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Oral presentation

Exposé oral

Written submission from Sunil Nijhawan

Mémoire de Sunil Nijhawan

In the Matter of the

À l’égard d’

Ontario Power Generation Inc.

Ontario Power Generation Inc.

Application to extend the operation of Pickering Nuclear Generating Station Units 5 to 8 until December 31, 2026

Demande visant à prolonger l’exploitation des tranches 5 à 8 de la centrale nucléaire de Pickering jusqu’au 31 décembre 2026

Commission Public Hearing

Audience publique de la Commission

June 2024

Juin 2024

Submission to the CNSC Public Hearing on Ontario Power Generation's Application to extend the Reactor Operating License for Pickering B, Units 5,6,7 and 8 to 2028

Sunil Nijhawan, Ph.D. P.Eng

April 29, 2024

As a CANDU nuclear design and safety engineer for over decades, I am surprised that the Commission has even been asked by OPG to consider supporting extension of operating life of Pickering B reactors, knowing very well by their own estimates that the degradation of fuel channels is widespread; a number of component and system failure mechanisms are fast converging to put the reactor into unsafe operation territory. It is well known to all, and even documented in safety reports that the Pickering B reactors cannot mitigate age related failures in ever thinning secondary side pipes without a severe overpressure resulting in channel failures that can fail the Calandria vessel and hence create an unmanageable accident. We also have long demonstrated by various analyses that the station is poorly equipped to mitigate a blackout without causing enormous off-site consequences in this most densely populated part of Canada. My fear is that the Commission will once again rubber stamp an irresponsible industry request. I see us all sleep walking towards a disaster and request one final opportunity to present evidence to dissuade all concerned parties from facilitating a national disaster. My country deserves better.

I start with a summary of who I am (‘My DNA’ below) so that the commission members recognize that I am technically qualified and am from within the CANDU industry and have the evidence that points to an impending disaster from an avoidable accident at Pickering that will likely cripple Canada for decades. Only time will tell if the associated CNSC staff as well as Commission members will be personally held accountable after an accident, if as is feared, this submission is summarily ignored irresponsibly.

My DNA

- Senior nuclear reactor safety engineer / analyst with over 43 years consulting experience in nuclear reactor safety, licensing, and design analysis for CANDU PHWRs, US PWR/BWRs. Russian RBMKs as well as research and medical isotope reactors.
- Specialized in deterministic evaluation of accident progression and consequence assessments following severe accidents, design review of advanced reactors, regulatory issues and safety analysis for licensing.
- Professional engineering consulting activities over the last 43 years with Atomic Energy of Canada Ltd, Ontario Power Generation, Canadian Nuclear Safety Commission and overseas PHWR industry on advanced reactor design evaluations, methods development, deterministic safety assessments, consequence analyses for severe core damage accidents and severe accident management. Conversant with the full cycle of reactor response analysis from core design, thermal hydraulics, fuel behavior, fission product release and transport, containment response, onto dose evaluations.

- Developed a large number of CANDU specific original computer software packages in *Fortran* for reactor licensing safety analyses submissions and for evaluation of severe accident progression & mitigation in research and power reactors. These include integrated codes for severe core damage assessments, fuelling machines, spent fuel pools, fuel channel leaks and cracks, fuel channel ageing with creep and many others, e.g. MAAP-CANDU, MINAR, ROSHNI, CIFTA, CHANNEL-CRACK, PHWR-CHANNEL r of international computer codes used for reactor thermal hydraulic analyses and in interpretation of data from experiments.
- Extensive computational modeling, research and hands-on experimental background. Academic background includes 5 years of undergraduate and graduate school teaching and research in the U.S and Canada.
- Deeply committed to real public safety from nuclear power operation and a vocal proponent of responsible fact based nuclear regulation in public interest alone.

INTRODUCTION

As a professional nuclear engineer with 42 years direct experience in PHWR safety analyses and design evaluations, I have engaged with Canadian Nuclear Safety Commission members for over 10 years and made various submissions in public interest. I did so in interest of public safety alone and as it has resulted in great personal and business cost to me. I have done significant original work on independent, dispassionate design assessments and deterministic analyses methods development. I have published a dozen external papers outlining the risk significant design errors and emerging safety issues with PHWRs, their regulators and industry. As I inch towards retirement, a few things have become clear:

1. *The Canadian Nuclear Safety Commission members routinely rubber stamp any and all requests by industry.*
2. *Staff stretch truth regularly in their presentations to Commission members who more or less have no clue as to veracity of technical information packaged by staff for them.*
3. *Even when information contrary to presentations made by the staff and industry is unearthed and presented through the CNSC channels of communications, the Commission Registrar blocks its transmission to the members who then make their 'decision' based on faulty information that contained misrepresentations during hearings that the public has no way of refuting at that time and must do so at a later date and to no avail.*

I oppose this license exception application by OPG for a number of technical reasons and do so in the spirit of my professional obligation to put public safety first. Designed over 50 years ago, Pickering B reactors have been allowed to operate for far too long; and well past their safe & useful life. Today these reactors compromise the safety of millions of people.

The 4 operating Pickering reactors pose an unacceptable risk as the condition of outdated station equipment, industry changes and safety culture make this plant ripe for an accident that may cripple not only the adjacent communities but also, because of the enormity of potential radioactive releases and an unfortunate location in the middle of a huge population center, a large productive part of upper Canada including metropolitan Toronto.

I focus on the following in this intervention:

a) The Pickering reactor internals (specifically, but not limited to, pressure tubes and feeders) have aged gracefully because of the professionalism of OPG personnel and procedures in place but from a safety engineering stand point they have deteriorated beyond redemption. They long ago reached the end of their life and are thus susceptible to uncontrolled failures with unintended consequences (*sections 1 and 2*).

b) When these reactors were built, the design principles, design targets and resources were so different. While these reactors represented pioneering work that made us all proud to be part of the CANDU industry, our knowledge-base about accidents and their consequences, and effect of radiation, loads and fluids on materials, equipment and systems was far from mature. We were experimenting; we were creative and fearless. However, as a result of where we started and the design decisions we made, Pickering reactors are now old and unfortunately also demonstratively poor in severe accident mitigation; have not undergone desirable design upgrades; sport one of the weakest and leakiest power reactor containments in the world and will cause unacceptable off-site radiological consequences following an event as simple as a sustained loss of AC power leading to severe core damage (*section 3*). No other jurisdiction in the world with a responsible regulatory regime would license such a reactor today or allow one to continue to operate.

3) The few in-reactor design improvements incorporated after Fukushima do not have the benefit of good supporting analyses and are practically useless in reality as they are unable to prevent, limit or mitigate off-site damage adequately. For example, the recombiners for 'hydrogen' mitigation systems should have specifically been qualified for D_2 instead of H_2 and their numbers and type should have been exhaustively analyzed such that anticipated Deuterium source term can be controlled and explosions can be avoided. New analyses show that these systems are way off their intended mark and as potential sources of ignition, actually dangerous. Similarly, the filtered venting systems need to have been designed for more severe fission product source terms and higher thermal loads. There are a dozen other accident scenario mitigation proposals and design fixes still awaiting sound engineering disposition and implementation. While OPG may have the right public profile, it has not had the mission, vision, right resources, opportunities or regulatory impetus to do so. Some are as simple as system over-pressure protection whose principles engineers have understood for a hundred years but the benefit of such understanding has not precipitated to Pickering or has been lost in blindly defending erroneous decisions of the past. In any event, the inherent impending failures in aged and deteriorated reactor internals compel that permanent shutdown of Pickering plant be implemented and alternate power sources investigated and deployed. The Pickering site should never have another nuclear reactor.

4) The rationale to shutdown Pickering B the stations is more compelling than the rationales used to justify either closing (Gentilly-2) or 'refurbishing' (as at Bruce and Darlington) other CANDUs much before they reached the age of these Pickering reactors. A very fundamental question you, the Commissioners, must also ask is this : *why we now ignore all the arguments used for justifying expenditure of tens of billions of dollars refurbishing reactors at Darlington and Bruce stations and yet let these oldest and weakest of CANDUs go on operating in the middle of what is essentially a continuous mega-metropolis of greater Toronto?*

In recommending that these plants be retired, you will find support in arguments summarized by Hydro Quebec management in shutting down Gentilly-2 even after they had been given a license by CNSC Commissioners, on recommendation of senior CNSC staff to continue operation and to refurbish. Hydro Quebec wisely walked away after spending ~$900 million on refurbishment activities with message by their CEO to the effect that he would not fly in an unsafe airplane either. *And* Gentilly-2 was a relatively more robust design that Pickering.

There is ample data that shows that the feeders and pressure tubes in Pickering B units have long passed the end of their safe life due to erosion and corrosion induced thinning. Pressure tubes have elongated, thinned, sagged and increased in diameter at places along with other changes like blister formation and hydride inclusions that initiate incipient cracks and ruptures. I will present you some data on pressure tube elongation, deuterium intake, and feeder thinning. We have tons more.

We cannot also ignore deterioration of carbon steel feeder pipes which have definitely thinned to close to 60% of the original thickness in certain places according to the test data and models that are open knowledge. New thermal-hydraulic models that we developed have produced whole core feeder failure maps (section 1) that show that a smaller than 1 cm^2 crack in an inlet feeder will cause the fuel downstream to overheat quickly and cause an in-core rupture with severe consequences. Fluid discharge from such cracks can remain undetected long enough for channel failure which will occur in a couple of minutes due to fast fuel heatup at full power. I also summarize in section 1 relevant data from public papers on feeder wall thinning rates due to a number of reasons that engineers lump under 'flow accelerated corrosion'. Of course, there is more data available from OPG, CNL and other utilities.

I have serious concerns about any more life extensions for the old Pickering reactors for a number of reasons that are based on a number of simple yet well established engineering principles. First of them is pure obsolescence. These reactors were designed about 50 years ago when we worked with slide rules and math tables and had a religious trust in and a fervor about indigenous nuclear power. (Almost nothing else today in our lives is that old or that risky). We did not have the benefit of research into and modeling of severe accidents or exposure to the horrors of Fukushima and Chernobyl. We did not know then and still have only limited knowledge of phenomena behind the effects we see of radiation fields and aging on materials. We did not expect fuel to get too hot for too long in an accident and thus equipped the reactors with cheap carbon steel feeders that we now know will oxidize quickly to produce copious amounts of 'hydrogen' that will likely blow up the flimsy and leaky Pickering containment buildings just like it happened in Fukushima.

Even now we operate these reactors with obsolete control systems, outdated hardware and grandfatherly computers. On top of obsolescence, the Pickering reactors also do not have a stellar operating record and are not too well designed for accident mitigation. As a result they fail to meet today's public expectations of risk and their design details scare me, in spite of my nuclear engineering background and the great confidence I have in the integrity and professionalism of OPG personnel at the operations level.

While more details are presented in Section 3, here is a list of some of the issues related to severe accident vulnerability of reactors at Pickering:

- *These 50 year old design reactors did not consider severe accidents in the design process. Unreasonable to expect easy severe accident mitigation.*
- *Containment 500 times leakier and 10 times weaker in than normal PWRs and unable to handle a severe accident in more than one inter-connected units*
- *Current overpressure protection design inherently forces a reactor damage even before an ECC loss leading to severe core damage.*
- *No provisions for manual depressurization of primary circuit after a sustained loss of power. No super high pressure ECC or makeup intervention / injection.*
- *Onset of a severe core damage puts activity and combustible gases directly into the relatively weak containment in 3-4 hours. There is no holding of activity in a vessel like in a PWR pressure vessel.*

- *Significantly higher sources of "hydrogen" from large amounts of carbon steel in feeders and Zircaloy in fuel sheaths and pressure tubes / Calandria tubes.*
- *The limited number of planned recombiners will cause explosions.*
- *Enhanced potential for energetic interactions of melting fuel with enveloping water*
- *Pressure relief in ALL relevant reactor systems is inadequate (PHTS, Calandria, Shield Tank, Containment)*
- *The containment amongst the weakest in the world for pressurization (~0.5 Atm);*
- *The Calandria Vessel, long heralded as a core catcher, is a thin ~1" thick stainless steel welded low pressure vessel that has been assessed to fail catastrophically at welds and not able to contain hot molten debris.*
- *This Calandria vessel failure can not only lead to enhanced combustible gas production but also severe energetic explosions leading to failure of structures*
- *Current Shield Tank cannot contain pressure upon boiling and can fail. Restoration of cooling after water depletion problematic.*
- *Inadequate instrumentation and control after a severe accident.*
- *Poor equipment survivability as severe accident environments very unfavorable due to a lack of containing pressure vessel*
- *Currently planned PARS inadequate and potentially dangerous.*
- *No dedicated operator training / simulators for severe accidents.*
- *Severe accident simulation methods are outdated, crude and inadequate.*
- *No significant design changes implemented. Known problems ignored.*
- *Current SAMGs are inadequate. Many Emergency hookups not implemented*
- *High risk potential from external events*
- *Need to reconsider malevolent actions and sabotage at spent fuel storage pools as well.*

Contrary to some misdirected and uninformed CNSC publications and OPG submissions, severe accidents really do not have benign consequences and there is very little time (~3 hours) before a simple loss of power incident develops into a serious core damage accident at Pickering. Less than a mile away, a large number of people will get exposed to harmful radiation before they panic to swarm into the parking lot and trap the Hwy 401 will become. The engineering issues that I have been putting in public domain for years need to be revisited with greater professionalism, team spirit and integrity. When similar safety culture issues were ignored by NISA regulatory staff and powerful utility TEPCO in Japan prior to 2011, the result was an accident which will take them a century to cleanup , a half a trillion dollars of expenditure and permanent changes in the destiny of hundreds of thousands of unsuspecting citizens living normal lives up to tens of km away.

Similarly at Pickering, any economic benefits to OPG are dwarfed by the hundreds of billions of dollars in on and off-site damages, a simple sustained loss of power accident, however initiated, can cause. We also have a reactor with much lower number of real barriers to radioactive releases to environment and a much higher source of explosive gases (see section 3). A Fukushima like loss of power accident will affect hundreds of thousands of my countrymen in the precious pretty towns that, but for the lake Ontario, surround this plant. Insurance on the properties they own are, as a rule, not covered by any nuclear accident and OPG liability is limited to a mere $1 billion.

Pickering reactors are some the most obsolete of all operating power reactors in the world. The obsolescence of this 50 year old design goes all the way from the limitations of the single shutdown system at Pickering A to the leakiness and low failure pressure of their common containment. The list is reflected in the design and other fixes I have summarized in section 3. A loss of regulation has a higher probability at these plants and the resulting Chernobyl like consequences are more likely here than at other CANDU plants that sport 2 independent, diverse shutdown systems, absent at Pickering A.

OPG staff need to be educated more in state-of-the-art severe accident issues and the corporation needs better in-house engineering resources of the caliber that had once made Ontario Hydro pride of the nation. Depending on compliant external engineering service providers to give the most convenient and management pleasing answers to tough questions, it is yet to implement any 'best of our knowledge and abilities', comprehensive, serious design enhancements at the ageing Pickering A & B stations (and Darlington) to reduce risk to the unsuspecting millions living in the greater Toronto area, many thousands just a short bicycle ride from the Station.

OPG also needs to diligently communicate the whole range of accident risks to public and first responders. The information provided so far to first responders (EMO was told of a very low release 100 TBq cs-137 (0.15% of total) from a severe accident) for emergency preparedness is not consistent with facts. Analyses by us and some CNL/CNSC staff have shown that the releases from a failed containment will likely be hundreds of time higher for a simple station blackout scenario.

OPG also needs to communicate to people the inadequacy of not only the limited area in which KI pills are distributed (to preemptively try to avoid absorption by thyroid of inhalation of radioactive I-131 releases), but also of their true effectiveness. It still has to implement most of the lessons learnt from Fukushima, even though CNSC staff have managed to declare closure of all Fukushima Action Items.

The 3 severe accidents in less than 15,000 reactor years of operation in 3 well-designed reactors, in 3 most technologically advanced countries should give OPG management impetus for improving their now obsolete reactors. Right now we likely are at a point of no return; of no good other options that will keep the plant running with reduced risk.

I will talk at length in my oral presentation about my research into the technical issues related to pressure tube and feeder end of life criteria; life extension opportunities and pressure tube & feeder failure mechanisms. My conclusions so far render another 4 year extension to Pickering B reactor operation unsupportable. I concentrate in the next sections on the three main issues that are important reasons for suggesting a denial of application.

I can competently present for your consideration information and arguments on these topics because during my now 43 years of still active, professional career as a reactor safety analyst in this very industry, 10 of which were at the old innovative and dynamic Nuclear Safety Division of Ontario Hydro for some brilliant technical managers, I have further specialized in critical evaluations of PHWR reactor designs for severe accident mitigation. Besides other contributions of analytical methods for CANDU safety assessments in support of licensing of many different CANDU stations, I developed comprehensive accident progression and consequence analyses codes for single and multi unit PHWRs, fuelling machines, spent fuel pools and research reactors for a number of organizations.

I have now developed and use a number of new state-of-the-art computer models (ROSHNI series of codes, see Reference 1 for example) to analyze progression and consequences of CANDU severe accidents; this time in unprecedented detail and complexity. These 'integrated' and modern codes are about a 1000 times more computational intensive, model the whole reactor response and require a higher understanding of not only the design of scores of reactor systems but also of various interdependent thermo-chemical phenomena. Many accident progression pathways we analyze are new and results very different than OPG has produced with an old code of mine (MAAP-CANDU) that I developed in late 1980s for them by marrying my CANDU models with an old US code called MAAP.

Three decades of experience in CANDU accident modeling and analyses has given me a unique insight into not only the vulnerabilities but into also engineering fixes that can help reduce risk. I have setup code development teams, analyzed accident scenarios, evaluated old and new reactor designs and liberally drew from research and development work by others to propose engineering solutions that will reduce risk from our unique CANDU reactors.

As a professional nuclear engineer, I insist on safe nuclear. This intervention is purely in public interest and driven by my sacred professional obligations to public safety and long defiance of the tradition of not airing in public the problems that plague our old reactors, just like the problems that plague other reactor designs of the same vintage.

I intervene with renewed understanding that currently there really are no better, more convenient avenues than a sane CNSC decision to immediately remove for the sake of our children the risk posed by operating Pickering reactors beyond their safety envelope. The province of Ontario unfortunately will bear all the consequences but is not in direct consultation about the enormous risks this and many of the past CNSC decisions expose it to. With no expertise or independent advise of it's own, the Ontario government needs to be educated more about risks and consequences of continued operation of these old Pickering reactors. OPG management is directly responsible for the decisions it makes for Ontario by proxy.

Vulnerabilities of the Pickering design are known to many for long and mostly to those educated and trained in the industry. There are hundreds of very honest and highly professional personnel at these organizations who care and mostly do their best to keep their positions going. I have the benefit of being able to vocalize what many, many others cannot do publically. Scores of them are my colleagues who support my intervention.

1. DEGRADATION OF CARBON STEEL FEEDERS AFTER 40 YEARS

Feeders connect the fuel channel end fittings at each end to common headers. There are about 7 km of 760 feeder pipes, 38 to 75 mm in diameter. The, as new, wall thickness of the feeder pipes (5 to 8 mm) is consistent with the 11 MPa design pressure. To put it in perspective, the ~1700 m^2 of surface area on each of inner and outer surfaces is more than the ~1516 m^2 surface area of a NHL hockey rink.

Feeders corrode from inside and rust from outside. Under accident conditions they may oxidize both from inside and outside. The Flow accelerated corrosion on the inside surfaces is higher for higher power channels that are least orificed and have larger flow velocities. The unfortunate choice of low carbon steel (ASTM SA106-Grade B) with actual chromium content at the lower end of the 0.04 to 0.4%, is one of the most critical life limiting factors for Pickering feeders and hence the reactors. This steel is easy to work with but corrodes, erodes and oxidizes easily. Steels with higher Chromium content do not.

These feeder pipes are composed of a number (~ 3000) of bends and a number of elbows (~1500) that are particularly prone to erosion and flow accelerated corrosion. About 8% of wall thinning already occurs during bending at the manufacturing plant. Eroded iron dust is regularly seen deposited on spent fuel bundles. Flow assisted / enhanced corrosion/erosion leads to thinning of feeders by an average of about 0.1mm/year and their inaccessibility leads to near impossible comprehensive monitoring.

Figure 1 : Feeders oxidized inside a feeder cabinet at Bruce CANDU

Thinned feeders can leak, crack or fail catastrophically. The main issue with a gradually developing or sudden loss of coolant after failure of a stressed and thinned feeder under normal operation is that it can result in break discharges into the containment that are not immediately detected. For a leak in an inlet feeder between the header and the inlet end fitting, the remaining flow into the fuel channel can stagnate leading at full power to fuel melting and channel rupture. The rate of fuel heatup at full power can be of the order of 100 C/s such that the fuel melting and channel destruction can occur in minutes.

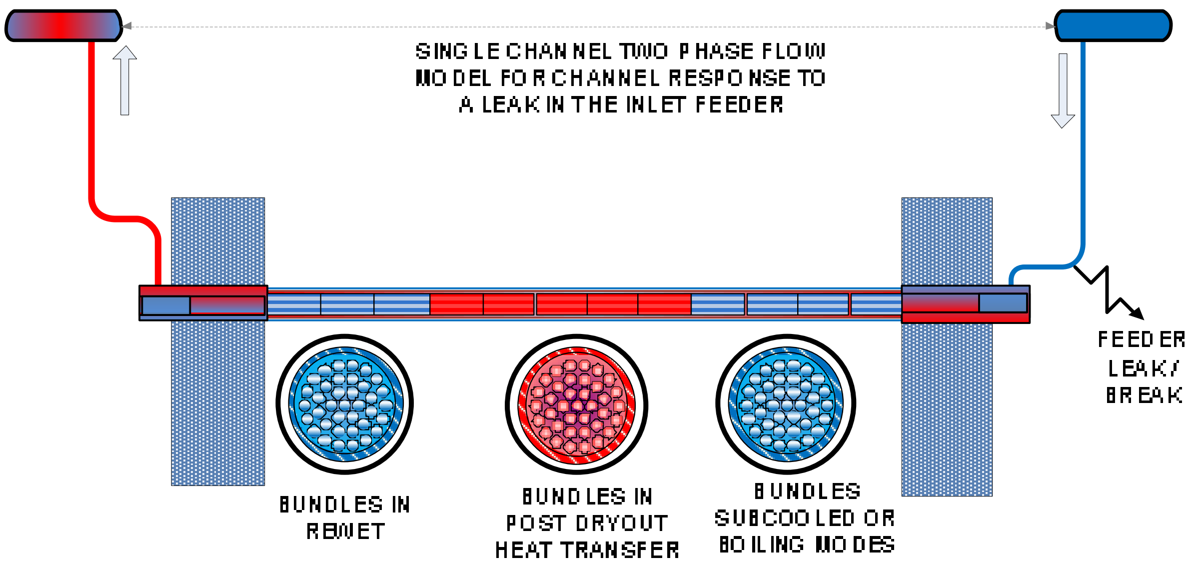


Figure 2: ROSHNI model used to evaluate feeder break sizes that can cause fuel heatup, undetected by reactor systems

The break discharge that can stagnate the flow is calculated for Pickering B header conditions (using CANDU 6 feeder data for convenience to make a point) by our integrated suite of ROSHNI series of severe accident codes that model all reactor channels. The core wide map of break discharges at full stagnation and break sizes for onset of fuel heatup are shown in Figure 4 and Figure 4 . A critical feeder break in a feeder can keep leak detection systems oblivious of the progressing accident. The 'leak' is of magnitude (<10-30 kg/s for half the channels) that is not detectable within a minute or two that it will take for the fuel to heatup and to fail the channel. It is more instructive to look at feeder leak sizes that may cause fuel heatup. Figure 4 maps out the results of analyses and shows that break sizes as small as 1 cm^2 can create flow degradation causing fuel dryout.

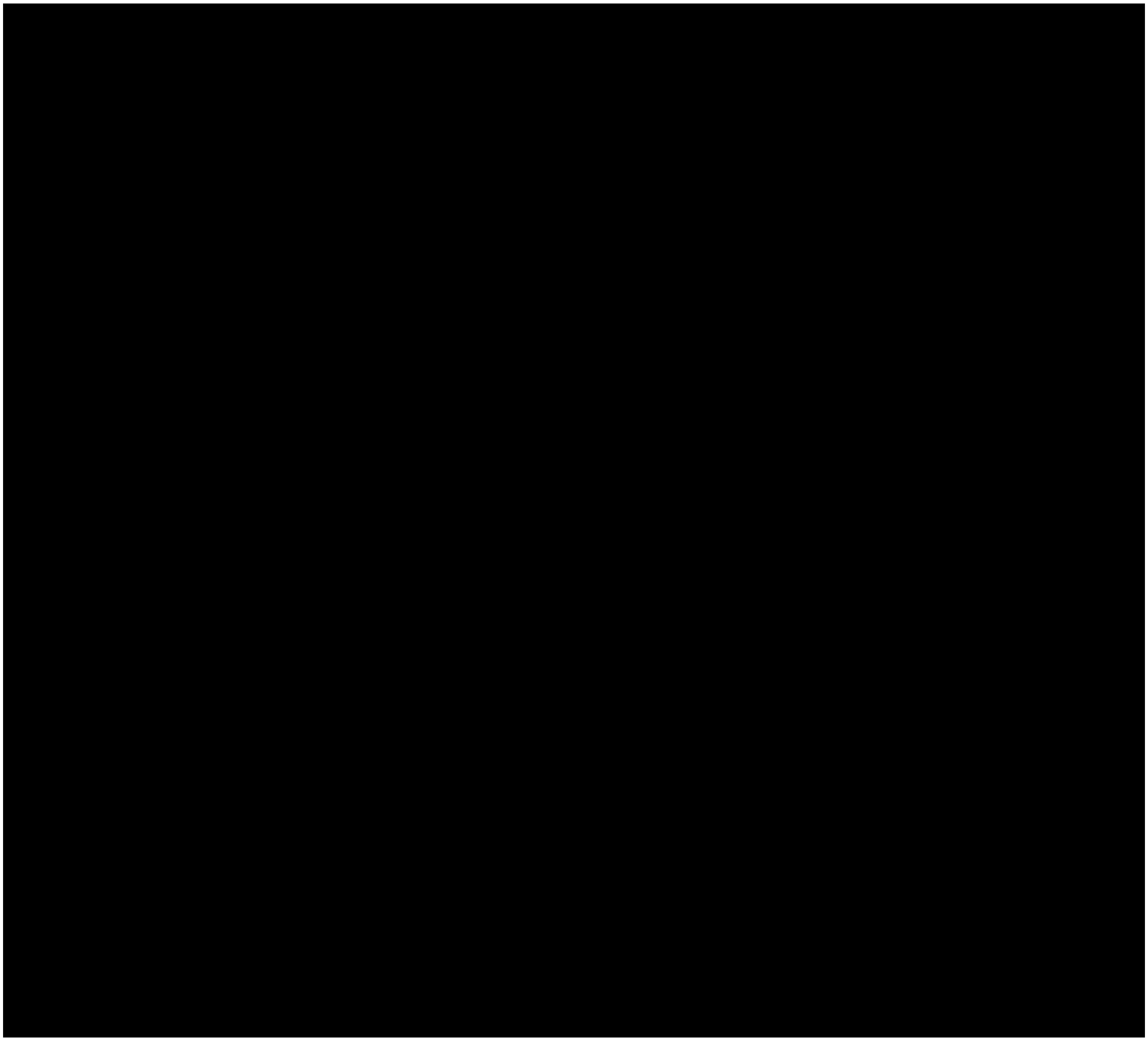
Therefore a continued exploitation of these feeders that have been known to erode at the rate of ~0.1mm/year and may already have thinned to half the original thickness at certain places is not recommended by good engineering practices or applicable codes under which the reactor class 1 components are designed and licensed.

Reference 2 includes data on wall thinning in CANDU feeders and proposes a correlation for 'remaining life'. Using the CANDU 6 data used in Figure 5 from that reference, we see that the feeders will thin to less than 50% wall thickness in 30 years, well beyond their safer operating envelope and code requirements.

Figure 6 taken from Reference 3 further demonstrates that more than 0.1 mm wall thinning per year is unsustainable in Pickering B reactors for another 5 years. A 2.5" feeder pipe would thin its wall thickness by 50% in 27 years of operation at a bend. That is below the ASME requirements and outside the realm of good engineering practices.

Thus there are no arguments that can be made to convince a professional engineer that all Pickering B feeders will last the sought after life extension. It is mind boggling that a request for life extension was even made with professional engineering support.

Figure 3 : Feeder break discharge (kg/s) causing flow stagnation in Pickering fuel channels (CANDU6 feeder data with Pickering header conditions used or illustration))



BREAK SIZE (SQ CI

10	4	
4	3.5	
3.5	3	
3	2.5	
2.5	2	
2	1.5	
1.5	1	
1	0.5	

Figure 4 : Inlet feeder break sizes (Sq cm) that cause onset of fuel heatup; slightly bigger cracks cause flow stagnation (CANDU 6 channel layout used for expedience).

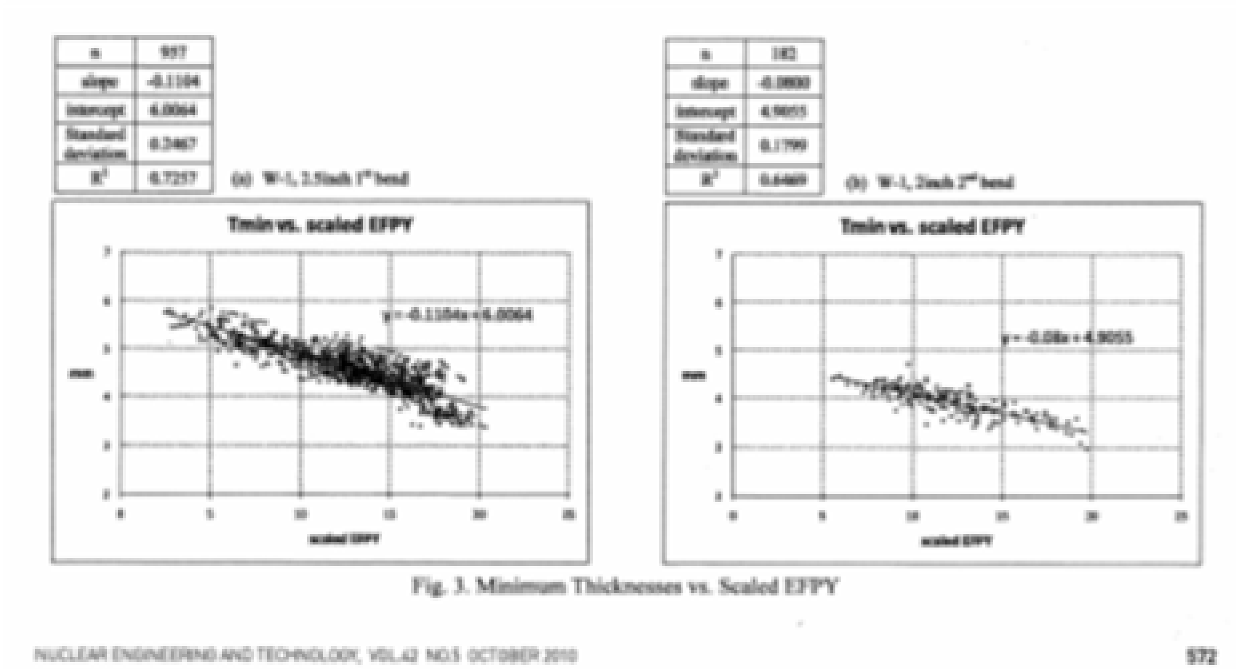


Figure 5: Feeder thinning correlated to remaining life for CANDU reactor

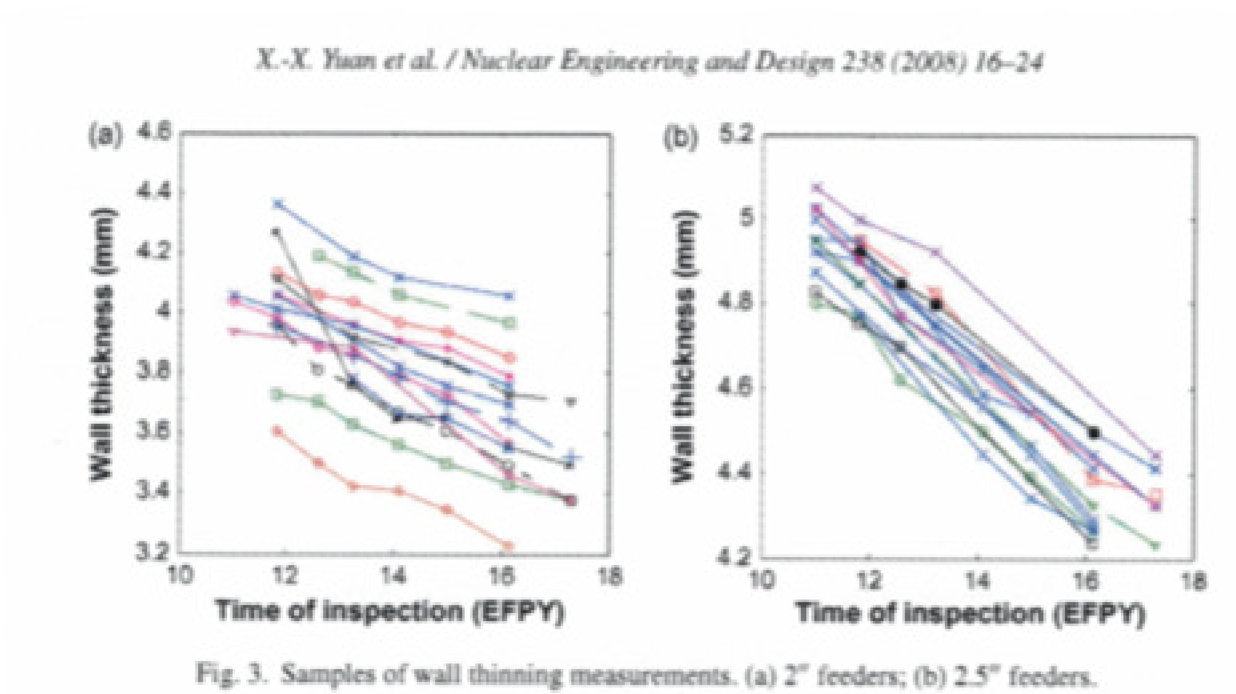


Figure 6: Sharp decreases in wall thickness pointing to a critically low wall thickness well before 35 EFPY

2. PRESSURE TUBE FITNESS FOR SERVICE AND END OF LIFE ISSUES THAT DO NOT ALLOW LONGER EXPLOITATION OF PICKERING

Geometric and material changes under radiation, thermal, chemical and fluid fields that occur over the years have an effect on operational behavior as well as safety functions of the fuel channels. The effects become more pronounced with time and use to a point that equipment becomes unfit for use. Just like everything else in our life. Over the years a number of fuel channel degradation mechanisms have been identified and compared against some fitness-for-service criteria that have been developed by the industry itself.

Hydriding of pressure tubes after dissolution of deuterium beyond the terminal solubility limit is commonly discussed first because of it's propensity to cause ruptures. Hydride inclusions are precipitated once the free deuterium D diffused from surface oxide layer into the body of the metal reaches a concentration greater than the terminal solubility limit (TSS). Hydrides become the nucleus for cracks and potential failure.

$$Zr + 2\ D_2O = ZrO_2 + 4D$$

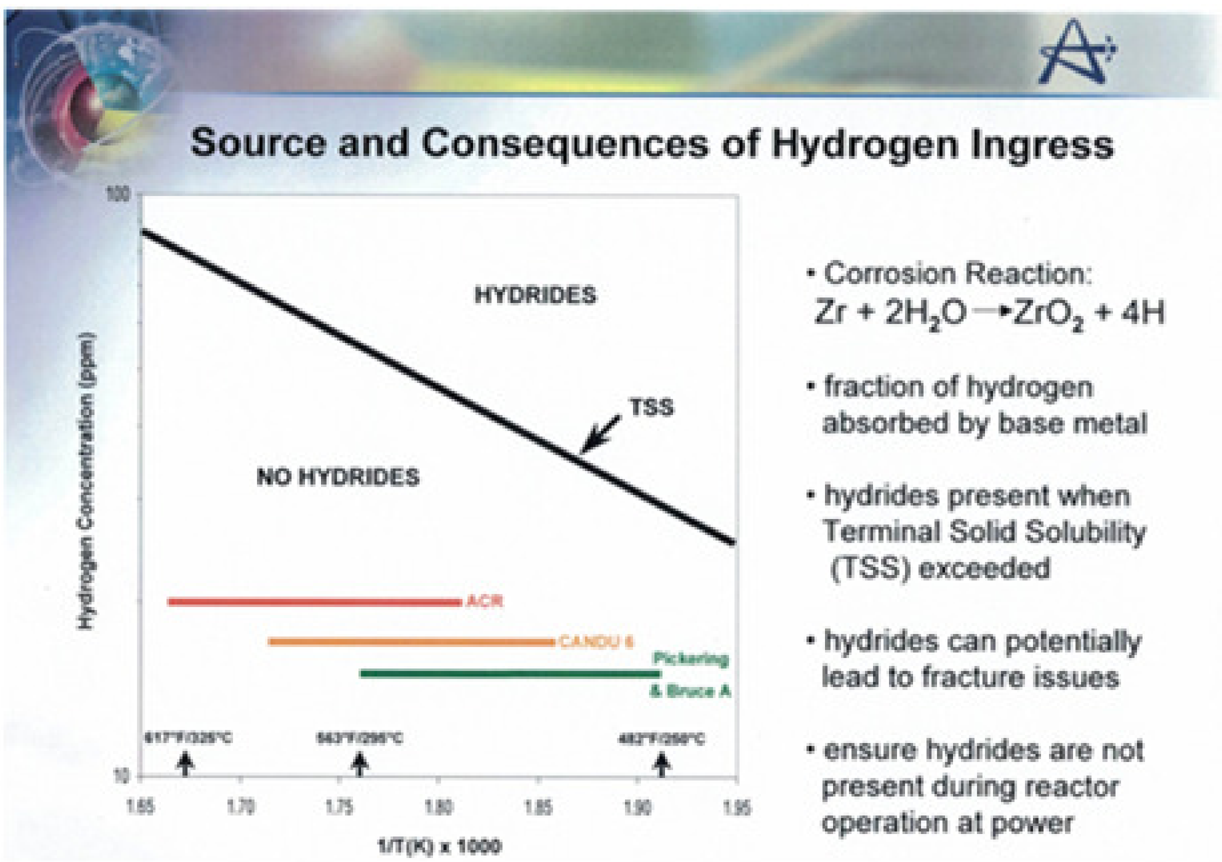


Figure 7: Terminal solubility limits (courtesy : AECL presentation to USNRC DEC 2002, Reference 4)

Recall that there have been a number of pressure tube ruptures and these ruptures do not cover all operating states and all potential consequences. Yet the data is alarming enough to force refurbishment (mass pressure tube and Calandria tube replacement) costing many billions of dollars. This was not anticipated even in 1981 when I joined AECL as we were happily licensing third of a dozen new reactors. The necessity of a premature 'refurbishment' to prematurely replace the reactor core internals was the first crack in the infallibility of the CANDU design we were taught to accept. Many more were still to come but soon a new CNSC came to the rescue and tacitly promised the industry automatic license renewals, irrespective of any data to the contrary. Industry started covering more and more of CNSC staff salaries.

We also knew early that the Deuterium ingress was greater at the rolled joints of pressure tubes with stainless steel end insert end fittings. CSA complied with Deuterium limits specifically for rolled joints, although there were other difficult to correlate factors like stress levels in play there as well. Data showed that even after 10 years the solubility limit was likely exceeded at the rolled joints as shown in Figure 8 presented to US NRC by AECL in 2003 (reference 4).

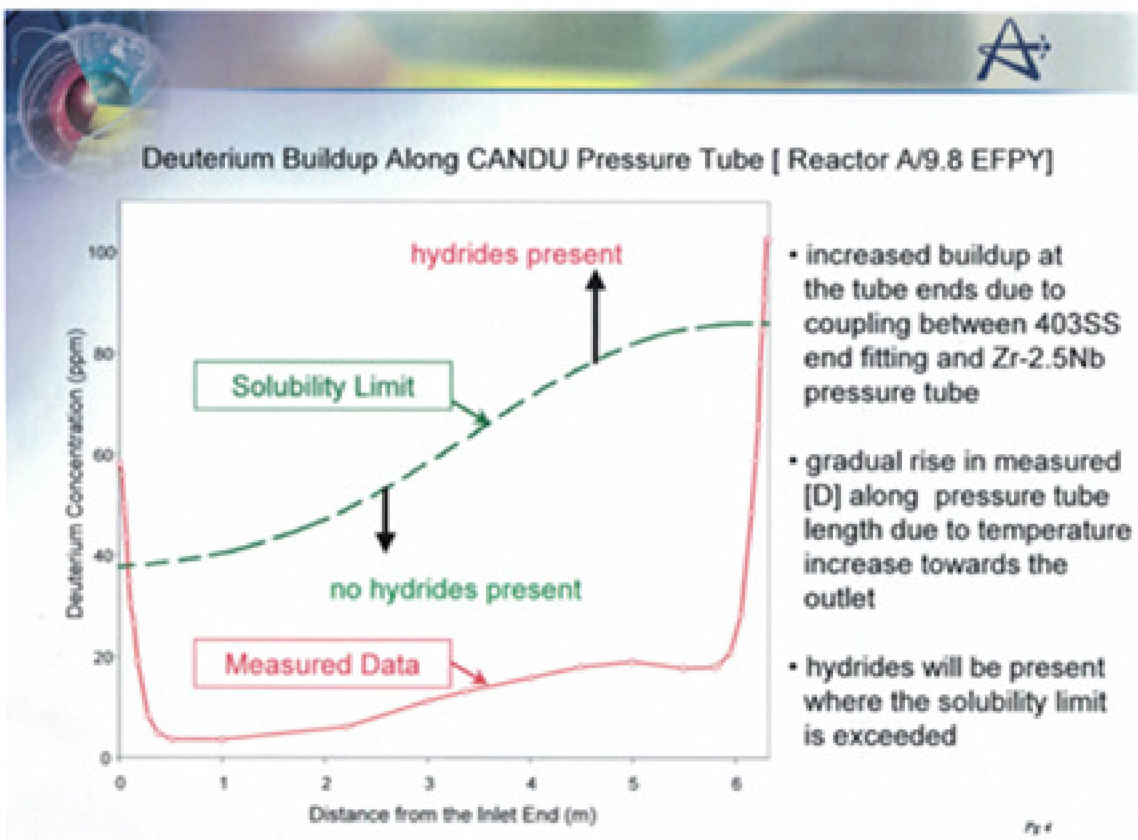


Figure 8: Data on variation of Deuterium ingress

There are other mechanisms such as radiation damage induced increase in diameter (diametral creep), increase in length (axial elongation) that can also be life limiting for pressure tubes. The former can result into a contact of pressure tube with the Calandria tube; into creating a bundle flow bypass & decreased margins for fuel cooling (unit de-rating by August 2017 we were told in 2015), as well as increased hydriding. Pressure tube elongation (an effect of radiation fields) can result in the channel failing off the end fitting bearings that support them at two ends. Both result in wall thinning and reduction in margins to safety. Additional deteriorations include increased channel sag whose measurement using available AECL designed tools, I am told by technicians who do those measurements , is highly inaccurate.

Other aging related fuel channel changes that have given the industry a lot of grief included garter spring spacer dislocations and reduction in pressure tube to Calandria tube gap. Monitoring and sampling programs were initiated to not only get data on the absolute values of the changes but also on rates of changes of these parameters (Hydride concentration, diametral changes, lateral creep, wall thinning, garter spring movements, PT/CT gap). Industry used their own people to set up standards for setting up fitness for service criteria (e.g. max 100 ppm of D in local concentration). Some criteria were based on end fitting design. These included total elongation of the pressure tube with limitation being the elongation that would still permit the end fitting to stay on the bearing pads.

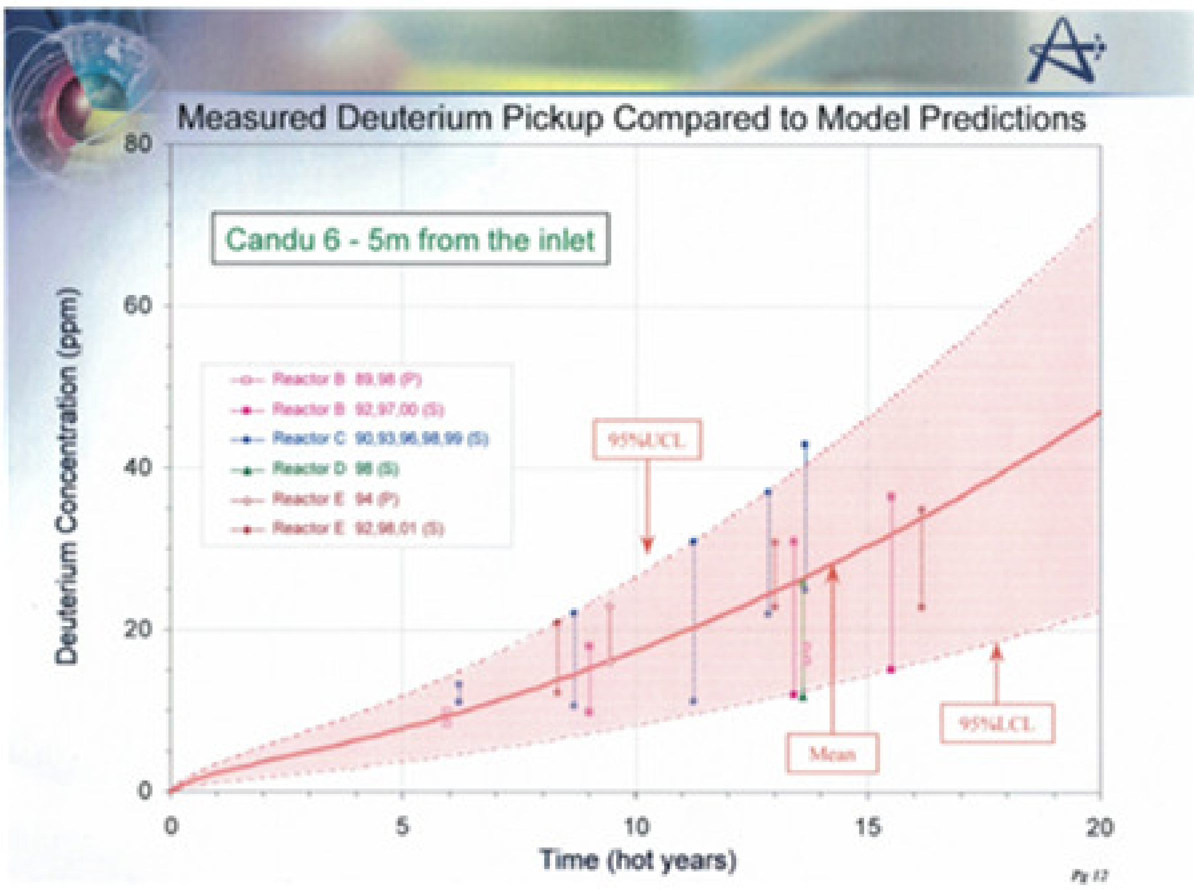


Figure 9 : Deuterium concentration increase data from a CANDU 6 reactor; note the location. this is not at the rolled joint. (courtesy : AECL presentation to USNRC DEC 2002, Reference 4)

With that back ground, here are the reasons why the Pickering pressure tubes will not meet fitness for service criteria.

To start with, here is SOME data on Pickering B fuel channels that is problematic. Table 1 presents an optimistic correlation between measured channel elongations until 2014 and the time left to projected end of life for Pickering B.

Table 1: estimates of channel end of life due to elongation (time at which for Available Bearing Travel ABT causes an end of life) data from NK30-REP-31100

1	2	3	4	5	6	7	8	9	10
	EFPH [1000S] Dec 2014	Dec 2014 West face ABT limit kEFPH	ABT after both East and West ends made free kEFPH	Remaining Time assuming linear elongation rates kEFPH	Elongation Rate mm/ 7k EFPH (one year at 80% power)	project date at 80% power of early end of life	project date at 80% power of late end of life	project date at 100% power of early end of life	project date at 100% power of late end of life
PICKERING B UNIT **5**	209	255	268	59	4.22	Jun-21	May-23	Feb-20	Aug-21
PICKERING B UNIT **6**	214	248	267	52	4.2	Oct-19	Jun-22	Oct-18	Dec-20
PICKERING B UNIT **7**	207	241	264	56	4.31	Oct-19	Jan-23	Oct-18	Jun-21
PICKERING B UNIT **8**	196	239	296	99	4.06	Jan-21	Mar-29	Oct-19	Apr-26

reference date	FACTOR	EFPH / YEAR	FACTOR	EFPH / YEAR
Dec-14	80%	7008	100%	8760

Please pay attention in this table to columns 5, the 'Time at which ABT (Available Bearing Travel) will be reached even after a second 'reconfiguration' to free the east face of the reactor. Without that project, the time to reach total ABT is next year (column 7) at 80% power and this year at 100% power (column 9).

In 2014, the projected time at which end of bearing travel times would be reached were as low as 268k EFPH with remaining time to safe operation predicted to be as low as 52k EFPH which would make the reactor pressure tube end of life at Dec 2020, if not sooner. See table above.

Data show that at 80% power the total channel elongation of 165 mm will occur as early as October 2019 and as only as late as June 2022 if another channel readjustment is done (making the other end free). Otherwise the end date is in October 2018 and December 2020 respectively at 100% power.

Therefore the pressure tubes are not good for any more life extension. I have a feeling, given my experience in this industry that this data will now be 'reconfigured' to show that the channels are good until all the cows come home. Sharpening of pencils to err on the side of management decision to prolong the life of these obsolete reactors is an unprofessional conduct on part of consultants who support now a different end of life time.

With additional data and additional information, I will make a more detailed presentation orally about the PT issues with more in-house information that contradicts the story now being propagated, time permitting. In looking for how these issues were treated in the CMDs, I am alarmed that CNSC staff CMD-18-H6 reads like a propaganda pamphlet for high school students rather than a technical case for Pickering plant life extension. Very important issues have been left out in a similarly worded CMD H6-1 by OPG.

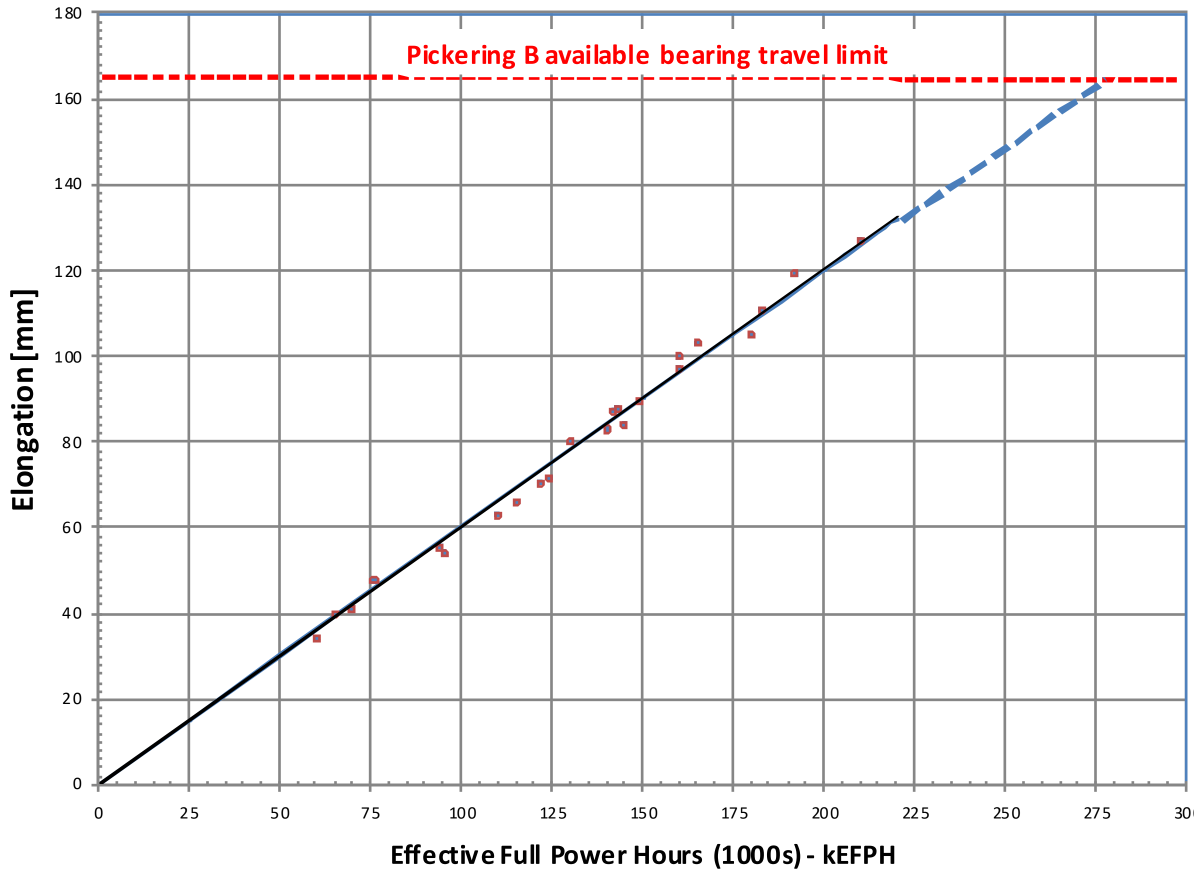


Figure 10: Pickering B Channel elongation data related to EFPH. Note the end of life or Target Operating Life should be earlier than reaching the limits. Data from NK30-REP-31100

Given the reality of having to admit out that the there is really no technical basis for extending the operating life of certain Pickering pressure tubes and having to admit that the real limitation is in axial elongation based on design bearing travel limits for all Pickering units, the CNSC document steps in to glorify the OPG commitments with "*OPG has a well-developed management strategy incorporated into the aging management programs to address this issue*". Point is simple, the channels in Pickering B are way past their useful life and those in Pickering A will need to be 'reconfigured, or shifted and are <u>not</u> fit for service very soon.

But even after 'reconfiguration', a tedious and expensive process conveniently omitted from the license application for Pickering A units, the channels cannot be in service in the period covered by the applicant's application.

One of the biggest deviations from engineering truth by the CNSC and OPG staff has been in presenting acceptable plant life in terms of the so called Effective Full Power Hours (EFPH). There is no credible data on plant life (especially the life of pressure tubes) dependence upon power level. (A human's age is not calculated based on hours she is awake either). As a result the actual 262,980 hours in a 30 year span have now been counted as 212,000 EFPH at 80% power level in RD-360 !!. Actual exposure of pressure tubes to hot D_2O and other oxidizing media and high neutron flux is actually larger than the erroneously glorified 212,000 EFPH. There are NO data to suggest that power level has any linear effect on plant equipment degradation or that neutron flux has any linear effect on pressure tube degradation after an initial burst. Therefore, the plant life extension application, consistent with the industry supported 212,000 hours representing 30 years of plant life as being of any meaning is a big departure from engineering truth to put it mildly.

All though the constructive, innovative, developing years of AECL, the Pressure Tube degradation was measured in hot years, hot hours and not some effective full power hours - EFPH. Figure below, used in an AECL presentation to US NRC in their unsuccessful foray into the US markets shows the deuterium uptakes as a function of hot years. Perhaps they dared not talk about EFPH in front of an intelligent technically savvy audience or perhaps they were still abiding by a different code of professional conduct.

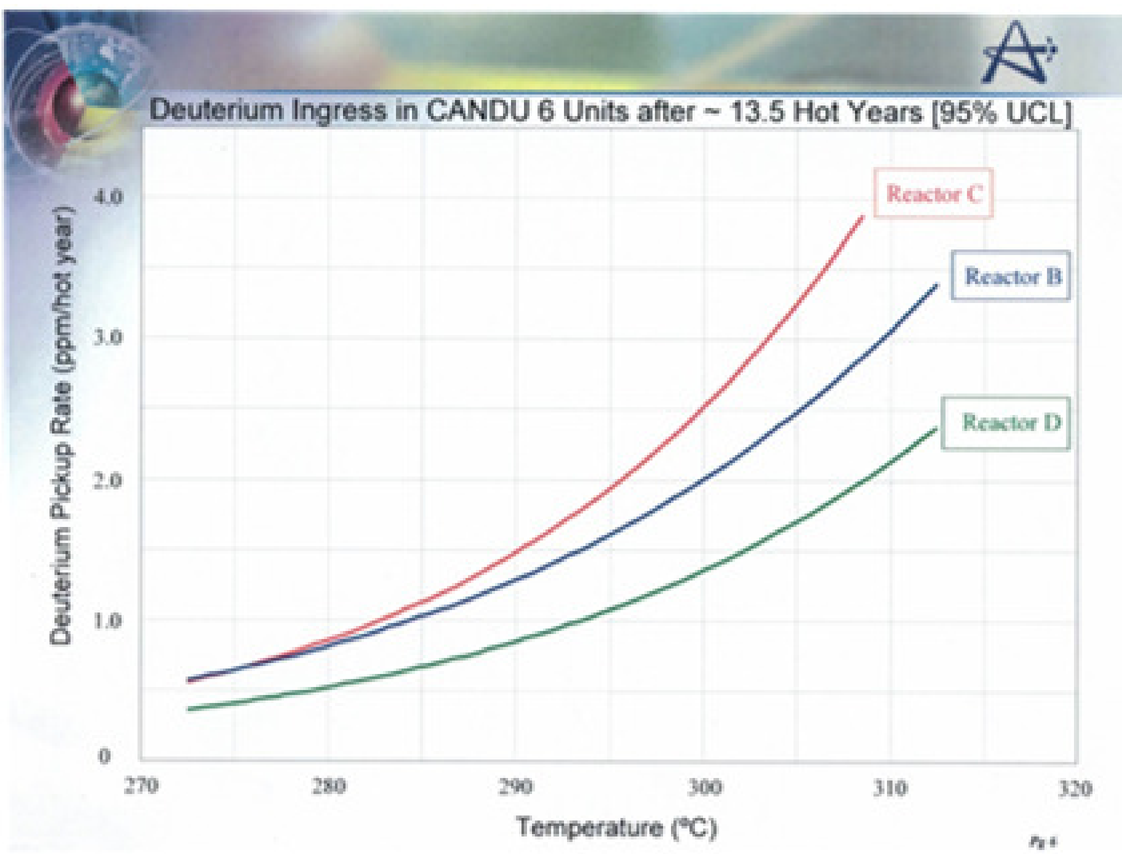


Figure 11: Increase in rate of Deuterium ingress with time. (courtesy : AECL presentation to USNRC DEC 2002, Reference 4)

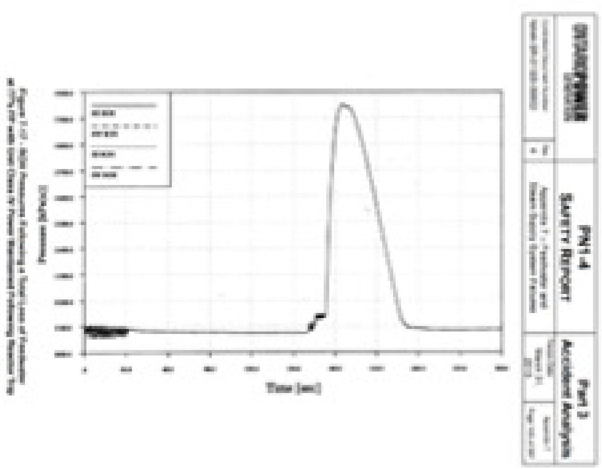
If one used the upper curve in Figure 9 and projected it to 35 'hot' years, one would see that Pickering pressure tubes, albeit operating at marginally lower temperatures, will also well exceed the CSA 'limit', now magically extended by a compliant CSA from 100 ppm to 120 ppm. What is also interesting is that it has always been understood that the rate of Deuterium ingress increases with age. While the rate of ingress decreases with temperature, so does the solubility limit.

This means that now claims for a life extension to 300,000 EFPH are based on alternate truths. Figure 8 from a presentation made by AECL to the USNRC in 2003 illustrate that even after 10 years of EFPY reactor operation the Deuterium concentration exceeded 100 ppm and thus the Pickering pressure tubes are ripe for failure anytime right now.

MAJOR DESIGN ERROR THAT FURTHER

The safety relief valves of primary heat transport system overpressure protection systems at Pickering stations seriously challenge reactor safety norms and pose undue risk to public safety and utility interests. At a combined steam relief capacity of 120 grams of steam (as opposed to equivalent PWRs sporting 120 mKILOGRAMS of steamn telief capacity) , they contravene applicable code requirements and world practices for proper sizing for valve energy relief capacity, unobstructed & adjacent placement to pressure source, redundancy and periodic testing. The liquid and steam relief capacity of the relief valves are functionally inadequate by orders of magnitude; valves are located too far (>40m) from the pressure source; require large accumulation to fully open for steam; the two, already undersized,'50% capacity' valves provide no redundancy and leave no room for single failure; and there is no periodic relief valve testing program like one mandated right after TMI by the US NRC. Components of the pressure relief system have itself caused *at least* 4 LOCAs by spurious actuations and faulty design. It does not take one long to realize that these are dangerous, risk significant design errors that together expose utility and public to significant un-necessary risks. A failure to mitigate an uncontrolled reactor overpressure can initiate pressure boundary failures with serious public consequences. Overall, these systems fail to meet many of the fundamental design criteria and code requirements for an overpressure system at a nuclear power plant. If left unchanged, a simple and a not too infrequent loss of a feedwater injection into a steam generator can cause loss of HTS pressure boundary with huge economic consequences to the utilities and the country. We will present findings of a joint investigative project with the CNSC and the torturous path to resolution of this simple to understand issue over the last 22 years.

Pickering safety report clearly talks about this inadequacy but the learned CNSC staff totally overlooked the simulation results of the following figure.



3. SEVERE ACCIDENT RELATED RISK TO CANADA FROM PHWR CANDU REACTORS AT PICKERING

3.1 Summary

TMI, Chernobyl and Fukushima accidents have exposed vulnerabilities to severe accidents in all reactor designs and nuclear safety cultures. All reactors operating today were designed prior to development of any useful understanding of severe accident instigators, progression and consequences. In addition, issues of willful ignorance, complacency and collusion between the utilities and the regulators have surfaced during the post mortem examination of the 3 severe core damage accidents that have occurred in less than 15,000 reactor years of operation in 3 of the technologically most advanced countries in the world. These accidents have resulted in near demise of the nuclear industry and contamination of thousands of sq. km of pristine land in Ukraine and Japan and, a luxury that Canada cannot afford but seems to risk unknowingly and without the benefit of an impartial regulator or of a good public discussion. Industry regulator CNSC and reactor owner OPG have known of the issues for years but have done nothing meaningful. Other Canadian reactor operators benefitting from the same lax regulator have done nothing as well.

The Pickering CANDU reactors have additional vulnerabilities that are specific to horizontal channel PHWR design. For example, with no LWR like pressure vessels to isolate the core debris, immediate discharge of un-attenuated radioactivity directly into containment would occur as soon as a fuel / core damage starts. The reactor cores have far too much Zircaloy (~ 40000 kg) in fuel channels and too much carbon steel (> 8 km of cheap carbon steel feeders with over 1800 m^2 of surface area) in feeders that would produce flammable deuterium in amounts that would be unavoidably explosive in short order and cause reactor building breeches exposing the unsuspecting population to radioactivity long before any evacuation can be affected.

The reactor process systems including the PHTS, moderator, shield tank have inadequate over-pressure protection for severe accident thermal loads and thus vulnerable to uncontrolled ruptures and containment bypass (see also 5) . The Pickering reactors have a relatively segmented and small containment with various locations for trapping gases and an early failure potential at poorly designed door seals. Inevitability of early failure of containments and of reactor structures and release of huge amounts of activity outside the reactor boundary is easy to demonstrate by responsible analyses.

The Pickering Nuclear Reactor represents a 50 year old technology that is not licensable today in any new jurisdiction in the world. The US NRC refused to even consider for licensing even a much newer version, the Advanced CANDU Reactor (ACR 700), developed about 30 years later. The refurbishment process for Pickering reactors did not update the design of the reactor in any significant way and the abysmal severe accident mitigation capabilities of the original design remained largely unchanged. Opportunity to upgrade these reactors after Fukushima have also been largely missed and the few design enhancements such as Passive Autocatalytic recombiners (PARS) are actually dangerous because there are not enough of them.

Independent deterministic analyses of system response and severe accident progression after a station blackout scenario for CANDU reactors have unveiled a number of design vulnerabilities that cause uncontrolled pressure boundary ruptures; premature expulsion of coolant from main loops and from the moderator heat sink; direct exposure of core debris and fission product releases to the containment; thermo-mechanical failure of the thin shell Calandria vessel welds; accelerated production of hydrogen with containment boundary failures by steaming as well as by sparsely populated PARS units potentially exposed to high concentration deuterium / hydrogen they are unable to adequately mitigate.

A number of design enhancements can however still be undertaken to minimize risk from a severe accident by eliminating or avoiding some of the undesirable system responses. This has to be a national priority and not a research topic or a political tussle. Discussion below walks through an accident scenario, highlights the risks and offers suggestions for improvement (see Reference 6, 7 as well).

Of equal concern is the lack of emergency planning for a realistic severe accident fission product source term. CNSC has provided a source term to EMO that is wrong and exposes their management to charges of willfully misleading the public and exposing the police and other personnel to undue radiation exposures in case of an accident they must help mitigate.

2.2. Station Blackout Scenario to Illustrate Risk.

All jurisdictions with responsible regulatory regimes require that progression of accident and consequences of a Station Black Out (SBO) scenario be evaluated to demonstrate effectiveness of existing systems and containment structures for at least 24 hours. The following summarizes some of the Pickering safety issues by stepping through such a scenario. It is shown that consequences to Korea can exceed those at Fukushima, an accident initiated by nature but considered totally avoidable and blamed on human errors including regulatory incompetence and industry complicity, & collusion for its consequences (reference 8, 9):

1. *After all AC power is lost, the reactor trips and reactor thermal power drops to about 5% in 5 seconds, 2% in about 20 minutes and 1.5% in about 1 hour.*

2. *Feedwater injection into the boilers drops and then stops a few minutes after loss of power. Steam relief through MSSVs starts. Heat transport system depressurizes to just above the secondary side pressure but continues to circulate coolant through the fuel & boiler tubes due to density difference induced flows so that fuel remains adequately cooled.*

3. *Boilers inventory of light water is depleted due to removal of stored heat in the fuel and piping, losses through inlet check valves and by any continuing boiler blow-down along with decay heat removal. The boilers remain an effective heat sink as long as they have sufficient water estimated to be more than ~4m height in each boiler. Thereafter (~1.5 hours) the heat transport system re-pressurizes.*

4. *<u>A high pressure steam driven turbine (installed in PLGS) can help avoid boiler dryout without operator action.</u>*

5. *At this time the first unintentional error in CANDU design becomes critical. The system re pressurizes and attempts at this time to avoid an over pressure by rejecting the decay heat through safety relief valves but an inadequate steam relief capacity leads to a continued over pressurization[1].*

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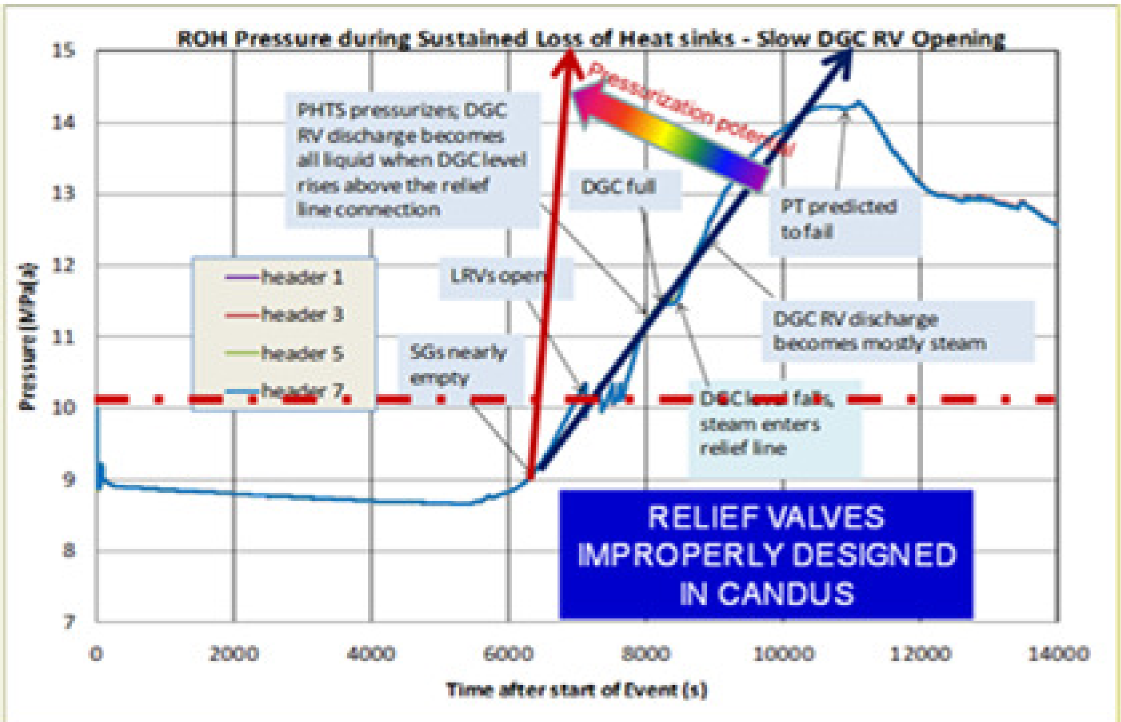
1. [1] *These pressure relief valves were reportedly properly designed in the original Pickering units but erroneously mis-sized in 1996 after a knee jerk reaction (and poor engineering decision) to a 1995 event at Pickering in Canada.*



Source : AECL 2011

Figure 12: PHTS over pressurization after the relief valves cannot cope with steam load

6. So, a boiler dryout leads to an unusual for a nuclear power reactor, over-pressurization of the Heat Transport System and an unavoidable, uncontrolled failure of a pressure boundary component. The failure is most likely to be in ever so vulnerable boiler tubes, resulting in a potential containment bypass and early population exposure to fission and activation products. Analyses at AECL and a recent COG report point to a potential failure of a fuel channel instead of a bunch of boiler tubes. There is ample data to dispute that outcome. In any case, in a properly designed reactor, <u>nothing</u> should fail upon an over-pressure transient.

7. This unplanned rupture of the pressure boundary occurs long before there is any severe core damage and a benign outcome that can be terminated by ECC, transforms into a serious accident whose economic consequences can be prohibitive even if a subsequent mitigation, for example by ECC injection upon this forced depressurization, is successful.

8. The uncontrolled failure can also be at any other location within the heat transport system. It could be in the pump or piping. Were it to occur at a fuel channel the effects can be catastrophic economically as a high pressure in-core rupture can cause extensive damage to other channels and in-core devices. Onset of a severe core damage will then likely be accelerated by draining the moderator with a potential end fitting ejection following a channel rupture. <u>Properly designed relief valves can help avoid uncontrolled rupture of HTS boundary.</u>

9. Even with proper safety relief valves installed, a gradual depletion of PHTS inventory will soon reach a state that no amount of restoration of water into the boilers will see restoration of natural circulation (thermo-syphoning) through the channels. <u>Depletion of only about 20% of PHTS inventory can cause a state of no return to adequate core cooling in CANDUs. Identification of this point is important for proper severe accident management and proper instrumentation is required.</u>

10. With boilers no longer a heat sink, gradual voiding of individual fuel channels and sequential onset of fuel heatup in the 380 fuel channels (depending upon individual feeder size and channel power) leads to heatup of the heavy water moderator and light water in end shields and shield tank.

11. A partial voiding of the Calandria vessel occurs as actuation of rupture disks cause partial moderator expulsion upon onset of boiling. This exposes a number of fuel channels and accelerates their disassembly. <u>A properly designed relief valve on the moderator could delay onset of severe core damage and provide an operator additional opportunities to avoid a disaster.</u>

12. <u>An emergency measure high pressure injection of water into PHTS is not available and there is no way of manually depressurizing the heat transport system without jeopardizing the heat sink (so avoid boiler crash cool).</u> Coolant inventory in the reactor continues to deplete.

13. An initial high pressure failure of an overheating channel into the moderator can also expel a part of the liquid moderator by carryover if the initial overpressure induced failures in boiler tubes rupture just enough tubes to relieve the stresses but maintain high PHTS pressures. A properly designed PHTS relief valve would also maintain high pressure in the system and an initial high temperature failure of a fuel channel at high pressures cannot be precluded. Combined with other design changes accident can be easily maneuvered to end favorably but not so in the current design.

14. Overheating channels, fed by steam circulating through the heat transport system also contribute to a natural consequential heatup of downstream end fittings and feeders. Different channels void at different times depending upon their decay power and volume of water in their feeders.

15. With some channels exposed following moderator depletion and losing all significant heat sinks, conditions form for accelerated fuel bundle overheating, deformations and bundle dissociation at low pressures. For all channels, the downstream end fittings and insulated feeders start oxidizing upon heatup by high temperature steam exiting channels.

16. A breech of a channel within the moderator space creates path for interaction of moderator water with dry channels and for a long time thereafter steam is supplied by the underlying moderator for fuel bundles and feeders to oxidize.

17. Channel segments begin to disassemble and supported by underlying channels and constrained by in-core devices continue to cascade down and heatup during holdup periods.

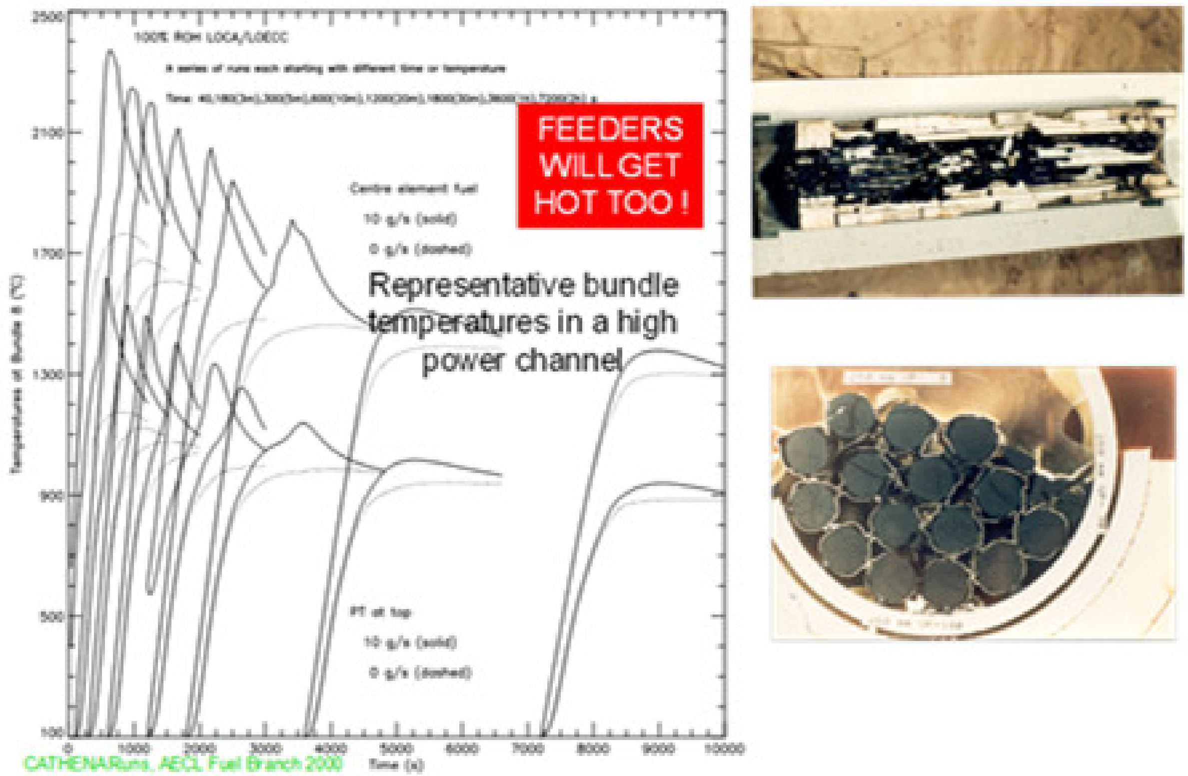


Figure 13: Channel heatup as a function of onset of heatup. note fuel bundle geometry after significant heatup.

18. Internal sources of water remaining in the end fittings, pump inlets, fuelling machines also contribute to oxidation of fuel and feeders. In addition water, steam ingress from the moderator into disassembling channels will contribute to steam availability.

19. As the fuel in the channels begin to heatup so do the end fittings and feeders. Oxidation of feeders starts at about 550^oC while fuel oxidation starts at about 800^oC. Over 7 km of carbon steel feeders provide over 1800 m^2 of carbon steel surface area for oxidation. Carbon steel oxidation to FeO /Fe_3O_4/Fe_2O_3 (in 95/4/1 ratio of Wusite, Magnetite and Haematite) is faster than that for Zircaloy at the same temperatures and the iron oxides have a propensity to peel off and expose fresh steel carbon surface for accelerated oxidation. Stainless steel end fittings also join in the oxidation process, albeit at a rate that is at times 10 times slower. Part of end fittings also include a heat sink to the end shields. Heatup of feeders will likely start fires in the feeder cabinets.

20. Flammable gas production from carbon steel oxidation may well exceed that from Zircaloy oxidation, especially for low power channels that do not disassemble but continue to circulate dry steam and oxidize the feeders over a long period of time.

21. With no pressure vessel to completely isolate the hot fuel from the containment, the overheating fuel & channel debris heatup further and their uncovery in steam over next few hours results in a direct expulsion of un-attenuated fission products into the containment. Fission product release may overheated fuel may be fast (~1%/min for Cs-137) and release of large fraction of fission products into the containment inevitable.

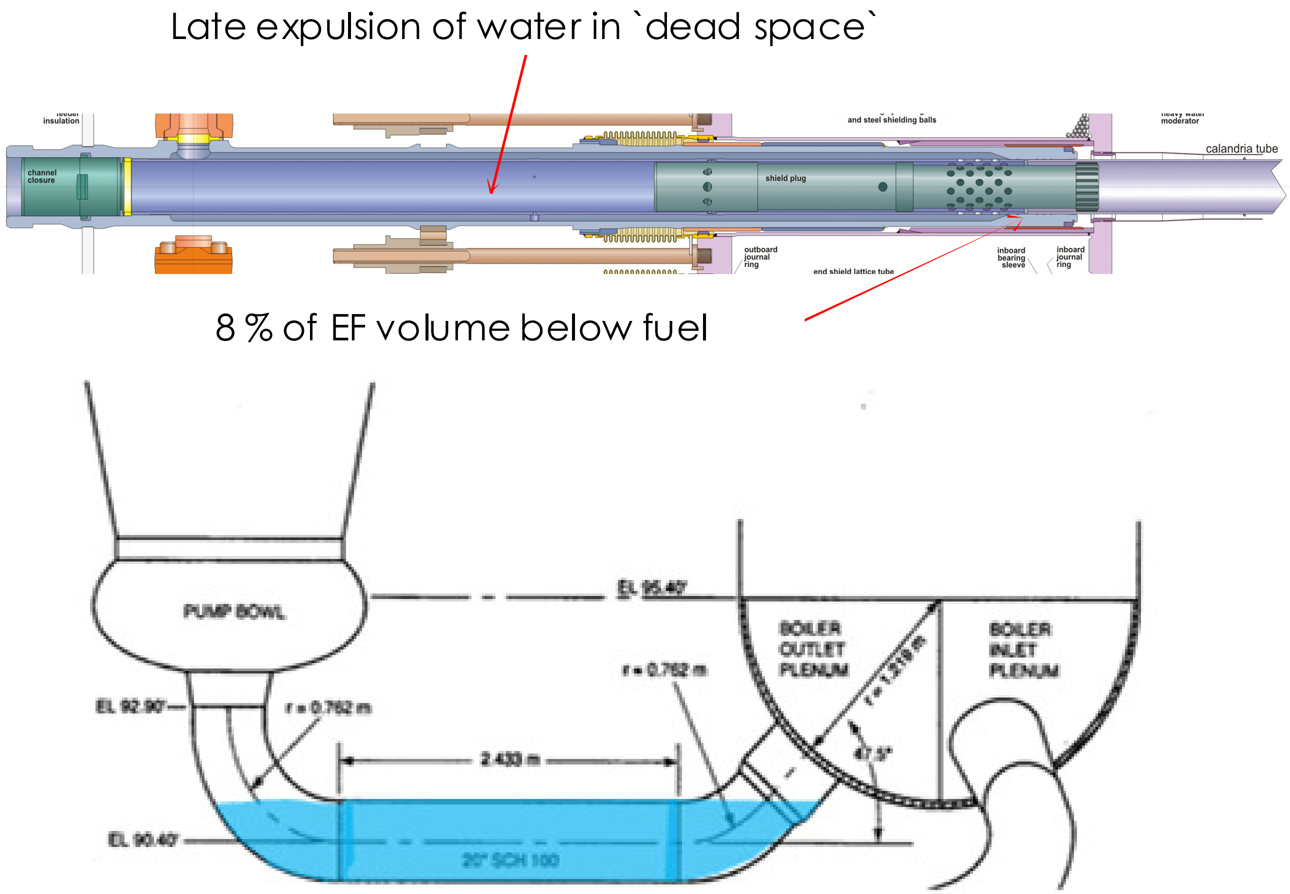


Figure 14; Water holdup in end fittings and piping as source of fuel and feeder oxidation. Moderator water also a source after channel perforation.

22. *More deuterium (isotope of hydrogen) is produced by intact carbon steel feeders than by intact fuel bundles. This has been confirmed by analyses using a new computer code ROSHNI.*

23. *Containment integrity becomes an important safety concern. Fuel sheath failures cause the free inventory of fission products to release followed by diffusion releases from grain boundary and grain bound species. All fission products find an easy path to the reactor building. Releases to the environment, accounting for settling and re-volatilization inside the building, depend upon time at which building failure is initiated.*

24. *Feeder oxidation by air and steam is exothermic and the heatup initiates fires in the feeder cabinet insulation. This also triggers burns and explosions of the Deuterium generated in the channels and released from failed channels into the Calandria vessel and ultimately into the small reactor vault.*

25. *Pickering reactors will have special issues with capture of flammable Deuterium in the fueling machine vaults.. As the containment pressure settles to just over atmospheric pressure and intra compartmental air flows subside, the release of Deuterium into the vault will occur through the Calandria vessel rupture disks. The flammable gas will tend to accumulate inside the vault and reach very high local concentrations.*

26. Analyses confirm that the whole CANDU core cannot just collapse down after a certain amount of debris have formed. The erstwhile MAAP-CANDU assumption of a 'core collapse' is a convenient way of decreasing source term to please ourselves. It is the channels that do not fail that contribute most to hydrogen source terms, ROSHNI analyses now reveal. A large number of fuel bundles (~33%) may remain in stubs at the end of channels that do not experience rolled joint pullout. Oxidizing feeders in channels that disassemble will cool down relative to feeders in channels that remain intact.

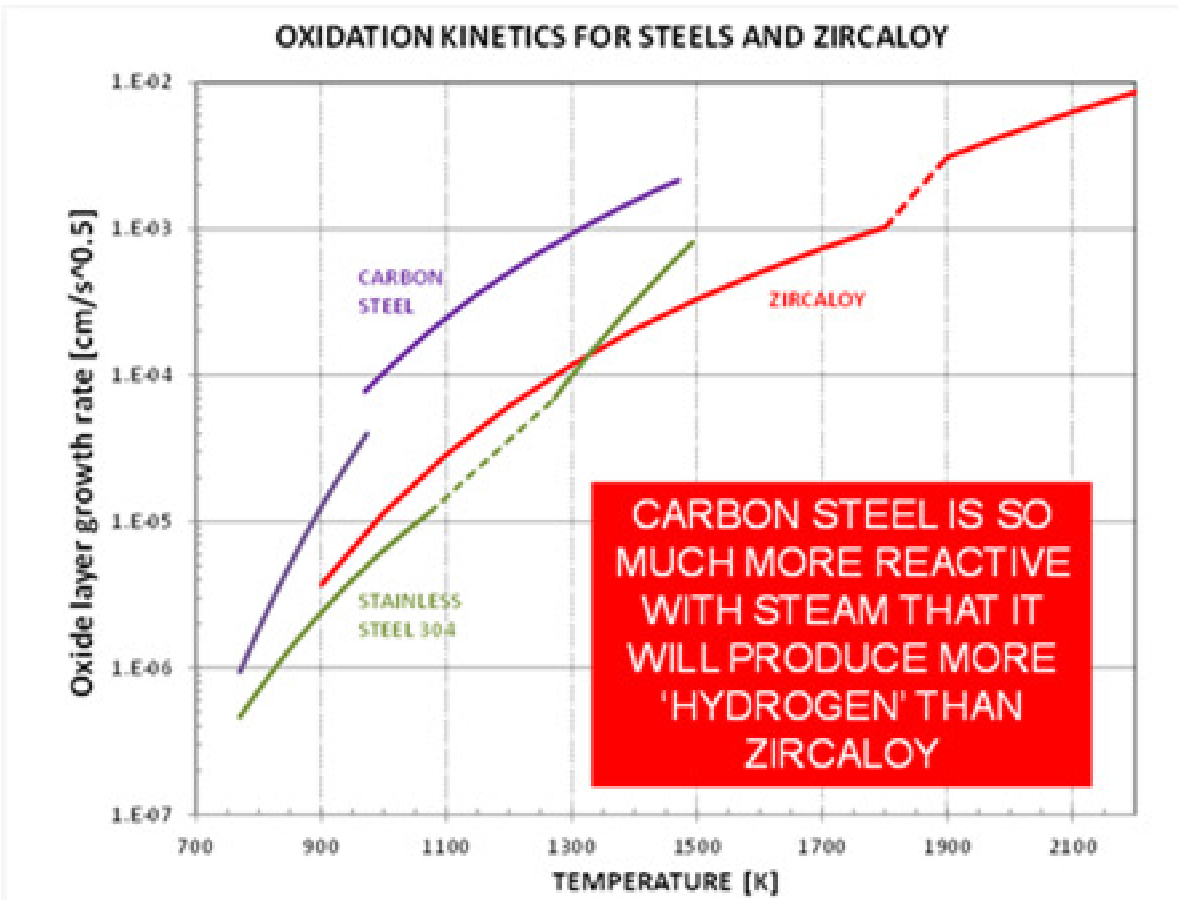
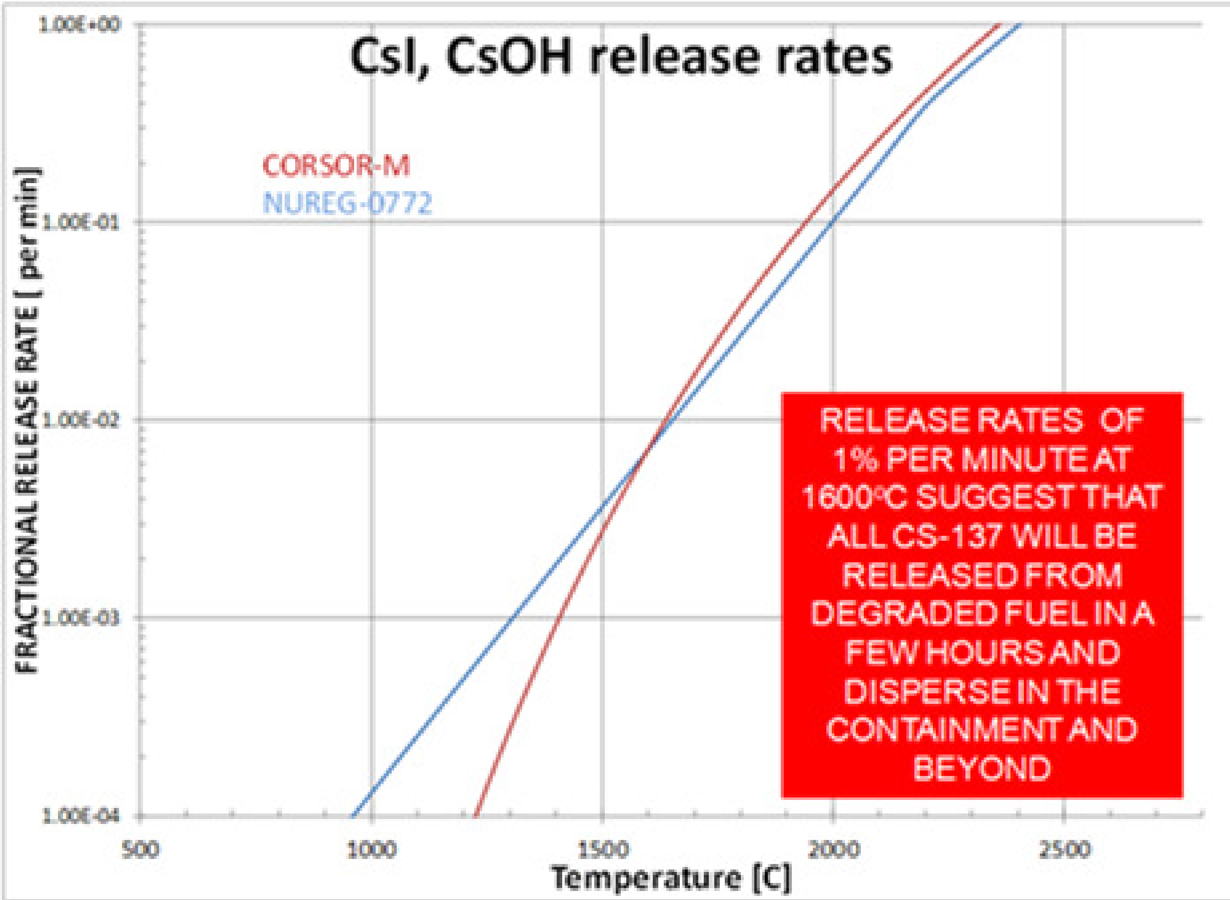


Figure 15: Oxidation rates of different channel materials

27. Deuterium gas releases into the containment readily exceed the local detonation limits as the small (19) number of passive recombiners, where present and interactive to the stream of combustible gas, are not only unable to arrest the increase of deuterium concentration but also introduce additional ignition potential leading to gas detonation at concentrations above 5 to 6%.

28. The debris formation in a CANDU reactor is in solid chunks of channel and its eventual retention upon melting in the Calandria vessel cannot be guaranteed as the relatively thin walled stepped and welded vessel (wall thickness varying between 19 and 28 mm) may fail at welds thus introducing water from the shield tank onto hot debris.



29. The effect of Calandria vessel weld failure can vary from additional hydrogen production, accelerated FP releases as one mode of outcome to catastrophic vessel failures by energetic interactions with the hot and molten solid-liquid debris at the bottom of the Calandria vessel as the other mode.

30. Shield tank relief valves cannot remove decay heat equivalent in steam as they are designed for a smaller gas relief capacity. An onset of boiling in the shield tank has a potential to cause it's failure.

31. Reactor building failure at any one of 2-3 different events coincident with energetic interaction of fuel and water is possible.

32. Inevitable failure of thin walled Calandria vessel will cause water from the shield tank to energetically react with debris and cause structural failures in these vessels as well as the containment structure mechanically joined to them and just overhead.

33. Large releases of activity into the environment are inevitable.

Opportunities to arrest the progression of accident early can only be availed by significant immediate investment into understanding the accident progression and instituting design changes to incorporate intelligent recovery actions. There are a number of other design, analysis and accident management enhancements that have not been discussed here but have been presented elsewhere (see a summary in section 2.3). These include improved instrumentation, improved off-site detectors, improved operator training by simulators, replacement of carbon steel feeders by stainless steel feeders during any refurbishment, stronger containment gate seals, deuterium specific instrumentation and mitigation, etc.

Far too much emphasis has been placed on Level 1 PSA in Canadian risk assessments and the actual processes required to evaluate accident progression (research, code development, analyses) have been neglected in deference to speculative hyperbole about CANDU superiority. Related progress in other CANDU utilities is also problematic.

3.3. ISSUES RELATED TO STORAGE OF SPENT FUEL POOLS

The Spent Fuel Pool and its storage racks serve the safety functions of cooling the irradiated fuel bundles removed from the reactor, while maintaining them in a sub-critical configuration and providing safe means of their reception, storage, and removal to a longer term storage solution. The potential for fuel heat up and subsequent Zircaloy fires in overheating fuel assemblies upon a sustained, accidental loss of cooling water from spent fuel storage pools is a safety concern and has required special investigations for all reactor designs after Fukushima.

The CANDU fuel bundles are rather densely stored in stacked a 'fish-basket' like relatively tightly packed, congested configurations that may severely limit the potential heat removal by air natural circulation after a loss of liquid water envelope. A potential exists for significant off site doses, should an accidental and sustained uncovery in air of spent fuel bundles cause energetic reactions leading to sheath failures and releases of radioactive fission products.

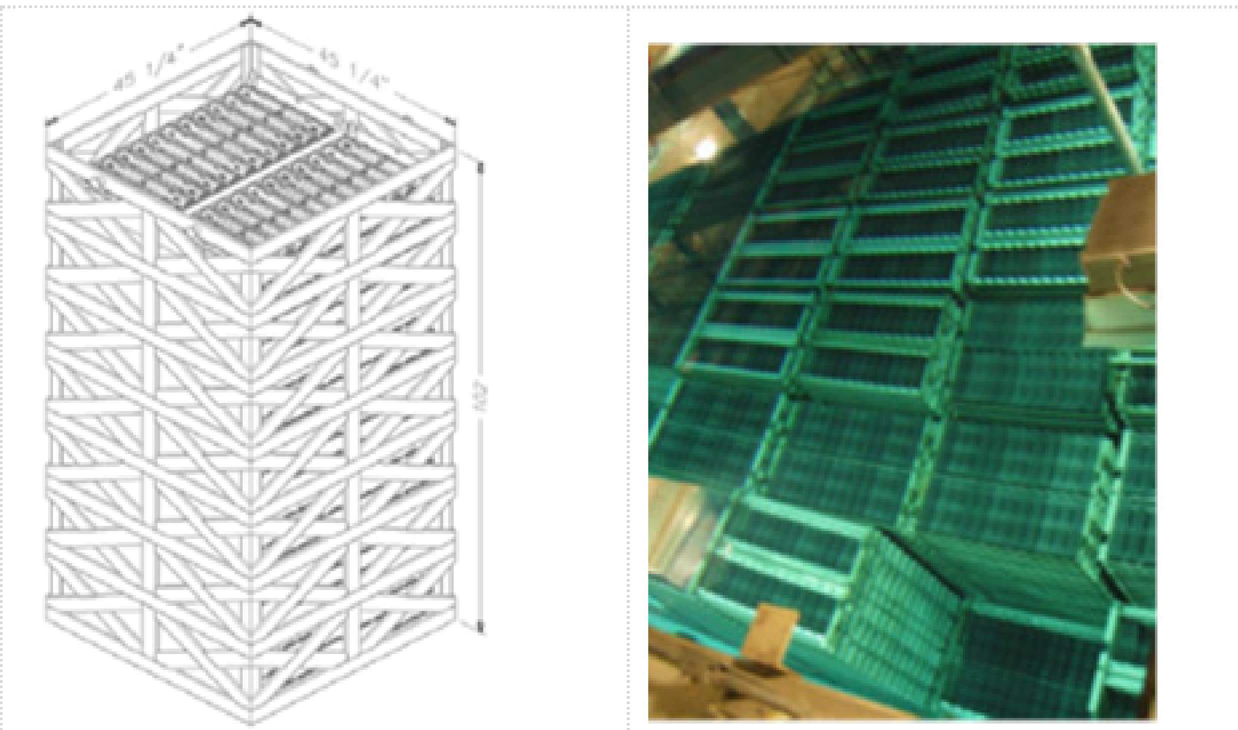


Figure 16 : CANDU fuel tray towers - structure and typical placement within the pool. Pictures not from Pickering.

For PHWRs, related evaluations have largely been qualitative and dismissive, citing the substantially lower than that for LWR decay heat owing to their substantially lower burnup. However, freshly discharged fuel bundles are added almost daily to the pool. Similar to LWRs, their dense horizontal stacking in tightly packed vertically piled trays creates a potential for adverse consequences following an accidental, sustained fuel uncovery in air. This also makes the effectiveness of any recovery / mitigation measures likely to be very challenging. Thermo-chemically significant oxidation of zirconium in air starts early (at below 600° C) and is about twice as energetic and appreciably faster than the zirconium oxidation in steam or even pure oxygen. Early nitrogen reactions (also exothermic as with oxygen) produce porous sheath nitride layers, degrade the fuel-cladding's "protective" oxide layer, and accelerates the oxidation. In addition, an energetic, runaway zirconium oxidation signifying a loss of protective layer will start in the 850°C range in air as opposed to the 1150°C range for steam. The resulting sharp increases in Zircaloy temperatures and accelerated exothermic oxidation are described as 'Zircaloy fires' that may propagate to adjacent fuel bundles by energy transfer through conduction, convection, and radiation. This may cause cascading fuel failures and large environmental releases of radioactivity, likely more severe than from a severe accident in the reactor itself.

Pickering spent fuel pools pose a much greater risk than that for single unit reactor stations.

3.4 OPPORTUNITIES FOR DESIGN IMPROVEMENT FOR ENHANCED SEVERE ACCIDENT MITIGATION

The following list of generic CANDU related design improvements were proposed for Pickering as well; none were dispositioned in a professional manner.

1. Further reduce the likelihood of a station blackout scenario that starts with a loss of off-site power or a malevolent act.
2. Reduce the likelihood of events and failures that create permutations of failures that may lead to severe core damage accident from other internal and external events
3. Reduce the likelihood of incidents progressing to a core damage state by measures such as external and internal hookups for adding power and water; daerator hookup.
4. Reduce the likelihood of an uncontrolled rupture of heat transport system pressure boundary at the onset of boiler dryout in case of a station blackout as at Fukushima.
5. Correct the inadequacy of heat transport system over pressure protection
6. Reduce the likelihood of containment bypass in boilers
7. Reduce the likelihood of containment failure by pressure, temperature, radiation and fluid/gas interactions with containment penetrations given that certain reactor units have weak confinement structures and no pressurizable containments.
8. Evaluate and document the effect of recovery actions including power restoration, water injection as a function of time since onset of core damage
9. Install additional and independent of that available before Fukushima, instrumentation to detect and help control the progression of a severe core damage accident
10. Reduce likelihood of recovery actions exasperating the accident consequences by enhanced severe accident specific instrumentation and display of state of the reactor
11. Reduce likelihood of fuelling machine adversely affecting the outcome upon restoration of cooling functions
12. Modify Calandria vessel overpressure system to avoid fluid loss through rupture disks; delay onset of severe core damage
13. Modify moderator cooling system to install recovery system hookups for inventory replenishment and reinstatement of cooling functions
14. Investigate potential of in-situ design enhancements to avoid Calandria vessel failure by hot debris to avoid catastrophic failure of reactor structures
15. Increase the likelihood of successful external water injection by manual depressurization of the heat transport system
16. Increase the likelihood of core inventory degradation by ultra high pressure water addition to pressurized HTS before core degradation and prior to an in-core rupture
17. Increase the likelihood of reactor heat transport system heat removal by thermosyphoning by adding systems to remove non condensable gases that can degrade thermosyphoning

18. Reduce the likelihood of ECC injection failure
19. Modify shield tank over pressure protection system to conform to anticipated heat loads to avoid catastrophic failure of shield tank vessel.
20. Install hookups for water addition to the shield tank
21. Obtain a more realistic evaluation of accident progression by using analytical methods that are more modern than the MAAP4-CANDU code that is 25 years old and obsolete in light of new information; and model the event with :
    - More detailed modelling of reactor core by differentiating between different bundles by modelling all reactor channels and in-core devices
    - More appropriate modelling by using D_2O properties
    - More appropriate modelling by evaluating Deuterium (D_2) gas production, transport, recombination and burns. Has the utility considered that Deuterium gas properties differ greatly from hydrogen (H_2).
    - Considers oxidation of end fittings and feeders as sources of flammable D2 gas during a severe accident
    - Consider a more representative inventory of fission products
    - Consider concurrent fires (e.g. In feeder cabinets) as core voids, heats up and degrades
    - Consider failure of Calandria vessel at welds with hot debris
    - Consider failure of Calandria vessel penetrations at the bottom of the vessel (moderator outlet)
    - Consider explosive interaction of water with melt in Calandria vessel
    - Consider explosions caused by interaction of deuterium gas with PARS
22. Consider alternate hydrogen mitigation measures as PARS may become ignition sources; consider upgraded catalyst plates with electrolytic deposition that limit gas temperatures.
23. Installation of measures to avoid ignition in existing PARS
24. Consider D_2 mitigation system optimization for a 100% Zircaloy oxidation (also to include effect of feeder oxidation)
25. Consider enhanced deuterium concentration monitoring systems within containment and Calandria vessel
26. Consider advanced video surveillance systems
27. Consider measures for mitigation of consequential fires during the progression of core disassembly
28. Consider post accident monitoring system instrumentation and control survival and functionality for severe accident conditions
29. Consider emergency filtered containment venting for severe accident loads
30. Consider improvements to pressure suppression system in reactor building as the vacuum building may be inadequate to avoid building failure for multi unit accidents
31. Consider reactor building reinforcements to avoid building failure; special emphasis on confinement on top of reactivity decks in multi unit station

32. Consider deploying on-site and off-site radiation detection equipment that actually detects the source characteristics and differentiates between incident radiation species by measuring the energy of incident radiation; does not get saturated by incident particulates as happened for Chernobyl at Leningrad station a thousand km away.
33. Develop methods and acquire instrumentation to help deduce source terms from radiation measurements so that prediction of radiation effects can be made for different locations and changing weather conditions
34. Develop simulators to train the operators in progression of a severe core damage accident and develop experimental basis & analysis to help avoid potential adverse outcomes of various mitigation measures

4. CONCLUSIONS

A serious re-examination of Pickering safety case for severe accident mitigation is necessary unless Fukushima like accidents with long term consequences are a publicly acceptable risk to a short period of generating electricity from Pickering. Ignoring the well-known issues and chronic inaction is not an acceptable option.

A denial of application for life extension is the only viable option that serves public interest. This will make OPG instill design changes with comprehensive new programs for training, analyses and research for their reactors at Darlington before a future CNSC shuts them down for a similar lack of safety improvements for reduction of risk from severe accidents and ageing.

It is the obligation of the new CNSC Commission to act solely in public interest. The intervention allows only a brief interaction with the Commission but I will be happy to make myself available for a day-long seminar on all the technical issues that compel me to recommend that Pickering reactors need to be retired gracefully and retired now. The risk benefit ratio is just far too high.

I also request you, the Commissioners to recommend that OPG begin dismantling of these reactors immediately and certainly well within the life time of the remaining people who designed them and built them. My OPG colleagues who have operated them so brilliantly, in spite of a number of unhappy incidents, equipment failures, unit closures and design shortcomings, will do best to help remove them from service safely without delay.

CNSC members must be mindful of the effect their decision can have on the destiny of my country for the mere convenience of an automatic license renewal for this aged, obsolete, dangerous reactor at the edge of Toronto. I have done my part and you must do yours now with due diligence.

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