CANADIAN NUCLEAR SOCIETY

Bulletin

DE LA SOCIÉTÉ NUCLÉAIRE CANADIENNE

SEPTEMBER 2007 SEPTEMBRE VOL. 28, NO. 3

• "Degradation 2007" • Kroll Zirconium Medal Award
• Molten Fuel Moderator Interaction • Student Papers

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Allan Norsworthy, Director, Business Development
al.norsworthy@slnuclear.com

Is AECL For Sale?

The Great Canadian Nuclear Renewal is upon us and not surprisingly the "Big Three" are showcasing their products in CANDU territory. Competition is strong but do we need to have our own Canadian institutions roll out the red carpet for them? The CNSC is welcoming foreign vendors by advocating "international rules" for licensing while the McGuinty Liberals, posturing for an October election take the politically correct stance that "we want the best deal for Ontarians".

Encouraged by the "LWR Welcoming Committee", Areva, General Electric and Westinghouse are reportedly having "inside talks" with Natural Resources Minister Gary Lunn to purchase the federally owned CANDU designer. AECL is no stranger to divestment – it spun off its medical isotopes production business in 1991. But is the Flagship for sale? Gary Lunn says "no".

Apparently he did not say it loud enough. Continuing its bid, Areva Canada president Armand Laferrere claims that building EPR in Canada through an Areva/AECL partnership would enable AECL engineers to compete for LWR services in other countries. He adds that Areva would also help sell CANDU abroad.

This may be so, but a new partnership would disrupt the existing relationships within the Canadian nuclear industry. One partnership is Team CANDU, a consortium with AECL, General Electric, Babcock and Wilcox, Hitachi and SNC Lavalin. General Electric and Westinghouse (both are competitors of Areva) have invested heavily in Canada building the infrastructure to support CANDU technology. An upset in these current relationships could put a dent in CANDU economics because owners need a continuing infrastructure to refurbish and maintain existing reactors. It is also unlikely that government funding would continue to support R&D for a foreign-owned CANDU designer, again to the detriment of current CANDU owners, the future of ACR (Generation III+) and any pursuit of Generation IV development in Canada.

Would Canadians benefit from a new partnership? Maybe. However, prospects for Team CANDU are strengthening - a feasibility study to build an ACR in New Brunswick is under way, and Energy Alberta Corporation has applied for a site preparation licence near Peace River, Alberta to build a twin ACR station. Unlike Areva's EPR project in Finland that is now 18 months behind schedule, AECL built their last half-dozen CANDU reactors within budget and ahead of schedule. Areva needs this ability to sell 30 new reactors in China.

Continuing attempts by foreign vendors to buy or partner with AECL is proof that CANDU technology is strong. Despite the distractions caused by our LWR welcoming committee, the CANDU appears best suited for new build in Canada and abroad.

Ric Fluke

In This Issue

Let me express my gratitude to Fred Boyd for "holding my hand" during the production of the first Bulletin with me as its Editor-in-Chief. This edition represents the "changing of the guard" and I welcome you to read Fred's new regular feature called **"From the Publisher"**.

This edition focuses on degradation of materials, the understanding and management of which is fundamental to safe and reliable operations, to the economics of large-scale plant refurbishment projects, to an expectation of a 60-year plant life for new build and to Generation IV development.

Bill Schneider leads off with his "easy reading" account of the **13th International Conference on Environmental Degradation in Nuclear Power Systems ("Degradation 2007")** followed by the presentations made by the Plenary Speakers Paul Spekkens (OPG) and Peter Ford (GE, retired).

Next we recognize two Canadians, winners of the **Kroll Zirconium Medal Award**.

In keeping with the CNS goal of raising the profile of our younger members two student papers dealing with material degradation issues are included, reprinted from the **31st CNS/CNA Student Conference**. One of these is the 1st place winner of the student conference – congratulations to Emily Corcoran, Doctoral candidate at the Royal Military College. A third paper from the **28th CNS Annual Conference** entitled "Environmental Assessment: Challenges And Opportunities" is a useful account of Bruce Power's experience under the new Canadian Environmental Assessment Act, which applies to all major refurbishment projects in Canada as well as to new build, even on existing nuclear sites.

Our fourth paper presented at **ICAPP '07** earlier this year in France is an account of the final (hopefully) experiment needed for closure of a long-standing regulatory issue. Entitled "**Results From the Second High-Pressure Melt Ejection Test**", it supports the industry's contention that ejection of molten fuel from a burst pressure tube into the moderator does not result in a vapour explosion.

We also have **General News** and one **Obituary**. Under **CNS News** we have Fred Boyd's **Meet the President**, and an article by the President himself, Eric Williams, which hopefully will be a regular "corner" to keep members up to date on CNS Council activities and initiatives.

And last but not least, we have Jeremy Whitlock's exospheric view of our world, which may differ from other views, in **Endpoint**.

Your comments and letters are invited.

This issue of the *Bulletin of the Canadian Nuclear Society* (to use the official name) marks the first change in the operation of the publication in 15 years.

As announced in the last issue, Ric Fluke has taken over as Editor-in-chief while I have moved to the still uncertain role of "publisher". There will be further developments as we pursue the recommendations of the task group that was convened a year and a half ago to look at the future of our Society's publications.

Reflecting the resurgence of nuclear activities and the growth of the CNS, the *Bulletin* has attracted more readership and more advertising over the past few years. The growth of advertising raises questions. While it still does not cover the cost of publication and mailing there are concerns about how much advertising is appropriate.

Further, there is the on-going question of whether or not to continue the mixture that has evolved over the past several years of combining technical papers with general news of the Canadian nuclear program and news of the Society. That evolved partly because there is no other Canadian publication devoted to our nuclear program.

There have been suggestions that the CNS should publish a "journal" of refereed papers. That was tried over two decades ago but it had to be terminated less than two years after its launch because of the paucity of subscriptions. Others have suggested that a "general" magazine focussed on the Canadian nuclear program is needed. Still others have suggested discarding the printed *Bulletin* in favour of totally web based communication

You are invited to join the debate. Let us know your feelings and thoughts about the *Bulletin* and about the broad question of communication within the Society and with the whole Canadian nuclear community.

WNU-SI-2007

That acronym stands for the World Nuclear University, Summer Institute of 2007, where I spent six weeks during July and August as a "mentor".

The World Nuclear University was created in 2003 by the World Nuclear Association, in cooperation with the International Atomic Energy Agency, the OECD Nuclear Energy Agency and the World Association of Nuclear Operators. WNU is a "virtual" organization intended to improve communication between nuclear education organizations around the world. UNENE (the University Network of Excellence in Nuclear Education) is the Canadian associate.

In 2004 WNU decided to create a "summer institute" to provide an intensive program on broad nuclear issues for young professionals from around the world. After some struggles the first Summer Institute was conducted at Idaho Falls in the USA. Mark McIntyre, of Atlantic Nuclear and a very active member of the CNS, was one of the Canadian participants. *(See his account in Vol. 26, No. 3, September 2005, issue of the CNS Bulletin.)* The success of that SI led to another, held in Stockholm, Sweden, in the summer of 2006. *(Again there was an account by Dominic Rivard in Vol. 28, No. 1, March 2007 issue of the Bulletin.)*

The third WNU Summer Institute was held in Korea, sponsored by the Korea Atomic Energy Research Institute (KAERI), Korea Hydro and Nuclear Power (KHNP) and other organizations.

A typical day started at 8:00 a.m. with lectures until past noon. After lunch the young professionals (called "Fellows") worked in groups of nine or ten to discuss the lectures or to work on related projects. It was in these group sessions that the "mentors" played a role, to facilitate, encourage, provide advice and, generally, to assist the fellows as much as possible. Over the six weeks each fellow was in three groups and, similarly, each mentor was assigned to three different groups. Considerable emphasis was given to the concept of working with others. Given that the working language, English, was the third or fourth language for most fellows, their eagerness to particpate was impressive.

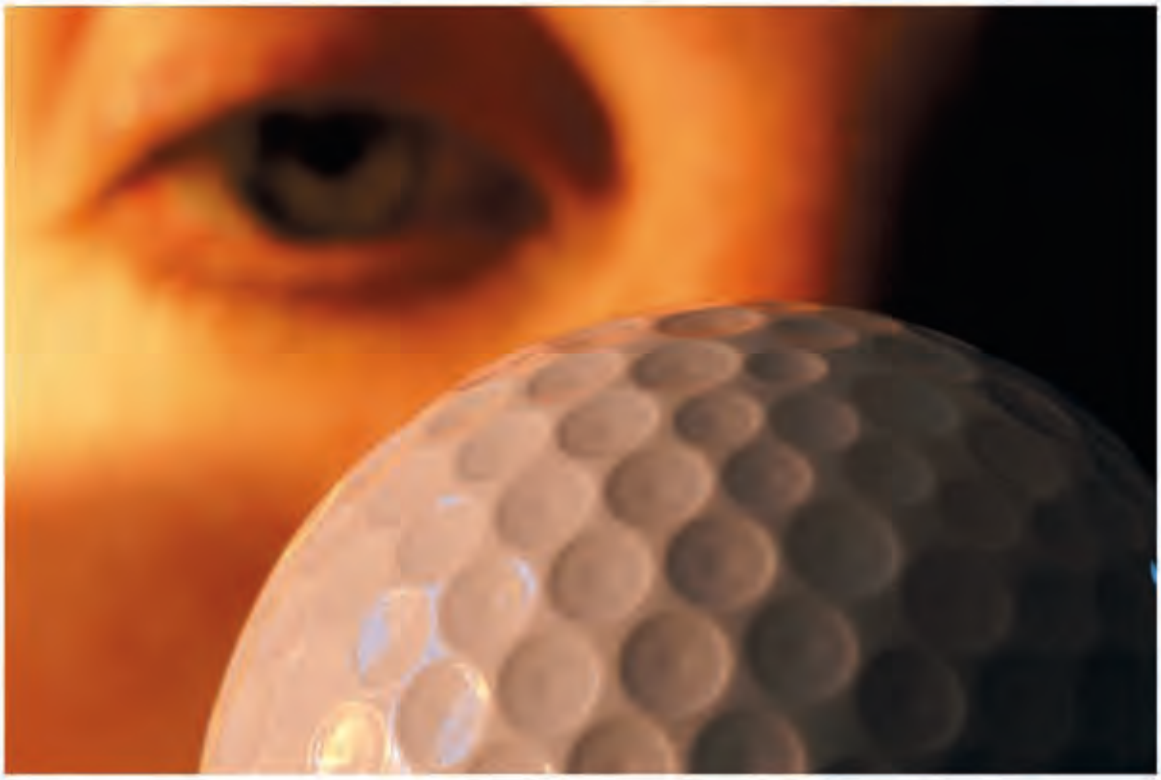
The curriculum was intended to expose the fellows to the many dimensions of the world nuclear scene, including: economics, international law, safeguards, proliferation, communication, knowledge transfer, resources, enrichment, reprocessing and more.

One week was devoted to tours of nuclear and related facilities, including the Wolsong and Kori NPPs but also Doosan Heavy Industries, one of just two plants in the world capable of manufacturing the pressure vessel of a 1400 MW PWR (one was on the floor when we visited) and the Posco Steel plant, one of the largest in the world. All of which emphasized how much Korea has surpassed Canada over the past couple of decades.

Despite the unevenness of the lectures and lecturers most of the fellows expressed strong appreciation for the experience.

Next year the WNU Summer Insitutue will be held in Canada. If you are, or know, a young professional, I strongly recommend looking into the program.*(Go to the WNU website for details.)* It is a once in a lifetime expereince.

Fred Boyd



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~ Cover Photo ~

The giant Mammoet, the Dutch word for Mammoth, towers over the Bruce A Station ready to lift the 250 tonne steam drums with the greatest of ease. The last time a crane of this magnitude visited a nuclear generating station was during construction of Darlington NGS in the 1980s.

– Photograph courtesy of Bruce Power.

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ISSN 0714-7074

The *Bulletin of the Canadian Nuclear Society* is published four times a year by:

The Canadian Nuclear Society
480 University Avenue, Suite 200
Toronto, Ontario, Canada, M5G 1V2
Telephone (416) 977-7620
Fax (416) 977-8131
e-mail: cns-snc@on.aibn.com

Le Bulletin SNC *est l'organe d'information de la Société Nucléaire Canadienne.*

CNS provides Canadians interested in nuclear energy with a forum for technical discussion. For membership information, contact the CNS office, a member of the Council, or local branch executive. Membership fee for new members is $75 annually, $44.00 for retirees, free to qualified students.

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Les frais annuels d'adhésion pour nouveaux membres sont 75$, 44$ pour les retraites, et sans frais pour les étudiants.

Editor / Rédacteur
Ric Fluke Tel./Fax (905) 823-9060 ext. 6511
e-mail: fluker@aecl.ca

Publisher
Fred Boyd Tel./Fax (613) 592-2256
e-mail: fboyd@sympatico.ca

Printed by The Vincent Press Ltd., Peterborough, ON

Canada Post Publication Agreement #1722751

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Degradation 2007 – the conference

Bill Schneider, CNS Program Chair

Bill Schneider,
Event Organizer Chair

Peter King,
General Conference Chair

Todd Allen,
Technical Program Chair

The 13th International Conference on Environmental Degradation of Materials in Nuclear Power Systems of August 19 – 23, 2007 has just wrapped up at Whistler BC. It was a huge success – over the top in terms of registrations, papers, corporate sponsorships and the general level of interest and focus. More of a scientific meeting than a convention, this conference series is the premier nuclear industry corrosion meeting where the world experts of the field from utilities, engineering and service organizations, manufacturers, research establishments and universities gather biennially to listen to papers on new work and to explore new insights into corrosion mechanisms in the many water cooled systems in nuclear power plants.

As this article is being written for a broader CANDU audience and in view of the traditional focus of the CNS which is heavily toward the core and its design and R&D, I must be-labour the importance of materials and degradation. They have long been the focus in some areas of the broader CNS and CANDU constituencies, but seldom as core CNS values.

Some highlights of the event are included later in this article, followed by the profiles and papers of our two plenary speakers. But before we get to that, let us reflect on the significance of this field of study to reactors everywhere and to CANDU in particular.

What can defeat the nuclear industry going forward (?)

Now that every public commentator and opinion hack "knows" that the answer to the "…reduction of the foot-print of man upon the earth." is nuclear power (for which we have many broken arms from self-congratulation), there is only one thing that can defeat a sound nuclear future.

That thing is … **"some sort of self-inflicted screw-up".**

It could be a big nuclear accident, some natural calamity which leads to radiation release or a compromised safety system, or possibly terrorist activity.

While calamity is a possibility not to be dismissed, screw-ups – things which should be manageable but are fumbled – are always a possibility. In fact, given past experience, they are likely. They could be such things as huge construction or re-furb cost over-runs (as has happened many times in the past) – or a massive, pre-mature degradation of critical materials (of which there are numerous examples throughout the nuclear industry). If we are honest with our-selves, we have to admit that over the years of research, design, construction, operation, maintenance and re-furb of CANDU we have mostly done a less than adequate job on that front – there are notable exceptions, but unfortunately "exception" is not really an exaggeration here.

The schedule and cost-containment break-down which results from pre-mature degradation (and the inspection, maintenance, repair and fitness-for-service work that goes with it), are totally intolerable to any modern private or public utility, inside or outside of Canada. In today's highly competitive utility market, tolerance for the 60% or so availability levels we have often had is just not there.

I fully expect a blast from those who consider such talk blasphemous – if that is a temptation, I recommend a bit of soul-searching – we have been endlessly tolerant of poor reliability and of the technical risk that leads to that – the future will not be so kind. We are in an era of new opportunity which is at the same time highly competitive – competition sharpens the mind. In the new competitive era there will be no more second chances – get it right or down the flusher we go.

Degradation – eh !?!?

As we design or work to confirm fitness-for-service of any plant old or new, all we expect is that it will operate trouble-free for another 30 to 60 years. Given that mother nature does not like to have metals hanging around in their pristine metallic state – and that the oldest surviving plants have barely reached the short end of that range, that is quite a challenge.

Going forward, we will have plant with surviving original equipment, plant with newly replaced equipment and new plants with recently produced equipment. Having done a huge amount of degradation research on those materials we may well ask whether or not all has been resolved (?). Will the new materials and equipment have problems too (?). If it does, will its degradation be the same as that of the past (?). – highly unlikely, I would say (?).

Degradation and CANDU

In CANDU we seem to have had a very curious attitude towards plant reliability and the degradation upon which reliability depends. We forget that in a power plant, unlike in nature, things always and only degrade – they never grow better – so the more components and systems, the more problems with degradation – and the more monitoring and inspection, repair, replacement, outage time, dose up-take, cost, etc.

We also have a lot of complexity – we seem to have start-up and shut-down systems stacked one on top of the other – each with its complex system of heat transfer equipment, pumps, controls, valves, instrumentation, etc. etc. Also, anyone who has ever gone into a CANDU plant could not but be aghast at the number of valves – big valves, small valves, high and low pressure valves, instrument valves, etc, etc. – thousands – literally.

In addition to the vast number and complexity of systems and components we have core components for which the available state-of-art materials still require matter-of-course mid-life replacement (e.g. pressure tubes). Somehow the problems of those components are viewed as wondrous phenomena to be pursued in the fullness of time by our best scientific minds.

We have other components which, being outside of the "core focus" area are viewed somewhat as a nasty trick perpetrated by the unwashed on an unsuspecting nuclear community.

We have degradation in components common to non-CANDU systems – such as primary pumps, steam generators and fuel.

And we have degradation in equipment which performs a necessary (to CANDU) function, but where other non-CANDU systems don't even have such equipment. That includes things like moderator heat exchangers, feeder tubes, zirconium-based pressure tubes, etc.

And we have degradation in equipment which we don't need to have and would not have except for blunders of concept design when the plants were first being conceived (which in those days was after the construction had started). The most incredible example of things for which we have no need is the two-zone reactor concept (which some CANDU reactors have) and the four SG-like pre-heater heat exchangers (and the extra headers, piping, valves, controls, instrumentation, engineering qualification, etc) that come with that – all of those components are totally un-necessary to the plant design. Despite their uselessness, the burden to the utility of their enormously expensive and dose intensive maintenance programs goes on and on and on.

The bottom line for CANDU is – we have many types of degradation in many areas of the plant including the critically important primary and secondary heat transport components for which maintenance is very difficult, costly and dose intensive.

Also, in CANDU plants generally, not only have we had many different types of degradation, we have had a large quantity of it.

Degradation and Future Nuclear – Gen IV Etc.

Advanced nuclear cycles are of utmost importance to the nuclear future. Nuclear fuel supplies are not un-limited and other sources of GHG-free energy are unlikely to materialize to anything like the wished-for capacity. We therefore need those advanced designs.

A characteristic of most of those advanced cycles is incredibly difficult material environmental conditions – temperatures which go from very high to exceedingly high (recall that 10C is a big temperature increase where corrosion is concerned), supercritical pressures, etc, etc. While Gen IV systems are a ways off, it will take all of that time and more than a few technological break-throughs to get the material performance needed for reliable operation.

Such future materials challenges were the subject of the second plenary paper at this conference.

Degradation Avoidance

The first and most powerful tool in degradation avoidance is simplification. Whenever a system can be deleted, its entire construction and life cycle management cost are avoided – think about it – no system – no degradation and no life cycle management costs.

Where systems cannot be avoided altogether, they can always be made simpler and with fewer, more elegantly simple components. Again, where systems and components are simplified or avoided altogether, their degradation management costs are reduced accordingly.

Having achieved the most elegantly simple configuration possible, degradation management comes down to dealing with the three parameters of; i) material susceptibility, ii) stress (both residual and operating stress) and iii) the respective operating environment.

Material susceptibility must be dealt with in the years of material selection and corrosion research work which precedes the material's selection, manufacture and installation. Once the material specification and the related manufacturing process pre-qualification work has been completed, there is no further possibility of adjustment of the material properties relating to corrosion susceptibility or to anything else. That means that the necessary R&D on candidate materials needs to be under way years in advanced of the spec. writing. Note also that there are numerous detailed requirements within the generic specs, which must be resolved to optimize degradation resistance as well as the structural strength and manufacturability. In other words, spec. writing, which pulls together all of that research and operation and manufacturing experience, is a very demanding task and you only get one shot at it – after that you've got what you've got.

Applied and residual stresses in operation are a major factor in many kinds of degradation. Stress minimization

Plenary Speakers, left to right: Todd Allen, Paul Spekkens, Peter Ford, Peter King.

requires great attention to detail at the design stage. While such stresses materialize during manufacture and operation, their minimization can only be achieved during the design stage. That is easy to say – but not easy to do.

Operating environment is the only parameter which is available for degradation management optimization after equipment is designed and built. Despite the importance of the outcome, the ability to avoid degradation solely by environment adjustment is limited. Discussion of the corrosion degradation of materials in their operating environments is the subject of the "Environmental Degradation Conference".

Degradation 2007 Conference Focus

The purpose and focus of this conference is best described in the following foreword taken from the conference's Advanced and Final Programs. It reads as follows:

13th International Conference on Environmental Degradation of Materials in Nuclear Power Systems

Technical Scope and Thrust – The Way Forward

The twelve Environmental Degradation Conferences to date have intensively pursued corrosion of the nuclear power plant materials for the current generation of plants – an appropriate focus given the widespread nature and the huge economic impact of such degradation. At this point, much of that older material has been or will be replaced or is being effectively managed and new and advanced materials and systems are on the way. It is therefore important that these conferences develop a forward looking focus. The alternative may be a diminution of the relevance of this conference – as has already befallen other events with such a singular focus.

Looking forward we see a very large fleet of existing plant and a similarly large fleet of new-build of advanced versions of current reactors. Beyond that there will be the next generation of reactors (GENIV) with its even more advanced materials and degradation challenges. We expect a mere 30 to 60 years of trouble-free, high availability service from such equipment. Will

Plenary Speakers' gifts "The Charging Bears".

it have degradation of some sort (?) – absolutely; will it be of the same nature as last time (?) – highly unlikely.

A further worthy question at this transitional stage is – "Whom are we here to serve (?)"; i.e., whose needs drive the research required to address such newly evolving degradation (?). Clearly the ultimate customers are; i) those who own all of that equipment and all of its problems and ii) the regulatory bodies who become the owners of those thick fitness-for-service assessment reports which they must ultimately approve. By the way, both of these groups become the owners of all of those requests for research funding as well.

From the above we see that; i) we need to stick our heads up and look to the murky future for newly emerging degradation issues which will surely evolve as time and system design progresses and ii) in order to maintain the high value of the work, we need to re-focus on the upcoming priorities and future needs of the technology's ultimate customers. In that regard, the fact that this conference focuses on several newly-breaking degradation issues is a good thing.

Degradation 2007 Highlights

The Plenary Session – which is described below was one of only three times the entire conference was together – the others being the Sunday reception and the Wednesday evening banquet. Its ultimate purpose was to encourage everyone to stick their heads up and look to what is coming down the road, both in terms of newly evolving mechanisms in existing systems and for the major materials challenges of the future advanced systems. For such advanced systems, by the way, a bushel of materials development work will be required for every pinch of physics work. The intro and papers for the Plenary Session are given below.

Special Session – Alloy 800 Steam Generator Tubing – a session of four invited papers, was directed specifically at the long range interests of the CANDU community which now finds itself somewhat of an orphan as the sole user of Alloy 800 tubing in new steam generation equipment. Alloy 800 was selected by Siemens in the late 1960s and they built many SGs with it in the ensuing years – up to the point where its new-build program was brought to a halt. At this point Siemens Nuclear is part of Areva and it is not clear whether or not any of their future SGs will use this material.

Alloy 800 has a remarkable record of reliability over its 3 1/2

decades of operation in Siemens plants and 2 decades in CANDU. With the exceptions noted below, it has not been susceptible to cracking in service. It has had a small amount of pitting and wastage in a number of plants. As discussed in the first paper of the session, cracking has to date been observed in only a very small number of tubes at one plant in Germany. This cracking is located in peripheral tubes at their mid-span within the tube-sheet. This is an area not associated with degradation in plants anywhere else.

Ken Sedman, Bruce Power.

The CANDU program now finds itself in the midst of replacement SG manufacture and on the brink of new-build. Now is the time to take a hard and critical look at the long term reliability prospects for this critically important material. Such long range discussions are expected to flow from this session at some time during the next few months.

The four papers whose titles reflect the scope of the session were; i) Operating Experience with Alloy 800 SG Tubing in Europe, Renate Kilian, AREVA NP GmbH, ii) A CANDU Utility Perspective on Using World Experience to Manage Alloy 800 SG Tube Degradation, John Slade, NB Power, iii) A Manufacturer's Perspective on SG Tubing Selection, SG Design and Fabrication, Peter King, B&W Canada, and iv) Alloy 800 SG Tubing: Current Status and Future Challenges, Robert Tapping, AECL.

The Balance of the Technical Program – consisted of about 130 papers by researchers from around the world on the range of environmental degradation subjects of these conferences. All papers will be available in the Proceedings CD of the conference which will be out in about October.

The Event – was fantastic if I do say so myself. The venue was the Whistler Westin Hotel in the bustling and impressive Whistler resort area. The banquet was held at Roundhouse Lodge at the top of Whistler Mountain, accessed by a 30 minute gondola ride on a beautiful sunny August evening. By all reports, the arrangements were excellent in all respects. Because of the excellent and highly organized team working on this event, every detail was planned and executed with great precision and timeliness.

The Numbers – The conference met or surpassed all of its budget targets including; registrations (225 including 6 students (200 budgeted) plus 30 guests), hotel room up-take (vs numbers guaranteed), technical papers (144), sponsorship revenue and net revenue to CNS, the conference sponsor.

Conference attendees came from around the world – 66% were non-US (up from 54% in 2005). Of those, 63% were from industry (utilities, engineering and service companies etc.), 21% were from universities, 6% were from regulators and 10% were from research labs. There were 30 Canadians (14%), some of whom were no doubt attracted by the Alloy 800 session. Of those from the industry category, 6 (3%) were from utilities (an increase from prior years). As utilities are the owners of all of that equipment and the identifiers of the technology needs of the future, their involvement is very important and must be encouraged.

The Team – which organized the technical program and the event itself was truly remarkable. The achievements of the combined team was all the more remarkable considering that we met for the first time on Sunday August 19th – all org was done entirely by tele-conference meetings and e-mail.

The work was split between the technical program side and the event organization. Those teams were as follows:

The "Conference" Chairs – were Peter King, B&W Canada, General Conference Chair; Todd Allen, U Wisconsin-Madison, Technical Program Chair and Jeremy Busby, Oak Ridge National Labs, Assistant TP Chair. The technical program is the heart of the conference – nobody decides to come because of the food – people only come for the substantial technical content. And that was provided exceedingly well by Todd and Jeremy - excellent work

The TP committee under Todd and Jeremy did the enormous job of issuing the CFP, receiving and reviewing first the abstracts and then the papers, organizing the sessions, etc. etc. - a huge task for a highly scientific conference like this one. All abstract and paper management, session setup, etc. was orchestrated by Todd and Jeremy using the START Conference Management System provided by CNS. While there were a few limitations, this certainly is the way to run a conference technical program - the authors submit their material to START after which all review and program assembly is done within the system. Among other things, it made un-necessary the traditional November abstract review and session planning meeting of the TP Committee - all of that was done on line.

Because of the relentless demands of his day job, Peter was unable to participate extensively in the pre-event organization of the conference (for these conferences the Executive Chair task is normally done by the General Chair). Nevertheless, at the conference itself, while the rest of us dealt with the details, Peter brought his years of involvement with this event, the fact that he knows just about everyone and his energy, vision and insight into the field to the important role of motivating and energizing the technical discussions – an important role well done.

The Event Org Team – consisted of Bill Schneider, B&W Canada, Event Org Chair serving as conference executive chair on behalf of conference sponsor CNS, Derek Lister, U New Brunswick, Student Team Supervisor (Derek's team did the loading of the presentations and the Q&A documentation), Peter Angell, AECL, Publicity and Promotions Chair, Steve Fluit, B&W Canada, START system technical back-up and Jim Harvie, Conference Treasurer. The volunteer Student Team consisted of six students from various universities including Derek's student Mahsa Khatibi who did much of the task of loading the many

presentations. All on the team did remarkable work, considering that almost all have demanding day jobs.

The presentations loading and questions and answers documentation (for the Proceedings CD) tasks are critically important for a conference like this - the presentations can't be given if they are not set up and the CDs can not be issued until all Q&As are documented therein (at past runnings of this and similar conferences, the lateness of Q&A documentation has held up the issuance of the CDs for 1/3 of a year or more). This is a lot of work and required extraordinary effort on the part of Derek and his team (many presentations were only submitted at the last minute and there were an enormous number (~300) of long and technically complex Q&As).

As for the students doing the loading of presentations and the handling of Q&As; while it was a lot of work, this was a once in a life-time opportunity to rub shoulders with the legends of the nuclear corrosion industry (and to rough them up a bit if they were not getting their Q&As done on time) - an experience they will remember for the rest of their careers.

The Event Administration Team – was led by Elizabeth Muckle-Jeffs of The Professional Edge who was contracted to do the Event Administration work for this event. Elizabeth was assisted by her associate Lisa Carmody. Kathy Davies of B&W Canada assisted with the on-site registration, and with the Q&A typing as well as with many other things before and during this event.

Event administration is an enormous job. It was performed with great energy and efficiency and with such meticulous care that every little thing was not only organized but specifically authorized before its execution. An indication of Elizabeth's performance was the standing ovation she received at the banquet – not something one usually expects.

Plenary Program

As indicated above, the plenary program was set up specifically to provide some "getting on with the future" inspiration for the discussions of the rest of the conference and to the ongoing vision of its participants.

The first speaker, as an executive with a major utility represents the "owner" of all of that equipment and of all of its problems. He is also responsible for the choices that need to be made on the technology investments needed to resolve those problems.

The second speaker, as a long-time leader in the field of responding to such degradation problems, brings insight into the process of addressing what may evolve in future; and in particular to the change of management approach that will be needed if we hope to realize the technical, cost and schedule objectives of advanced reactor types.

The first Plenary Session paper – (of two) was presented by Paul Spekkens as follows:

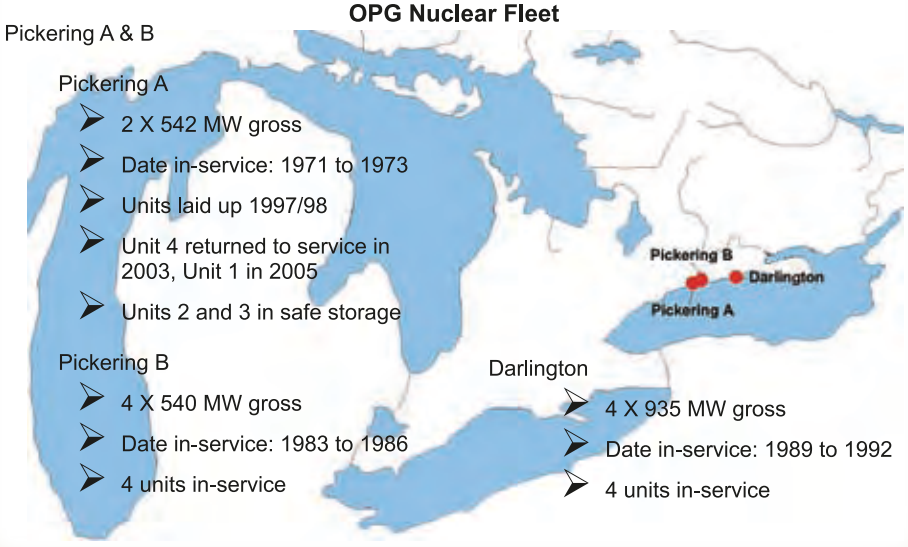
Paul Spekkens, Vice President, Science & Technology Development, Ontario Power Generation

Dr. Paul Spekkens is responsible for managing OPG's investment in nuclear Research & Development and technology development. He is also responsible for the Feeder Integrity Program, whose goal is to minimize the impact of feeder degradation on the safe and reliable operation of OPG's ten CANDU nuclear units. Dr. Spekkens worked for many years in the Research Division of the former Ontario Hydro and specialized in issues of chemistry and materials degradation in primary and secondary systems, and in steam generators.

Material Degradation – A Nuclear Utility's View

Good morning, ladies and gentlemen. It is a pleasure to be invited to speak to you at this conference. Conferences like this are very important to the nuclear power industry. We have one of the best public and employee safety records of any industry, and understanding materials degradation helps the industry protect this excellent achievement. Like all of you, I'm looking forward to the next few days of interesting and detailed technical presentations. The organizing committee has, I think, done a remarkable job of pulling together a very strong program of papers that will deal with many aspects of corrosion and material degradation. But before the conference sessions get into the details of the material degradation work that is being carried

Figure 1
OPG Nuclear Fleet



out, I'd like to spend a bit of time in this opening presentation addressing the "why" surrounding some of this work, at least from the point of view of one of the end-users of the information, that of a nuclear utility. My background includes fifteen years in corrosion and materials research. But about fifteen years ago, I joined the management team of the nuclear operations side of my company. I am now the senior manager responsible for technology development, and it is from this point of view that I'm going to be speaking here today. I will use my company as an illustrative example, although I believe that we are not very different from many other nuclear utilities in this area.

I'll start by describing the significance that material degradation has for us, and outline how we use information about material degradation, and more specifically the different time-frames that we need to consider. And coming out of that I'll summarize what I think the nuclear utilities need out of the R&D program.

I work for a company called Ontario Power Generation, which is the largest producer of electricity in the Province of Ontario. OPG has a generating fleet consisting of nuclear, hydraulic and fossil plants with an installed capacity of approx. 22,000 MWe. We operate 10 CANDU PHWR nuclear units at 3 sites as shown in Fig 1, two at Pickering A, four at Pickering B and four at Darlington with a total installed capacity of about 6600 MWe.

One of the significant features of our nuclear fleet is that the age of the units spans roughly two decades, from the Pickering A units which came into service in the early 1970's (and were refurbished recently) to the Darlington units which came into service in the early 1990's. The CANDU reactor design evolved considerably during this period. The result is that we have a variety of materials and designs in our reactor systems. This variety ranges from

- relatively exotic materials, such as zirconium alloys, to
- less exotic materials being used in uncommon applications, for instance, Monel 400 steam generator tubes at our Pickering A and B units through to
- common materials such as carbon steel and 304 stainless being used in a variety of environments. These environments range from relatively benign, for example the low temperature, high purity water of the moderator system, through to relatively aggressive environments, such as the acidic conditions that can form in the moist air that sometimes exists in some of our reactor enclosures.

So we have a diverse set of materials, in a broad range of environments. The challenge for us as a nuclear utility is to understand enough about material integrity for all of these materials in all of these environments to be able to achieve our business objective. And an illustration of why this is important is shown in Figure 1. At Pickering A, two of the four original units, Units 2 and 3, were placed into safe storage rather than being refurbished like Units 1 and 4. This decision was made in 2005 largely on the basis of the more severe material degradation in some of the major components in these two units, which made the business case for refurbishment unattractive. Material degradation was at the root of that major business decision.

So what is our business objective in operating our nuclear plants? Quite simply, it is to operate our nuclear fleet safely, reliably and cost competitively, now and going forward into the future. What makes the topic of this conference so relevant to us in the nuclear utilities is that if material degradation is poorly managed, it can have a deleterious effect on all three of these business drivers for our company.

- Clearly, safe operation is a non-negotiable attribute when operating a nuclear fleet. The nuclear industry has an excellent safety record and we need to maintain it that way. Our public franchise depends on it. Material degradation can impact on the safety of our nuclear facilities in two ways, either directly or indirectly. Direct effects are by jeopardizing the structural integrity of safety significant components and systems, for example cracking or thinning of the heat transport system pressure boundary such as SG tubes, the piping, the reactor pressure vessel in a LWR or the pressure tubes in a CANDU, etc. Indirect effects on safety can happen when degradation affects safety-critical process parameters, for example, corrosion products fouling critical heat transfer surfaces or dimensional changes of critical components through creep. You simply cannot run a nuclear plant unless the safety-critical systems are in good condition.
- Reliability is important when selling electricity into a competitive marketplace that values predictability. OPG's units have experienced a variety of types of material degradation that have produced major unexpected outage time, both through forced outages and extensions to planned outages. Steam generator tube leaks at our Pickering B plant, carbon steel thinning in the heat transport system at Pickering A, 304 stainless instrument line cracking at Darlington are all examples of unexpected degradation of relatively common materials that have had an impact on the reliability of our plants by causing unplanned outages.
- And finally, cost competitiveness. Apart from the loss of revenue during outages, there are other major costs associated with material degradation, such as the extensive inspections that are required to monitor degraded equipment, and the costs of repairing or replacing components that are degraded beyond their serviceability limit. This is ultimately what drove the decision on Pickering 2 and 3. Technically, we could have refurbished the units, in other words we knew how to repair and refurbish the degraded components. But the extent of the material degradation was such that we could not convince ourselves that the refurbished units could run cost competitively enough to support the multi-billion dollar investment that would have been required.

Material degradation has been the leading cause of unplanned incapability and major expenditures over the life of the OPG plants. Re-tubing of the Pickering A reactors, chemically cleaning the steam generators at Pickering, the turbine issues we've had at several of our units, the vast amount of inspection we do in all our units, all these things are the result of material degradation. They have cost us hundreds and hundreds of millions of dollars. As the electric sector becomes increasingly competitive, it is clear that our future business success will depend to a large extent on our ability to manage the material integrity issues that we know we are going to face. That's where you in the technical community come in. The knowledge and understanding generated by the work being discussed at this conference provide us the capability to manage degradation phenomena in our nuclear units. That is why nuclear utilities invest money and resources to support R&D activities.

Let me turn to how we use information on material degradation. Nuclear utilities need to be able to manage materials issues over three distinct time-frames: short, medium to long, and very long.

1) Short-term, measured in months to a few years, to address Fitness for Service questions, i.e. "is the system or component able to operate safely through its next duty cycle?"
2) Medium- to long-term, say 5 to 25 years, to address Life Cycle Planning questions such as "what combination of actions will provide the best, most cost effective means of optimizing the life of the plant?"
3) Long- to very long-term, say 20 to 60 years, to support decisions related to plant refurbishment or the building of new plants.

I want to look at each of these in turn, starting with short term fitness for service. As operators of a nuclear power plant facility, we have an obligation to assure ourselves that our units are fit for service, in other words, that all the systems, structures and components are able to perform their credited safety functions if called upon during the upcoming operating cycle. This confirmation of fitness for service is made every time the reactor is restarted from an outage and it extends at least until the next shutdown opportunity to inspect or monitor the condition of the components. Given that outages are every 1 to 3 years apart depending on the units and reactor type, fitness for service declarations look at material condition on that sort of time scale. Clearly the fitness for service declaration requires two key pieces of information:

1. what is the condition of the system or component currently, and
2. how is the material condition changing over time?

The first question is answered by inspection, for example, using non-destructive evaluation techniques or other methods of monitoring the condition of the equipment. The answer to the second question can come from a few different sources. Having a fundamental understanding of the degradation mechanism of the particular material in the specific environment is the most desirable means of doing this. In addition, we can use empirical observations of the behavior of the system during previous periods of operation. For a relatively short extrapolation, deducing the rate of progress of degradation on the basis of data gathered over the previous two or three operating cycles is often sufficiently reliable to allow fitness for service to be established. This works well for mechanisms such as thinning, where the progression is relatively slow and predictable, at least in the short run. Clearly, however, there are some degradation mechanisms such as cracking which are sufficiently random that even a short-term extrapolation is not a reliable indicator of future plant behaviour. If neither extrapolation from previous performance nor fundamental understanding is available, then fitness for service can only be established by demonstrating that the consequences of the degradation are acceptable. However, in general, and I know there are exceptions, but in general, short term fitness for service declarations can be managed without precise predictions of the rate of degradation except where a component is close to the end of its acceptable service life.

Now let's look at the medium- to long-term activity of Life Cycle Planning. In this case, utilities address the same question as in a fitness for service determination, but for the remainder of the expected plant life, as shown schematically in Figure 2. We are monitoring some sort of degradation, shown by the curve, since it was first observed at the Limit of Detection. We know from analysis the Limit of Failure, in other words the point at which the degradation is no longer acceptable. In Life Cycle Planning, we need to determine whether the curve is going to stay below the Limit of Failure long enough for the plant to reach its Expected Life. In the example in Figure 2, there is not much margin, so we would be asking ourselves what actions we can take, such as system modifications, chemistry changes, cleaning operations, etc, to slow the rate of degradation and assure ourselves that the plant will reach its intended end of life. This is considerably more difficult than a short term Fitness for Service determination, simply because the time scale over which the prediction is being made is much longer than one or two cycles of operation, but rather over several decades. If the Life Cycle Planning activities are being carried out early in plant life or if the degradation has only recently reached its Limit of Detection, the period over which the prediction is being made will be much longer than the period of plant experience that the prediction is based upon. In this case, the precision with which the degradation rate needs to be known is even more onerous. It doesn't take a very large change in propagation rate to cause the curve to intersect the Limit of Failure before the plant reaches its Expected Life. Unlike the short term Fitness for Service determination, a projection to the end of life of the unit cannot generally be made with confidence based solely by extrapo-

Figure 2
Schematic of Degradation vs Time

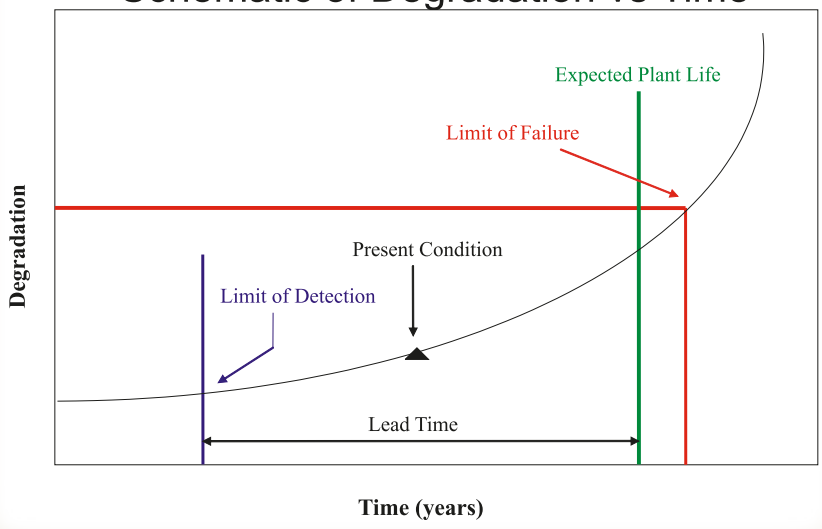
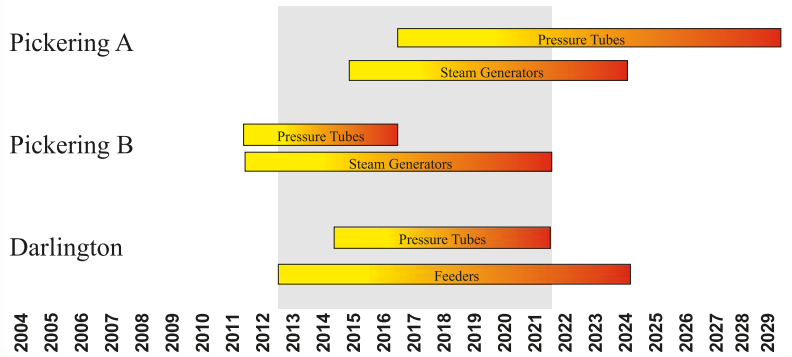


Figure 3
OPG Nuclear Major Component End of Life Outlook in 2005[1]



lating observed plant behavior. Again, this is where you come in. R&D which establishes the types of degradation which will occur in plant systems and their rates is of fundamental value to effective Life Cycle Planning. It is generally true that you cannot carry out effective Life Cycle Planning on something which you don't understand. That is why R&D is important to us.

Even when you generally understand the degradation that's going on, utilities can still be left with a high degree of uncertainty in decisions they need to make. Figure 3 illustrates the point. In a CANDU plant, end of life of the unit is generally defined by the service limit of the major components, namely fuel channels, feeders and steam generators. When one of these components reaches the point at which it cannot be operated without undertaking a large amount of remedial work, then we consider that the unit has reached the end of its life. A decision is then required to be made on whether to refurbish or whether to move into a storage or decommissioning mode. Figure 3 shows the situation at the OPG plants in 2005, as presented by our Chief Nuclear Officer to the Canadian Nuclear Society annual conference[1]. It shows the projected service limits of the major components in each of our plants based on our understanding of the degradation in them and of the planned mitigation measures that we were taking. Because of the uncertainty in this understanding, the projection shows "optimistic", "pessimistic" and "most likely" dates at which the components will reach their end of life, as shown by the bars in the figure. These life projections range over very long periods of time. For example, for the steam generators at Pickering B, there's an enormous difference between reaching "end of life" in 2011 or so, versus lasting to 2021, which is well beyond end of life of another major component, the fuel channels. In the former case, additional life extension measures would be needed for the steam generators, while in the latter case no additional work would be warranted. It is very difficult for nuclear utilities to make sound investment decisions in the face of these uncertainties. In round numbers, a year of operation of a nuclear unit is worth $250-500M. Therefore, the difference between a unit that reaches its end of life in 2018 versus 2022 is one to two billion dollars. There would be a great benefit to the utilities if there were more certainty in these "end of life" predictions. And there is always the nagging doubt about whether we have missed something because we've been focused too closely on the known problems of the day. In 2005, we did not show feeders as a potential life-limiting component for Pickering A. These are plain carbon steel pipes in a well characterized environment, but we did not understand the system well enough at the time to realize that this component is indeed at risk for Pickering A. We have had to do a lot of catch up work and change our life predictions because of what we now know. So to be really helpful to us, the R&D program needs to focus on more than just the problems of the day, but maintain sufficient breadth to surface the less obvious issues in other components and other materials, and to produce more definitive life predictions.

The third time scale that many nuclear utilities are working on is 20 to 60 years, associated with projects to refurbish existing plants to provide an additional extended operating life and with "new build" projects. In each case, the same schematic of degradation vs time in Figure 2 is still the critical question. But now the expected plant life has moved outwards to the end of the next operating interval, 50 or 60 years from the beginning of plant life. In a refurbishment project, the answer to the question of whether the degradation rate of the existing materials is sufficiently slow to permit these materials to remain serviceable for the extended life of the plant is critical. An overly pessimistic answer to this question will cause unnecessary work to be carried out during the refurbishment outage. An overly optimistic answer is even worse, in that it will cause unexpected failures in the refurbished plant prior to its expected extended plant life. A nuclear utility must base these difficult judgments in large measure on the confidence in the understanding that exists in the technical community at the time. A relevant example for us in OPG is the question of whether the steam generators at our Darlington plant which are tubed with Incoloy 800 will be capable of reaching 60 years of life.

For "new build", the problem has a somewhat different complexion. New materials and new configurations are available to the plant designer to attempt to avoid the material difficulties that earlier designs may have experienced. However, environmental conditions of temperature and pressure will generally also become more challenging in newer designs as utilities seek greater efficiencies from their advanced units. Design decisions made today can have a profound effect on the reliability of the new plants many years into the future. There are lots of examples of this in the nuclear industry. Even seemingly small decisions can have important consequences. Let me give you an example from our CANDU plants. In several CANDU units, a decision was made to not stress relieve the bends in the carbon steel feeders in the

1 "Nuclear Renewal at OPG" presented by P.R. Charlebois, at Canadian Nuclear Society Annual Conference, June 13, 2005.

heat transport system. This has had major impacts on one utility which has experienced cracking and on all the other utilities that have done major amounts of inspection because of the risk that they may also be susceptible. A small decision, a huge consequence. It has cost us millions of dollars in OPG in inspections for a phenomenon which we may never experience.

Any decision to change the material of a component in order to "design out" previously observed degradation mechanisms needs to be made very cautiously for fear of introducing a susceptibility to new forms of degradation that may even be more difficult to manage than the degradation that was being designed out. New materials or old materials in new applications need to have a sufficient "track record" of testing to provide a high degree of assurance to nuclear utilities that their investment in the new plants are based on realistic reliability projections. That's where you come in. The R&D programs need to look into the future, to prepare for and attempt to avoid the surprises that will otherwise emerge.

So in each of the three time frames, and particularly in the medium, long and very long time frames associated with life cycle planning, refurbishment and new build, the technical community has an opportunity to provide information to nuclear utilities to allow us to make the right choices for our business. You do this to a large extent by developing and conducting effective R&D programs. While much of this work is sponsored by governments and government agencies, nuclear utilities also support significant R&D activities.

In OPG, we invest approximately $18-20 M per year on R&D activities, mostly in collaborative research ventures such as those managed by the CANDU Owners Group in Canada and the Electric Power Research Institute in the US. The majority of this work is related to material degradation. This is either directly through projects to understand the incidence of the degradation phenomena, their mechanisms and their rates, or indirectly through programs aimed at helping utilities cope with the degradation, for example, development of inspection and repair technologies, or analysis to understand safety margins in systems that have suffered some sort of material degradation. Why does OPG support this kind of work? Quite simply, because it makes business sense. Make no mistake, we consider the money we spend on R&D to be an investment, not support to a charitable cause. We need to know what degradation is happening in our systems and how we can mitigate it in order to be able to manage our nuclear fleet.

So given this compelling need for information, it should be fairly straightforward for nuclear utilities to make their technology investment decisions shouldn't it? Unfortunately, the answer is no, not at all, for several reasons:

1) There are always more potentially good ideas than there are resources to pay for them. In my experience, there has never been, and probably never will be, an instance where there is more than enough money to thoroughly investigate every aspect of a materials problem to everyone's satisfaction. On the contrary, there are more good ideas in more different areas than any utility (or the entire industry, for that matter) is going to be able to support. So choices need to be made. And inevitably some good ideas will not be pursued to the disappointment of their proponents. But choices do need to be made. And in my view, there needs to be a conscious effort in those choices to maintain a breadth in the R&D portfolio to address a range of topics and issues, to avoid becoming overly focused on the problem of the day to the exclusion of everything else.
2) You can't tell which ones are the really good ideas until long after the work has been carried out and paid for. In other words, there are no guarantees with true R&D that the utility will ever get anything useful out of a particular piece of work. It is also unfortunate but probably inevitable that those good ideas that promise the greatest benefit will generally come with the highest risk of failure. The utility decision-makers are left with the difficult judgments of which proposals and ideas will ultimately have the greatest beneficial impact on their company's success. These judgments are often being made by individuals without the benefit of the detailed technical knowledge in the areas in which the proposals are being made.

As a result, utility managers need all the help they can get to understand the true significance of the work that's being proposed, the realistic likelihood of success and the most probable impact of the work. And that's where we need your help. We know that researchers and technical experts are committed to their particular discipline or area of specialty – that is what makes them good researchers. However, that is also what sometimes makes it difficult for them to recognize that a completely different approach or solution may be the most appropriate course for a utility to take in addressing a particular degradation situation. Utility decision-makers need to distribute the limited resources they have at their disposal across the range of current problems and future problems to produce the optimal R&D investment strategy. We are making choices with less than perfect knowledge, and we need objective insights and perspectives from you the technical community to help us to make the right decisions.

So let me conclude by re-iterating that success for us as a nuclear utility depends in large part on our ability to manage the material degradation that will inevitably happen in our systems. There are two basic messages I would like to leave with you.

1. Nuclear utilities are facing material degradation issues in a variety of systems and components. The technical community needs to develop R&D programs that include a broad range of materials, environments and time-frames. You need to pay particular attention to brand new materials to build up a sufficient degree of confidence to allow them to be used in new plant applications. The success of our future projects depends on the effectiveness of the work you are doing today to address tomorrow's issues.
2. Utility managers have to make difficult choices in what proposals to support in order to get the most out of their R&D investment. The technical community can help the utility managers make much more effective choices by providing balanced, objective information on the potential benefits and risks of the work being proposed, as well as its most likely outcome.

If you do these two things, you will be making the lives of the utility managers faced with all of these material degradation issues a great deal easier.

The second plenary paper – was presented by Peter Ford as follows:

F. Peter Ford,
GE Global Research Center (Retired)

Dr. Peter Ford has been associated with the power generation business for 35 years, initially at the Central Electricity Research Laboratories in the UK, and then as manager of the corrosion program at the General Electric Research and Development Center for 25 years. Since retiring from GE he has consulted for various organizations and has been a member of the Advisory Committee for Reactor Safeguards at the US Nuclear Regulatory Commission. He has been a member of this conference's Technical Program Committee since 1985 and has also been associated with other groups concerned with corrosion problems in water-cooled reactors.

Technical and Management Challenges Associated with Structural Materials Degradation in Nuclear Reactors in the Future

Abstract

There are active plans worldwide to increase nuclear power production by significant amounts. In the near term (i.e. by 2020) this will be accomplished by, (a) increasing the power output of the existing reactors and extending their life, and by, (b) constructing new reactors that are very similar to the current water-cooled designs. Beyond 2025-2030, it is probable that the new reactors will have designs that may be very different from those currently in service. Apart from gas-cooled reactors, there is relatively limited commercial experience with most of these innovative reactor designs.

A full discussion of the technical and management concerns associated with materials degradation that might arise over the next 40 years would need to address a wide range of topics. Quite apart from discussing the structural integrity issues for the materials of construction and the fuel cladding, the debate would need to also cover, for example, fuel resources and the associated issues of fuel cycle management and waste disposal, manufacturing capacity, inspection capabilities, human reliability, etc., since these all impact to one degree or another the choice of material and the operating conditions.

For brevity, the scope of this article is confined to the integrity of the materials of construction in the current water-cooled reactors, the evolutionary designs (which will dominate the near term new constructions) and the very different GEN IV reactor designs. In all cases the operating environments will be more aggressive than currently encountered. For instance, the concerns for flow accelerated corrosion and flow-induced vibration are increased under extended power uprate conditions for the current water-cooled reactors. Of greater concern, the design life will be at least 60 years for all of the new reactors, and for those current reactors applying for extended licenses. This automatically presents challenges with regard to managing irradiation damage in both metallic and non-metallic materials of construction, and for managing cumulative damage due to environmentally assisted cracking. This issue is compounded by the fact that some of the future innovative reactor designs involve fast neutron spectra, and all involve increases in temperature to the range 5000C - 12500C. Comparatively little is known of the effect of, for instance, creep-fatigue interactions in high irradiation fluxes on the structural integrity of the potential materials of construction.

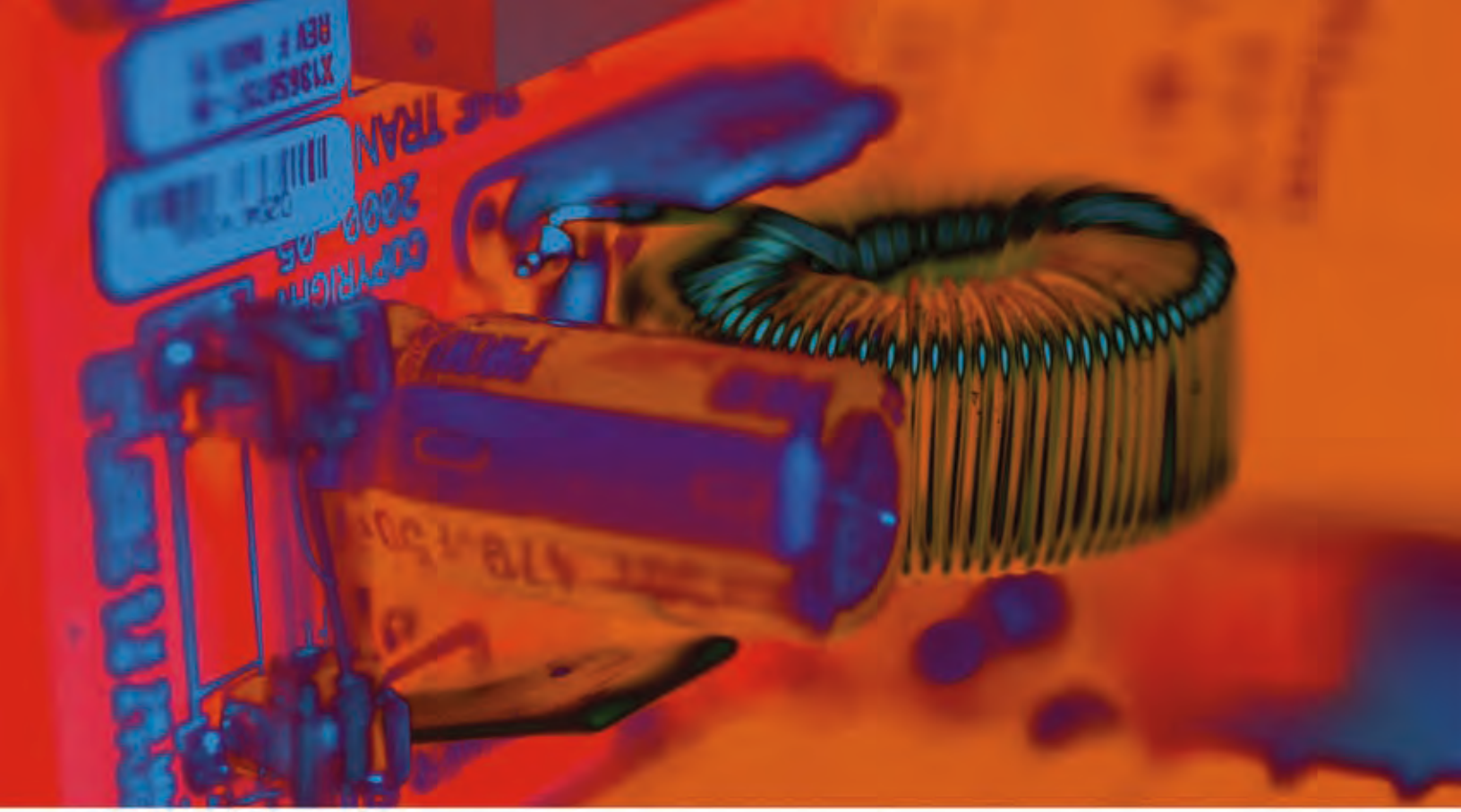
In spite of these technical concerns there is the business management expectation that all of these reactors will experience very few materials degradation problems that might affect the economics of operation.

The paper starts with a review of our present capability to predict the materials degradation modes encountered in the current BWR and PWR reactor designs. This capability is the basis for any analysis of the future degradation problems (and their mitigation) in the current reactors and in the evolutionary water-cooled reactor designs. This section concludes with an overview of assessments of future materials degradation issues that might be expected in these water-cooled reactors.

These preliminary discussions are then broadened to cover some of the more obvious technical problems likely to be encountered with the more advanced GEN IV designs, such as the Very High Temperature Reactor (VHTR) and the Super Critical Water Cooled Reactor (SCWR).

The article concludes with a brief discussion of some of the challenges facing the technical management/leadership, with some suggestions on how to overcome them. These challenges may become especially severe given the fact that the technical problems must be overcome in a time frame that is short compared with that taken to resolve the issues that have faced us over the last 30 years. Some specific management challenges include:

- The decrease in the number of experienced experimentalists and analysts over the last 10 years.
- The decrease in "institutional " memory as it relates to the operation of the current reactors and the design and construction of evolutionary water-cooled reactors.
- Financial constraints which topple the desired balance between shorter term operations-and-maintenance development programs and the longer term research programs.
- The vital need to ensure effective communications between diverse institutions such as national laboratories, universities, regulators, reactor vendors and utilities, during the design stages for the GEN IV reactor concepts, and a clear definition of the changing leadership roles during the various stages of development.



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The Kroll Zirconium Medal Award

Dr. Malcolm Griffiths, Manager of Deformation Technology, Chalk River, was awarded the 2005 Kroll Zirconium Medal.

Dr. Malcolm Griffiths, head of Deformation Technology Branch at Chalk River Laboratories, and Professor Richard Holt, Professor of Nuclear Materials at Queens University, were each presented the William J Kroll Zirconium Medal Award on June 27, 2007 at the ASTM 15th International Symposium on Zirconium in the Nuclear Industry held at Sun River, Oregon, USA.

The William J Kroll Zirconium Medal was established in 1975 to recognize outstanding achievement in the scientific, technological or commercial aspects of zirconium production and utilization, and to encourage future efforts, studies, and research. Once regarded as a rare metallurgical curiosity, this exotic metal has been proven to possess exceptional properties that make it suitable for use in nuclear reactors. Dr. Kroll and his colleagues developed the process that could be carried out on a commercial scale so that the one-time laboratory curiosity could become an article of commerce.

Dr. Griffiths is recognized for his significant contributions to the understanding of phase transformations related to radiation induced deformation of zirconium alloys and was the first to show that vacancy dislocation loops, and the conditions under which they occur are the defects associated with breakaway irradiation growth in pressure tubes. Although the award was presented at the June Symposium the awards committee selected him for the 2005 Award.

Professor Richard Holt, Professor and Chair of Nuclear Materials at Queens University was selected for the 2004 award. He formerly held various research and management positions at Ontario Hydro and Chalk River Laboratories, and is recognized for his extraordinary work in identifying and predicting in-reactor deformation mechanisms of zirconium alloys and for his contributions to the safe and efficient operation of CANDU and LWRs. With over 100 technical publications, Professor Holt laid the groundwork for developing better predictions of in-reactor deformation.

Call for Papers – IYNC 2008

"YOUTH, FUTURE, NUCLEAR"

The International Youth Nuclear Congress (IYNC) 2008 will be held in Interlaken, Switzerland, from 20 -26 of September 2008.

Deadline for abstracts is October 31, 2007.

For More Information contact the Technical Program Chair Yung Hoang: yung.hoang@iync.org or visit the IYNC website: www.iync.org.

Call for Papers – 16th PBNC

"Pacific Partnership toward a Sustainable Nuclear Future"

The 16th Pacific Basin Nuclear Conference (16PBNC) will be held in Aomori, Japan, from 13-18 October, 2008.

Deadline for abstracts is September 28, 2007.

For more information, please contact the organising committee: info@pbnc2008.org or visit the PBNC websire: http://www.pbnc2008.org/

Call for Papers – IRPA 12

The Argentine Radiation Protection Society is hosting the 12th International Congress of the International Radiation Protection Association. It will take place in Buenos Aires (Argentina), from October 19 to 24, 2008, at the Buenos Aires Sheraton Hotel and Convention Center.

Deadline for abstracts is December 1, 2007.

For more information please visit the host website: http://www.irpa12.org.ar

Computed Phase Equilibria and Material Studies for the Zirconia – Gadolinia – Dysprosia – Yttria System

by E.C. Corcoran; Supervisors: W.T. Thompson and B.J. Lewis[1]

Ed. Note: The following paper was awarded 1st place for student paper, presented at the 31st CNS/CNA Student Conference. The paper examines high-temperature material properties of Zirconia – Gadolinia – Dysprosia - Yttria alloys needed for advanced reactor fuels.

Abstract

Neutron absorbing elements are required in the center of advanced CANDU fuel bundle designs that make use of slightly enriched uranium in the surrounding elements. Dysprosia mixed with uranium dioxide is one such absorber that has been used for Low Void Reactivity Fuel (LVRF)[1]. An inert zirconia (with and without yttria) mixture is also being considered as a carrier for the neutron absorbers gadolinium and dysprosium. This quaternary oxide system was modeled from the binary oxide phase diagrams involving zirconia using interpolation methods to estimate the Gibbs energies of the multi-component phases. The model provides the solubility of Ln_2O_3 (Ln=Y,Gd,Dy) in the cubic structure of ZrO_2 as well as the temperature where the cubic phase is expected to melt or decompose into more stable solid phases.

1 Chemical and Materials Engineering in the Department of Chemistry and Chemical Engineering, Royal Military College of Canada, P. O. Box 17000, St. Forces, Kingston, ON, Canada K7K 7B4

1 Introduction

Fundamental thermodynamics can be applied to determine the most stable phases at specific conditions of composition and temperature [2]. This capability is especially useful at high temperature where experimental work is difficult to conduct. This paper provides a computed phase diagram for a proposed new material for the central element of a CANDU fuel bundle which contains zirconia (ZrO_2), and a mixture of yttria (Y_2O_3), gadolinia (Gd_2O_3), and dysprosia (Dy_2O_3) [1]. The central element does not provide fissionable material but rather is intended to capture neutrons to reduce coolant void reactivity when slightly enriched uranium oxide is used in the surrounding fuel elements.

The project first involved modelling the binary phase diagram using equations to represent the Gibbs energy of the phases as a function of composition and temperature for zirconia-gadolinia, zirconia-dysprosia and zirconia-yttria systems. Thereafter these thermodynamic models were combined using an interpolation scheme to generate the quaternary system.

2 Rare Earth Oxides (REO)

Dysprosium and gadolinium are members of the rare earth or lanthanide series of elements ($57 \leq$ Atomic No. ≤ 71). Another member of Group 3B, yttrium (Y, Atomic No. 39) is usually also included in discussing lanthanide chemistry. For simplicity, Ln will be used to represent all of the chemically similar lanthanide elements (La to Lu and Y). Those in this series are characterized by small chemical distinctions that stem from the partially filled 4f electron orbital. Yttrium has no "f" electron but has a similar ionic radius (Ln^{3+}) to other "true" lanthanides and generally similar chemistry as well [3].

The 3^+ oxidation state leads to the formation of the very stable Ln_2O_3 referred to as a sesquioxide. Depending on the ionic radius, sesquioxides generally crystallize into three main structures namely: hexagonal, monoclinic, and cubic [3]. The cubic structure bears many similarities to cubic zirconia.

3 Modelling Mixed Oxide Systems

When modelling an oxide solid solution, it is important to keep in mind that the cations (Zr^{4+} and $Ln3^+$) interchange on similar sites within a continuous lattice of oxide ions. This reality is best remembered by selecting as the formula mass of the lanthanide oxides $LnO_{1.5}$ rather than Ln_2O_3 so that a mole of each oxide component contributes the same number of moles of cations to the solid solution. The cubic solid solution of ZrO_2 and $LnO_{1.5}$ provides a good basis for discussion on this subject.

4 Crystal Structure of Cubic ZrO_2 and Cubic Ln_2O_3

4.1 Cubic ZrO_2

Zirconium oxide (like UO_2) is isomorphous with mineral CaF_2 (fluorite) as shown in Figure 1.

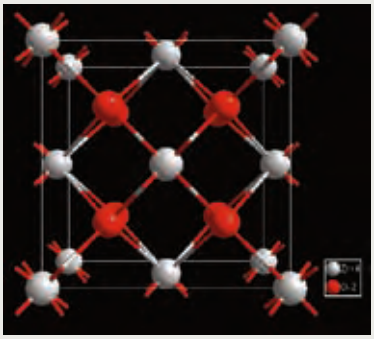


Figure 1. The cubic (fluorite) structure of ZrO_2 White atoms represent Zr^{4+} ions; red atoms represent O^{2-} ions [4].

Chemically, the actual unit cell contains 12 complete atoms (ions). In each unit cell there are eight O^{2-}ions, eight Zr^{4+}, 1/8th corner atoms, and six Zr^{4+} 1/2 atoms on each face (Zr_4O_8). This structure can be represented by the Pearson symbol of cF12 [5] or more uniquely by the Hermann-Maugin space group symbolism, Fm-3m [6].

4.2 Cubic Ln_2O_3 (C-type structure)

In cubic Ln_2O_3, the Ln^{3+} ions have similar radii as the Zr^{4+} ions, and form a similar fluorite-like arrangement as ZrO_2 provided the temperature is not too high. To maintain charge neutrality, one quarter of the oxide ion sites are vacant. These vacancies appear in a three-dimensional periodic pattern which distorts the placement of oxide ions as in fluorite [3]. This is shown on the left side of Figure 2 in which the true unit cell is difficult to visualize. Therefore, one-eighth of this unit cell is shown in the right side of Figure 2 where it is evident that only six O^{2-} ions (red) surround the Ln^{3+} ions (white). It is this side of Figure 2 that bears close comparison with Figure 1.

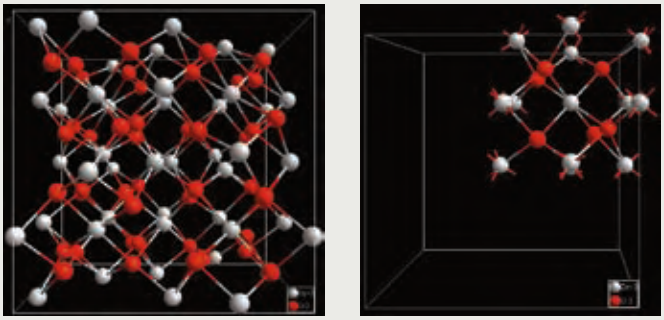


Figure 2. Bixbyite structure of gadolinium oxide. Complete cF80 unit cell (left). 1/8th of the unit cell (right). White atoms represent Gd^{3+} ions; red atoms represent O^{2-} ions. Note: the distortion of the oxide plane (as compared to Figure 1) is caused by the absence of O^{2-} ion[4].

The Pearson Symbol for cubic Ln_2O_3 is cF80 (32 Gd^{3+} ions + 48 O^{2-} ions). The Hermann-Maugin space group is Ia-3. These notational practices relay the otherwise close structural similarities of cubic ZrO_2 and cubic Ln_2O_3.

5 Mixing Oxides to Form Solid Solutions

When cubic solid oxide solutions of ZrO_2 and Ln_2O_3 are formed, the Zr^{4+} and Ln^{3+} ions interchange on cation lattice sites. The difference in charge affects the number and placement of vacancies throughout the crystal structure. This has a direct (although small) effect on the Gibbs energy of the system which affects the thermodynamic stabilities of the cubic solid solution relative to other potential phases at comparable conditions of temperature (and pressure).

As mentioned above, it is convenient to select the formula mass of the lanthanide sesquioxide as $LnO_{1.5}$. This formula mass contains 1 mol of cations as does the formula mass for zirconia, ZrO_2. Because of the potential confusion in representing composition, computed binary phase diagrams include three scales: mole fraction (X) of ZrO_2-$LnO_{1.5}$, mole fraction ZrO_2-Ln_2O_3, and weight percent (*wt%*) lanthanide oxide. A conversion table for zirconia-gadolinia mixtures is shown in Table 1 below. Note that the formula mass selected for the lanthanide sesquioxide has numerical affects on what is meant by the mole fraction of ZrO_2.

Table 1. Conversion table between the three compositional scales in the zirconia-gadolinia system.

wt% Gadolinia	$X_{GdO_{1.5}}$	X_{ZrO_2}	$X_{Gd_2O_3}$	X'_{ZrO_2}
0	0	1	0	1
26.9	0.2	0.8	0.11	0.89
49.5	0.4	0.6	0.25	0.75
68.8	0.6	0.4	0.42	0.58
85.5	0.8	0.2	0.66	0.44
100	1	0	1	0

6 Binary Systems

Phase diagrams are maps of the most stable phase(s) as a function of composition and temperature. The hydrostatic pressure is set at 1 atm although this variable is of little consequence unless the variation is extreme (several kbars). To illustrate the fundamentals of the subject matter [7], ZrO_2-$GdO_{1.5}$ will be examined at 2500°C where for simplicity only the relative stability of the cubic and liquid phases is considered.

The phase(s) may be determined by the process of Gibbs energy minimization. The phase(s) present at a given temperature and overall composition must provide the lowest Gibbs energy (G) for the system as a whole. For a binary isobaric system, the change in Gibbs resultant from mixing (dissolution) may be viewed as the summation of an ideal mixing term (ΔG_{ideal}) and an excess Gibbs energy term (G^E) [2]:

$$\Delta G_{mix} = \Delta G_{ideal} + G^E \qquad (1)$$

The ideal mixing term assumes cations [Zr^{4+}, Gd^{3+}] in the binary system randomly interchange on similar lattice sites. For one mole of solution the ideal mixing may be represented by [2]:

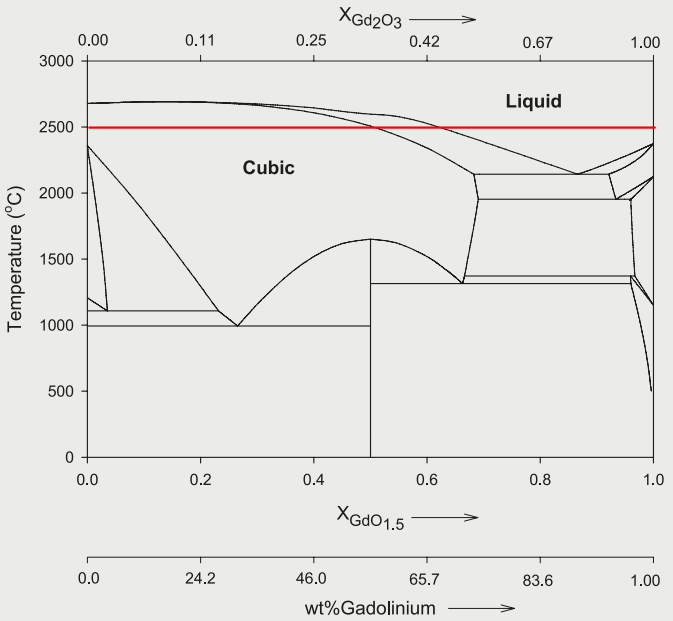


Figure 3. Binary phase diagram for ZrO2 and GdO1.5. Red line highlights the isotherm at 2500oC.

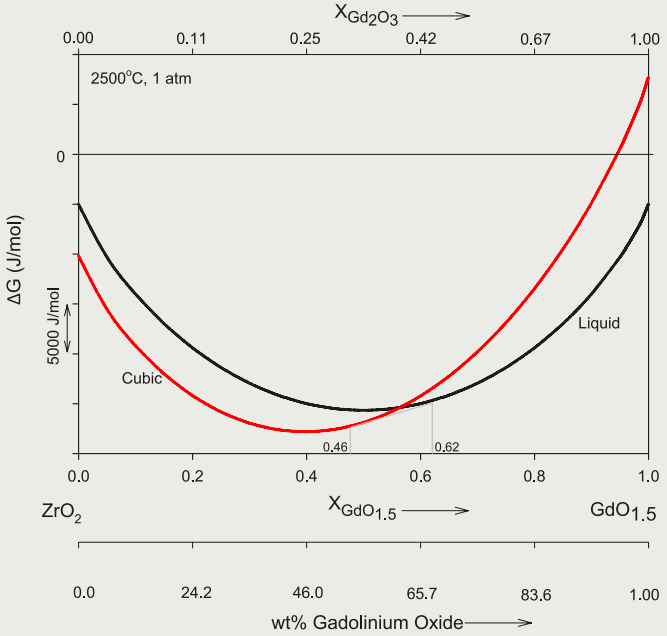


Figure 4. Gibbs energy curves for cubic and liquid phases at 2500oC and 1 atm.

$$X_{ZrO_2} ZrO_2 + X_{GdO_{1.5}} GdO_{1.5} = (ZrO_2 - GdO_{1.5})_{Solution} \quad (2)$$

The ideal Gibbs energy of mixing for this process based on the assumption of random mixing of the cations is closely represented by [2]:

$$\Delta G_{ideal} = X_{ZrO_2} RT \ln X_{ZrO_2} + X_{GdO_{1.5}} RT \ln X_{GdO_{1.5}} \quad (3)$$

where X_{ZrO_2} and $X_{GdO_{1.5}}$ are the mole fractions of ZrO_2 and $GdO_{1.5}$, respectively, R is the ideal gas constant, and T is the absolute temperature (K). The ideal mixing term overlooks the thermal effects associated with mixing. To adjust the Gibbs energy for any departure from the ideal term, an excess (G^E) term is added. The excess Gibbs energy may be represented as an empirical series [2]:

$$G^E = X_{ZrO_2} X_{GdO_{1.5}} (p_o + p_1 X_{GdO_{1.5}} + p_2 X^2_{GdO_{1.5}} + \ldots) \quad (4)$$

where the coefficients of p may be functions of temperature (often linear). To simplify, the empirical series in Equation 4 can be truncated to:

$$G^E = p_o X_{ZrO_2} X_{GdO_{1.5}} \quad (5)$$

where p_o is a constant. These are called "regular" solutions [2]. The Gibbs energy curves can be constructed for the cubic and liquid phases at 2500°C and 1 atm as functions of $X_{GdO_{1.5}}$ with an appropriate value of p_o for each phase.

Combined with a knowledge of the Gibbs energy difference between the cubic and liquid phases of the pure component oxides, the Gibbs energy curves for the cubic and liquid phases appear as shown in Figure 4.

By utilizing the methodology of Gibbs energy minimization, the phase with the lower Gibbs energy at a given temperature and composition is the phase that is most stable [2]. However, between the line of common tangency running from 0.46 to 0.62 $X_{GdO_{1.5}}$, two phases are more stable then either one separately. Accordingly, represented in Figure 3 (at 2500°C):

- between 0 and 0.46 $X_{GdO_{1.5}}$ cubic is most stable;
- between 0.62 and 1 ($X_{GdO_{1.5}}$) liquid is most stable;
- between 0.46 and 0.62 ($X_{GdO_{1.5}}$) a mechanical mixture of both liquid and cubic is most stable (two phase region).

6.1 ZrO_2-$GdO_{1.5}$ Binary Phase Diagram

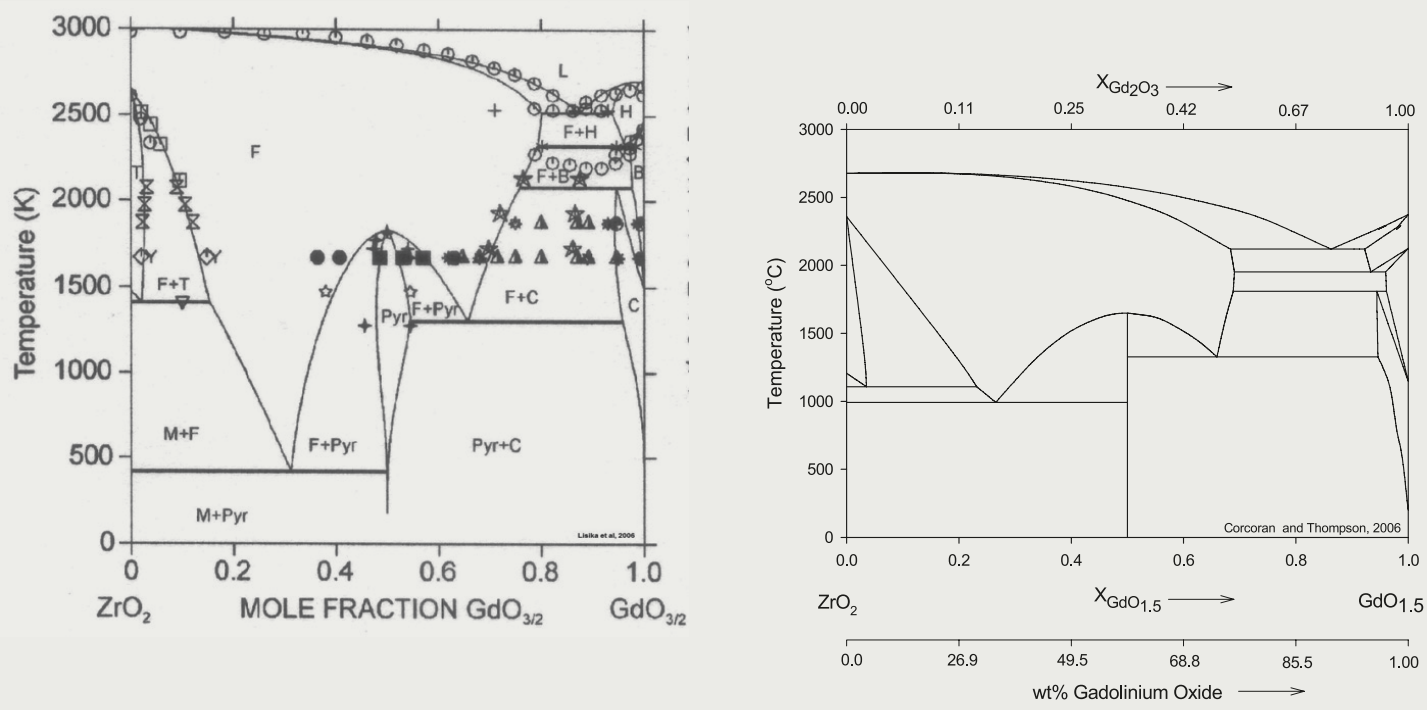
The thermodynamic methodology described above can be expanded to include all of the phases present in the binary system (these include: monoclinic, hexagonal, tetragonal, and bixbyite) and excess mixing parameters can be tuned to match the latest accepted version of the phase diagram. This approach does not preclude using measurements of the Gibbs energy of mixing or related properties but because this information is not currently available.

6.2 ZrO_2-$DyO_{1.5}$ Binary Phase Diagram

The process applied to the ZrO_2-$GdO_{1.5}$ system can also be adapted to the ZrO_2-$DyO_{1.5}$ system. The major difference is that ZrO_2 and Dy_2O_3 can form two oxide compounds whereas ZrO_2-Gd_2O_3 forms only one.

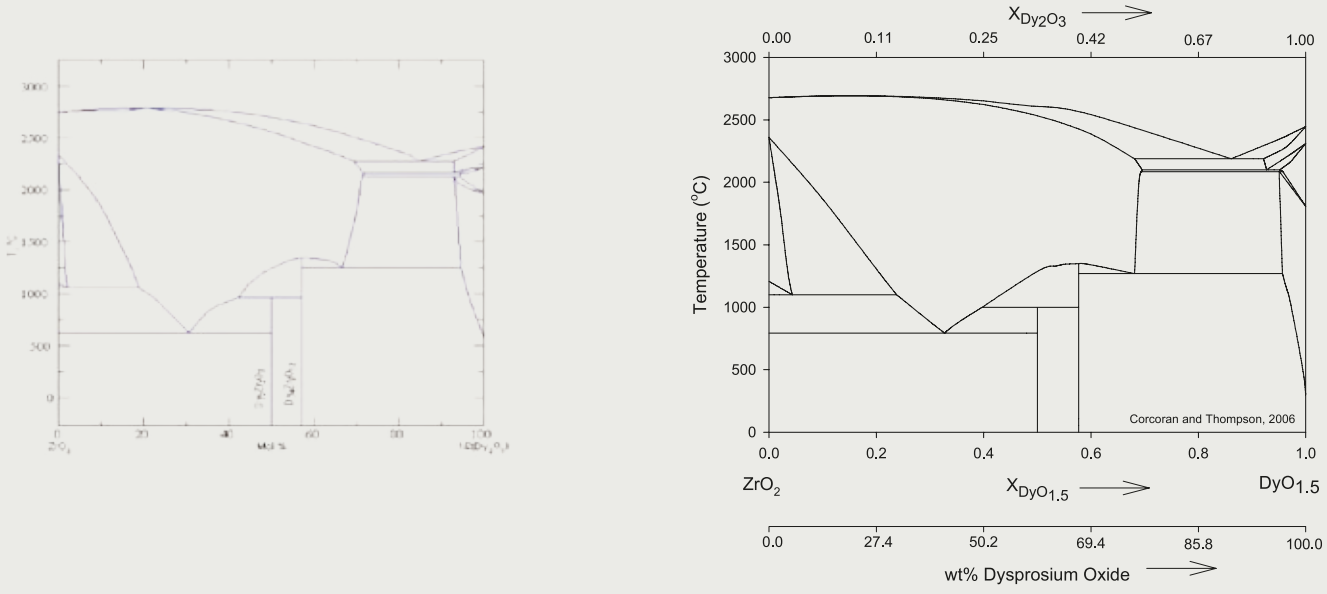
7 Estimated Phase Equilibrium in Ternary Systems

One system considered for the central element material is the ternary zirconia-gadolinia-dysprosia system. This ternary phase system is the integration of the binary phase diagrams for ZrO_2-$GdO_{1.5}$, ZrO_2-$DyO_{1.5}$ with ideal mixing assumed for the various



(a) **(b)**

Figure 5. (a) Recently proposed (2006) binary ZrO_2-$GdO_{1.5}$ phase diagram by Lakiza et al. Solid lines represent thermodynamic model.Data points represent experimental measurements [9]. (b) Thermodynamic binary model developed to match Lakiza et al.



(a) **(b)**

Figure 6. (a)Yokokawa et al. binary phase diagram for ZrO_2 and 1/2 Dy_2O_3 appearing in the ACeRs-NIST v3.0 Database. (Note: 1/2 Dy_2O_3 =$DyO_{1.5}$) [8]. (b) Thermodynamic binary model developed to match Yokokawa et al.

phases in the gadolinia and dyprosia mixtures.

The estimated Gibbs energy at ternary composition point *t* (Figure 7) includes contributions from ideal mixing and excess terms. The ideal term is expanded to be of the form [2]:

$$\Delta G_{mix}^{ideal} = X_{ZrO_2} RT \ln(X_{ZrO_2}) + X_{GdO_{1.5}} RT \ln(X_{GdO_{1.5}}) + X_{DyO_{1.5}} RT \ln(X_{DyO_{1.5}}) \quad (6)$$

For any of the solution phases the excess term is an interpolation from the three binary sub-systems as follows [2,10,11]:

$$G_T^E = \left(\frac{X_{GdO_{1.5}}}{X_{GdO_{1.5}} + X_{DyO_{1.5}}}\right) G_a^E + \left(\frac{X_{DyO_{1.5}}}{X_{GdO_{1.5}} + X_{DyO_{1.5}}}\right) G_b^E + (X_{GdO_{1.5}} + X_{DyO_{1.5}})^2 G_c^E \quad (7)$$

Table 2. Computed Gibbs energy of mixing for ZrO_2-$GdO_{1.5}$-$DyO_{1.5}$ in the cubic phase present at X_{ZrO_2}=0.40, $X_{GdO_{1.5}}$=0.20 and $X_{DyO_{1.5}}$=0.40 (1800°C). See Figure 7.

Point	ΔG_{mix}^{ideal} (kJ/mol)	G_T^E (kJ/mol)	ΔG_{mix} (kJ/mol)
a	-208.54	-6.15	-214.69
b	-211.38	-6.85	-218.23
c	-435.08	0.00	-435.08
t	-215.46	-6.62	-222.08

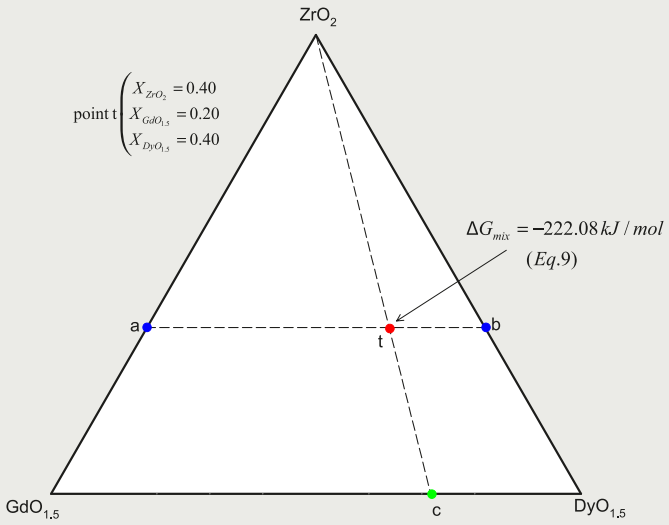


Figure 7. Toop interpolation [11] scheme for Gibbs energy of mixing for the cubic phase at 1800°C. Gibbs energies at *a*, *b*, *c*, and *t* appear in Table 7.

Table 3. Gibbs Energy of mixing for all phases present at X_{ZrO_2}=0.4, $X_{GdO_{1.5}}$=0.20 and $X_{DyO_{1.5}}$=0.40 (1800°C). Note: the lowest Gibbs energy corresponds to the cubic phase which is consistent with the phase field placement at 1800°C in Figure 8.

Phase	ΔG_{mix} (kJ/mol)
Cubic	-222.08
Tetragonal	-216.08
Liquid	-212.54
Monoclinic	-213.62
Bixbyite	-214.38
Hexagonal	-215.46

To numerically illustrate, the Gibbs energy (ideal and excess) for the cubic phase can be calculated at 1800°C for points *a*, *b*, *c*, and *t* on Figure 7. These are shown in Table 2.

The Gibbs energy can be calculated in the same manner for other ternary phases liquid, monoclinic (Ln rich), hexagonal, tetragonal and bixbyite. At point t in Figure 7, the cubic phase has the lowest Gibbs energy (Table 3) and corresponds with the phase field placement in Figure 8.

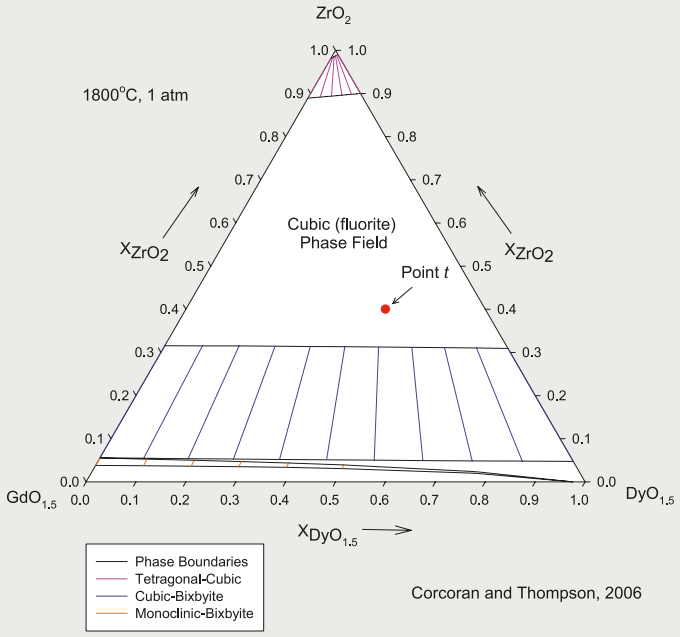


Figure 8. Computed ternary phase diagram for ZrO_2-$GdO_{1.5}$-$DyO_{1.5}$ systems at 1800°C.

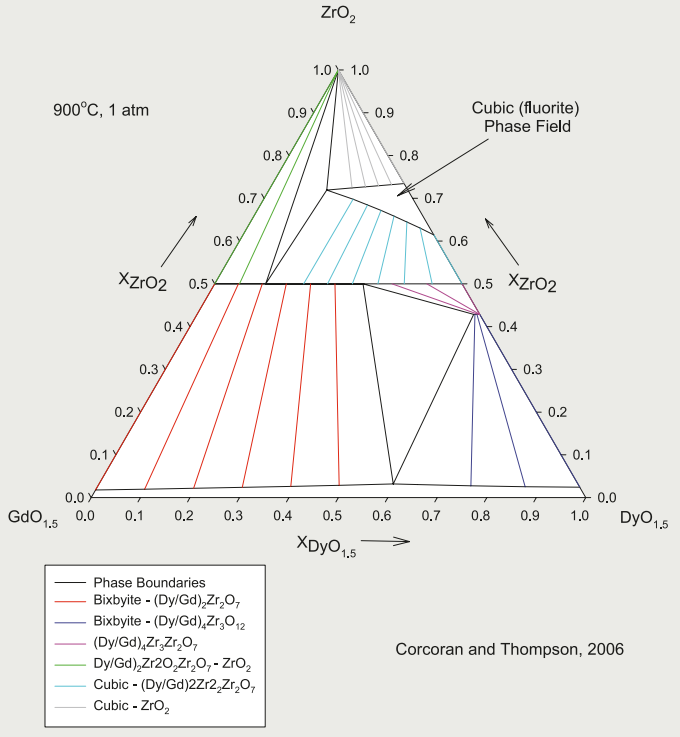
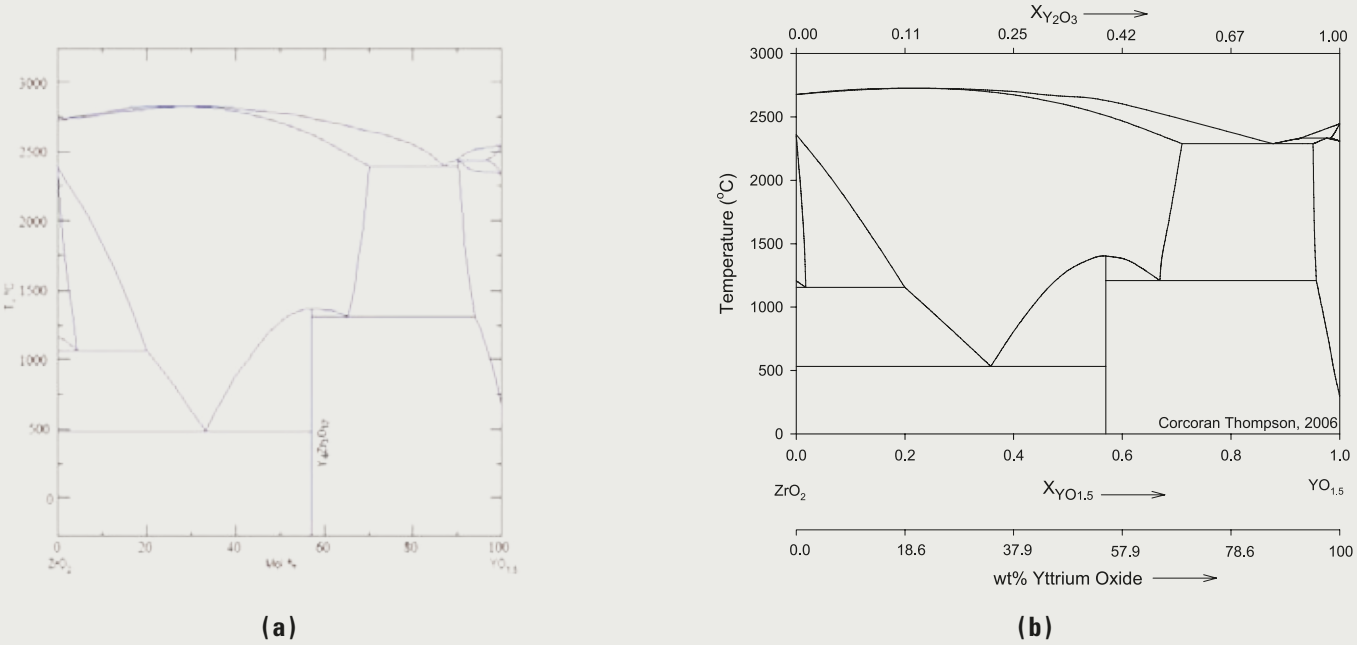


Figure 9. Computed ternary phase diagram for ZrO_2-$GdO_{1.5}$-$DyO_{1.5}$ systems at 900°C.

The computed ternary isothermal sections for the ZrO_2-$GdO_{1.5}$-$DyO_{1.5}$ system at 1800 and 900°C are shown in Figure 8 and Figure 9. These computations are based on the same methodology of Gibbs energy minimizations (used to develop the binary phase diagrams). When ternary Gibbs energy surfaces intersect, implying that ternary phases may co-exist, the



(a) **(b)**

Figure 10. (a) Yokokawa et al. binary phase diagram for ZrO_2 and ½ Y_2O_3 appearing in the ACeRs-NIST v3.0 Database. (Note: ½ Y_2O_3=$YO_{1.5}$) [8]. (b) Thermodynamic binary model developed to match Yokokawa et. al.

compositions for the co-existing phases (ends of tie-lines) can be established in concept by geometrically finding common tangent points associated with a doubly rolling tangent plane.

8 Estimated Phase Equilibrium in the ZrO_2-$YO_{1.5}$-$GdO_{1.5}$-$DyO_{1.5}$ (Quaternary) System

The quaternary integration scheme is much the same as the ternary system (discussed above) with the addition of the zirconia-yttria binary phase diagram (shown in Figure 10).

The Gibbs energy equations for the quaternary multi-components system are shown below; the ideal term is expanded to [2]:

$$\Delta G_{mix}^{ideal} = X_{ZrO_2} RT \ln(X_{ZrO_2}) + X_{GdO_{1.5}} RT \ln(X_{GdO_{1.5}}) + X_{DyO_{1.5}} RT \ln(X_{DyO_{1.5}}) + X_{YO_{1.5}} RT \ln(X_{YO_{1.5}}) \quad (8)$$

The Toop interpolation for the excess term is expanded for the four binary sub-systems* [2]:

$$\Delta G_T^E = \left(\frac{X_{GdO_{1.5}}}{X_{GdO_{1.5}} + X_{DyO_{1.5}} + X_{YO_{1.5}}}\right) G^E_{ZrO_2-GdO_{1.5}} + \left(\frac{X_{DyO_{1.5}}}{X_{GdO_{1.5}} + X_{DyO_{1.5}} + X_{YO_{1.5}}}\right) G^E_{ZrO_2-DyO_{1.5}} + \left(\frac{X_{YO_{1.5}}}{X_{GdO_{1.5}} + X_{DyO_{1.5}} + X_{YO_{1.5}}}\right) G^E_{ZrO_2-YO_{1.5}} + (X_{GdO_{1.5}} + X_{DyO_{1.5}} + X_{YO_{1.5}})^2 G_{GdO_{1.5}-DyO_{1.5}-YO_{1.5}} \quad (9)$$

It is not possible to graphically represent a quaternary system effectively in conventional phase diagram format. Therefore, the phases and their proportions present at a particular composition over a range of temperatures are represented on a bar graph (in which the oxides of Y, Gd, and Dy are lumped together as Ln in the resultant phases). As seen in Figure 11, if this mixture is sintered at 1200°C long enough, a homogeneous cubic phase would form. At reactor operational conditions, thought to be approximately 300-600°C, the thermodynamic model predicts that the sintered cubic compound would become a mixture of monoclinic and the stoichiometric compounds† (as seen in Figure 11).

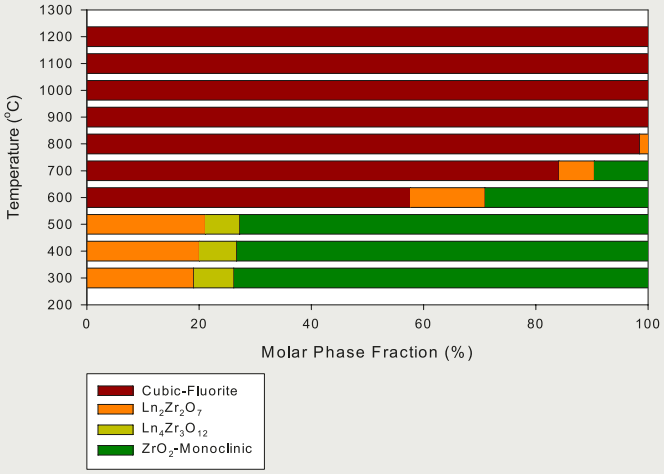


Figure 11. Phase compositional bar graph in mole percent. For an 80 mol% ZrO_2, 6.67 mol% $YO_{1.5}$, 6.67 mol% $GdO_{1.5}$, and 6.67 mol% $DyO_{1.5}$.

* Three binary systems discussed above and one system ($GdO_{1.5}$-$DyO_{1.5}$-$YO_{1.5}$) which assume ideal mixing between the component lanthanide oxides in each of the possible phases

† $Ln_2Zr_2O_7$ and $Ln_4Zr_3O_{12}$ (are near cubic structures)

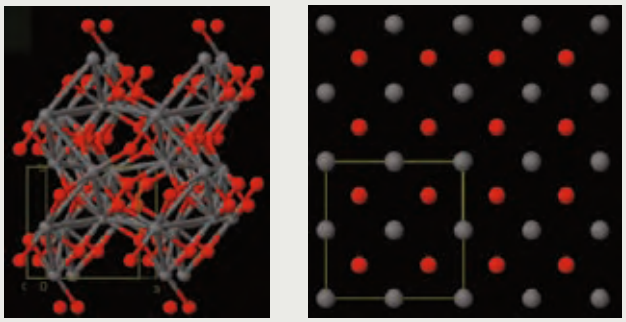


Figure 12. (a) Structural composition of monoclinic zirconia (baddeleyite) [12] versus (b) Structural composition of cubic fluorite zirconia [13]. Zr^{4+} ions are in red; O^{2-} ions in grey.

If this change in phase occurs, it is expected that a volume increase of the material might occur since the monoclinic structure of the pure ZrO_2 (predicted to form) is not as densely packed as the cubic structure (Figure 12) from which it was formed.

It is important to note that the thermodynamic model predicts what is most stable at a given temperature; it does not predict the rate at which the necessary phase changes occur. Currently an investigation using high temperature X-Ray Diffraction (XRD) on mixed oxide sample(s)‡ is underway. This work and related measurements of the thermal expansion coefficient of the cubic (fluorite) solid solution by lattice constant shift will be discussed during the presentation.

9 Summary

This work describes thermodynamic models for the binary systems of zirconia – gadolinia, zirconia – dysprosia, and zirconia – yttria. These were incorporated into ternary and quaternary systems using the Toop interpolation. Phase stabilities as a function of temperature and lanthanide composition are discussed in relation to reactor conditions.

10 Acknowledgements

The author recognizes and thanks AECL-CRL, especially, P. Boczar, L. Dickson, Z. He, R. Verrall, J. Mouris, and H. Hamilton for their continued collaboration and support which have made this project possible. The author also recognizes the continued support of Dr. W. T. Thompson and Dr. B. J. Lewis; their insight and guidance have been extremely valuable to this work.

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‡ Samples provided by AECL-CRL

Our Mistake

In the last issue of the Bulletin (June 2007)
we erroneously attributed the wrong university to
Emily Corcoran
(page 55 "Winners at CNS/CNA Student Conference").
Emily is a Doctoral candidate at the
Royal Military College. We apologise for the error.

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Electrochemical And Weight-Loss Study Of Carbon Steel Corrosion

by V.J. Thomas and R.P. Olive[1]
Supervisor: Dr. Derek Lister

Ed. Note: The following paper was presented at the 31st CNS/CNA Student Conference. The paper provides useful information needed to manage corrosion while the heat transport system is open to atmosphere for an extended period during a large-scale refurbishment project.

Abstract

The Point Lepreau Generating Station (PLGS) will undergo an 18 month refurbishment project beginning in April, 2008. During this time, most of the carbon steel piping in the primary loop will be drained of water and dried. However, some water will remain during the shutdown due to the lack of drains in some lower points in the piping system. As a result, it is necessary to examine the effect of corrosion during the refurbishment.

This study examined the effect of several variables on the corrosion rate of clean carbon steel. Specifically, the effect of oxygen in the system and the presence of chloride ions were evaluated. Corrosion rates were determined using both a weight-loss technique and electrochemical methods. The experiment was conducted at room temperature. The corrosion products from the experiment were analyzed using a Raman microscope.

The results of the weight-loss measurements show that the corrosion rate of polished carbon steel is independent of both the presence of oxygen and chloride ions. The electrochemical method failed to yield meaningful results due to the lack of clearly interpretable data and the inherent subjectivity in the analysis. Lepidocricite was found to be the main corrosion product using the Raman microscope.

1 University of New Brunswick, Canada

1. Introduction

The PLGS employs a CANDU-6 nuclear reactor. CANDU-6 reactors make use of natural uranium fuel and use heavy water as moderator and coolant. The PLGS will begin refurbishment of their reactor in just under one year. This refurbishment is scheduled to last 18 months during which the station will be under outage conditions. This is required in order to extend the service life of the station to approximately 2032.

During the 18 month refurbishment several parts of the reactor will be replaced. This includes: the pressure tubes, the calandria tubes, the end fittings, and the feeders. Thus, to successfully replace these parts, draining of the majority of the piping will be required. Some water will inevitably be trapped in the primary heat transport system; as a result, the effect of corrosion on these pipes must be evaluated. Should corrosion be a concern, the station can then take preventative measures to avoid major problems like build-up of corrosion product in the pipes and leaks.

Piping in this system is mostly made of SA106 grade B carbon steel with some stainless steel sections. Different variables can affect the rate of corrosion such as the presence of chlorides as well as exposure to oxygen from the atmosphere. The variation of pH in the water trapped in the piping can be an indication of the rate of corrosion. One of the worse types of corrosion for piping, pitting corrosion, must be considered. As it is a localised type of corrosion, it can be dangerous and cause damage to the piping or, in severe cases, leaks.

2. Weight-loss analysis of carbon steel

2.1 Experimental Set-up

The operating conditions of the PLGS were approximated in this study. The pH is kept at approximately 10.2; however, this is for heavy water. For simplicity, deionized light water was used in the experiment. The pH was thus taken as 9.8 to account for the difference in fluid. This was achieved using lithium hydroxide. The difference in radiation level between the station and the Head Hall laboratory was disregarded as it was insignificant to this experiment. The temperature in the laboratory was assumed to be similar to that of the station and thus not considered in the analysis of the results.

SA106 grade B carbon steel and SA312 304L stainless steel samples were obtained from NB Power. The samples were machined into small coupons measuring approximately 1 cm by 1 cm with a 1 mm thickness. For the electrochemical experiment, the samples were placed into Mason jars using protected stainless steel wire, in which the station conditions were simulated. For the weight-loss experiments, the coupons were hung on small hooks inside the jars.

Over the 18-month outage, residual water will likely evaporate completely; nonetheless, the experiment considered cases where the water was allowed to evaporate completely as well as cases where water remains in the piping. In order to account for evaporation of the water, two coupons were placed towards the top of each jar and two very close to the bottom. This is because two of the coupons were tested for weight-loss on a weekly basis, whereas the other two were only tested at the end of the experiment. The arrangement can be observed in Figure 1, section 2.3. To account for the 'worse case scenario', a concentration of 0.1ppm lithium chloride was used in half of the jars. The oxygen was also purged out of half of the jars using hydrogen. Thus,

Table 1 Test matrix for weight-loss experiment

Jar #	Chloride	Sealed/Unsealed	Experiment
1	Yes	Unsealed	Weight-loss
2	Yes	Sealed	Weight-loss
3	No	Unsealed	Weight-loss
4	No	Sealed	Weight-loss

four jars were required to represent four important cases. Their setup is shown in Table 1.

The sealed jars refer to the jars purged of oxygen, which were sealed after purging, and the unsealed jars were open to the atmosphere.

2.2 Descaling procedure and weight-loss analysis

The descaling procedure used was the ASTM standard procedure for cleaning corrosion test specimens [1]. Prior to the start of the experiment, the samples were weighed to determine their initial weights. Once a week, two samples from jars 1, 2, 3 and 4 were removed and tested: one sample from the top of the jar and one sample from the bottom. The weekly analysis was performed on the same eight coupons. The remaining 8 coupons were left undisturbed for 8 weeks.

The samples were removed from the jars using tweezers, dried with compressed air, and weighed. The samples were then descaled by being dipped for 25 minutes in agitated Clarke's solution, composed of 20g of antimony trioxide, 50g of tin chloride and one litre of high concentration HCl solution, as recommended by the ASTM Standards [1]. All samples were then rinsed under running de-ionized water, air-dried and weighed again before being returned to their respective positions in the Mason jars. This procedure was followed for 56 days for the eight samples that were analysed weekly. After the 56 days, the coupons were returned to their jars and left undisturbed until a final descaling was performed 13 weeks from the start of the experiment. The remaining 8 coupons were left undisturbed from the start of the experiment for 8 weeks and then descaled and weighed using the procedure described above.

2.3 Results and discussion

The experiment ran for 56 days under weekly testing conditions for eight of the samples, and then for an additional 35 days during which none of the samples were displaced, giving a total of 91 days. During that time, the samples corroded significantly. An example of a Mason jar containing carbon steel coupons, after 56 days, is shown in Figure 1. From this figure, the oxide can be clearly observed on the coupons, as well as at the bottom of the jar. This photograph only illustrates one example; nonetheless, after 91 days, all 8 of the jars had a similar appearance. The oxide formation was actually quite significant as early as 10 days after beginning the experiment.

Also unclear in the photograph is the distinction between the samples that were descaled weekly and the undisturbed samples; however, measured corrosion rates demonstrate the effects of weekly descaling as opposed to allowing corrosion product build-up to form on the coupons.

Figure 1 Carbon steel coupons in Mason jar after 91 days

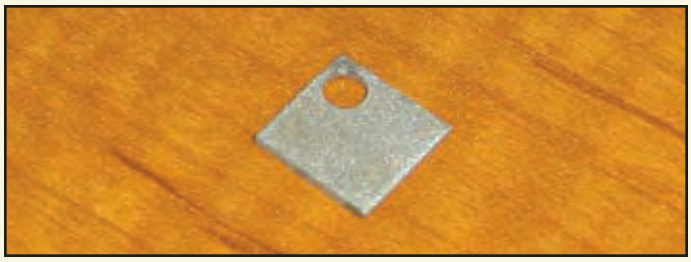

Figure 2 Carbon steel sample after descaling

Figure 2 illustrates a carbon steel coupon after undergoing descaling procedure. After descaling, the coupons returned to a state similar in appearance to their initial conditions. Although it is not very clear from this photograph, after descaling the coupons it was evident that visible pitting did not occur in the carbon steel samples. This can be explained by the high corrosion rates obtained.

Figure 3 shows the experimental results from the weight-loss experiment. The weekly descaled coupons are represented by two connected data points; one at 7 days and the other at 91 days. Due to problems encountered with the Clarke's solution, these were the only two valid data points obtained in the experiment. The undisturbed coupons are plotted as points at 56 days.

Several important conclusions can be drawn from Figure 3. First, the coupons in jars that were unsealed exhibited higher rates of corrosion than did the ones in sealed jars. This was as expected as the presence of oxygen generally accelerates corrosion. Secondly, the presence of chloride had no effect on the corrosion rates. As observed in Figure 3, there is no specific trend among the samples in contact with chlorides. In addition, the weekly descaled coupons exhibited higher corrosion rates then their undisturbed counterparts. One

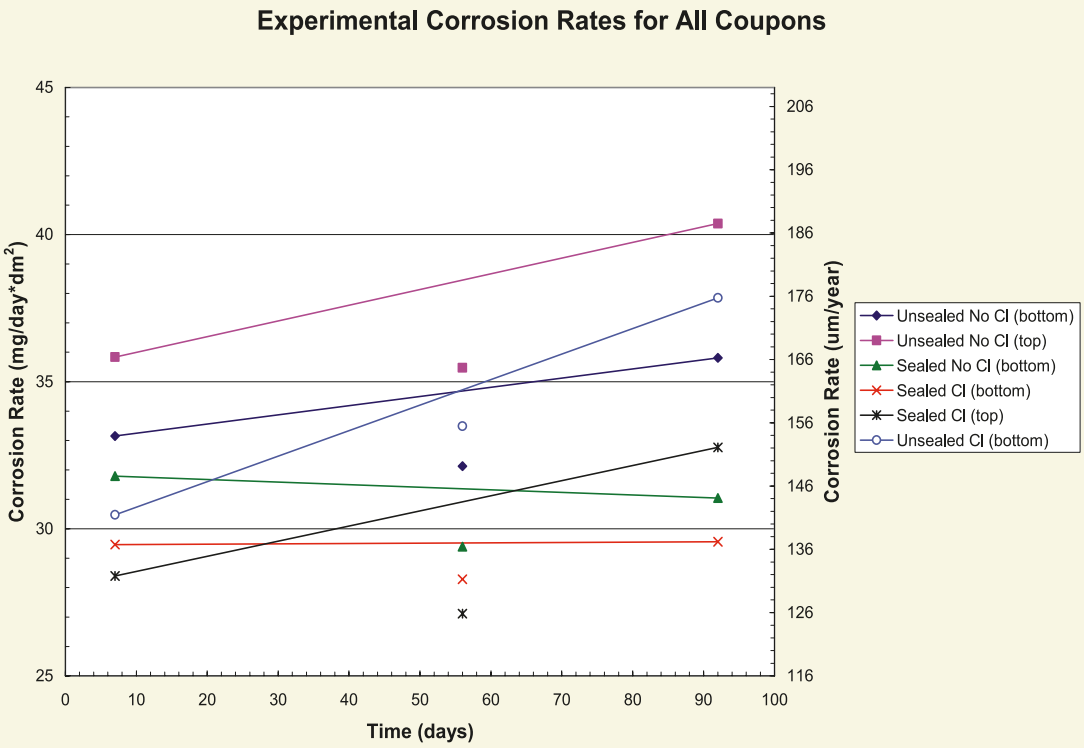

Figure 3 Corrosion rates of all carbon steel coupons

reason for this may be the continual removal of oxide film each week.

It should be noted that the pH of the jars was measured again at the end of the experiment. All of the jars had a pH of approximately 7.1. This explains the high rates of corrosion exhibited by the coupons. As well, the high corrosion rates may also explain why there was no observable pitting. Because pitting is a localized form of corrosion, the already high corrosion rates would not allow pits to develop [2].

3. Electrochemical analysis of carbon and stainless steel corrosion

3.1 Electrochemical corrosion experiments

For the electrochemical experiments, the arrangement of the coupons varied from the arrangement of the coupons subjected to the weight-loss analysis. In this case, only two carbon steel coupons per jar were used: one placed near the top of a Mason jar and one near the bottom. However, each coupon was paired with a stainless steel counter electrode. These were placed at the same height as and parallel to the carbon steel coupons, approximately half a centimetre away from them. These were connected to long wires to allow for connection of the testing equipment. Table 2 provides a summary of the specific conditions for each Mason jar.

Table 2 Test matrix for electrochemical experiments

Jar #	Chloride	Sealed/Unsealed	Experiment
5	Yes	Unsealed	Electrochemical
6	Yes	Sealed	Electrochemical
7	No	Unsealed	Electrochemical
8	No	Sealed	Electrochemical

Electrochemical analysis was performed on these samples using a PCI4 Family Potentiostat 300. This equipment enabled potentiodynamic scans on the carbon steel samples, using stainless steel as a counter electrode. These scans were meant to determine the corrosion potential. Pitting potential scans were also performed on the stainless steel samples only.

3.2 Results and Discussion

3.2.1 Potentiodynamic analysis of carbon steel

Potentiodynamic scans were run on each carbon steel sample weekly. The scans were then analysed using Gamry Echem Analysis Version 1.30 2003 software. The corrosion potential for each carbon steel sample was also determined each week. With this known, the anodic Tafel region was extrapolated back to the point of intersection. From this, the corrosion rate was determined.

However, the potentiodynamic scans proved unreliable for corrosion rate determination. Figure 4 illustrates the problem with this type of testing. From observation, there is no clear straight line

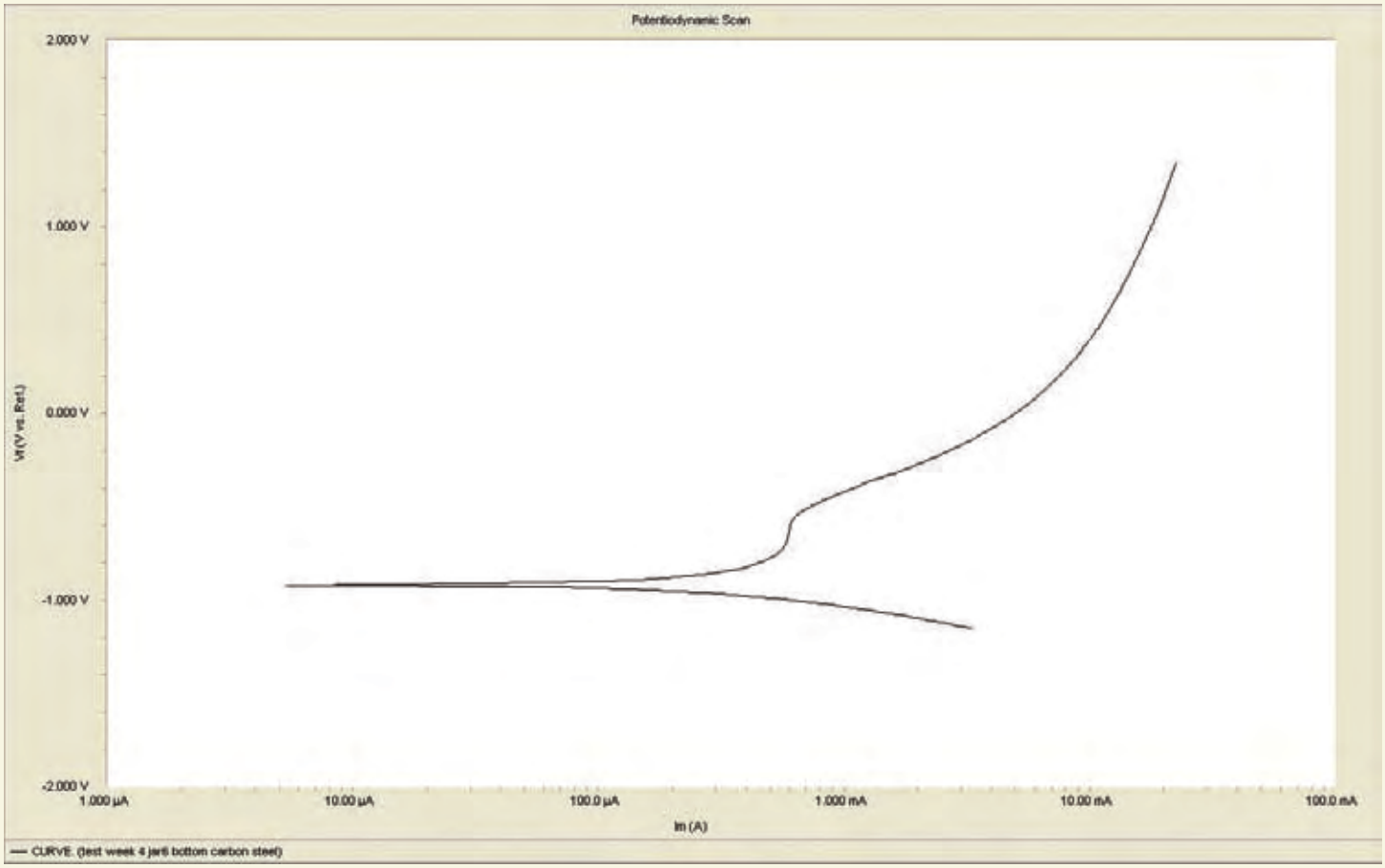

Figure 4 Potentiodynamic scan for unsealed carbon steel coupon, no chloride, located at the bottom of the jar, November 14, 2006.

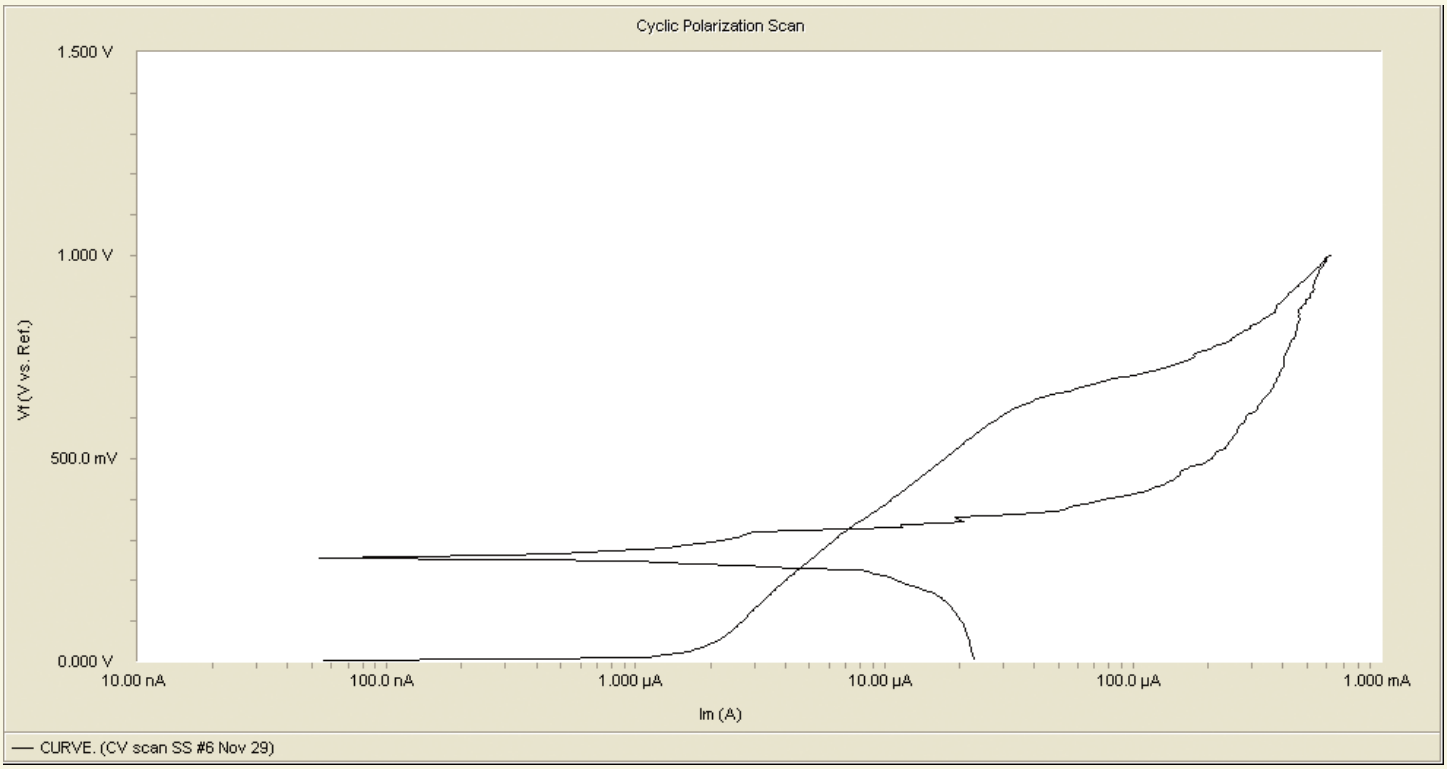

Figure 5 Pitting potential for stainless steel coupon with chloride, located at the top of a sealed jar, November 29, 2006. Pitting potential is 630mV.

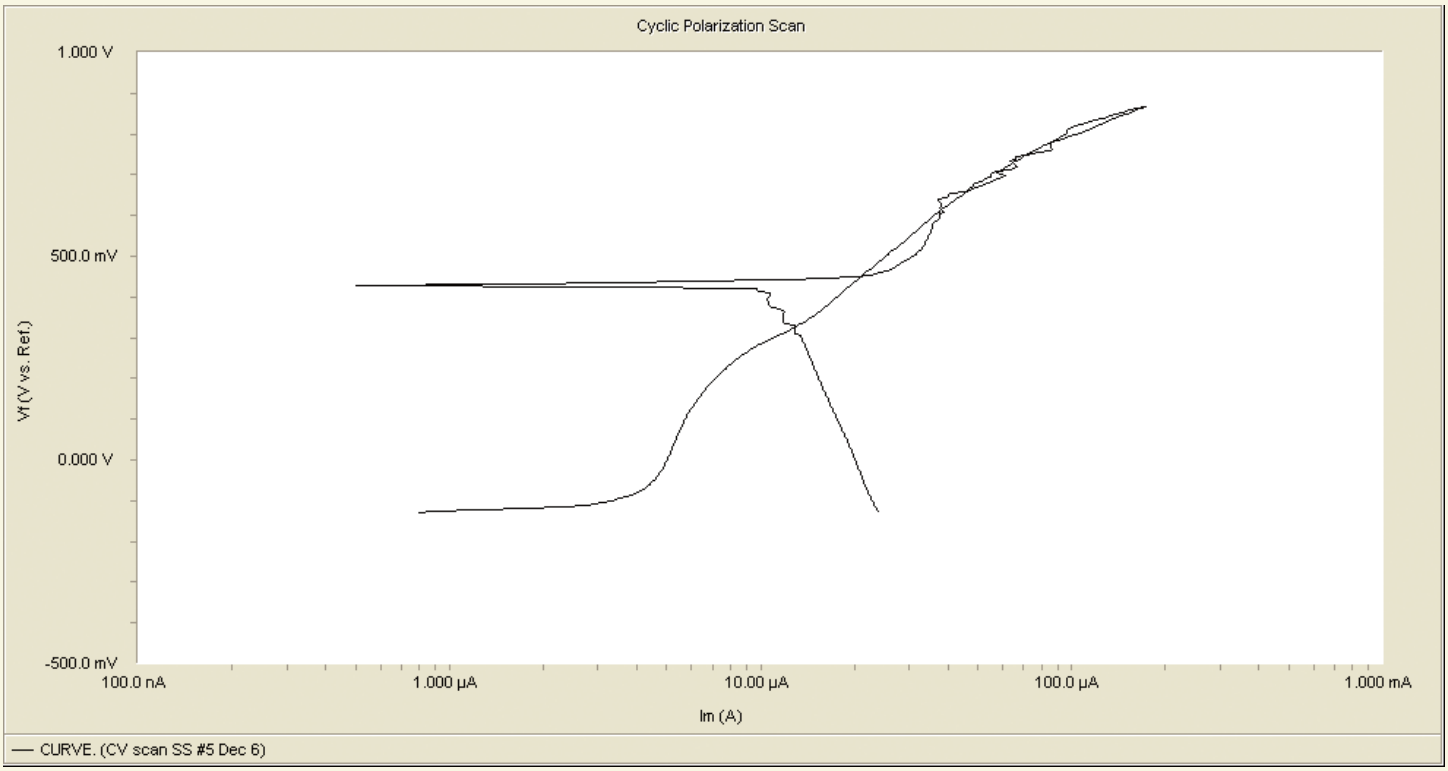

Figure 6 Pitting potential for stainless steel coupon with chloride, located at the top of a sealed jar, December 5, 2006. Pitting potential is ambiguous.

anodic Tafel region. Thus, it is left to the judgement of the experimenter as to where the Tafel region actually is. Because of the inherent subjectivity of this technique, the potentiodynamic scans on the carbon steel samples failed to produce any meaningful results.

3.2.2 Pitting potential of stainless steel

In addition to the potentiodynamic scans on the carbon steel samples, cyclic polarization scans were run on the stainless steel samples in order to determine pitting potentials. These scans were started 5 weeks into the experiment and were performed weekly for the next 3 weeks. Pitting potentials were determined using the method outlined by Roberge [3].

Figure 5 shows the cyclic polarization scan of the top stainless steel sample in the sealed with chloride Mason jar. From the scan, the pitting potential can be determined to be 560 mV. Figure 6 shows the same sample on week later. From this scan, the pitting potential cannot be readily determined. As a result, like the potentiodynamic scans on their carbon steel counterparts, the scans on the stainless steel samples also failed to yield any meaningful results.

4. Surface analysis

4.1 Results

After completion of the experiment, the contents of the Mason jars were analysed. The Laser Raman microscope was used to determine the substance formed in the jars. The results are shown in Figure 7.

From Figure 7, the corrosion product was determined to be Lepidocricite. The peaks at 250cm^{-1}, 378cm^{-1}, correspond to the reference spectrum of Lepidocricite [4].

5. Conclusion

The experiment showed that the presence of chlorides had no effect on the corrosion rates of the samples. As expected, the unsealed samples all exhibited higher rates of corrosion than those purged of oxygen. Also, pitting was determined not to be a concern for SA106 grade B carbon steel under the tested conditions due to high general corrosion rates and observations. The orange sludge corrosion product forming on the coupons was determined to be Lepidocricite using the Laser Raman microscope.

Finally, any electrochemical method should be carefully used. The majority of analysis of such data is very subjective to each individual researcher; as a result, these methods should always be supported or accompanied by other forms of analysis.

6. References

[1] Designation G1-90, "Standard Practice for Preparing, Cleaning and Evaluating Corrosion Test Specimens", *Annual Book of ASTM Standards,* Vol. 03.02 (Wear and Erosion; Metal Corrosion), 2000, pp.15-21

[2] Fontana M.G., "Corrosion Engineering", Second Edition, McGraw-Hill, NY USA, 1978

[3] Roberge P.R., "Handbook of Corrosion Engineering", McGraw-Hill, NY, USA, 2000

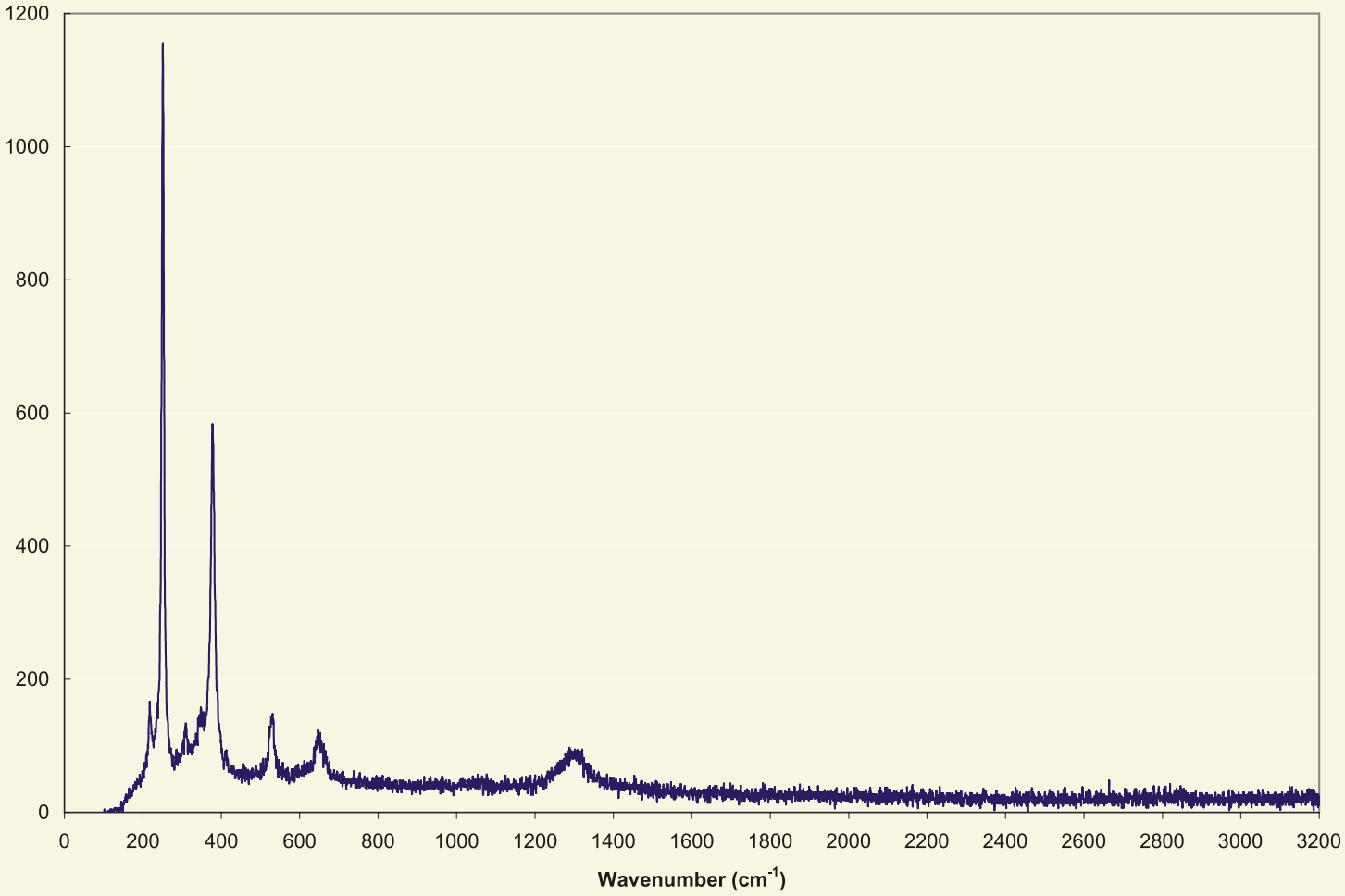

Figure 7 Raman spectrum of Mason jar corrosion product

[4] R.J. Thibeau, C.W. Brown, and R.H. HeidersbachRaman, "Spectra of Possible Corrosion Products of Iron Applied Spectroscopy", Volume 32, Number 6, 1978, pp. 532-535

7. Acknowledgements

I would sincerely like to thank my supervisor Dr. D.H. Lister for all his assistance and support throughout the experiment. I would also like to thank Robert P. Olive, co-author of this paper, and Dr. William Cook who also assisted me with the laboratory experiment and result analysis. Sincere thanks also go to Dave Loughead from NB Power for providing us with the steel samples used in the experiment, along with Andrew Justason from CNER for his involvement in the experiment. I would also like to extend gratitude to Andrew Feicht, the laboratory manager, Lihui Liu, the surface analysis specialist, and Jody Chessie, shop technician.

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Environmental Assessment: Challenges And Opportunities

by Jim Hilbig[1], Duncan Moffett[2], Kelly Beri[2]

Ed. Note: The following paper was presented at the 28th Annual Conference of the Canadian Nuclear Society. The paper describes the Bruce Power experience in conducting the Environmental Assessment for the restart of Units 1 and 2 and provides useful information to guide future assessments under the Canadian Environmental Assessment Act.

Abstract

As part of a $4.5 billion investment, Bruce Power is refurbishing Bruce A Units 1 and 2, having successfully completed an environmental assessment to return these units to service after a lay-up of almost 10 years. The project includes implementing a series of refurbishments and upgrades which will enhance safety, increase electricity generation capacity and improve reliability for the 30-year extended life of the units. This paper describes four challenges that were successfully managed during the extensive environmental assessment: (i) defining the scope of the Project; (ii) understanding the EA trigger under the *Canadian Environmental Assessment Act;* (iii) maintaining an effective relationship with the regulatory agencies; and (iv) managing stakeholder communications.

1 Bruce Power L.P., P.O. Box 3000, Tiverton, Ontario, Canada

2 Golder Associates Ltd., 2390 Argentia Road, Mississauga, Ontario, Canada

1. Introduction

In December 2005, an environmental assessment (EA) was completed for the Bruce A Refurbishment for Life Extension and Continued Operations Project (the Bruce A Refurbishment Project) as proposed by Bruce Power LP (Bruce Power) [1]. The Bruce A Refurbishment Project comprises the return of Units 1 and 2 at the Bruce A nuclear generating station (Bruce A) to service (i.e., to operational status for an extended period through the end of a potential Bruce Power lease extension to 2043), potentially refurbishing Bruce A Units 3 and 4 to enable these units to produce power until 2043, and the potential use of low void reactivity fuel (LVRF), also referred to as New Fuel, in all four units at Bruce A. Progress on the refurbishment can be reviewed on Bruce Power's website (www.brucepower.com).

Bruce Power's proposal to return Bruce A Units 1 and 2 to service from their temporary lay-up required implementing a series of refurbishments, upgrades, and enhancements at Bruce A. In this regard, Bruce Power's proposal had three main goals:

1. Enhance the safety and reliability of the Bruce A station;
2. Increase Bruce A's capacity to generate electricity; and
3. Ensure the station remains safe and fit-for-service through the end of a potential Bruce Power lease extension, i.e., through 2043.

Following a public hearing held on May 19, 2006, the Canadian Nuclear Safety Commission (CNSC) announced its conclusion that Bruce Power's proposed project for the return to service of Units 1 and 2, the refurbishment for life extension of Bruce A, and the potential use of New Fuel in all four units at Bruce A, taking into account identified mitigation measures, is **not likely to cause significant adverse environmental effects** [2]. The CNSC's decision was based on its consideration of an EA of the project that was prepared in accordance with the requirements of the *Canadian Environmental Assessment Act* (CEAA). The Commission was then in a position, under the *Nuclear Safety and Control Act* (NSCA), to consider a licence amendment for the proposed project and Bruce Power was authorized to proceed with the refurbishment.

During the extensive EA process, challenges arose that were successfully managed. Among these challenges were: defining the scope of the project; understanding the EA triggers under the CEAA; maintaining an effective relationship with the regulatory agencies; and managing stakeholder communications. This paper discusses each of these challenges as they relate to the EA for the Bruce A Refurbishment Project.

2. EA schedule

The EA schedule for the Bruce A Refurbishment Project is shown in Table 1. The EA process began with the submission of the draft project description to the CNSC and ended with the CNSC decision on the EA for a total duration of 21 months. Although the implementation of mitigation and a follow-up is defined as a phase of EA under CEAA, it has not been included in the EA schedule presented in Table 1.

3. Defining the scope of the project

Defining the project scope is an essential element in managing a project. The project scope sets the boundaries for the assessment, ensures that the EA focuses on the correct issues, identifies exactly what is being assessed (i.e., defines what is included or excluded in the EA), and establishes the environmental effects that should be considered in an EA.

Table 1: Bruce A Refurbishment EA Schedule

Schedule Item	Date
Draft Project Description submitted to CNSC	October 2004
Final Project Description submitted to CNSC	December 2004
Draft EA Guidelines (Scope of Project and Assessment) issued by CNSC	December 2004
Commission Hearing on the Draft EA Guidelines	May 2005
Final EA Guidelines (Scope of Project and Assessment) issued by CNSC	July 2005
Draft EA Study Report issued by Bruce Power	August 2005
Final EA Study Report issued by Bruce Power	December 2005
Draft Screening Report issued by CNSC	January 2006
Final Screening Report issued by CNSC	March 2006
Commission Hearing on the Screening Report	May 2006
Announcement of CNSC Decision on EA	July 2006

The scope of the project should consider the life-cycle of the project (e.g., site preparation, construction, operations, maintenance, modifications, etc.), malfunctions and accidents, normal operations, and input from stakeholders. Scoping is one of the most critical and challenging aspects of an EA.

In determining the project scope for an EA under the CEAA, the proponent (or delegate) must first determine the physical works involved in the project and the specific activities to be carried out. Most often, these details are provided in a project description, which is submitted to the responsible authority. Under the CEAA, the responsible authority must "determine the scope of the project for the purposes of EA; ensure that all proposed undertakings for a project (e.g., construction, operation, decommissioning, etc.) are included in the EA; ensure consideration of the factors outlined in the Act (e.g., environmental effects, significance, public comments, etc.); and determine the scope of factors to be considered" [3].

For the Bruce A Refurbishment Project, defining the project scope was the joint responsibility of Bruce Power and Golder Associates Ltd. (Golder). Defining the project scope included:

1. Completing a draft project description;
2. Facilitating comments from the CNSC (the only responsible authority identified for the EA); and
3. Completing a final project description.

The scope of the project included a refurbishment phase and an operations phase. The refurbishment phase accounted for site preparation, refurbishment works and activities including upgrades, refurbishment waste management, associated employment, and malfunctions and accidents during refurbishment (radiological and conventional). The operations phase accounted for the normal operations of plant systems, maintenance, waste management, and malfunctions and accidents during operations (conventional, nuclear and criticality events). The assessment scenarios were defined to include some overlap (double accounting) for conservatism. Using conservatism is a wise strategy in EA, since it is likely that the effects identified in the EA as resulting from the project will be greater in magnitude than those experienced during the actual project undertaking. It also allows for additional flexibility in the project schedule.

It is important to ensure agreement between the scope of the project as understood by the proponent and the scope as identified by the responsible authority. Although the project description for the Bruce A Refurbishment Project explicitly stated what work could not be done without the EA being completed, the scoping document issued by the CNSC did not explicitly state this. Based on this decision that the EA scope superseded the existing licence, no work associated with the EA could be initiated until the EA was approved. Accordingly, activities permitted under the existing power reactor operating licence were captured in the EA and therefore could not take place until the EA was completed. This represented a significant challenge to Bruce Power in maintaining the refurbishment schedule since the company had assumed some "pre-project" activities could take place, although at financial risk in the event the EA for the Bruce A Refurbishment Project was not approved. CNSC staff concerns related to the fact that, should they permit Bruce Power to proceed with work related to refurbishment without an approved EA, they could be seen as pre-empting the Commission's decision regarding the EA and the requirement that the EA be approved before a licensing decision is made. The CNSC ultimately decided that, while no major refurbishment work could take place until the EA was successfully completed, some planning and preparation for refurbishment could be undertaken at Bruce Power's financial risk. It is notable that the EA must be completed and approved before the responsible authority can give permit, licence, or licence amendment for the project activities to take place.

The project description for the Bruce A Refurbishment Project benefited from extensive discussions with the CNSC in the form of weekly teleconferences, which allowed for discussions in advance of preparing the project description. Weekly EA team conferences were also beneficial.

In order to improve the defining of the project scope for future nuclear EA projects, a proponent should:

- Ensure that the EA guidelines (scoping document issued by the responsible authority) are consistent with the proponent's project description (documented consistency is key);
- Explicitly recognize existing licence conditions to maintain activities permitted under the licence;
- Understand that the EA guidelines approved by the Commission supersede the project description accepted by CNSC staff;
- Recognize that the scope allowed by the current licence may be undertaken "at owner's risk"; and
- Define what works may be undertaken before the EA is approved by the responsible authority.

4. Understanding the EA triggers under the CEAA

For an EA to be required under the CEAA, it must be determined whether the project triggers the legal requirement for a federal EA. Figure 1 shows the decision process for determining whether an EA is required under the CEAA [3].

As shown in Figure 1, there are four questions to answer to determine whether an EA is required under the CEAA [3]:

1. Does the project meet the CEAA definition of a "project"?
2. Is the project excluded from having to undergo an EA?
3. Does the project require a federal authority[1] decision or action?
4. Is the federal authority obligated (triggered) to ensure that an EA is conducted as a result of the federal decision?

The federal environmental assessment process is triggered whenever a federal authority (FA):

- Proposes a project;
- Provides financial assistance to a proponent to enable a project to be carried out;
- Sells, leases, or otherwise transfers control or administration of federal land to enable a project to be carried out; or
- Provides a licence, permit or an approval that is listed in the *Law List Regulations* that enables a project to be carried out.

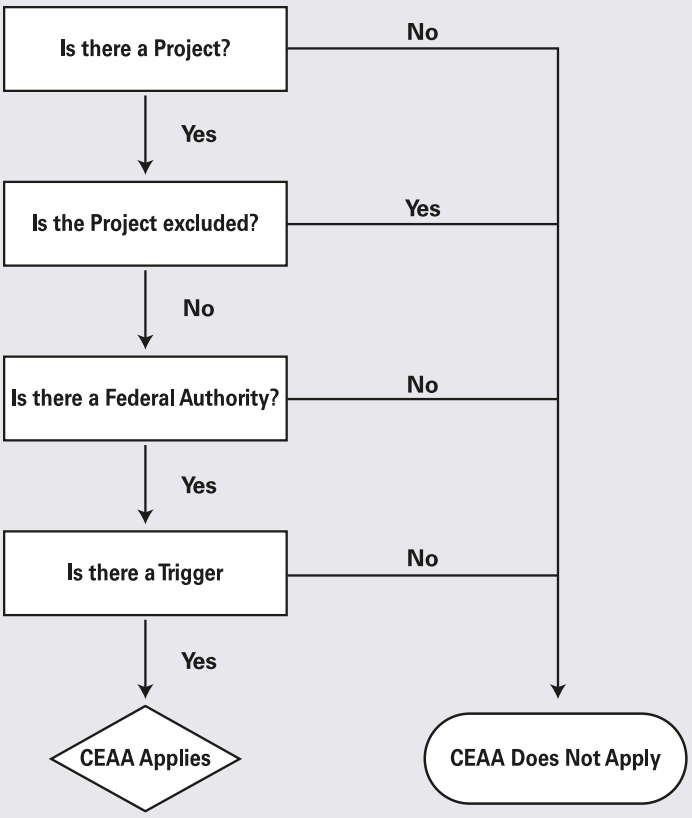

Figure 1: CEAA EA Determination Process

Implementing the Bruce A Refurbishment Project required licensing decisions and amendments to the existing Bruce A power reactor operating licence issued by the CNSC. The CNSC determined that a federal EA was required under paragraph 5(1)(d) of the CEAA before the existing power reactor operating licence could be amended to give Bruce Power the authorization to restart Bruce A Units 1 and 2. (The CNSC determined that, under the NSCA, an amendment of the operating licence for the Bruce A station is a 'trigger' for the CEAA under the *Law List Regulations*). It was determined that there were no other CEAA triggers, such as being a proponent, funding, or disposing of a land interest to support the project, that involved the CNSC.

It is currently the position of the CNSC that in order to amend an operating licence, an EA is required under the CEAA paragraph 5(1)(d) as follows: *"An environmental assessment of a project is required before a federal authority exercises one of the following powers or performs one of the following duties or functions in respect of a project, namely, where a federal authority[:] (d) under a provision prescribed pursuant to paragraph 59(f), issues a permit or licence, grants an approval or takes any other action for the purpose of enabling the project to be carried out in whole or in part."* However, this raises the question of whether an EA is required for each and every operating licence amendment that meets the first three EA requirements as outlined in Figure 1, above.

Under subsection 24(1) of the CEAA, *"where a proponent proposes to carry out, in whole or in part, a project for which an environmental assessment was previously conducted and (a) the project did not proceed after the assessment was completed, (b) in the case of a project that is in relation to a physical work, the proponent proposes an undertaking in relation to that work different from that proposed when the assessment was conducted, (c) the manner in which the project is to be carried out has subsequently changed, or (d) the renewal of a licence, permit, approval or other action under a prescribed provision is sought, the responsible authority shall use that assessment and the report thereon to whatever extent is appropriate for the purpose of complying with section 18 or 21."* With the completion of the Bruce A Refurbishment EA [1], Bruce Power has completed EAs for the Bruce A station and the Bruce B station [5]. Accordingly, there are existing EAs for the Bruce Power facilities; therefore, under subsection 24(1) of the CEAA, future amendments to the Bruce A and Bruce B operating licences should use these EAs to the extent that is appropriate.

Under subsection 24(2) of the CEAA, *"where a responsible authority uses an environmental assessment and the report thereon pursuant to subsection (1), the responsible authority shall ensure that any adjustments are made to the report that are necessary to take into account any significant changes in the environment and in the circumstances of the project and any significant new information relating to the environmental effects of the project."* Therefore, if Bruce Power proposes a project in the future, which involves either of the Bruce A or Bruce B stations, and the project works and activities identified in the proposal differ from those identified in the completed EA, an amendment in the form of an addendum could be prepared to satisfy all of the responsible authority's requirements.

1 Under the CEAA, a "federal authority" is a federal agency or department that has expertise or a mandate relevant to the project and includes federal departments, agencies and ministers of the Canadian government.

This proposed addendum would first identify the new project works and activities under the new proposal and compare them to those from the previously completed EA. Next, a screening of measurable change would be conducted wherein the measurable changes in likely effects between the new proposal and the completed EA are identified in order to focus the assessment on areas where the environmental effects differ. (Measurable changes identified as beneficial are not advanced for further assessment.) A comparison of residual adverse effects would then be conducted on the activities resulting in a measurable change in effects, and a screening recommendation would be made regarding the effects of the project. Where the residual adverse effects of the proposal have been identified as being likely different from the completed EA, the effects would be advanced for an assessment of significance. A recommendation could be made based on the assessment completed in the addendum. The addendum could be submitted to the responsible authority who would be responsible for a formal decision regarding the significance.

It is important to state in the addendum that the reader should be familiar with the previously completed EA in order to avoid any confusion.

5. Relationship with regulatory agencies

For the Bruce A Refurbishment Project EA, the CNSC was identified as the only responsible authority under the CEAA; however, other federal regulatory agencies were contacted and asked to determine if their role in the EA was that of a responsible authority or an expert federal authority (FA). In addition, the Ontario Ministry of the Environment was contacted to determine their role in the EA. In consideration of the CEAA 'Federal Co-ordination Regulations', the following federal departments were identified as federal authorities for the purpose of providing expert assistance during the assessment: Health Canada, Environment Canada, Natural Resources Canada, and Fisheries and Oceans Canada. There were no provincial EA requirements under the *Ontario Environmental Assessment Act* that were applicable to the project. Once the EA was underway, the Ontario Ministry of Natural Resources was provided with information on the various EA studies.

Expert federal authorities have specialist knowledge that can be applied to a project. The expertise of an FA can be used during any stage of an EA, but most often the FAs are called upon to review the EA and provide their input on the validity of the assessment. If a proponent (or delegate) has a good relationship with the FAs, it is likely that they can predict the issues that a FA would find most relevant, and address these issues accordingly in the EA. A good relationship with the FAs enables the proponent to discuss issues with the FAs directly, thereby expediting the comment disposition process.

Golder has working relationships with the EA and project officers at the CNSC, and staff at Environment Canada and Fisheries and Oceans Canada, which have been built through many years of working in cooperation on federal nuclear EAs under the CEAA. These relationships allowed for open communication between Golder and the agencies, providing confidence that the EA studies and the EA Study Report could meet agency expectations.

In order to create or maintain relationships with regulatory agencies for future nuclear EA projects, a proponent should open lines of communication via federal and provincial regulator workshops. These workshops could focus on past EAs conducted for projects on the proponent's site(s), implementation of these projects demonstrating good environmental performance and diligence, lessons learned from past EAs, and proponent operating performance.

6. Managing stakeholder communications

Public participation is not mandatory for screening EAs conducted under the CEAA. It is the responsibility of the responsible authority to determine if public consultation is appropriate. The decision on public consultation may depend on the level of public interest in the project, the potential of the project to cause environmental effects, the potential for gaining local or traditional knowledge, and whether the project is perceived as controversial.

The purpose of public participation is to share information with the public and gather local knowledge and input on public concerns, ultimately improving the EA and promoting good decision making on the part of the responsible authority. For the purposes of EA, the public may be defined as local residents, community groups, local businesses, provincial and municipal governments, non-governmental organizations (NGOs), local aboriginal groups, etc., which could be potentially affected by the project.

The EA guidelines approved by the CNSC as the FA, and issued for the Bruce A Refurbishment Project [4] required that the EA include notification of, and consultation with, potentially affected stakeholders including the local public. As part of the EA, Bruce Power carried out community and stakeholder consultation activities, which were significantly more extensive than those required by the CEAA or the Canadian Environmental Assessment Agency. Public and stakeholder consultation activities carried out during the EA process included:

- Notification letters for upcoming events such as workshops and open houses;
- Newsletters detailing the progress of the EA;
- Community updates;
- Radio and newspaper advertising for open houses;
- Open houses comprising display panels, information handouts, and presentations;
- Stakeholder briefings and presentations targeting specific stakeholder groups;
- An EA consultation workshop;
- Consultation with government agencies including a workshop and meeting;
- Public library repositories containing EA information and materials;
- Project website containing EA information and materials;
- E-mail consultation comprising email addresses set up for the public to contact Bruce Power about the project, and submit questions and comments;
- A toll-free information line; and

- Bruce Power employee communications through internal publications and displays.

The local First Nations and Métis communities were contacted to determine their desire to participate in the EA process. Information was provided in the form of letters detailing the EA process, invitations to open houses and workshops, and a presentation to the Joint Council.

The open houses held in the smaller communities lend to the sense of community ownership. Through the experience with the EA for the Bruce A Refurbishment Project, it was determined that in order to improve the management of stakeholder communications for future nuclear EA projects, a proponent should:

- Continue to hold open houses in smaller communities to maintain the community sense of ownership;
- Better identify key groups in each of the surrounding communities;
- Solicit invitations from community groups and organizations to hold information sessions;
- Participate in existing community events or co-ordinate open houses with other community events to take advantage of public attendance;
- Develop a First Nations communication plan in co-ordination with the First Nations to improve engagement;
- Develop an NGO communication plan and hold an NGO workshop to improve engagement;
- Avoid over-saturating the public with information, which can lead to lower open house attendance;
- Delay conducting radio interviews until after open houses to encourage the public to attend the open houses instead of only obtaining their information from a radio broadcast;
- Develop publicly accessible project displays to be shown for the duration of the EA or project; and
- Potentially bring EA/project information into local schools.

In some recent EAs, the CNSC staff have themselves conducted some consultation activities in the local community in addition to monitoring the proponent's activities. This move to greater participation in consultation by the CNSC provides the opportunity for enhancing the overall consultation on the project.

The current approach to consultation often results in a process that is too focused on the proponent as advocate of the project and fails to acknowledge that conduct of the EA, though delegated to the proponent as allowed under Section 17(1) of the CEAA, remains the responsibility of the CNSC as responsible authority. CNSC staff approves the proponent's consultation and communication plan at the outset of the EA process and monitors consultation throughout the EA (including attending open houses) but generally plays a minor role in the consultation itself.

Recently, CNSC staff has been directed by the CNSC tribunal to conduct open houses during the EA guidelines review process and prior to the CNSC tribunal decision hearing. At best, these open house events have been limited in terms of participation and input from the community, likely because of the extent of public communications carried out by the proponent within the same time frames. The goal of this independent CNSC consultation appears to be to confirm that the proponent's consultation process has been fair and credible. This two-tiered approach to consultation has come about because of the desire for CNSC staff to remain independent from the proponent, despite the fact that they remain responsible for making recommendations to the CNSC tribunal regarding the decision on the project.

Communication and consultation throughout the EA process could be significantly improved through a stronger community-based focus by the CNSC in partnership with the proponent. Since the EA studies have been delegated to the proponent and the EA guidelines are issued by the CNSC, it is reasonable that CNSC staff participate fully in the communication and consultation of progress of the EA and results of the EA studies.

Specific measures to improve communication and consultation could include:

- Joint design and implementation by the CNSC and proponent of all community communication and consultation events;
- Demonstration, through shared communication activities such as newsletters and project updates, that the CNSC and the proponent are working toward a common goal (i.e., completion of an EA that fully meets the requirements of the EA guidelines) and are not involved in an adversarial process; and
- Enhancement of community accessibility and user friendliness by more extensive use of high-end web sites which provide easy access to information and coming events. The proponents often have highly developed communication vehicles available, know the community that they work in, and know how to communicate effectively with stakeholders. These vehicles should be adapted for joint use by the CNSC and proponent.

Community focused consultation, conducted jointly by the CNSC and the proponent, would ensure that information on the EA process and EA studies is made available to the community. Joint participation explicitly recognises that the proponent best knows the community and that both regulator and proponent are focused on ensuring a transparent EA involving the community throughout. Finally, it would remove the public perception of an adversarial process and favour consensus.

7. References

[1] Bruce Power LP. *Bruce A Refurbishment for Life Extension and Continued Operations Environmental Assessment Study Report.* December 7, 2005.

[2] Canadian Nuclear Safety Commission (CNSC). *Record of Proceedings, Including Reasons for Decision.* July 5, 2006.

[3] Canadian Environmental Assessment Agency. *Screenings under the Canadian Environmental Assessment Act – Participant Manual.* 2006.

[4] Canadian Nuclear Safety Commission (CNSC). *Screening Report on Environmental Assessment of the Bruce A Refurbishment for Life Extension and Continued Operations project, Bruce A Nuclear Generating Station, Kincardine, Ontario.* March 2006.

[5] Bruce Power LP. *New Fuel Project for Bruce B Environmental Assessment Study Report.* October 22, 2004.

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1456 Rivermont Avenue, Lynchburg, VA 24503 USA
Phone: 434 847-4925 | Fax: 434 847-0885
grae@energysolutions.com

United States | Canada | United Kingdom

The Results From the Second High-Pressure Melt Ejection Test Completed in the Molten Fuel Moderator Interaction Facility at Chalk River Laboratories

by T. Nitheanandan, G. Kyle, and R. O'Connor[1]

Ed. Note: The following paper was presented at the International Congress on Advances in Nuclear Power Plants held in Nice, France, 13-18 May, 2007. This paper is significant because it confirms the non-existence of a vapour explosion following a large melt injection to the moderator.

Ed. Additional Note: A similar paper was published in the December 2006 edition of the Bulletin, but it did not contain the "definitive result". The present paper contains much of the same descriptions, so we apologize for the overlap.

Abstract

The Canadian nuclear power generation industry, represented by the CANDU Owners Group (COG), is funding an experimental program at Chalk River Laboratories to study the interaction between the molten material ejected from the fuel channel and the moderator. These experiments are designed to address one of the very low probability postulated accident events considered for CANDU® Pressurized Heavy Water Reactors (PHWRs), where an array of fuel channels contain the nuclear fuel and high-temperature, high-pressure coolant. Under severely restricted flow blockage conditions postulated in a fuel channel, the temperature excursion could result in fuel melting, consequential failure of the fuel channel, and ejection of the molten fuel at high pressures into the heavy water moderator at near atmospheric pressure. The objective of the experimental program is to demonstrate that a highly energetic Molten Fuel Moderator Interaction (MFMI) and associated high-pressure pulse can be ruled out.

The second high-pressure melt ejection test using 22 kg of prototypical corium was completed recently at Chalk River Laboratories. The second test consisted of heating a thermite mixture of U, U_3O_8, Zr, and CrO_3, simulating the molten material expected in a fuel channel, inside a 1 m length of insulated pressure tube. Once the molten material reached the desired temperature of ~2400°C, the molten material was ejected into the surrounding tank of 63°C water. At the time of melt ejection, the static pressure in the test section was 3.35 MPa. The confinement vessel pressure reached a peak value of 201 kPa following the rupture of the test section. The peak dynamic pressure measured on the inner vessel walls ranged between 0.7 MPa and 1 MPa. The dynamic pressure history, debris size, and the effects of the material interacting with tubes representing neighbouring fuel channels were investigated.

I. Introduction

A limited number of melt-water interaction experiments have been funded by the Canadian nuclear power generation industry to address one of the very low probability postulated accident events considered for CANDU®[2] Pressurized Heavy Water Reactors (Fig.1). A description of the postulated event is provided in Reference 1. In these experiments, molten corium is ejected under high-pressure differential through a "fish mouth" opening of the pressure tube, finely fragmenting the melt particles. The melt particles transfer the energy to the water as it is dispersed, creating a modest pressure pulse in the vessel with peak pressure lower than the driving pressure differential in the pressure tube. This method of melt-water interaction is called the forced interaction mode.

In preparation for these tests, a chemical mixture called a thermite, that produces a simulated molten fuel when ignited, was developed [2]. The thermite mixture consisted of U, U_3O_8, Zr, and CrO_3, designed to be representative of the chemical constituents of molten material expected in a CANDU fuel channel [2]. Following the thermite development, two base-case reference tests [3] and two high-pressure melt ejection tests using prototypical corium were completed in the Molten-Fuel Moderator-Interaction (MFMI) facility at Chalk River Laboratories.

For the base reference cases, pressure tubes with a machined flaw were pressurized to ~10 MPa, using a mixture of steam and helium, and ruptured under water to obtain a reference base to be compared against the melt ejection tests. These reference tests helped interpret the results of melt ejection tests, and study the effect of molten corium interaction with water, over and above the effect of the shock wave created by the pressure-tube rupture in the absence of corium. Following the base-case reference tests, two melt ejection tests were conducted. The first melt ejection test, completed with ~5 kg of molten corium, was reported in Reference [4].

The objectives of this paper are to present the results obtained from the second melt ejection test completed in the MFMI facility using 22 kg of corium.

1 Chalk River Laboratories, Atomic Energy of Canada Limited, Chalk River, Ontario, Canada, K0J 1J0

2 CANDU – CANadian DeUterium

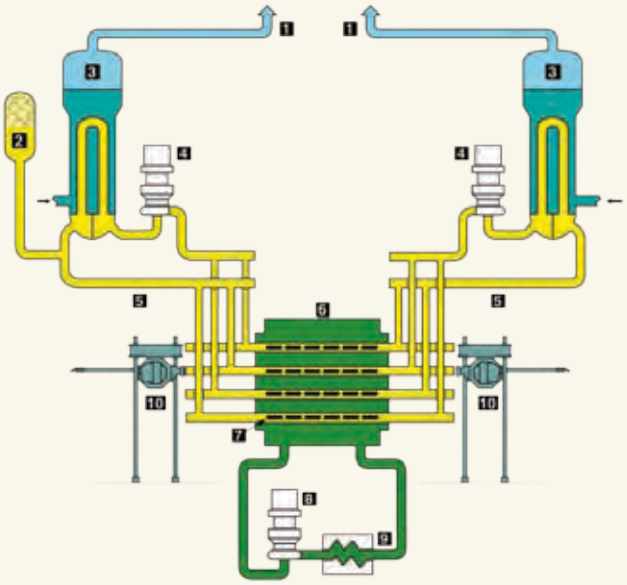

Fig.1. A schematic of a CANDU primary heat transport system (Legend: 1. Steam line leading to electric turbines, 2. Pressurizer, 3. Steam generator, 4. Pumps, 5. Inlet headers, 6. Calandria vessel, 7. Fuel channel, 8. Moderator recirculation pump, 9. Heat exchanger, and 10. Online refueling machines).

II. Test Apparatus and Facility

The test section is a 1.14-m long section of Zr-2.5 Nb pressure tube (0.1038-m ID and 0.0043-m thick wall) placed concentrically inside a 1.04-m long, 0.13-m ID, and 3-mm thick quartz tube. The quartz tube insulates the pressure tube from the surrounding water. The pressure-tube/quartz-tube assembly is submerged in 63ºC water at a depth of 1.4 m. Two end hubs, attached to the ends of the pressure tube with an O-ring seal between, have penetrations for two pressure transducers, a thermite fill port, two ignition wires, CO_2 gas inlet and outlet ports, steam-injection lines, and two Type-C thermocouples. A schematic of the assembled test section with the penetrations is shown in Fig.2. Four Type-K thermocouples were spot welded on the outside surface of the pressure tube along the bottom to measure the wall temperature.

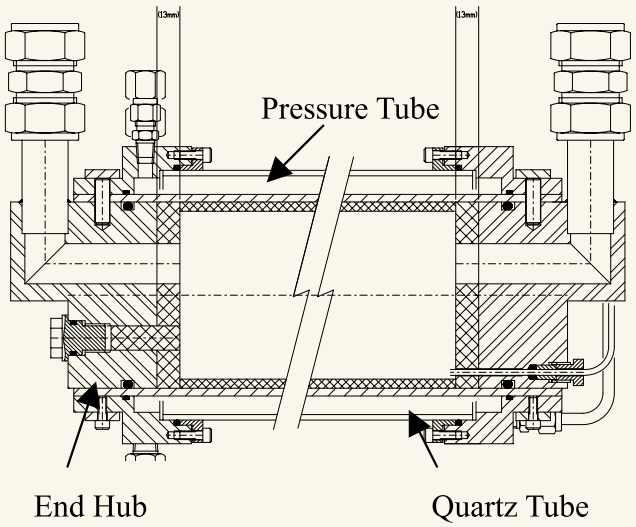

Fig.2. A schematic of the test section.

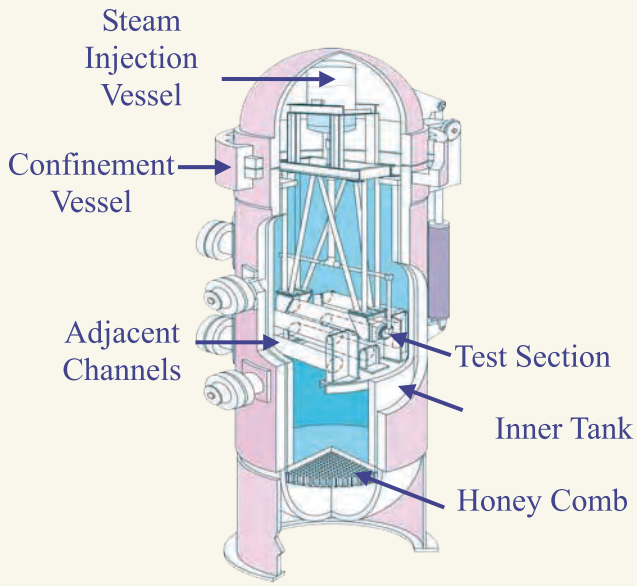

Fig.3. A schematic of the steam-injection vessel, confinement vessel, inner tank and the test section.

The pressure-tube/quartz-tube assembly was bolted to a stainless steel frame that provided structural support to the steam-injection vessel, the steam-injection line, and the quick-acting

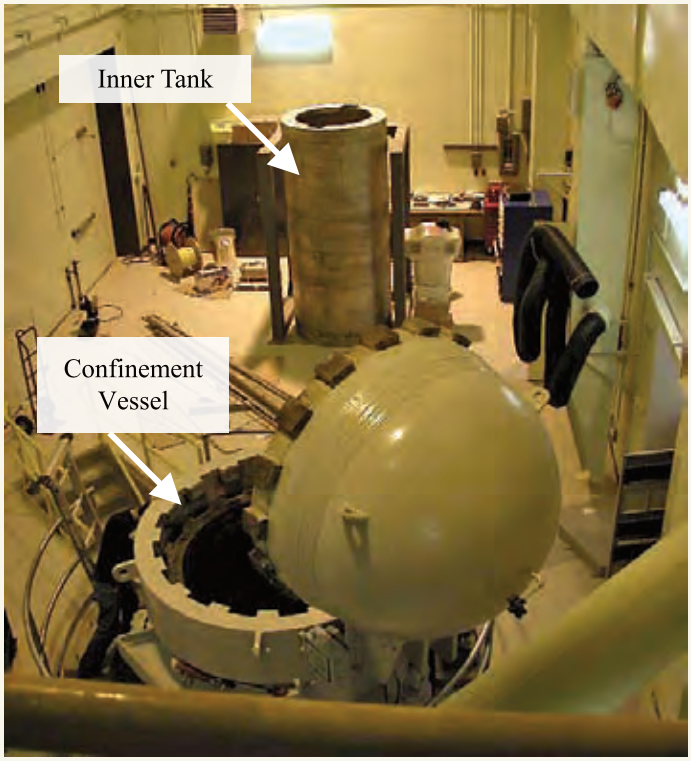

Fig.4. A photographic view of the inner tank and confinement vessels in the MFMI facility at Chalk River.

ball valves. A schematic of the structural frame supporting the test section and the steam-injection vessel is shown in Fig.3.

A V-shaped groove was machined on the outside surface of the pressure tube to weaken a section of the wall at the 6 o'clock position to ensure a predictable rupture at a defined location. The defect is a 450-mm long section with a 60° groove, leaving a wall thickness of 1.03 to 1.06 mm at the weakest section.

II.A. Confinement and Auxiliary Systems

The confinement for the MFMI experiment consists of an outer confinement vessel, and an inner tank. The outer vessel is cylindrical, 1.5 m diameter and 5.0 m high (Figs.3 and 4). The top of the confinement vessel is hinged providing full diameter access to the vessel interior. The bottom of the confinement vessel is filled with concrete and an energy absorbing 38-mm thick honeycomb sheet is placed on top of the concrete to absorb part of the pressure pulse from the pressure tube rupture and distribute the load evenly over the bottom head of the vessel. The confinement vessel has flanged connections for process and instrumentation feed-throughs. The lines from the pre-heated water tank, the helium supply lines, and steam vent lines penetrate the confinement tank via these feed-throughs in the flanges. A vent line off the top of the confinement vessel, the vacuum line at the bottom of the vessel and the rupture disc placed in one of the flanges are available to depressurize the vessel through the room ventilation system fitted with HEPA filters.

Inside the confinement vessel is the inner tank (Figs.3 and 4), which holds water (simulating the moderator), instrumentation, simulated adjacent fuel channels, and the structure supporting the test section. It is an open top tank which has an inside diameter of 1.25 m and a height of 2.9 m. Instrument sensors for level, pressure, and temperature measurement are located at various locations inside the tank. The locations of the pressure transducers are shown in Fig.5. The concrete filled 0.132 m diameter fuel channels were placed at a 0.286 m square pitch (Fig.3).

The steam-injection vessel, designed to provide ~17 L of saturated steam at 10 MPa, is connected to the test section via a steam-injection line that consists of two quick-acting ball valves (Fig.3). The piping throughout the steam-injection system is nominally 38 mm diameter and connected in parallel below the quick-acting ball valves to each end of the pressure tube. The purpose of the steam-injection vessel and piping is to deliver steam to the test section just prior to pressure tube rupture, simulating the coolant flow to the channel.

II.B. Data Acquisition and Instrumentation

The data acquisition system (DAS) configuration included both high-speed sampling and low-speed sampling during the test. The low-speed sampling of data occurs at 10 Hz. The high-speed system could acquire 28 channels simultaneously at a rate of 100 kHz.

Eight piezo-electric pressure transducers were used to monitor the pressure within the water and on the walls of the inner tank. The dynamic range of four transducers was 0 to 35 MPa and the dynamic range of the remaining four was 0 to 69 MPa. The sensors had a 0.69 kPa resolution, ≥ 500 kHz resonance frequency, and a ≤ 1 μs rise time.

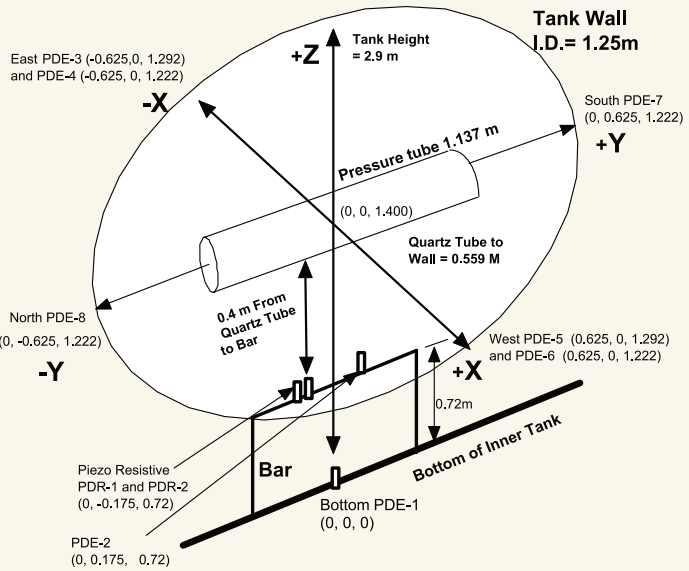

Fig.5. A schematic of the piezo-electric and piezo-resistive dynamic pressure transducer locations in the inner tank.

Two piezo-resistive pressure transducers with 0 to 34 MPa range were also used. These transducers have a resonance frequency > 1 MHz. The piezo-resistive pressure transducers were placed on a rigid bar, 0.35 m apart from each other and 0.4 m below the test section as shown in Fig.5.

Three precision quartz shear accelerometers of ±100g, ±250g, and ±500g were used on the inner tank with the ability to measure accelerations of up to 500 times the gravitational acceleration. Six strain gauges were used, 3 on the inner tank and 3 on the confinement vessel. In addition, six standard resistance temperature detectors (RTDs) are used to monitor the water temperature at various elevations within the inner tank. These RTDs have a maximum operating temperature of 300°C.

During the MFMI test, a series of sequential operations were performed remotely using a Programmable Logic Controller (PLC). The entire test sequence was divided into six stages. The PLC is programmed to execute a set of instructions when stage switches were activated. These stages represent discrete operations such as pumping water into the inner tank, pre-pressurization of the pressure tube, injecting steam into the pressure tube, etc. A time delay between instructions had also been pre-programmed into the PLC. The stage switches provide operational flexibility during the test.

III. Experimental Procedure

The following is an overview of the test procedure followed during the test:

1. Mix and load the thermite into the pressure tube.
2. Connect the test section to the steam injection vessel, place the apparatus into the inner tank, and close the confinement vessel lid.
3. Evacuate the confinement vessel three times and back fill with helium to reduce the oxygen concentration in the vessel.
4. Turn the power on to the moderator heating system and

heat the water to 86°C.

5. Turn the power on to the steam injection vessel.
6. When steam injection vessel temperature reaches 265°C, turn the low speed DAS on and flip Stage 1 switch to pump water into the inner tank.
7. Pump turns off automatically on high-level float switch.
8. When steam injection tank water reaches 305°C, turn on high-speed data acquisition system.
9. When steam-injection vessel temperature reaches 310°C, turn on the master switch, which would enable automatic thermite ignition by the PLC.
10. Flip Stage 2 switch to pressurize the steam injection vessel to 10 MPa using helium gas and energize the ignitor power supply to charge the capacitors and maintain on standby mode.
11. Flip Stage 3 switch to supply nitrogen gas to energize quick acting ball valves, maintain them on standby mode.
12. Flip Stage 4 switch to ignite the thermite using PLC.
13. When the gas pressure inside the pressure tube exceeds 0.3 MPa, PLC automatically increases the pressure-tube pressure to 10 MPa and after 1.7 s delay opens the quick acting ball valves to direct 10 MPa saturated steam into the pressure tube.
14. If ball valve does not open automatically, activate Stage 5 switch to direct 10 MPa saturated steam into the pressure tube.
15. If pressure tube does not rupture within 30 s, activate Stage 6 switch to automatically bring the bump system at 13 MPa helium gas pressure to the pressure tube via the steam injection vessel.
16. If pressure tube does not rupture with the bump system for 30 s, turn Stage Manual to vent the pressure from steam injection, pressure tube and the confinement vessel.

IV. Experimental Results

The results obtained from the 22 kg corium melt ejection test are described in this section.

During the test, thermite was ignited and the corium temperature reached ~2470°C. The pressure tube pressurized to 4.7 MPa[3] due to thermite ignition. The pressure tube melted through 51 ms before steam entered the pressure tube. Steam was injected into the pressure tube ~3.4 s after the pressure in the pressure tube exceeded 0.3 MPa. The pressure tube melted through when the pressure-tube pressure was 3.35 MPa and the average inner tank water temperature was 63°C. When the pressure-tube melt-through occurred, the peak shock wave pressure measured on the inner tank wall ranged between 0.74 MPa and 1.0 MPa.

IV.A. Experimental Uncertainty

The measurement uncertainties were estimated using standard techniques recommended for estimating experimental uncertainties [5]. A summary of the instrument uncertainties, including the DAS uncertainties, at the maximum measured values in the 22 kg corium test is given here.

- Pressure-tube thermocouples (Type K) at 600°C ±4.7°C
- Corium temperature using Type-C thermocouples ±50°C[4]
- Steam injection vessel temperature at 317°C ±3.3°C
- Static pressure (pressure tube) at 10 MPa ±0.21 MPa
- Tank water temperature using RTDs at 63°C ±2.3°C
- Dynamic pressure at 1.0 MPa (Piezo-electric) ±0.06 MPa
- Dynamic pressure at 1.0 MPa (Piezo-resistive) ±0.19 MPa
- Vessel strain $\pm 7.2 \times 10^{-4}$ %
- Accelerometers 5880 m/s^3 (600g) ±58.8 m/s^3

IV.B. Steam and Water Temperature

The Stage 1 switch was turned on at time = 251.4 s and the moderator water, pre-heated to 86°C and stored in the insulated storage tank, was pumped to the inner tank. The water pump took ~18 minutes to pump 3100 L of water into the inner tank. The response of the RTDs placed at different elevations in the inner tank is shown in Fig.6. As shown in the figure, the RTD temperatures started at 21.6°C (measuring ambient air temperature) when the water pump was turned on. When water encountered the first RTD, it began to record a sharp increase in temperature reaching 64°C. A similar trend was shown by the other three RTDs indicating the water filling into the tank. The water temperature cooled down gradually to an average of 63°C just before pressure-tube rupture. At time = 2084.4 s, all four RTDs recorded a temperature between 62.9°C and 63.4°C. The barometric pressure at the time of the test was 99.6 kPa. The saturation temperature and the subcooling at the mid plane of the pressure tube, including the height of water, were 103.1°C and 40°C, respectively.

As water was pumped into the inner tank, the vacuum pump was started to evacuate the confinement vessel when the confinement vessel pressure exceeded 20 kPa (Fig.7). This operation was required to keep the tank from pressurizing, as water was pumped into the tank, compressing the air trapped in the confinement

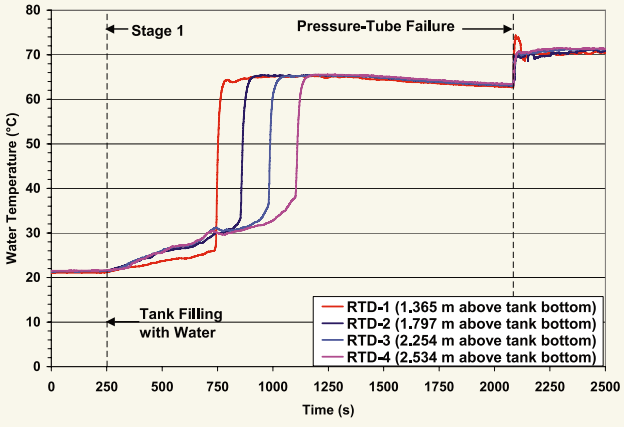

Fig.6. The Measured Water Temperatures Inside the Inner Tank During the Second Corium Test with 22 kg Melt

3 The measured pressures reported in this document are in gauge pressures.

4 The uncertainty stated here was estimated at 2470°C and reflects only the estimated calibration uncertainties. Actual measurement uncertainties, depending on the environment where the measurement is made, can be higher.

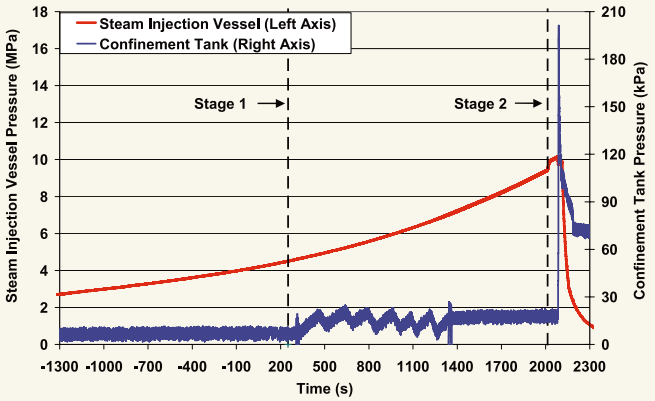

Fig.7. The Measured Confinement Vessel Pressure, Steam Injection-Vessel Pressure, and the Quick-Acting Ball Valve Position During the Second Corium Test with 22 kg Melt.

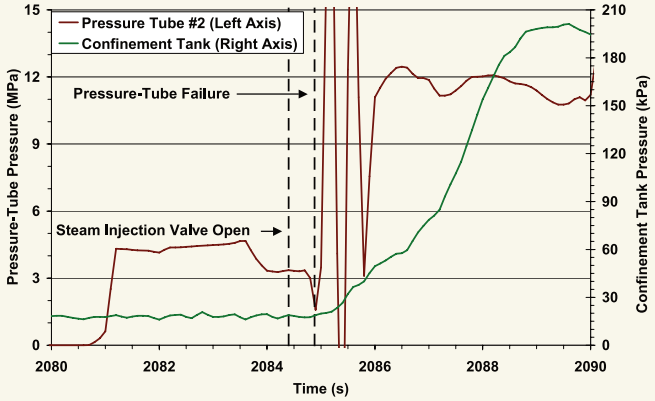

Fig.8. The Pressure Transients in the Pressure Tube and the Confinement Vessel

vessel. The confinement vessel pressure exceeded 20 kPa six times and each time the vacuum pump was turned on to evacuate the confinement vessel to bring the pressure down to ~15 kPa.

Once the steam injection vessel pressure and temperature reached 9.5 MPa and 316°C, respectively, the Stage 2 switch was flipped (time = 2013.8 s) as shown in Fig.7. In this stage, the steam-injection vessel pressure was increased to 10 MPa using helium gas. The steam-injection vessel pressure gradually increased to the target pressure of 10 MPa (Fig.7). In Stage 3, the nitrogen gas supply to the quick acting ball valves was also turned on by the PLC (time = 2041.2 s). The steam pressure and temperature just prior to pressure tube failure were 10 MPa and 317°C, respectively.

IV.C. Thermite Ignition and Pressure-Tube Rupture

Once the steam-injection vessel pressure and temperature reached the target values, the Stage 4 switch was flipped to ignite the thermite at time = 2056.2 s. In Stage 4, the current to a nichrome wire circuit was supplied to ignite the thermite. When thermite ignited, the pressure inside the pressure tube reached 4.7 MPa due to the non-condensable gases released during thermite reaction (Fig.8). The ideal gas law calculation performed prior to the test indicated the peak pressure-tube pressure following thermite ignition to be 4.8 MPa. The pressure switch connected to the pressure tube closed when the pressure-tube pressure exceeded 0.3 MPa, triggering the PLC to open the quick acting ball valves with a 1.7 s delay. The time when the quick acting ball valves opened is shown in Fig.8.

The time when thermite ignited was determined from the surge in the ignitor current. The time when the increase in the current flow occurred was selected as the thermite ignition time. The time taken for the thermite to ignite since flipping Stage 4 switch was 23 s.

There were two Type-C thermocouples in the test section; however, only the tungsten-sheathed thermocouple responded to the aggressive corium environment. The second thermocouple failed as soon as thermite ignited. About 2.2 s elapsed between the application of power to the ignitor and the thermocouple responding to the temperature escalation. The initial response of the tungsten sheathed Type-C thermocouples was an overshoot beyond the output range of the thermocouple. Once the overshoot disappeared the thermocouple responded to the corium temperature quite satisfactorily. The corium temperature was between 2259°C and 2869°C during the measured period (except the initial overshoot) with an average of approximately 2470°C.

Following thermite ignition and activation of the pressure switch at 2081 s, the PLC opened the helium supply to pressurize the pressure tube to 10 MPa. Then the quick acting ball valves on the steam injection line were expected to open after a timed 1.7 s delay. The time difference between the activation of the pressure switch and the quick acting ball valves opening was 3.4 s, which is twice as long as expected. The quick acting ball valves opened at 2084.4 s.

The peak pressure in the test section, measured by transducer #2, was 4.7 MPa (Fig.8). The peak pressure was reached within 3 s of first indication of pressure rise. The measured pressure decreased to 3.3 MPa after reaching the maximum value within 0.5 s. The pressure then remained steady for about 1 s before the pressure-tube ruptured at 2084.881 s.

The pressure-tube rupture time was estimated from measured data using five different instruments. A photographic view of the pressure-tube rupture is shown in Fig.9. The first set of instruments is the RTDs measuring the tank water temperature. The thermal response of the water in the inner tank to the ejected melt from the pressure tube was used to estimate the rupture time. All

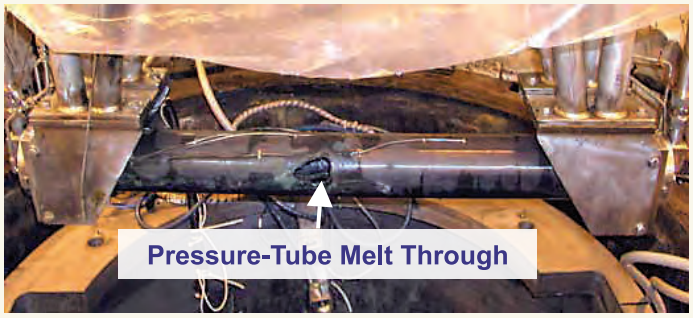

Fig.9. The Post-Test View of the Ruptured Pressure Tube

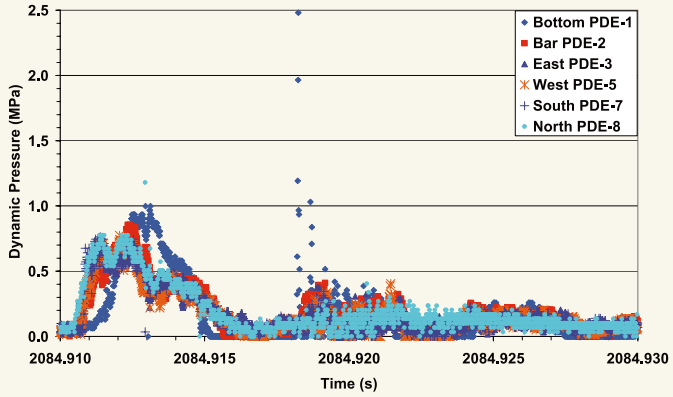

Fig.10. The Dynamic Pressure Measured by the High Speed DAS in the Inner Tank Following Pressure-Tube Rupture

four RTDs responded to a temperature increase at 2085.4 s, which is the pressure-tube rupture time indicated by the water temperature. The second set of instruments used to determine the pressure tube rupture were the thermocouples placed on the outside surface of the pressure tube. The thermocouples indicated an abrupt escalation in temperature at time = 2085.4 s, which was coincident with the temperature escalation indicated by the RTDs. The third set of instruments was the confinement vessel pressure which started to increase at 2084.9 s. This is 0.5 s earlier than the time indicated by the RTDs and the pressure-tube outside-surface thermocouples. The fourth set of instruments that indicated the pressure-tube rupture was the accelerometer response to melt ejection. The inner tank acceleration suddenly increased from zero to 189g at 2084.881 s. This is very close to the rupture time indicated by the confinement vessel pressure. The fifth set of instruments used to indicate pressure-tube rupture is the dynamic pressure transducers. The dynamic pressure transducers started to register an increase in pressure at 2084.9105 s, which was 29.1 ms later than the rupture indicated by the accelerometers. The time difference of 29.1 ms, between the increase in dynamic pressure and pressure-tube rupture, was in the same order of magnitude as in non-corium tests (1 - 15 ms) and 5-kg corium test (30.9 ms) [4]. Therefore, the pressure-tube rupture time was estimated to be at 2084.881 s.

Once the pressure-tube rupture time is established, the pressure-tube pressure at the time of rupture can be determined from Fig.8. At the time of rupture, the pressure-tube pressure was 3.35 MPa. Within 0.4 s of rupture, the pressure increased to >10 MPa. Since the first hole from the melt through was relatively small and the supply of helium and steam at 10 MPa continued, the pressure-tube pressure is likely to have reached >10 MPa. The pressure history (Fig.8) indicates that when the helium gas line was opened by the PLC to pressurize the pressure tube (as soon as pressure switch was activated), helium gas did not enter the tube. Had helium gas entered the tube, pressure in the tube should have approached 10 MPa. Then the quick acting ball valves were opened by the PLC and yet, the pressure did not increase in the pressure tube. This is possible only if the steam-injection-port plugs[5] were stuck in the end hub. During post-test examination, the steam-injection-port plug on end hub #2 was found to be stuck.

Once the pressure tube ruptured, the melt quench generated steam and the non-condensable gases released from the pressure tube increased the confinement vessel pressure to a maximum of 201.2 kPa within 4.7 s (Fig.8). The confinement vessel pressure dropped gradually to a steady pressure of 69 kPa, about 310 s after reaching peak pressure (Fig.7). The difference between the peak confinement-vessel pressure and the steady-state confinement vessel pressure was the drop in vapour pressure due to steam condensing on the vessel walls.

IV.D. Dynamic Shock Wave Pressure

The dynamic pressure on the inner tank water increased 29.1 ms after pressure-tube rupture. The dynamic pressure history recorded by the transducers is shown in Fig.10. The dynamic pressure wave reached the bar at 2084.91071 s. The bar was placed 0.4 m below the quartz tube. The dynamic pressure wave reached the bottom of the inner tank at 2084.911296 s. The distance between the bar and the bottom transducers was 0.72 m. Considering the distance and the time (i.e., 2084.911296 – 2084.91071 = 0.586 ms), the pressure wave speed is calculated as 1229 m/s. This is in good agreement with the wave velocity calculated in the non-corium and 5-kg corium tests (1074 m/s to 1220 m/s).

One data point in the measured dynamic pressure at the bottom of the vessel (PDE-1) indicated a peak pressure of 12.2 MPa (not shown in Fig.10) approximately 6 ms after the dynamic pressure reached the transducer. Since the period between successive data points is 1/100,000th of a second, the energy represented by a single data point is insignificant (~0.122 kPa.s impulse). Such spurious data points, from a piezo-electric transducer, are consistent with electronic noise. Since only one transducer indicated 12.2 MPa, while none of the other transducers responded at that time, it is considered electronic noise in that transducer.

The peak pressure measured on the east-side wall of the inner tank (Fig.10) was 0.78 MPa. The peak pressures measured on the west-side wall of the inner tank was 0.78 MPa. Similarly, the peak pressures measured on the north and the south walls were 0.78 MPa and 0.76 MPa, respectively. The transducer on the bar indicated a peak pressure of 0.85 MPa. The dynamic pressure transducer at the bottom of the inner tank recorded a peak dynamic pressure of 1.0 MPa. The dynamic pressure wave period was between 5.2 ms and 6.0 ms.

All of the dynamic pressure measured showed a "hump" or irregularity on the exponential decay of the peak pressure. Similar behaviour was observed in the 5-kg corium test [4]. Although the origin of these humps has not been conclusively determined, it is presumed that they are due to internal reflections of the melt interacting with water at the gas bubble interface. Although not conclusive, the thermite ignition and ejection is similar to underwater explosions and thus the humps appearing in these measurements may have a similarity to the humps observed in the underwater explosions [6].

5 The steam-injection-port plugs were placed to prevent non-condensable gas, produced during thermite ignition, from entering the steam injection line.

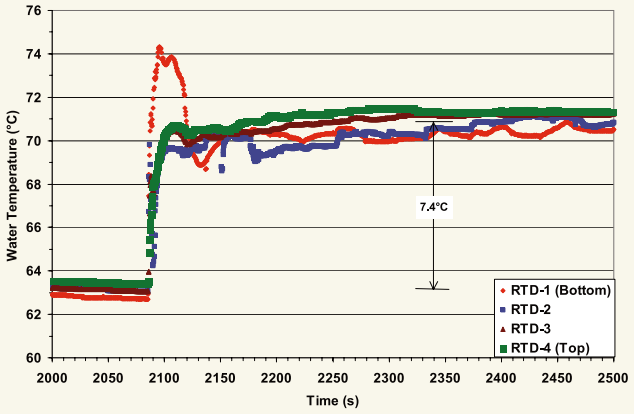

Fig.11. The Water Temperature Reaching a Quasi-Steady State Following Melt Ejection.

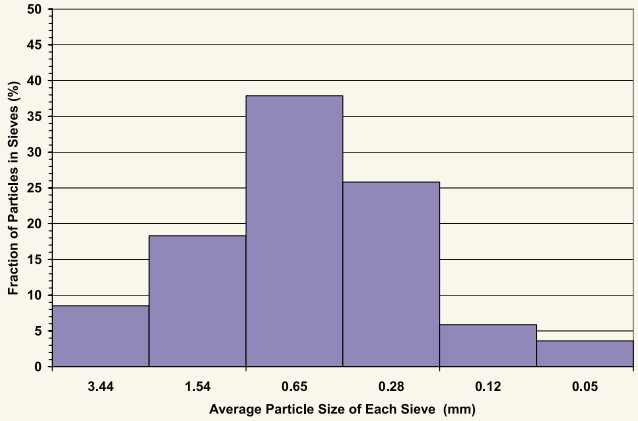

Fig.12. The Debris Particle Size Distribution Obtained From the Second Corium Test with 22 kg Melt.

IV.E. Long-Term Debris Cooling

The melt cooling is determined from water temperature. Fig.11 shows the water temperature for up to 1.14 h after melt ejection. The tank water reaches a quasi-steady state temperature at around 2300 s, approximately 215 s after pressure-tube rupture and melt ejection. The steady water temperature is indicative of establishing a balance between heat loss from the tank and heat gained from the melt debris. The time where the steady temperature is reached is the end of the quenching process. After the steady state, the bottom RTD cools rapidly compared to the top RTDs. The bottom RTD is cooled by 5.5°C whereas the top most RTDs are cooled only by 3.5°C in the same period.

Fig.11 also shows the average temperature increase of the inner tank water. The temperature increase was calculated using the average water temperature immediately before pressure-tube rupture and the average temperature when the quasi-steady state water temperature was established after rupture. Since the quasi-steady-state indicates the end of melt cooling, the calculated average temperature increase can be used to assess the energy gained by water during cooling. Using 3100 L of water and 22 kg of thermite, the enthalpy transferred per kilogram of thermite is calculated to be 4.3 MJ/kg (= 7.4°C x 3100 L/1000 L/m^3 x 4.2 kJ/(kg·K)x 980 kg/m^3/22 kg). Within this amount of energy in the water the steam injection vessel at 10 MPa and 317°C would have added 1.9 MJ/kg (= [2759.4 kJ/kg – 264.1 kJ/kg] x 17 kg/22 kg). The values within the square brackets represent the vapour enthalpy at 10 MPa and 317°C and liquid enthalpy at 63°C, respectively. When the energy added by the steam is subtracted, the energy from the melt is calculated to be 2.4 MJ/kg, which is very close to the adiabatic melt energy [2].

IV.F. Post-Test Hydrogen Measurement and Debris Characterization

A 150 mL gas sample bomb was filled with the confinement-vessel gas inventory immediately after the test and the sample was analysed. The gas indicated that there was 17.6% hydrogen.

A post-test debris analysis was carried out using a set of sieves after the obvious "non-corium" debris was removed. The obvious non-corium debris is quartz and Zircar. A view of the equipment used for particle size analysis is shown in Fig.12. The entire corium mass of 22 kg could not be recovered as some amount of corium deposited on the dummy channels, Zircar discs, and tungsten cups could not be removed and accurately estimated. The corium mass lost in all these components is estimated to be 200 g. Approximately 1.17 kg of corium remained inside of the test section[6]. This corium was not used in the particle size analysis.

The total debris collected was 22.65 kg, while the thermite used in the experiment was 22 kg. It is difficult to separate the particles below 1 mm into quartz, Zircar, tantalum and corium owing to the fact that the debris in the experiment contained quartz, Zircar, and tantalum besides corium. As such, the particle size distribution presented in Fig.12 should be used with caution.

The mean particle size was calculated on a weight basis using the geometric mean of the diameter openings in two adjacent sieves in the stack [7]. The mean particle size was then calculated using the following equation:

$$d_{gw} = \log^{-1}\left[\frac{\Sigma(W_i \log d_i)}{\Sigma W_i}\right]. \qquad (1)$$

In this equation, d_{gw} is the mean particle size (mm), W_i is the weight of particles (kg), and d_i is the logarithmic mean of two adjacent sieves sizes (mm) calculated using $d_i = (d_u \times d_o)^{0.5}$, where d_u is the diameter opening through which particles will pass (sieve proceeding ith sieve) and d_o is the diameter opening through which particles will not pass (ith sieve). The calculated mean diameter of the debris particles using Equation (1) was 0.581 mm.

A random sample of the debris particles were obtained from 2.36 mm, 1 mm, 425 µm, 180 µm, 75 µm, and 38 µm size sieves for SEM[7] and EDX[8] analysis. Fig.13 shows the particles magnified fifty times. The corium particles were black in colour and of different sizes and shapes. The > 1 mm diameter particles were mostly flat pieces and those < 1 mm were mostly spherical and long irregularly shaped particles. The particle shown in Fig.13 was spherical and had somewhat smooth surface. There were few pores on the sphere especially at the top right hand corner. The appearance of the surface is similar to a volcanic surface with holes

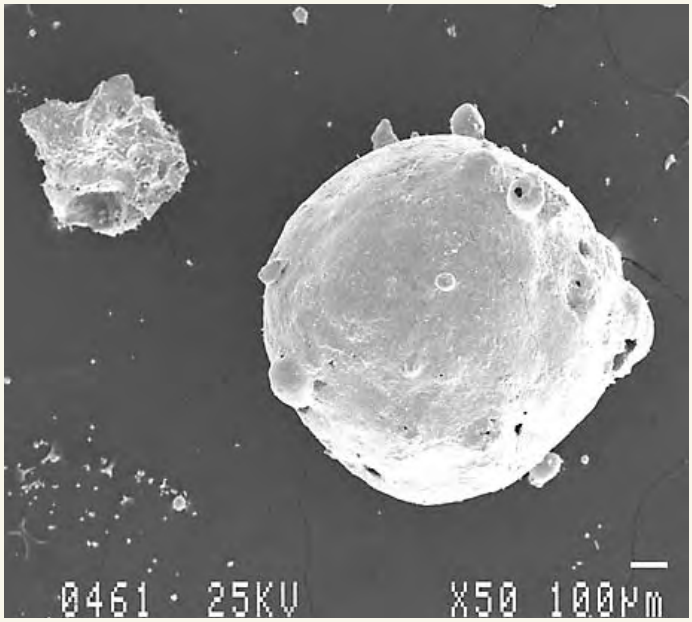

Fig.13. A Random Sample of Debris Particles From the Second Corium Test with 22 kg Melt Magnified Fifty Times.

similar to geysers in a volcano resulting from gas ebullition.

Similar surface morphology was reported in the corium particles obtained from TROI-36 test [8]. This experiment used 0.7 UO_2 and 0.3 ZrO_2 (wt%) mixture. In comparison, the corium composition expected from the MFMI experiments is 0.73 UO_2/0.11 Zr/0.06 ZrO_2/0.10 Cr (wt%). The TROI-36 test produced 1.42% of hydrogen and 40.8% of the particles from the test were below 0.425 mm in diameter. The holes observed in the corium particles in the TROI-36 test were attributed to geysers resulting from hydrogen production.

A significant amount of particles showed areas of smooth surface and some areas with precipitated material deposited on the surface. The surface also gives an appearance of a frozen puddle of melt, which is indicative of high-temperature melt quench.

An EDX analysis of the particles was completed to assess the corium composition. The EDX results performed on the material indicated that the material in the particles consisted of about 0.5725 U, 0.2413 Zr, 0.0294 Cr, 0.0964 O (wt%). The remaining were impurities such as Mg, Al, Cu, and Si. The Mg, Al, Cu, and Si impurities in this sample may have come from the ignitor wire, thermocouple leads and the mineral insulation. The target corium composition is 0.73 UO_2/0.11 Zr/0.06 ZrO_2/0.10 Cr (wt%).

IV.G. Impulse From Measured Dynamic Pressure Transients

The specific impulse at the inner tank wall can be calculated by integrating the measured dynamic pressure transients over the time duration of the pressure transient. The numerical relationship used in obtaining the specific impulse from measured dynamic pressure is given in Equation (2):

$$I = \sum_{i=1}^{n} P_i \times \Delta T_i \,, \qquad (2)$$

where I is the specific impulse (kPa·s), i index for the data points, n is the total number of data points within the pulse width, and ΔT is the time interval between two data points. There are 100 data points per milli-second. The time durations of the measured first pressure transient in the non-corium and corium tests were approximately 3 ms and 5 ms, respectively. Therefore, the specific impulses for the non-corium tests were calculated on the basis of 3 ms time duration, whereas for the corium tests 5 ms time duration was used in the calculation.

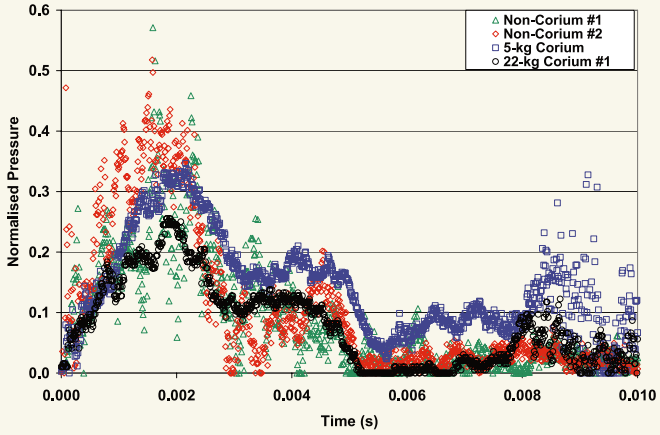

Fig.14. A Comparison of Measured Dynamic Pressures From Non-Corium and Corium Tests. The details of the non-corium and 5-kg corium test are provided in Reference 3 and 4, respectively.

The specific impulses from the 22 kg corium test ranged between 2.1 and 2.4 kPa·s. The average specific impulse calculated for the 22 kg corium test was 2.2 kPa·s, which is lower than the average specific impulse of 10.4 kPa·s calculated for the 5-kg corium test due to the lower pressure-tube pressure during melt ejection.

The impulses from various tests can be compared directly provided they are normalized to their rupture pressure. The normalized impulse (I_N), represented in milli-seconds, is obtained by integrating the normalized dynamic pressure (P_i/P_{PT}) over the time duration of the pressure transient as:

$$I_N = \sum_{i=1}^{n} \left(\frac{P_i}{P_{PT}} \right) \times \Delta T_i \,, \qquad (3)$$

where P_{PT} is the pressure-tube pressure at rupture. The normalized impulse can be assumed a period when the measured dynamic pressure, provided as a ratio of pressure-tube pressure, is applied on the inner tank walls. The range of normalized impulses in the 5-kg corium test is between 0.48 ms and 1.04 ms, which is comparable to the range of normalized impulse (0.62 ms to 0.72 ms) in the first 22-kg corium test.

A comparison of the normalized dynamic pressure transients

6 The debris mass inside the test section consisted mainly of corium smeared Zircar insulation. An estimate of the amount of corium was made based on the corium layer thickness.

7 Scanning Electron Microscope

8 Energy Dispersive X-ray

obtained from non-corium and corium tests are shown in Fig.14. The pressures in the figure have been normalized to the pressure-tube pressure before rupture. For the 22 kg corium test, the measured dynamic pressure was normalized using 3.35 MPa, i.e., the pressure-tube pressure at the time of rupture. An examination of Fig.14 reveals that the peak dynamic pressure characteristics of the two corium tests are similar and indicates the reproducibility of corium tests. The measured peak dynamic pressures between the corium tests and the non-corium tests are similar; however, the pulse widths of the corium tests are wider compared with the non-corium tests. Because the measured peak dynamic pressure in the inner tank water was less than the driving pressure, it can be concluded that there was no steam explosion when 22 kg of corium was ejected into water from a pressure tube at 3.35 MPa driving pressure. In addition, the average particle size from the test (0.581 mm) being less than 1 mm demonstrates that at the end of the melt ejection process the particles were not large enough to indicate that there was a potential for steam explosion.

V. Summary

The second high-pressure corium ejection test using 22 kg of corium was completed in the MFMI facility at AECL's Chalk River Laboratories. The test consisted of target calandria tubes, placed beneath and on the sides of the test section, to simulate a three by two matrix of the reactor core. The steam injection vessel was pressurized with 17 L of water to 10 MPa and 317°C. The 3100 L of inner tank water was at an average temperature of 63°C during pressure-tube rupture. The pressure tube, loaded with 22 kg of thermite, was ignited and the corium temperatures reached ~2470°C. After thermite ignition, the pressure tube was pressurized to a maximum of 4.7 MPa by non-condensable gases released from the thermite. The pressure in the test section at the time of failure was 3.35 MPa. The peak dynamic pressure measured on the inner vessel walls ranged between 0.74 MPa and 1.0 MPa. The peak dynamic pressure measured 0.4 m below the test section was 0.85 MPa, while the peak dynamic pressure at the bottom of the vessel was 1.0 MPa.

The total debris collected inside the inner tank was 22.65 kg. The debris inside the inner tank consists of corium, quartz, and Zircar. Although quartz and Zircar particles interfered with the particle size analysis, the debris from the test showed that the majority of the corium particles were less than 1 mm in diameter. The mean size of the debris was calculated to be 0.581 mm. An analysis of the confinement vessel gas inventory indicated 17.6% hydrogen. Based on the measured peak dynamic pressure transient in the inner tank water and the debris particle size, it can be concluded that there was no steam explosion when 22 kg of corium at 3.35 MPa pressure was ejected into 63°C water.

Acknowledgments

The work reported in this paper was funded by the CANDU Owners Group R&D Safety and Licensing Program, Work Package No. 21302 under the joint participation of AECL, HQ, NBP, OPG, and BP.

The contributions of the following past and present members of the CANDU Owners Group working group on Fuel and Fuel Channel Safety are gratefully acknowledged: M. Ali (OPG), M.H. Bayoumi (NSS), P. Chan (BP), F. Doria (AECL), R.A. Gibb, S. Girgis (AECL), F. Iglesias (BP), J.H.K. Lau (AECL), G. Parent (HQ), Y. Parlatan (OPG), D. Radford (AECL), M. Richards, L. Sun (NBP), and N. Wahba (NSS).

Nomenclature

AECL	Atomic Energy of Canada Limited
BP	Bruce Power
CANDU®-PHWR	CANadian Deuterium Uranium-Pressurized Heavy Water Reactor
DAS	Data Acquisition System
HEPA	High-Efficiency Particulate Air filter
HQ	Hydro Quebec
ID	Inside Diameter
MFMI	Molten-Fuel Moderator-Interaction
NBP	New Brunswick Power
NSS	Nuclear Safety Solution
OPG	Ontario Power Generation
PLC	Programmable Logic Controllers
R&D	Research and Development
RTD	Resistance Temperature Device

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Royal Military College of Canada Department of Chemistry and Chemical Engineering

Faculty Position in Nuclear Engineering

The Department of Chemistry and Chemical Engineering at the Royal Military College of Canada in Kingston, Ontario, invites applications for a tenure track position at the level of assistant professor or at a more senior level for an appropriately qualified candidate. The Department desires applicants with a recent PhD, or one nearing completion, in Nuclear Engineering or a closely related field, who would be able to teach graduate courses in Nuclear Engineering and courses in the undergraduate Chemical Engineering program. The candidate must be able to establish and maintain an independent and competitively funded creative research program.

The Department is involved in research funded by national granting councils, other government agencies and industrial partnerships at a level of $19.7M in 2006/07. More information on the Department can be found at http://www.rmc.ca/academic/chem/index_e.html. Current activity in the nuclear field includes: nuclear fuel engineering, nuclear fuel management, nuclear reactor materials, novel techniques of radiation dosimetry, reactor instrumentation and control, nuclear reactor design, non-destructive testing, neutron activation analysis and radiochemistry, nuclear counter-terrorism, radiological contamination and remediation, and nuclear waste management. The Department operates a SLOWPOKE-2 nuclear research reactor.

Industrial or postdoctoral experience in the nuclear field is desirable. Candidates with expertise in the nuclear fuel area are especially welcomed. The ability to read, comprehend and communicate orally in both official languages will be considered an asset. The candidate is expected to supervise MASc, MSc and PhD graduate students and professional/technical research staff.

Kingston is a beautiful historic city on the shores of Lake Ontario, easily accessible from Toronto, Ottawa and Montreal, at the mouth of the St. Lawrence River and at the start of the Thousand Islands, with diverse and eclectic offerings of music, art, history, theatre and live entertainment that create a unique cultural environment. Kingston has been designated as an official bilingual city by the Government of Ontario.

For an Assistant Professor, the salary is up to $84,515 per annum plus an annual Terminable Allowance of $3,300 for 2006/07. The starting salary will be in accordance with professional qualifications and experience. The expected starting time for the position is the spring of 2008. The closing date for the competition is 15 December 2007. The Royal Military College of Canada is a coeducational and bilingual institution, and this position is offered equally to women and men. In accordance with the Public Service Employment Act, preference must be given to Canadian citizens, however applicants from outside Canada will be considered. Candidates with foreign educational credentials are required to provide proof of Canadian equivalency. You may consult the Canadian Information Centre for International Credentials at http://www.cmec.ca/cicic/ for further information.

Interested candidates should forward their *curriculum vitae* showing their qualifications for the position, including a statement of current citizenship status, proof of education, research activity with selected publications and relevant teaching experience, in addition to the names of three references to:

Mrs. Doris Meade
Civilian Human Resources Office (Kingston)
Canadian Forces Base Kingston
P.O. Box 17000, Station Forces
11 Mercury Cres., Kingston, Ontario K7K 7B4
Phone: (613) 541-5010, ext 2218
Fax: (613) 541-4496
Email: Meade.DM@forces.gc.ca

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Le Département de chimie et de génie chimique du Collège militaire royal du Canada invite les candidatures à un poste menant à la permanence au niveau de professeur adjoint ou de professeur de niveaux supérieurs par des candidats ou candidates dûment qualifiés. Le Département désire des candidats et candidates ayant récemment obtenu un diplôme de doctorat, ou sur le point de terminer leur programme de doctorat, en génie nucléaire, ou dans un domaine très connexe, et qui sont capables d'enseigner des cours de niveaux supérieurs en génie nucléaire ainsi que des cours de premier cycle en génie chimique. Le candidat ou la candidate doit être capable d'établir et de maintenir un programme de recherche innovateur et pouvant recevoir du financement via les programmes d'octrois compétitifs.

Le Département est impliqué dans des programmes de recherche financés par les organismes nationaux d'octroi de recherche et par des partenariats industriels à un niveau de $19.7M en 2006/07. On pourra trouver plus de renseignements sur le Département au site internet suivant : http://www.rmc.ca/academic/chem/index_f.html. Les domaines de recherche présentement actifs en génie nucléaire incluent les suivants: génie des combustibles nucléaires, gestion du combustible nucléaire, matériaux des réacteurs nucléaires, techniques innovatrices en dosimétrie des rayonnements, instrumentation et contrôle des réacteurs, désign de réacteurs nucléaires, essais non-destructeurs, analyse par activation neutronique et radiochimie, contre-terrorisme nucléaire, contamination et restauration radiologique, et gestion des déchets nucléaires. Le Département exploite un réacteur nucléaire de recherche SLOWPOKE-2.

De l'expérience industrielle ou postdoctorale dans le domaine du nucléaire est souhaitable. Les candidats avec de l'expertise dans le domaine du combustible nucléaire sont spécialement recherchés. La capacité de lire, comprendre et communiquer oralement dans les deux langues officielles sera considérée comme un avantage. Le candidat ou la candidate doit s'attendre à avoir à diriger des étudiants de cycles supérieurs en MScA, MSc ou au PhD, en plus de superviser des membres du personnel de recherche professionnel/technique.

Kingston est une splendide ville historique sur les rives du lac Ontario, d'accès facile de Toronto, Ottawa et Montréal, à la décharge du lac Ontario dans le fleuve Saint-Laurent et au début des Mille Îles, et est dotée d'un ensemble éclectique d'activités en musique, art, histoire, théâtre et spectacles divers, ce qui crée un environnement culturel unique. Kingston a été désignée comme ville officiellement bilingue par le Gouvernement de l'Ontario.

Au niveau de Professeur Adjoint, le salaire annuel maximum est de $84,515, en plus d'une allocation terminable annuelle qui était de $3,300 pour 2006/07. Le salaire initial sera selon les qualifications professionnelles et l'expérience. On s'attend à ce que l'emploi débute au printemps de 2008. La date limite pour soumettre les candidatures pour cette compétition est le 15 décembre 2007. Le Collège militaire royal du Canada est une institution bilingue dont les étudiants sont des femmes et des hommes, et la présente offre d'emploi s'adresse aussi bien aux femmes qu'aux hommes. Selon les règlements de l'Acte de l'Emploi de la Fonction Publique, on doit accorder la préférence aux Citoyens canadiens, cependant, on considérera les candidatures provenant de l'extérieur du Canada. Les candidats dont les diplômes et les relevés de notes proviennent de pays étrangers doivent fournir des preuves d'équivalence canadienne. Vous pouvez consulter le Centre d'Information Canadien sur les Diplômes Internationaux à l'adresse internet suivante: http://www.cmec.ca/cicic/ pour obtenir de plus amples informations.

Les candidats et candidates intéressés doivent envoyer leur *curriculum vitae* montrant leurs qualifications pour le poste, incluant un énoncé du statut de citoyenneté présente, des preuves de l'éducation académique reçue, des activités de recherche avec une sélection des publications et de l'expérience en enseignement pertinent, en plus des noms de trois personnes pouvant servir de références, à:

Mme Doris Meade
Bureau des Ressources Humaines Civiles (Kingston)
Base des Forces Canadiennes de Kingston
C.P. 17000, Station Forces
11 Mercury Cres., Kingston, Ontario K7K 7B4
Téléphone: (613) 541-5010, poste 2218
Fax: (613) 541-4496
Courriel: Meade.DM@forces.gc.ca

GENERAL news

(Compiled by Fred Boyd from open sources)

OPA issues plan for Ontario's electricity

On August 29, 2007, the Ontario Power Authority issued its Integrated Power System Plan (IPSP) which it describes as the first comprehensive review and plan for the electricity system in the province in 15 years.

The Integrated Power System Plan (IPSP) presents a prioritized implementation plan that responds to the Ontario government policy direction conveyed through the Supply Mix Directive of June 13, 2006.

The IPSP is based on a 20-year forecast, but is subject to updating and regulatory review every three years; this will enable the Ontario Power Authority (OPA) to revise the mid- and long-term plans based on near-term accomplishments and experiences.

In its June 2006 directive, the government stipulated that the province's electricity requirements should be met from the following resources in this priority order:

1. Conservation
2. Renewable resources
3. Nuclear power for remaining baseload requirements
4. Gas-fired generation for peaking, high-value and high-efficiency uses
5. Coal-fired generation replaced by cleaner sources in the earliest practical time frame

   (The government later determined through regulation that coal phase-out will occur by the end of 2014 at the latest)

The directive also indicated that the transmission system is to be strengthened to meet these goals.

The plan has five defining features:

1. Reduce demand by conservation
2. Replace coal by gas and renewable sources
3. Restore nuclear capability for base load by refurbishment or new build
4. Develop flexibility from gas and purchase options
5. Implement transmission that is necessary for reliability, incorporation of desired generation and system efficiency.

It calls for 45 percent of Ontario's electricity supply to be provided by conservation and renewable sources by 2025 (!!), eight percent from natural gas and 47 percent from nuclear power.

OPA states that the IPSP will offset any growth in peak demand arising from increased population and economic growth for next 10 years or more, but that Ontario will need to build almost as much generating capacity over the next 20 years as there is in the province today.

The Plan places its highest priority on conservation, which OPA states has priority over any form of new supply. OPA predicts that 6,300 MW of total peak demand will be reduced through conservation by 2025. The OPA will seek to identify and develop conservation opportunities that exceed the 2010 and 2025 goals. One-sixth of the projected budget – more than $10 billion – is to be spent on conservation alone.

A baseload gap of 85 TWh is predicted, to be met by either [new or refurbished] nuclear or combined cycle gas generation. However, the report states that OPA does not intend to procure any [additional] nuclear supply before 2010. The Plan calls for 15,000MW of new generation by 2015, 6,434 MW of which will specifically be to replace existing coal generation capacity.

The total cost is predicted to be roughly $60 billion, of which $26.5 billion is slated for nuclear.

In preparing the IPSP, OPA pursued a process of "engagement" with the public. Eight "critical-issue" discussion papers were issued, web-conferences and in-person meetings were held involving 2,200 residents of the province, 161 written submissions were reviewed, and 134 First Nations were contacted.

AECL Signs MoU with China and Argentina

On September 4, 2007, Atomic Energy of Canada Limited (AECL) announced that it had signed a Memorandum of Understanding (MOU) with the China National Nuclear Corporation (CNNC) and Nucleoeléctrica Argentina S.A. (NASA), to conduct a joint study of the potential for cooperation in design, manufacture, construction and operation of CANDU nuclear power plants on future projects in Argentina, Canada and China.

CNNC and NASA have also agreed to strengthen cooperation in sharing and exchanging their CANDU 6 nuclear power plant operational and maintenance experience. CNNC's Third Qinshan Nuclear Power Plant began full operation in 2003, while NASA's Embalse CANDU plant in Argentina has performed excellently for more than 24 years.

The signing ceremony in China was attended by Daniel Cameron, Argentina's Secretary of Energy, Eduardo Messi, President of NASA, Kang Rixin, President of CNNC, and Ken Petrunik, Chief Operating Officer of AECL.

During the meeting, the Argentinean delegation presented their country's active nuclear program, including the recent agreement between NASA and AECL to enter into commercial negotiations to build a new reactor similar to the Qinshan Phase III design.

Kang Rixin, of CNNC, China's most comprehensive and largest nuclear power corporation, said CNNC is interested in exploring cooperative efforts regarding various aspects of future CANDU projects, such as the new build in Argentina.

Argentina has two operating reactors in the country: Atucha-1, a 300 megawatt (MWe) net pressure heavy water reactor (PHWR) of the pressure vessel type, supplied by Germany and in operation since 1974; and Embalse, a CANDU 6 power reactor designed and supplied by AECL that was connected to the grid in 1983. NASA has implemented a proactive nuclear program, including plans to extend the life of the Embalse CANDU 6 power plant, build a new CANDU station, and complete Atucha-2, a second heavy water reactor of 700 MWe, originally supplied by Germany.

New CNSC regulatory documents

In August 2007, the Canadian Nuclear Safety Commission released two further regulatory documents:

Regulatory Standard S-210, *Maintenance Programs for Nuclear Power Plants*

This regulatory standard sets out the requirements for maintenance programs that licensees of nuclear power plants must implement. The CNSC stated that comments on a draft version that had been issued in April 2006 were taken into account in the final document.

Regulatory Guide G-323, *Ensuring the Presence of Sufficient Qualified Staff at Class 1 Nuclear Facilities – Minimum Staff Complement*

The CNSC states that the purpose of this regulatory guide is to assist licensees of Class 1 nuclear facilities and applicants to demonstrate to the commission that they will ensure the presence of a sufficient number of qualified workers to carry on the licensed activity safely and in accordance with the *Nuclear Safety and Control Act,* the regulations made under the Act and their licence.

Comments on a draft version, issued in October 2005 were taken into account in finalizing the document.

Both documents are available in either English or French on the CNSC website: www.nuclearsafety.gc.ca. Paper copies can be requested: e-mail: info@cnsc-ccsn.gc.ca

ANS names award after W. B. Lewis

The American Nuclear Society (ANS) has created the W. Bennett Lewis Award, "to honour individuals who make major lifetime contributions in nuclear science and engineering towards minimizing environmental footprint, attaining long-term global sustainable energy and development, and having shown great foresight in elucidating these goals as recorded in archival publications."

Dr. W. B. Lewis was vice-president of Atomic Energy of Canada Limited from 1946 to 1973 and headed the research and development that led to the CANDU design of nuclear power plants.

The ANS has another award named after a Canadian pioneer – the George C. Laurence award for outstanding contributions toward nuclear reactor safety. George Laurence was a senior director at AECL and became the first full-time president of the Atomic Energy Control Board (AECB), the predecessor of the Canadian Nuclear Safety Commission, in 1962. He led the development of the reactor safety concepts that were applied to all of the existing Canadian nuclear power plants.

Application for nuclear plants in Alberta

In late August 2007, Energy Alberta Corporation filed an application with the Canadian Nuclear Safety Commission (CNSC) for a licence to prepare a site for four CANDU ACR 1000 nuclear power plants near Peace River in northern Alberta.

The CNSC has pointed out that the first step in the regulatory process is an Environmental Assessment with which it has extensive experience. The CNSC states that it will work closely with the Canadian Environmental Assessment Agency and other federal and provincial agencies to ensure an effective and efficient EA process that follows the requirements of the Canadian Environmental Assessment Act (CEAA).

Energy Alberta Corporation president, Wayne Henuset, said his company plans to start with one two-unit plant with a target date for start-up of 2017.

Most of the output will be used for the extraction of hydrocarbons from the complex structures of the area, with the remaining power available for the Alberta electricity grid.

Bruce Power to refurbish Unit A-4

Bruce Power added another component to its large refurbishing project when, on August 29, 2007, it announced that it would refurbish Unit 4 of its Bruce A station .

Bruce Power will now replace all 480 fuel channels in Unit 4, extending its operational life until 2036. Under the original 2005 agreement with the Ontario Power Authority (OPA), Bruce Power intended to install new steam generators in all four Bruce A units and fuel channels in Units 1, 2 and 3. Without new fuel

channels, Unit 4 was only expected to run until 2017.

Under the revised plan, Bruce Power expects to invest an additional $1 billion, resulting in a total investment in the restart and refurbishment program of approximately $5.25 billion.

Bruce Power expects to complete the work on Units 3 and 4 by 2013. Under the revised agreement, the OPA can elect for a three unit restart program if it expects there will be insufficient transmission to accommodate all eight Bruce Power units by mid 2013. This election expires on April 1, 2008. In April, Hydro One announced plans to upgrade the transmission line between Bruce Power and Milton, Ont. and expects the new line to be in service by 2012.

Meanwhile, work to restart Units 1 and 2 continues and is expected to deliver an additional 1,500 megawatts to the Ontario power grid by late 2009 or early 2010. For stories and video clips on the entire project, go to the Bruce A Restart web pages at the Bruce Power website www.brucepower.com.

Cameco reports on Cigar Lake

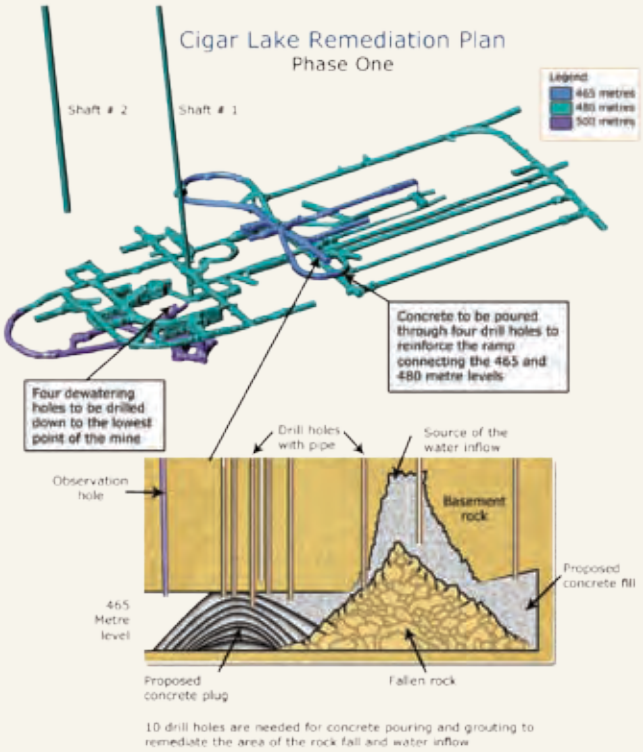
Cameco Corporation continues to make progress on its phased plan to restore the Cigar Lake project after a water inflow on October 23, 2006 flooded the underground development.

The first phase of the remediation plan involved drilling holes down to the source of the inflow and to a nearby tunnel where reinforcement is needed, pumping concrete through the drill holes, sealing off the inflow with grout and drilling dewatering holes. Subsequent phases for remediation include dewatering the mine, ground freezing in the area of the inflow, restoring underground areas and resumption of mine development. Regulatory approval is required for each phase of the remediation plan.

Reinforcement of the adjacent tunnel is now complete, and all of the holes required for pouring concrete to seal off the inflow have been drilled. Drilling of four larger-diameter holes required for dewatering is 90% complete.

Regulatory agencies have approved plans to flush sand and fine material away from the inflow area and to pour the concrete plug. The concrete mixture is designed to harden under water and will be poured in successive layers. If the concrete solidifies as planned, it should prevent or reduce water inflows sufficiently to enable mine dewatering. The effectiveness of the plug will not be known until dewatering is under way.

The second phase of remediation includes dewatering the underground development, verifying that the water inflow has been sufficiently sealed, and installing the surface freezing piping.



Cameco is now working to provide regulators with the information needed to secure approval for installation of dewatering pumps and infrastructure, and ongoing operation of water treatment facilities required for dewatering. An application will be made to the Canadian Nuclear Safety Commission for extension of the Cigar Lake construction licence that expires at the end of 2007.

Completion of the second phase had been expected by the end of the third quarter of 2007. Cameco now expects it will require a number of additional months to seal the inflow and dewater the mine.

There are about 285 people on site working on remediation and construction of surface facilities including the access road, piping infrastructure, load-out building and water treatment facilities.

Bruce Power orders steam generators for Unit 3

In early August 2007 Bruce Power signed a contract with Babcock & Wilcox (B&W) Canada for eight replacement steam generators for Unit 3 at the Bruce A generating station at a cost of more than $90 million.

The new steam generators will be engineered and built at B & W Canada's Cambridge facility and installed in Unit 3 as part of the ongoing, $4.25 billion Bruce A Restart and Refurbishment project.

In 2005, Bruce Power and B & W Canada signed a similar agreement for the manufacture of 16 replacement steam generators for Bruce A Units 1 and 2. To date, four of those generators have been installed in Unit 2.

B & W Canada, which built the original Bruce A steam generators more than 30 years ago, will manufacture the replacement vessels using Alloy 800 tubes and design enhancements learned from years of supplying replacement vessels for the U.S. and Canadian markets. Each of the new steam generators will weigh more than 100 tonnes and stand approximately 12 metres high.

Algae problems at OPG's Pickering B

During the hot spells of early August, Ontario Power Generation's Pickering B Generating Station was forced to reduce its electrical output as a result of algae blockage in the station's water intake systems.

The hot weather had resulted in an increased growth of algae in Lake Ontario. Wind and wave conditions increased the flow of algae, causing a larger than normal amount of algae to enter the station's water intake systems.

The algae accumulated on screens and filters, which temporarily reduces the station's ability to draw in the large volumes of water needed for cooling the steam in the station's turbine condensers; part of the conventional or non-nuclear side of the station.

Pickering Unit 5 was taken off line but was re-connected within a few days after the algae was cleaned away and normal water flow re-established. This condition was unrelated to the earlier maintenance outage of Unit 5.

Zircatec celebrates 50th anniversary

On Saturday September 8, 2007, Zircatec Precision Industries (now a subsidiary of Cameco Corporation) held an Open House to celebrate the 50th anniversary of the beginning of its fuel manufacturing plant in Port Hope, Ontario.

The plant was built by American Machine and Foundry in 1957 to research and develop fuels for Canada's nuclear energy program, which, at that time, was centred at the Chalk River Laboratories of Atomic Energy of Canada Limited. However, the design of the NPD demonstration nuclear power plant was underway.

In 1964, the plant was acquired by Westinghouse Canada Ltd. and the following year began commercial production of fuel bundles for CANDU nuclear power plants. In 1970 a sister plant was built in nearby Cobourg to manufacture reactor components.

A group of Canadian investors purchased the business in 1988 and renamed it Zircatec Precision Industries.

Two years ago, the uranium mining and refining company, Cameco Corporation, purchased the business.

Earlier this year Zircatec received a five-year operating licence from the Canadian Nuclear Safety Commission and is now going through the process to obtain approval to manufacture fuel bundles with slightly enriched uranium.

Because of security restrictions (set by the CNSC) the celebration was limited to families of employees and a few special guests.

Obituary

Ross Campbell

Ross Campbell, a former chairman and acting president of Atomic Energy of Canada Limited died August 15, 2007 at the age of 88.

Born and educated in Toronto, Campbell obtained a law degree in 1940 at the age of 21. He then joined the Royal Canadian Navy but was transferred to the Royal Navy because the RCN was not yet ready to train recruits. He served with the RN, first in the English Channel, then in the Mediterranean, and back to the English Channel for the invasion of 1944.

After the Second World War he joined the Department of External Affairs (now Foreign Affairs Canada). After assignments in Norway, Denmark and Turkey he became head of the Middle East division in Ottawa and later Assistant Under-Secretary. Subsequently, he was appointed ambassador to Yugoslavia and successively to Algeria, NATO, Korea and Japan.

After retiring from External Affairs in 1975 he was appointed Chairman of Atomic Energy of Canada Limited and, for a period, also served as acting President. That was a difficult time for AECL as a parliamentary committee began investigating the contract for the Wolsong 1 reactor in Korea. He survived that and led the company in its sales to Argentina and Romania.

Following his term as AECL chairman, he remained as a director of the company and was appointed president of AECL International, a subsidiary set up to sell CANDUs overseas.

Subsequently he ran his own consulting company in Ottawa.

Ross Campbell was an Officer of the Order of Canada and was among the first group of veterans named to the Veterans Hall of Fame.

He is survived by his wife, Pippa, and sons Hugh and Timothy.

Meet the President

To his role as president of the Canadian Nuclear Society for 2007–2008 **Eric Lyn Williams** brings extensive and varied experience from his many years in the Canadian nuclear power program and from his background in canoeing and other activities.

Born (on an undeclared date) in Toronto, Eric spent his early years in Burlington (about 40 km west of the big city). Showing early his athletic tendencies he played hockey (as a goalie) in the winter and participated in distance running in the summer. (He says he also tried figure skating but does not reveal how many quad jumps he achieved !!).

Before completing high school he joined the Royal Canadian Air Force in 1964 (it was still called the RCAF at that time) where he participated in the first solid-state electronics course offered by the force. Stationed near Sioux Lookout in northern Ontario as an Air Defence Technician, Eric continued his high school studies but also found time to (as he says) "court" (and marry) the Commanding Officer's daughter, Lynda Gay.

He was transferred to the NORAD base in North Bay, Ontario in 1967 to serve as a technician / underground controller for fighter aircraft and BOMARC missiles. Along the way he managed to complete high school (Ontario grade 13).

In the fall of 1968 Eric retired from the RCAF and enrolled in the co-op program at the University of Waterloo. That led to his initiation to nuclear power. Two of his co-op assignments were with Ontario Hydro stationed at the Babcock & Wilcox plant in Cambridge as an inspector for the Pickering A steam generators, while another was at OH's head office with Bruce A plant services. His wife, Lynda Gay, who had also enrolled at Waterloo, obtained her B.A. in 1972, despite raising son Brent who was born earlier that year. Eric obtained a B.A.Sc. in mechanical engineering in 1973.

Obtaining a position with Ontario Hydro on the construction of the Bruce A plant, he and his family moved to Kincardine in the fall of 1973. The following year he was named assistant Turbine Engineer, Bruce A construction, and daughter Larisa was born. In 1977 Eric was promoted to Bruce (A) Construction Scheduling Engineer, responsible for completing the construction planning.

He pursued postgraduate studies while working and obtained his M.A. Sc. in mechanical engineering in 1978.

Two years later, in 1980, Eric was loaned to Atomic Energy of Canada Limited to assist with the project planning of the Wolsong 1 unit and moved to Korea. The original short assignment stretched to 30 months and his responsibilities expanded.

In early 1983, with his task at Wolsong ended and Ontario Hydro Construction no longer needing him, Eric was offered a supervisor's position for the commissioning of Bruce B. That continued until the four units were turned over to Ontario Hydro Operations in 1988.

Eric then transferred to operations at Bruce A with the responsibility of reactor safety. That involved continuing contact with the regulator (then the Atomic Energy Control Board).

In 1993 he was assigned Bruce A Emergency Preparedness, responsible coordinating fire and medical response at the station and for liaison with local services. Five years later he moved to Bruce B to supervise the start-up of a full time Emergency Response Team, which subsequently had responsibility for the entire 2800 acre Bruce site.

After Bruce Power took over the Bruce A and B plants in 2001, Eric was assigned to the Bruce A units 3 and 4 restart program, with

Eric with grandchildren

Hitting the books in earlier times

Always working

Eric and Lynda Gay at the 2007 CNS Annual Conference

emphasis on fire protection. That continued until 2004 when he was appointed Senior Fire Protection Engineer for the Bruce site.

Eric retired from Bruce Power in August 2006, just in time, as 1st Vice-President of the Canadian Nuclear Society, to oversee the organization of the very successful 2007 CNS Annual Conference held in Saint John, New Brunswick.

Eric's non-nuclear activities have included a strong interest in canoeing, which he continues despite surgery on his knee. (Even his e-mail address is "canoe.about@bmts.com.) That involved several years as president of Paddle Canada and other national and provincial canoeing organizations.

He has also been a scout leader for over 20 years and an active member of the Anglican Church where he is still a lay reader, which involves conducting services from time to time. He has an extensive model train layout, which he claims is now primarily for his grandchildren.

As the CNS faces challenges of growth in the environment of a renewing nuclear program, Eric Williams brings extensive experience combined with still-youthful enthusiasm to the challenge of leading the Society over the coming months.

27 August 2007

"From Here To There" - The View From The CNS President's Seat

My personal thanks to each and every one of you for the very special opportunity you have afforded me to serve as the CNS President for the 2007 -8 year. My past few years on the CNS Council have been vision expanding, challenging, and have afforded me, a retired long term Operations type, the unique opportunity to work with a diverse cross section of knowledgeable and committed Canadian nuclear industry experts. And it has been enjoyable to say the least.

If you have not considered serving on the CNS Council, and or its related Divisions, Committees, Branches, and Conference Organizing Committees, I would urge you to reconsider. You won't regret it.

You asked the CNS Council to hold the CNS Annual Meeting in locations other than Toronto. We did, and you rewarded us with record attendance and support at the Saint John Annual Conference last June. While planning for the Toronto June 2008 Annual Conference is well in hand, I can share with you that Council has approved Calgary Alberta as the venue for the June 2009 Annual Conference. The long-term vision is for odd year conferences to be held at venues across Canada, with the even year ones in Toronto.

CNS President's have traditionally taken upon themselves a personal CNS enhancement project during their all too short term in office. Following on from the CNS's 2007 Annual Conference theme "Embracing the Future, Canada's Nuclear Renewal and Growth", I have chosen to focus my presidencies CNS improvement initiatives on the ***"Renewal and Growth of the CNS"*** itself. While membership has continued its upward trend over the past few years (and is now over 1140 members), recent CNS promoted conferences continue to grow and excel in all respects, and the CNS is in a very healthy situation in all respects; there are still many areas in which I think we can do better.

While our membership continues to grow, we still loose far too many new members after just a few short years of membership. The CNS's membership fees are very reasonable compared to like societies, and the CNS membership benefits rival the best. I do believe that the CNS could do better however in recognizing its' new members and making them feel more welcomed and part of the CNS team at an earlier stage. To that end new members will be contacted personally by a CNS Executive member or Division or Committee Chair within the first few months of membership to ensure the new member understands the benefits of their new membership, is linked to a segment of the CNS which reflects their interest in the industry, and is encouraged to get involved. This initiative will also help get more CNS members involved with the day-to-day workings of the CNS. The goal is to recognize the new member, and make them feel valued and wanted at an early stage.

While many of the new CNS members are younger professionals in our industry, this is not reflected on the CNS Council and its extended Council (the Divisions, Committees, and Branches) that do the majority of the work within the CNS. I will therefore be working to initiate both a student and young members position on the CNS Council itself, and on each and every Division, Committee, and Branch. I will also be endeavouring to garner support for this initiative from the many employers within our industry who will reap the benefits of such participation by their younger staff. It may take some time to get this fully operational throughout the CNS, but I herein challenge any students and young professionals reading this to contact me personally to express interest in taking a leadership role on any of the CNS's Divisions, Committees, and Branches. A word of explanation is in order. Most of these groups meet via email, telephone conference calls, and the odd face-to-face meeting usually associated with conferences and courses. I will encourage the CNS Division, Committee, and Branch Chairs to seize this opportunity and invigorate their organization with the hard working exuberance of youth. I will also encourage a motion at the 2008 CNS Annual Meeting to constitutionalize a youth (under 30 - 35) and student member on the CNS Council itself.

While the CNS Membership fees continue to be one of the best bargains in the nuclear industry today, for a new starting out member they can be a barrier to continuing membership with the CNS. Students can take advantage of free memberships up until the end of the calendar year in which they graduate with their first degree. My vision is for reduced CNS membership fees for the first few years of membership with the CNS.

These are just a few of the ideas I am pursuing on your behalf. This is on top of the continuing diverse industry backing program that the CNS fosters and supports.

If you have other ideas to enhance the operation and effectiveness of the CNS, please let me know. I would greatly appreciate hearing from you.

Eric L. Williams, P.Eng
canoe.about@bmts.com

CNS Membership Grows

Membership in the CNS has been increasing steadily over the years. As of 2007 August, the number of CNS members in good standing is 1141. The number of new members who joined in 2007 is 151. The graph below illustrates the steady growth in CNS membership since 1999.

Part of this growth can be attributed to the number of new hires into the Canadian nuclear industry in the last few years. However, membership was growing even before the nuclear renaissance, and I believe that this reflects a general recognition of the quality of CNS events and of the advantages of being a member. Some of the benefits accruing to CNS members are:

- Belong to a Professional Society with its mandate in the area of your professional career
- Take advantage of many excellent opportunities to grow professionally by meeting and networking with colleagues in Canada and internationally
- Take advantage of excellent opportunities to grow personally by volunteering, learning new things and new skills
- Receive quarterly CNS Bulletin
- Receive earliest notification of CNS Courses and Conferences, Branch Seminars and other Programs
- Receive early notices by e-mail of many other items of interest
- Receive CNS Membership Certificate, yearly sticker and membership card
- Receive special member registration fees to CNS Conferences and Courses
- At Conferences organized by another Society (e.g., ANS) and co-sponsored by the CNS, take advantage of same fee as member of the organizing Society
- Free posting of resume on CNS website
- Low membership fees compared to many other similar professional societies
- Possibility to always take advantage of lowest renewal fee by subscribing to automatic renewal service

Les adhésions à la SNC augmentent

Les adhésions à la SNC augmentent régulièrement au fil des années. Présentement (août 2007), la SNC compte 1141 membres en bonne et due forme. Il y a eu 151 nouveaux membres en 2007. Le graphique ci-dessous illustre l'augmentation constante des adhésions depuis 1999.

Une partie des cette augmentation est attribuable au nombre des nouvelles recrues dans l'industrie nucléaire canadienne ces quelques dernières années. Mais les adhésions augmentaient même avant la renaissance nucléaire, et je pense que ceci reflète le fait que l'on reconnaît de façon générale la qualité des conférences et des cours de la SNC, ainsi que le avantages d'être membre. Ces avantages incluent:

- Être membre d'une société professionnelle agissant dans le domaine de votre carrière professionnelle
- Bénéficier d'excellentes opportunités de croissance professionnelle rendues possibles par un grand réseau de nouveaux collègues canadiens et internationaux
- Bénéficier d'excellentes occasions d'accroître ses atouts personnels en devenant bénévole, en apprenant de nouvelles choses et en acquérant de nouvelles compétences
- Recevoir le Bulletin trimestriel de la SNC
- Recevoir en premier les avis de cours et conférences de la SNC, de colloques des chapitres locaux, et d'autres programmes
- Recevoir en premier, par courriel, toutes sortes d'autres avis d'intérêt et nouvelles diverses
- Le certificat d'adhésion à la SNC, le collant annuel et la carte individuelle d'adhésion
- Bénéficier de frais réduits d'inscription aux conférences et cours de la SNC
- Aux conférences organisées par d'autres sociétés (par exemple l'American Nuclear Society) et commanditées par la SNC, bénéficier des mêmes frais d'inscription que les membres de la société organisatrice
- Pouvoir inclure gratis son curriculum vitae sur le site internet de la SNC
- Frais d'adhésion moins chers que ceux de bien d'autres sociétés professionnelles
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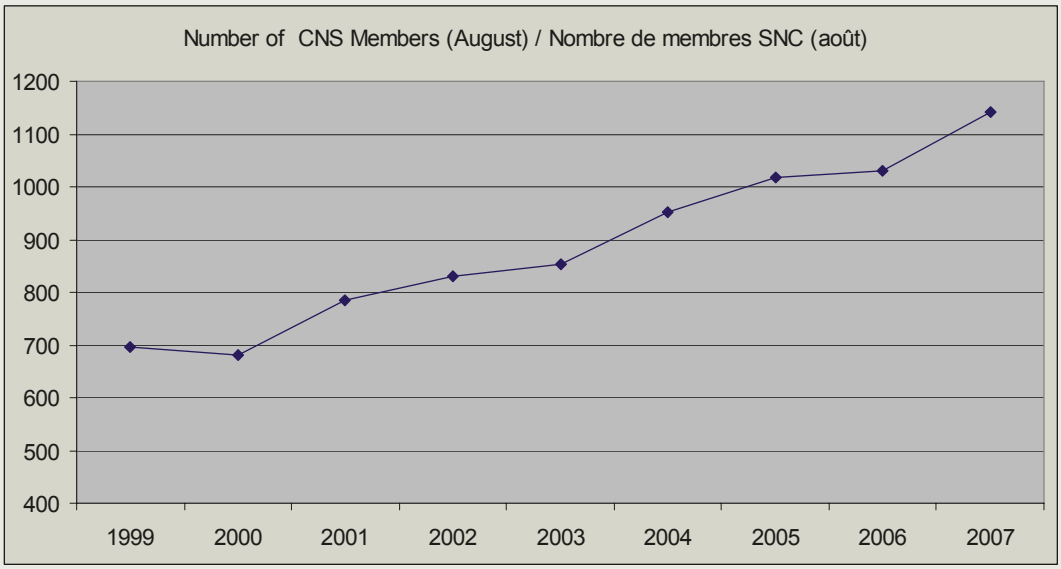

It Was The Best of Technologies; It Was The Worst of Technologies

by Jeremy Whitlock

And now, dear students, please cast aside all you've learned thus far in Social Behavioral Analysis 101, and accompany me on a journey to the erratic side of mass psychology.

Consider, if you will, the morning of July 16, 2007, when a magnitude 6.8 earthquake shook the northwest coast of Honshu, Japan, wreaking $100 billion in damage that killed 11 people, injured 1000 others, flattened hundreds of homes, and left about 9,000 refugees.

In one of the biggest non-events of that day, all four operating units at the nearby Kashiwazaki-Kariwa nuclear plant shut down without incident, as designed. Happily, this transpired despite the plant's location a mere 16 km from the epicentre, with local accelerations significantly exceeding the seismic qualification of the station.

Unhappily for thousands, the seismic qualifications were similarly exceeded for most structures in the region, and while that fact boggles the mind in one of the most earthquake-prone and technologically advanced countries on the planet, it remains a reassuring fact that the inherent conservatism of reactor design allowed the cores at the world's largest nuclear plant to safely shut down and remain protected despite the shortcomings of the seismic code.

But, to the crux: Ask anyone about Japanese earthquakes and you'll hear of the reactor that burned, the incompetent engineers, the radiation leaks, and how all this makes reactors anywhere on the planet an insane idea. ("We Almost Lost Niigata!", the inevitable rallying cry)

Not that there wasn't a fire (in a transformer), some structural damage, and a minor radiation leak that pales next to what is likely found in municipal wastewater. But in terms of human suffering, it was a non-event.

It was more than a non-event; it was soaring testament to the foresight and conservatism of reactor safety designers. It was a gushing good-news story that screamed: "If Mother Nature does the unexpected, probably the best place to be is inside a nuclear facility!" (And that goes for terrorism too, by gum.)

So why the mix of fortunes, an age of wisdom and an age of foolishness?

Quite simply, there are two nuclear technologies: one of Light and the other of Darkness. There is the nuclear world that most governments and scientists know, and the one that the people know. Both exist. Both are as real as a heart attack on a congested freeway during an unnecessary evacuation.

The virtual nuclear world (the one that killed millions after Chernobyl, almost destroyed half of Japan and not to mention Harrisburg, Pennsylvania, and routinely accounts for most of the deformities and disease within 100 km of each nuclear installation) is a powerful construct of the mass consciousness: a "memoid" if you will, held together by half-century-old memes as strong as the day Oppenheimer became a shatterer of worlds.

It was memoid marketing that drove Energy Probe to label the Canadian deep geological disposal plan for nuclear waste as "50% safe", prompted by a federal environmental review that declared the technology "technically sound" but "unsafe from a social perspective". That bridge crossed the two worlds, and upon that bridge the next many decades of Canadian nuclear used fuel management will be built.

Moreover - and this is the point - it is the only way that used fuel management can ever happen.

This yin and yang, after all, are deeply rooted in the nuclear psyche. One could easily argue that the memoid nuclear is much bigger and more real than the real nuclear.

Consider:

It is clearly good that we can detect radiation down to the decay of single atoms. It is also clearly bad that we can detect radiation down to the decay of single atoms (witness the worldwide angst over a handful of becquerels at Kashiwazaki-Kariwa).

It is clearly good that we can store all of our waste product in one place. It is clearly bad that we can store all of our waste product in one place.

It is good that fission offers millions of times higher energy density than any other energy source. It is bad that fission puts this much energy all in one place.

Ergo, nuclear technology has the best safety and environmental record of any energy source. Nuclear technology has the worst safety and environmental record of any energy source.

We have everything before us, we have nothing before us; we are all going directly to Heaven, we are all going the other way.

CALENDAR

2007

Sept. 30 - Oct. 4 — **NURETH-12: 12th International Meeting on Nuclear Reactor Thermalhydraulics**
Pittsburgh, PA
website:
www.ans.org/meetings

Oct. 15 - 19 — **SIEN '07 International Symposium on Nuclear Energy**
Bucharest, Romania
website: www.aren.ro

Nov. 11 - 15 — **ANS / ENS International Conference**
Wasington, D.C.
website:
www.ans.org/meetings

2008

June 1 - 4 — **29th Annual CNS Conference and 32nd CNS/CNA Student Conference**
Marriotte Eaton Centre
website:
www.cns-snc.ca

Sept. 20 - 26 — **IYNC 2008 International Youth Nuclear Congress**
Interlaken, Switzerland
website: www.iync.org

Oct. 13 - 18 — **16th PBNC 16th Pacific Basin Nuclear Conference (16PBNC)**
Aomori, Japan
website:
www.pbnc2008.org

Oct. 19 - 24 — **IRPA 12 12th International Congress of the International Radiation Protection Association**
Buenos Aires, Argentina
website:
www.irpa12.org.ar

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29th Annual Conference of the Canadian Nuclear Society and 32nd Annual CNS/CNA Student Conference

The 29th Annual Conference of the Canadian Nuclear Society and the 32nd Annual CNS/CNA Student Conference will be held in Toronto, Ontario, Canada, 2008 June 01-04, at the Toronto Marriott Eaton Centre.

Deadline for abstracts is December 1, 2008.

For more information contact the Conference Executive Chair, Jim Harvie: jdharvie@rogers.com or visit the Conference website: http://www.cns-snc.ca/conf2008.html

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CNA Liaison / *Agent de liaison avec l'ANC*

Colin Hunt 613-237-3010 huntc@cna.ca

CNS Office / Bureau de la SNC

Denise Rouben 416-977-7620 cns-snc@on.aibn.com

CNS Bulletin Editor / Rédacteur du Bulletin SNC

Rick Fluke 905-823-9060 x 6511 fluker@aecl.ca

CNS Bulletin Publisher / Éditeur du Bulletin SNC

Fred Boyd 613-592-2256 fboyd@sympatico.ca

2008 Conference Chair

Jim Harvie 613-833-0552 jdharvie@rogers.com

CNS Branch Chairs / Présidents des sections locales de la SNC 2007

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