudet and Bill Schneider. Dr. Beaudet presented the "Electricity Supply & Demand in the Québec Scene" and CNS President Bill Schneider talked about "The CNS – 25 Years of Success – On With the Future".

A Conference Banquet was organized with musical entertainment. During the banquet, Dr. Meneley shared with delegates, his vision on "Now that We've Arrived, Where Shall We Go?"

The Organizing Committee would like to thank Hydro-Québec, Ontario Power Generation, Bruce Power, Atomic Energy of Canada and Candesco Research Corporation for their financial support.

The yearly fall offering of the **CANDU Reactor Safety Course** was organized at the Best Western Governor's Inn Hotel in Kincardine, Ontario, October 25 - 27. There were 42 participants and 15 speakers. The CNS thanks all the speakers who contributed their time to present a lecture and ensured the course was, again, a success.

Nuclear Operations and Maintenance Division

The Nuclear Operations and Maintenance Division has 3 course offerings this fall and winter.

A course on **STEAM GENERATORS – Real Design and Potential Degradation** will be held November 22-23, 2004 at the offices of Babcock & Wilcox Canada in Cambridge, Ontario. This course is intended to provide insight into SG design and degradation for those on the front lines of SG inspection, maintenance and repair. While one can easily go look at the equipment, all that can be seen is insulation; this course presents the objectives and features of SG design and degradation that can occur in operation. It is intended to present:

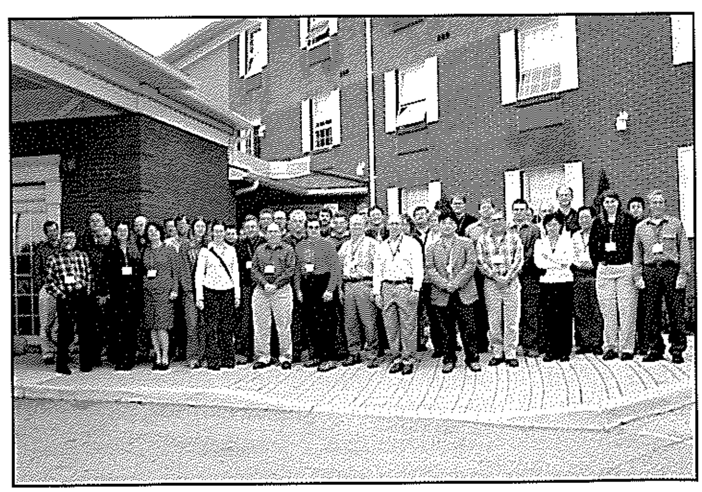
- Design and development objectives, methods and priorities
- To have the registrant do basic heat transfer calculations (to de-mystify the design process)
- To identify the range of possible in-service degradation conditions which has developed or may develop around the world
- To discuss the approach to their inspection, stabilization and repair

A new course on **EDDY CURRENT FOR NON-SPECIALISTS** will be held November 29-30, 2004 at the offices of Babcock & Wilcox Canada in Cambridge, Ontario. The objective of this course is to introduce eddy current theory and practice to ECT non-specialists (all the rest of us) who will never be ECT experts but who need to understand the fundamentals of the method, what it can do, what it cannot do, the limits of its accuracy, the probability of detection and the methodologies of data analysis, resolution, presentation and storage.

The course **CHEMISTRY Of Preservation And Degradation** will be offered January 31 – February 1, 2005 at the offices of Babcock & Wilcox Canada Cambridge, Ontario.

The objective of this course is to present to those who have an interest in the design, operation, maintenance, manufacture and repair of CANDU power reactors, their systems and equipment:

- CANDU chemistry fundamentals
- Overview of plant systems
- Current chemistry practice and specifications for the major process systems, including:
  - Primary Heat Transport System, Auxiliaries
  - Moderator System and Auxiliaries
  - Steam, Feedwater and Condensate System
  - Service Water Systems Course Materials



Attendees at the fall 2004 CNS CANDU Reactor Safety Course pose outside the Governor's Inn Hotel in Kincardine, the venue for the course.

Chalk River

Following on the success of the 2004 Chalk River Branch Essay Contest, Blair Bromley has initiated it again for 2005. This contest is open to secondary school students in Renfrew County. Blair has assembled an information package on the contest for consideration by other CNS Branches. The CRB Annual General Meeting is planned for Tuesday Dec 7 2004. The speaker, Mr. Dick Bourgeois-Doyle has written a book on George Klein (www.nrc-cnrc.gc.ca/highlights/0409klein_e.html), the engineer who headed up the ZEEP engineering design team.

(See also www.sciencetech.technomuses.ca/english/about/hallfame/u_i19_e.cfm)

Darlington

Darlington Nuclear and the Darlington Branch of the CNS are hosting the 1st CANDU Life Cycle Management Workshop, Feb 21-22, 2005. Preparations are under way to welcome some 60 participants to a very interesting and timely session on HTS Aging Management. This event will also be an opportunity to promote the CNS and boost interest for Branch activities at Darlington.

New Brunswick

The New Brunswick Branch of the CNS held its annual dinner On Saturday October 16th. This event was sold out as CNS members, consulting firm representatives, Saint John area politicians, and UNB staff and students gathered to hear keynote speaker Jerry Grandey, President and CEO of Cameco, deliver a strong message on the need for and obligations to expand global nuclear energy options and respond to the growing need for reliable energy in the face of climate change.

The evening began in a typical Southern New Brunswick fashion with guests gathering to mingle and enjoy smoked salmon with their cocktails. Dinner was preceeded with greetings from Saint John Deputy Mayor Michelle Hooton and remarks from CNS President Bill Schneider. Ms. Hooton received a spontaneous round of applause when she told the audience that Saint John Council strongly supports PLGS refurbishment. Following dinner, Rod Eagles, PLGS Refurbishment Project Director provided a brief overview of the project status. The New Brunswick Branch CNS Award was presented to Paul Thompson. Jerry Grandey then provided the audience with a clear strong message on the role he feels the nuclear industry can and must play in providing clean, reliable power options to a global community that places increasing strain on resources and the environment. He sees an increasing role for both Cameco and Canada in this regard. His vision that - there is a strong future for the nuclear industry - was well received by the audience.

It was a good night out for the New Brunswick nuclear community. The CNS NB branch wishes to thank the organizing committee, the various corporate sponsors, and especially Jerry Grandey and Al Shpyth who made the trip from Saskatoon, and Bill and Lynda Schneider who travelled from Cambridge for making the evening a success.

Ottawa

Jim Harvie is the new Chair of the Ottawa Branch.

Toronto

The Nuclear Power Group is a University of Toronto student organization that is committed to developing understanding and support for nuclear power. The goals of the organization are twofold, to promote nuclear power in the predominantly anti-nuclear student body at U of T and provide a source of reliable information on nuclear power. For further information, contact Justin Alizadeh, President, justin.alizadeh@utoronto.ca.

Prize Draw at CNS Fuel Technology Course

A prize draw was held on 2005 October 20 at the end of the CNS Fuel Technology Course, at the Holiday Inn, Mississauga, ON.

Three prizes of a complimentary CNS membership, valid to 2005 December 31, were drawn from among the badges of Course attendees. Winning badges were drawn by Erl Køhn, one of the Course organizers.

The winners were:

- Joe Berney
- Colleen Polley
- Doug Burton

By the luck of the draw, the affiliation of all three winners was Zircatec Precision Industries!

Tirage de prix au Cours de la SNC sur la Technologie du combustible

On a fait un tirage au sort le 20 octobre 2005 pour trois adhésions gratuites à la SNC, bonnes jusqu'au 31 décembre 2005. Le tirage eut lieu à l'hôtel Holiday Inn, à Mississauga, ON, à la fin du Cours de la SNC sur la Technologie du combustible.

Trois noms ont été tirés au sort, à partir des porte-insigne des inscrits au Cours. Les noms des gagnants ont été tirés par Erl Køhn, un des organisateurs du cours.

Les trois gagnants sont:

- Joe Berney
- Colleen Polley
- Doug Burton

Par le jeu du hasard, tous les trois gagnants sont affiliés à Zircatec Precision Industries!

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.

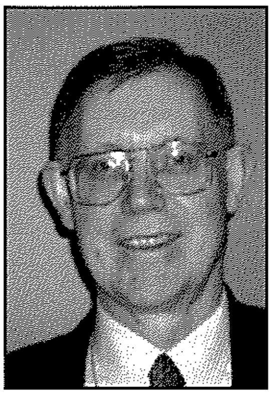
Imtiaz Ahmed, UOIT
Syed Sarmad Ahmed, UOIT
Hassan Albasha, Bruce Power
Fawaz Hassan Ali, UOIT
Mohamed H. Ali, UOIT
Mingwang An, Atlantic Nuclear Services Ltd.
Saravanan Ananthalingam, MCGI
Fanomezantsoz Pierre H. Andrianirina, UQTR (Laboratoire Hydro-Québec)
Farina Baig, UOIT
Joe Berney, Zircatec Precision Industries, Inc.
Doug Burton, Zircatec Precision Industries, Inc.
Catherine E. Campbell, International Safety Research
Doug Chambers, SENES
Joseph G.Chaput, UOIT
Kalyani V. Chari, National Physical Laboratory
Evan Charles, Canada Border Services Agency (Co-op)
Ramy George Chehade, U of T, Chem. Eng. Dept.
Cornelia Chilian-Turtoi, Ecole Polytechnique de Montréal
Adrian F. Connolly
Michael A. Cormier, AECL
Zoë Lewis Coull, U of T
Andrew L. Daley, University of New Brunswick
Richard J. De Klerk, Ryerson University
Lillian V. De Melo, U of T
Harsh Singh Deol, UOIT
Raymond S. Dickson, AECL
Paul J. Dinner, OPG - NWMD
Ian Dovey, AECL
Martial Doyon, Hydro-Québec
Corinne A. Draesner, Bruce Power
Rodney Eagles, New Brunswick Power
Stephanie M. Eisan, UOIT
Chris Elliott, Bruce Power
Mohamed Arafat El-Mansi, UOIT
Esteban A. Estévez
Christine Anne Fahey, AECL
Kevin J.W. Fice, University of Western Ontario
Jinghui Gao, U of T
Spencer J. Gill, UOIT
René Girard, Hydro-Québec
Sandeep Gopaul, UOIT
Joshua M.A. Guin, UOIT
Yujun Gao, AECL
Vimmi Gupta, UOIT
Michelle Hall, Framatome ANP Canada Ltd.
Steve Hamilton, AREVA Framatome-ANP Canada Ltd.
Shane W.D. Hart, UOIT
Navid Hasanzadeh, MCGI
Yung C. Hoang, Nuclear Safety Solutions Ltd.
Dave G. Ingalls, Cameco Corporation
Kevin D. Jayawardene, UOIT
Jiantao Jiang, UOIT
Tae-Cheol Jung, UOIT
Xun-Keun Jung, UOIT
Alexandre L. Kearnan, UOIT
Stephanie C.E. Kelley, UOIT
Anas M. Khaial, McMaster University
Jayden T. Kilbourne, Nuclear Safety Solutions Ltd.
Michael J. Knaszak, Acres-Sargent & Lundy
John P. Krasznai, Kinectrics Inc.
Bart H. Kreps
Peter M. Lang
Milena Lazaroski, Nuclear Safety Solutions Ltd.
Jennifer P. Leung, Nuclear Safety Solutions Ltd.
Laurence Kim-Hung Leung, AECL
Hsiao-Tsu Ryan Lin, UOIT
Cory R.H. Linton, UOIT
Mike Liska, Bruce Power
Liaohui Liu, UOIT
Benjamin R. Lootsma, UOIT
Carlos Lorencez, Ontario Power Generation
Ravi Mahadevan, Valcor Eng. Corp.
Peter W. Mason, General Electric Canada
Cheryl D. McCulloch, Bruce Power
Kirk A. Megra, UOIT
Hany Michael, Wardrop Engineering Inc.
Ron Mitchel, AECL
Dimitri D. Moisseev
David R. Moore, Babcock & Wilcox Canada Ltd.
Rafael P. Moya, Nuclear Safety Solutions
Waqas Amjad Mughal, UOIT
Nguyen-Chuong (Tom) Nguyen, AECL - CRL
Jeffrey L. Norton, MDS Nordion
Ronald C. Oberth, Flipside Solutions Inc.
Alan L. O'Brien, Acres International
Pat O'Cain, Bruce Power
Carlos Jr. Alberto O'Donell, UWO
Chris O'Reilly, Summit Controls Ltd.
Samuel L. Orr, U of T
Anand N. Panditrao, Bruce Power
Wladimir Paskievici, École Polytechnique
Robert Pollock, AREVA-COGEMA Resources Inc.
Evon P.D.B. Reynolds, Nuclear Safety Solutions Ltd.
Bo Wook Rhee, KAERI
Michael J. Rhodes, AECL
Julia A. Richman, Merlin Gerin Corporation
Cole William Roberts, UOIT
Tarek Saghir, U of T
Johan Saladeen, UOIT
Tracy V. Sanderson, AECL
Mujahid P. Saqib, UOIT
Derek M. Sawyer
D. Paul Schroeder, Bruce Power
Hamdi Ahmad Seid, UOIT
Thomas E. Shannon, Parsons/MMM
Gaurav Sharma, UOIT
Sat N. Sharma, AECL
Sergiy Shaula
Terrance M. Slobodian, UOIT
Prabhu Srinivasa Raghavendra, University of Ottawa
Keith P. Stratton, New Brunswick Power
Jennifer L. Suddard, UOIT
Ho-Chun Suk, Canadian Nuclear Safety Commission
Taeyong Sung, Canadian Nuclear Safety Commission
Eva E. Sunny, UOIT
Taha Shabhir Husain Sutarwala, U of T
Jawad Zahid Uppal, UOIT
Raynald Vaillancourt, Hydro-Québec
Alexey Voevodskiy
John G. Waddington
Timothy F. Walker, Nova Machine Products Corporation
William D.Warnica, Babcock & Wilcox Canada Ltd.
Brian Whiffin, CH2M HILL Canada Ltd.
Jeffrey Chun Wai Yau, University of Waterloo
Shahrokh Zangeneh, Kinectrics Inc.
David G. Zekveld, UOIT

AECL: Atomic Energy of Canada Ltd.
MCGI: Marc Garneau Collegiate Institute
UOIT: University of Ontario Institute of Technology
U of T: University of Toronto
UWO: University of Western Ontario

Errata

The following write-up was inadvertently omitted from the article on Nuclear Achievement Awards in the last issue of the CNS Bulletin. Our apologies to Dr. Luxat.

Outstanding Contribution Award - Dr. John Luxat



John Luxat has made significant contributions to safety analysis of CANDU reac-tors through his ability to specify and interpret R&D findings and apply them to analytical codes to develop safety cases.

John Luxat received his B.Sc. and M.Sc. in electrical engineering from the University of Cape Town, South Africa, and his Ph.D. in electrical engineering from the University of Windsor. He held a number of positions in consulting firms in Canada before joining Ontario Hydro in 1977. In 25 years at Ontario Hydro, Ontario Power Generation and Nuclear Safety Solutions, John has been a key contributor in the areas of spatial reactor kinetics, trip assessment, multiphase thermal hydraulics and interpreting R&D results to practical purposes through his understanding of nuclear science and engineering. He has initiated and conducted cooperative projects in nuclear safety and nuclear technology that have contributed to the successful licensing and safe operation of CANDU reactors in Canada.

John has also represented Ontario Hydro, OPG and Canada on international projects and in the activities of international organizations such as the IAEA. In the near future, Dr. Luxat will take up a new professorship at McMaster University.

PURPOSE OF THE AWARD:

The Outstanding Contribution Award recognizes Canadian-based individuals, organizations or parts of organizations that have made significant contributions in the nuclear field, either technical or non-technical.

Highlights of CNS privacy policy

The CNS Council has adopted a **Privacy Policy** to comply with the ***Personal Information Protection and Electronic Documents Act*** (PIPEDA), the Canadian federal privacy law, which came into effect the beginning of 2004. Following are highlights of that policy. The full document can be found on the CNS website <www.cns-snc.ca>.

CNS Membership Information ("Information")

The Information is gathered to allow the CNS to do business with, and supply services, to its membership.

The Information will be used

to distribute CNS mailings and e-mails (the Bulletin, Nuclear Canada, notices about Conferences and Courses, etc.) to members.

to list CNS members in the CNS Annual Membership Directory.

to allow CNS Branch Chairs to contact their Branch Membership.

Credit-card information is used strictly for the payment of CNS membership fees, for the payment of course, seminar, workshop, and conference fees, and for the purchase of requested products (e.g., Conference Proceedings).

The CNS Membership list will not be sold or provided to organizations other than occasionally to the Canadian Nuclear Association (CNA), for the purpose of mailings to CNS members.

This Policy shall be posted on the CNS web site.

CNS members are required to update their Information on an annual basis at the time of membership renewal, or when there is a change in their address, telephone number, email address, etc. CNS Office Staff will update CNS records within ten (10) working days of the Information being received.

Members may contact the CNS Office Manager at any time to confirm their Information details.

Information is stored in password-protected computers and is not left accessible or unprotected.

The physical membership files are kept in locked filing cabinets in the CNS Office Manager's home office.

Award for Deep River Science Academy

The Deep River Science Academy has received a 2004 Michael Smith Award. The award, sponsored by Science and Engineering Research Canada (NSERC), was created in honour of the late Canadian biochemist who won a Nobel Prize in 1993 for his pioneering work in genetics.

The Michael Smith Awards honour individuals and groups who make outstanding contributions to the promotion of science in Canada, through activities encouraging popular interest in science or developing science abilities.

The $10k prize accompanying the award will help the non-profit organization which was founded 17 years ago by AECL employees to encourage high school students to pursue science or engineering careers by exposing them to real-life research.

CGS IEEE Canada CSEM CNS CSME CSSE

Call For Papers
Engineering Institute of Canada Conference
Climate Change Technology:
Engineering Challenges and Solutions in the 21st Century
May 9-12, 2006 - Ottawa Congress Centre, Ottawa, Ontario Canada

FACT: The engineering implications of global climate change are a challenge and an opportunity for engineering innovation.

The EIC 2006 EIC Climate Change Technology Conference - Engineering Challenges and Solutions in the 21st Century will examine engineering solutions that either mitigate or adapt to climate change. This three-day conference will interest engineering and environmental technology practitioners of all disciplines; delegates from industry, manufacturing, academia, government agencies and regulators; consulting engineers, and special interest groups; economists, financial, and legal experts and other specialists working in the climate change field.

Interested authors/panelists are invited to submit a proposal for a manuscript for presentation at a Paper or a Poster Session or for an electronic presentation to be part of a Panel Discussion. Proposals should include:

- The title.
- An abstract (< 400 words) providing a synopsis of the central theme.
- The best suited Conference Track and Topic(s) (*see web site for full list*).
- A statement regarding previous publication of papers.
- A list of the full names, affiliations, and contact information for the authors or panelists.
- A designated primary contact person for the proposal with full contact information.

Conference Tracks: 1. - Policy, Strategy and Regulations. **2.** - Monitoring & Recording GHG Emissions and Climate Indicators. **3.** - Engineering for Mitigation (Reductions and removals of GHG.). **4.** - Engineering for Adaptation (Allowing for CC in infrastructure design). **5.** - Financial and Risk Management. **6.** - Continuing Education and Engineering Roles. **7.** - Standards and Protocols. **8.** - Modeling and Analysis.

Full information and guidelines for proposals are available on the Conference website: www.ccc2006.ca or from Terrance Malkinson at malkinst@telus.net or 403-282-1065.

Important Dates:

Proposal Submission	**March 18, 2005**
Notification of Acceptance	**May 31, 2005**
Authors Submit Original Manuscript	**August 31, 2005**
Notification of Acceptance to Authors with any changes	**October 31, 2005**
Authors submit final manuscript submission for CD ROM	**January 31, 2006**
Panelists submit presentation for CD-ROM	**February 28, 2006**
Presentation at Conference	**May 10, 11 or 12, 2006**

In collaboration with:

Canadian Council of Professional Engineers / Conseil canadien des ingénieurs
Association of Consulting Engineers of Canada / Association des ingénieurs-conseils du Canada
Ontario Society of Professional Engineers
Professional Engineers Ontario

Twenty-Sixth Annual Conference of the Canadian Nuclear Society

The New Nuclear Generation

Le nouveau nucléaire et la génération de relève

2005 June 12-15

Toronto Marriott Eaton Centre, Toronto, Ontario, Canada

Call for Papers

The Canadian Nuclear Society's 26th Annual Conference will be held in Toronto, Ontario, Canada, 2005 June 12-15, 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 technical aspects, challenges and opportunities of nuclear technology as it prepares to undertake the task of supplying a much larger share of the world energy demand. As usual, papers are solicited on technical developments in all subjects relating to nuclear technology.

Conference Web Page: http://www.cns-snc.ca/conf2005.html

Deadlines

- Notice of Intent to present: 2005 January 21.
- Receipt of full papers: 2005 March 21.
- Notification of paper acceptance: 2005 April 21.

As a minimum, the Notice of Intent must include a paper title and the author's commitment to submit a full paper by the deadline of AprilMarch 21.

Guidelines for Papers

Papers should present facts that are new and significant, or represent a state-of-the-art review. They 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. Proper reference should be made to all closely related published information. The name(s), affiliation(s), and contact information of the author(s) should appear below the title of the paper. A short abstract of 50-100 words must be placed at the beginning of the paper, after the title and author's names. Abstracts will be collected in an Abstract Book for use by Conference attendees as a guide to presentations. The author should also provide separately a few key words for the paper.

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 (2005 May 1).

Paper Submission Procedure (Full Papers)

The required format of submission is electronic (Word 2000). Submissions should be made through the Conference web page.

Questions regarding papers and the technical program should be sent to: e-mail: cns2005@aecl.ca

General questions regarding the Conference may be addressed to

Denise Rouben, CNS Office Manager
e-mail: cns-snc@on.aibn.com
Tel: 416-977-7620

Slightly Semantic

by Jeremy Whitlock

This industry has had an image problem since they first coined the phrase "going critical" for a self-sustained fission chain reaction. Simply put, we're scientists and engineers, and not PR specialists. To this day there are still very few PR specialists in the industry, and, sadly, getting fewer all the time.

"Going critical" is what a reactor does, and frankly that's a much better phrase than "self-sustained chain reaction". Try to tell your friends that your reactor will soon be going critical, however, and watch how the conversation turns. Effuse about Qinshan going critical ahead of schedule and you'll soon learn who has been harbouring conspiracy theories about the nuclear industry all along.

Our capacity to frighten knows no bounds: We speak with confidence of the "Most Exposed Individual", a mysterious unfortunate living amongst the public. We laud a reactor's "containment", clearly a serious measure invoked when accidents equal disasters, last seen failing miserably in the likes of Jurassic Park and Ghostbusters.

We seek solutions for our nuclear "waste", apparently more dangerous than anything else on earth judging by our unprecedented approach - notwithstanding, of course, the fact that we currently store the material in "swimming pools".

The public requires "defense in depth", in lieu of protection from the need to defend. We design mighty coolant tubes that "leak", with the consolation that this happens before they "break". We have legislation that limits our "liability" to the public (rather than our indemnification) in case of an accident.

The list goes on, but the point is that we did not invent our terminology with a view to public perception.

Nowhere is this more acutely evident than in Port Hope, Ontario, where for over a year the uranium giant Cameco Corporation has been nurturing public understanding of "Slightly Enriched Uranium", or SEU. The backdrop is the company's quest for a license to down-blend foreign LEU ("Low Enriched", or LWR-grade, uranium) into SEU, soon to be a fuel of choice in the CANDU market.

Now, a sillier term than "slightly enriched uranium" is hard to imagine but of course it makes perfect sense from the technical point of view. Here is a fuel somewhere between natural and traditional "low" enrichment (lower than low, if you will).

To the suspicious eye, however, somewhat accustomed to the wiles of natural uranium (it may be evil, but it's a natural evil), any kind of "enriched" uranium is a step in the wrong direction. We note the irony in "depleted" uranium being a step in the wrong direction as well, but that's another story.

So, rather than admit its complicity in a global conspiracy to blow everything up, it is natural that Big Uranium would attempt to pass off its new product as "slightly" enriched. Tobacco, of course, is slightly addictive, and SUVs slightly polluting.

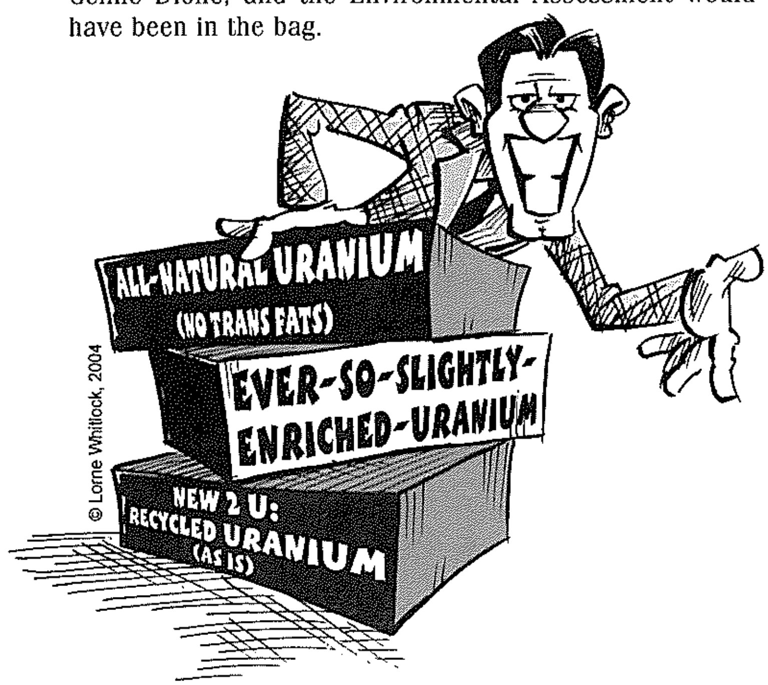
Factor in the risk of enriched uranium "going slightly critical", and you get a sense of the PR challenge facing Cameco.

After a slightly pregnant pause to reflect, one imagines slightly more market-friendly product labelling. It is possible, after all, to be technically complex and publicly tolerable at the same time. Witness the microwave: "micro" because small is good, and "wave" because only things that come in particles are dangerous. Put the two together and you've got a socially acceptable technology capable of beaming porn into every home on the planet.

Consider the "laser": a friendlier product with a more ambiguous name you'd be hard-pressed to find.

Perhaps, then, the concerned citizens of Port Hope may have found "down-graded LEU" easier to accept than "slightly enriched". Or what about "LEU-Lite"? Or, getting right to the point, how about "high-efficiency uranium"?

A personal favourite is "Uranium Jazz", with apologies to Air Canada. Throw in an endorsement contract with Céline Dione, and the Environmental Assessment would have been in the bag.



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2005

Mar. 9 - 10 — **CNA Annual Seminar**
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website: www.cna.ca

Mar. 13 - 17 — **Symposium on Radioisotope Production and Application (at 229th National Meeting of American Chemical Society)**
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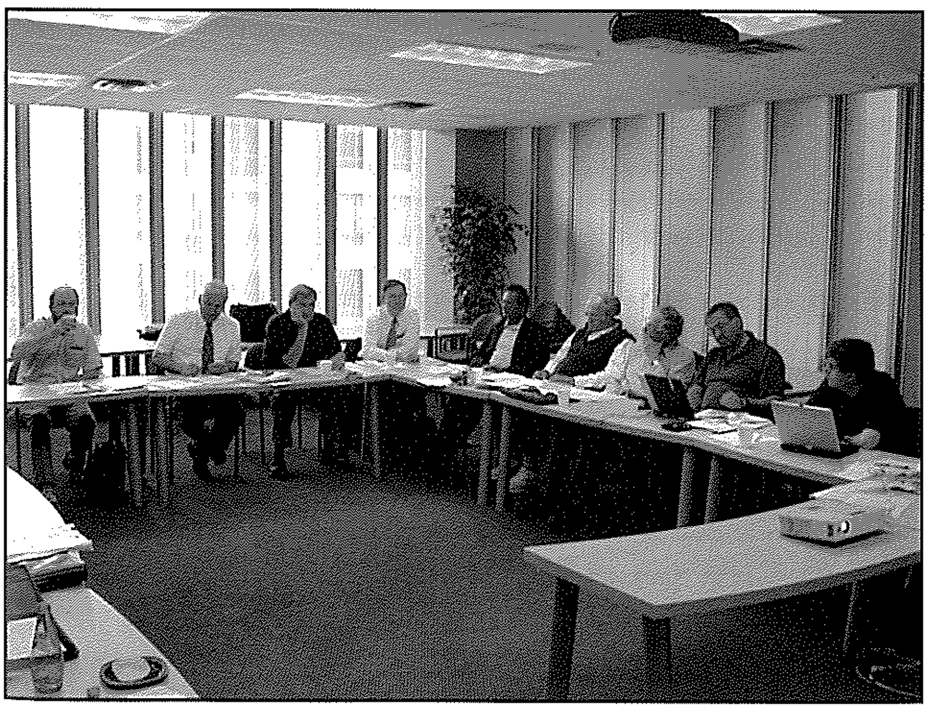

A view of part of the 2004 - 2005 CNS Council at work at its first meeting in July 2004 with President Bill Schneider, second from the left, in the chair.

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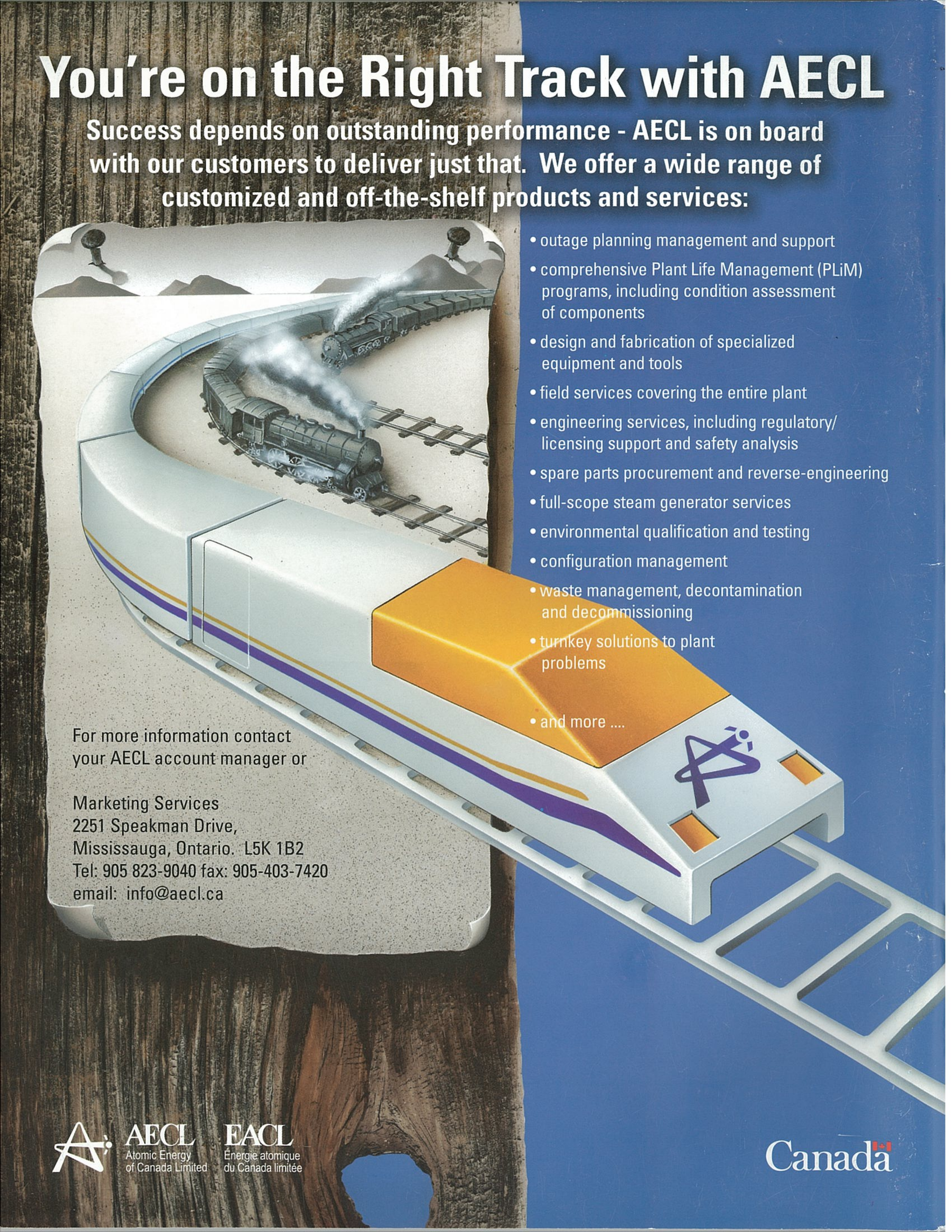
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