Submission to the CNSC Public Hearing on NB Power's Application to Renew the Reactor Operating Licence for Pt. Lepreau NGS

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The Pt. Lepreau Nuclear Reactor, although refurbished just about 5 years ago, 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 newer version, the Advanced CANDU Reactor (ACR 700), developed about 30 years later. The refurbishment process in New Brunswick did not update the design of the Pt. Lepreau reactor in any significant way and the abysmal severe accident mitigation capabilities of the original design remained largely unchanged. It should therefore not be relicensed in New Brunswick and the current application for a 5 year license extension should be denied.

Independent deterministic analyses of system response and severe accident progression after a station blackout scenario for CANDU6 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 be undertaken to minimize risk from a severe accident by eliminating or avoiding some of the undesirable system responses. NB Power staff contributed to a COG report that denied ALL suggestions I made for design and process improvements to the Pt. Lepreau reactor for improved mitigation of severe accidents focused to reduce liability to NB Power stakeholders as well as risk to public. In doing so they perpetuated a number of untruths that will be challenged. The CNSC should deny a license extension based on the apparent lack of safety culture at NB Power that allowed such irreverent attitude to reactor safety and public welfare.

TMI, Chernobyl, Fukushima accidents have exposed us of vulnerabilities to severe accidents in all reactor designs as all nuclear reactors operating today were designed prior to development of any understanding of severe accident instigators, progression and consequences assessments. The Pt. Lepreau CANDU reactor has additional vulnerabilities that are specific to horizontal channel PHWR design. For example, with no PWR like pressure vessels to isolate the core debris and would thus immediately discharge un attenuated radioactivity directly into containment as soon as a core damage starts; 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; the Pt. Lepreau has a relatively segmented and small containment with various locations for trapping gases and an early failure potential at door seals; and the reactor cores have far too much Zircaloy (~ 43000 kg) in fuel channels and too much carbon steel (> 8 km of 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. Inevitability of early failure of containments[1] and of reactor structures and release of huge amounts of activity outside the reactor boundary is easy to demonstrate.

The Pt. Lepreau reactor represents a design that is very poor in its mitigation capabilities for severe core damage accidents and thus is not consistent with the current expectations of risk by public. It is expected that the Commissioners should by now be familiar with the vulnerabilities of these reactors to large releases from the core into the containment, large production of flammable gases and early potentials for containment failure and thus large releases of activity into the atmosphere given that an in-vessel retention of activity from disassembled fuel (as in a typical PWR) is absent in a CANDU reactor that forces a direct, un-attenuated expulsion activity into the containment after channel disassembly. Economic consequences of a severe core damage accident at Pt. Lepreau, as erroneously claimed for Ontario reactors, may not be small for New Brunswick, Nova Scotia or the State of Maine and the capabilities of the Emergency Management Organizations in these jurisdictions are too limited for the scale of damage that a simple accident such as a Station Blackout can be caused by the obsolete reactor unit at Pt. Lepreau.

Should the CNSC feel compelled to ignore public safety concerns outlined above and feel compelled to give a short term license extension, it should consider the following:

- CNSC commissioners should require New Brunswick Power to demonstrate that the Pt. Lepreau CANDU reactor has been comprehensively analyzed for it's transient response to events that lead to severe core damage accidents and that accident progression, source terms for flammable gases, fission products, energetic interactions as well as off-site health and economic consequences have been analyzed considering all hazards and full detail, using state of the art technology.
- New Brunswick Power must also demonstrate that all necessary mitigation measures have already been taken consistent with challenges resulting from largest flammable gas, fission product, energy releases from the reactor core to ensure maximum risk reduction and that all possible avenues of risk reduction have been thoroughly examined and independently confirmed in interest of public safety. Unless the Commission members and the New Brunswick Power management totally absolve themselves of the responsibility vested in them, necessary upgrades to Pt. Lepreau reactor and a serious re-evaluation of accident progression leading to simulator development and direct operator training in severe accident issues should be a condition to their continued operation under a new limited time licence renewal. As a refurbished reactor, Pt. Lepreau should meet advanced risk reduction requirements, design requirements and risk targets significantly more detailed than now.
- New Brunswick Power should demonstrate that the operator training included severe core damage accidents on simulators that include effects of recovery actions.
- New Brunswick Power should demonstrate that off-site emergency planning basis have been developed from large releases of core inventory consistent with the containment source term of

[1] Containment failure pressure for single unit containment has never been properly evaluated; especially with consideration of ageing issues. The Air Lock door seals are designed for only about 124 kPa(g) and may fail earlier with elevated air temperatures after severe accidents.

over 50% of core inventory of risk sensitive fission products released into the containment in the first day and a large fraction of that released into the environment. This level of releases have been confirmed by independent analyses by the intervener and a paper published by a team of CNL/CNSC engineers in late 2016.

- New Brunswick Power should demonstrate that that an early containment failure based on unavoidable hydrogen explosions due to inadequate hydrogen mitigation equipment at Pt. Lepreau and due to early failure potential of the access door seals have been included in the planning basis of the off-site emergency measures.
- New Brunswick Power should demonstrate that radiation detection and emergency planning measures including evacuations and KI pill distributions have been developed for Nova Scotia as well which is about 60 km from the Pt. Lepreau generating station. The responsibility for emergency planning should lie with New Brunswick as the Emergency Planning Nova Scotia may have no expertise or financial resources for a nuclear emergency.
- NB Power must demonstrate that it has internal expertise independent of others taht is able to understand reactor response to severe accident initiators.
- New Brunswick Power should demonstrate that it has understood and applied the lessons learnt by the Japanese from the still ongoing consequences of the Of the March 2011 accident at Fukushima. Of many investigations that followed, the one by the National Diet (parliament) of Japan Nuclear Accident Independent Investigation Commission (reference 1) stands out in its conclusions:

  > *The TEPCO Fukushima Nuclear Power Plant accident was the result of collusion between the government, the regulators and TEPCO, and the lack of governance by said parties. They effectively betrayed the nation's right to be safe from nuclear accidents. Therefore, we conclude that the accident was clearly "manmade." We believe that the root causes were the organizational and regulatory systems that supported faulty rationales for decisions and actions, rather than issues relating to the competency of any specific individual.*

- New Brunswick Power should use an independent forum to demonstrate that its interactions with CNSC have nothing in common with the above conclusions.
- New Brunswick Power should use a competent, independent, impartial, external review organization to demonstrate that it does not and/or not likely to suffer from any of the root causes pointed out by review (reference 2) that summarized the root causes of Fukushima as:

  - Institutional and regulatory failure
  - Inappropriate safety culture; over confidence on NPP safety
  - Insufficient expertise with decision makers
  - Insufficient understanding of severe accident phenomenology & progression
  - Improper accident management
  - Improper and insufficient understanding of reactor conditions
  - No timely advice sought or available from external experts
  - Insufficient exchange/transfer of information among and within organizations

- New Brunswick Power should document and have an independent forum certify the measures taken by it for
    - Strengthening of safety culture, including an independent assessment system
    - Practical countermeasures against severe accidents
    - Improvement of NPP procedures, covering up to extreme severe accident scenarios
    - Enhancement of NPP instrumentation
    - Improvements in diversity & reliability of emergency power supply systems
    - Reliable decay heat removal by strengthening passive safety
    - Improvement and strengthening of defense in depth strategy
    - Increased contributions to effective nuclear safety research and sharing of research outputs
    - Enhancement of its understanding of regulatory standards
    - Contributions to strengthened independence & expertise of regulatory organizations
    - Emphasized role and enhanced capability of operating organizations
    - Enhanced operator training

If New Brunswick Power is unable to demonstrate in good faith that they have acted expeditiously and without reservations in this matter, a licence extension should be made contingent upon their addressing severe accident related weaknesses in design and preparedness within a specified, but short period of time (~6 months).

It is hoped that CNSC and the utility will finally review their commitment to public safety and undertake concrete actions rather than the smoke and mirror, show and tell attitude of hoping that no technical challenge to their decision of doing as little as possible, is forthcoming.

Whether the two entities (New Brunswick Power & CNSC) can recognize previously un-availed opportunities in increasing station safety from ideas raised by this and all interventions will decide whether public interest is safeguarded or New Brunswick Power is again rubber stamped a licence extension without conditions related to necessary severe accident related upgrades to design, operations, safety assessments, emergency planning and off-site support for risk reduction.

Given what we know now about consequences of severe accidents in general (worldwide there have been 6 reactor units lost to severe accidents in about 15000 reactor years of nuclear power reactor operation) , I do not believe that there is any justification for continued unfettered operation of Pt. Lepreau reactor unless significant upgrades are made immediately in a number of critical areas related to developing further understanding of accident progression and demonstrable risk reduction from severe accidents. The response of the NB Power management to rectifying known issues is abysmal so far and should be a basis under a regulator for denying them a license to continue operations beyond the current license.

SEVER ACCIDENT PROGRESSION PATHWAYS THAT IDENTIFY DESIGN VULNERABILITIES AND RISK

I will summarize some of the issues by using an easy to understand Station Blackout (SBO) scenario. Just because we cannot have an ocean tsunami at Pt. Lepreau reactor does not mean that a sustained loss of AC power event cannot be caused to happen and consequences cannot exceed those at Fukushima, an accident initiated by nature but considered totally avoidable and blamed on human errors including regulatory incompetence and industry arrogance for its consequences (reference 1). All jurisdictions with responsible regulatory regimes require that progression of accident and consequences of such an event be evaluated to demonstrate effectiveness of existing systems and containment structures for at least 24 hours. 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.

Here is a summary of overall progression of the station blackout accident in a CANDU reactor at Pt. Lepreau[2]:

> *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.*
>
> *Feedwater injection into the 4Pt. Lepreau boilers drops and then stops a few minutes after loss of power. Heat transport system that circulated heavy water around the fuel channels depressurizes to just above the secondary side pressure but continues to circulate coolant through the boilers due to density difference induced flows (thermo-syphoning). Fuel remains adequately cooled at decay power levels. Boilers (also called steam generators) remain an effective heat sink as long as they have sufficient inventory of light water.*
>
> *As soon as the depleting boiler secondary side inventory falls too low to remove heat from the thermo-syphoning water flowing in fuel channels (~1-2 hours) the heat transport system re-pressurizes. Recall that no operator action is credited in this scenario and no addition of water into boilers from feedwater train considered.*
>
> *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 (tests for Pt. Lepreau safety relief valves confirm this) leads to a continued over pressurization. These pressure relief valves were reportedly properly designed in the original Pt. Lepreau units but erroneously mis-sized in 1996 after a knee jerk reaction (and poor engineering decision) to a 1995 event at Pickering.*

[2] A station Blackout scenario includes loss of all AC power, including emergency equipment. No cause necessarily specified. No operator actions credited.

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 points to a potential failure of a fuel channel instead of a bunch of boiler tubes. There is ample data to dispute that outcome. Any uncontrolled rupture due to over pressurization at this stage is an unfortunate outcome.

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.

The uncontrolled failure can also be at any other location within the heat transport system. It could be in the pump and cause a containment bypass at Pt. Lepreau. 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 is likely accelerated by draining the moderator with a potential end fitting ejection following a channel rupture.

With boilers no longer a heat sink, gradual voiding of individual fuel channels and sequential onset of fuel heatup in the 480 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.

A voiding of the Calandria vessel occurs as rupture disks cause partial moderator expulsion upon onset of boiling. The fluid expulsion may be smaller than previously modelled, yet an avoidable artifact. A properly designed relief valve on the moderator could delay onset of severe core damage.

A high pressure injection of water into PHTS is not available and there is no way of manually depressurizing the heat transport system. Inventory in the reactor continues to deplete.

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 manoeuvred to end favourably but not so in the current design.

*Overheating channels (*Figure 2*), 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.*

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. An early 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.

Figure 3 illustrates channel power distribution in the reactor. The high power channels typically heatup and disassemble early but the low power channels may contribute more to Deuterium gas production in their feeders. The channel heatup is accelerated as moderator depletes and uncovers rows of channels. 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.

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.

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.

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. Figure 6 *shows that the fission product release may overheated fuel may be fast and release of large fraction of fission products into the containment inevitable. 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 (not to be confused with the traditional containment that regular single unit PWR and PHWR reactor sport). Releases to the environment, accounting for settling and re-volatilization inside the building, depend upon time at which building failure is initiated.*

Pt. Lepreau reactor will have special issues with capture of flammable Deuterium in the fueling machine vaults. The gas production by oxidation of fuel and feeders will occur after the dousing system has cycled to reduce the containment pressure. 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.

Analyses confirm that the whole CANDU core cannot just fall 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, 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.*

Given the large amount of Zircaloy in reactor channels and carbon steel in the CANDU feeder pipes, stainless steel in end fittings and vessels, accelerated 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%.

Early breech of the confinement pressure boundary by simple overpressure pulse by just above 3 atmospheres gage cannot be avoided.

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.

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.

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.

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

Here is a rehash of phenomenology and design features that affect consequences:

1. *As soon as the boilers dryout, the primary heat transport system at Pt. Lepreau will repressurize and an uncontrolled rupture of the pressure boundary will occur because the PHTS over pressure relief valves are far too small to handle decay heat at boiler dryout of about 30 MW. If the rupture is in a channel the shareholders are in for a billion dollar surprise even if the ECC system actuates (best case scenario) and further progression of accident is avoided. If instead, the ever so vulnerable boiler tubes burst to relieve the excess energy and ECC does not come in (worst case scenario) a most undesirable containment bypass occurs and public is potentially exposed to un attenuated releases from overheating fuel in 480 fuel channels gradually and sequentially running out of water. New Brunswick Power liability and damage to environment becomes unfathomable. See page 16 for a partial discussion of the over pressure protection issue*

that has remained unresolved for 14 years and has included 10 years of OPG/New Brunswick Power misinforming about relief valve capacity and 5 years of accepting that error in judgment and now maintaining a position that a channel rupture is an acceptable outcome. Combined with an inability to manually depressurize the system (as PWRs can) or add emergency coolant at high pressures, a potentially benign event of a loss of power is turned into a reactor damage accident. Fix is in replacing two $38k valves that are not only inadequate but termed 'bad actors' by industry OpEx .

2. *As the fuel in the channels begin to heatup so do the end fittings and feeders. Oxidation of feeders starts at about 550 C while fuel oxidation starts at about 800 C. Over 10 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.*
3. *As channels use the moderator to reject the heat, the moderator begins to boil and its rupture disks actuate in absence of an adequate relief system. Core uncovery is accelerated and Calandria tubes and pressure tubes begin to deform, sag and initiate cracks. This exposes the internals to steam produced in the Calandria vessel. Parts of channel disassemble, copious amounts of flammable deuterium gas are produced from reaction of steam with Zircaloy in fuel, pressure tubes and Calandria tubes. 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.*
4. *Feeder oxidation by air and steam is exothermic (gives out enormous amounts of heat) and the heatup initiates fires in the feeder cabinet insulation. This also triggers burns and explosions of the heavy hydrogen generated in the channels and released from failed channels into the Calandria vessel and ultimately into the small reactor vault. Accumulation and concentration of flammable gas inside the individual unit reactor vaults is very likely with local concentrations of deuterium exceeding flammable concentrations easily.*
5. *A part of the overheating and disassembling core makes it to the bottom of the Calandria vessel. A large number of low power peripheral channels do not fail and attain temperatures that continue to cause oxidation of fuel and feeders but avoid gross failures.*
6. *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.*
7. *Large releases of activity into the environment are inevitable.*
8. *Opportunities to arrest the progression of accident early can only be availed by significant investment into understanding the accident progression and instituting design changes to incorporate intelligent recovery actions.*

PUBLIC EXPECTATIONS OF RISK REDUCTION FROM THE UTILITIES

While it is recognized that Pt. Lepreau nuclear power plant was not designed with severe accidents within their design basis, the public perception of risk has changed since Fukushima and an outcome akin Fukushima to a sustained loss of power, however caused, is not an acceptable outcome. No industrial activity should be allowed to have a risk attribute that significant. Thus a decision should be made to quantify the risk and undertake concrete actions to reduce it as soon as possible. The CNSC Action Items (reference 3) were a good start in that direction but have been incomplete in scope, ineffectively planned and poorly implemented. The haste with which a number of Fukushima Action Items were declared 'closed' by CNSC staff in 2013 reminds one of the tacit agreement between the Japanese regulator NISA and the utility TEPCO that Fukushima investigation report for the Japanese parliament blames the lack of Fukushima station preparedness on. The onus in Canada should be on the licensees to demonstrate to the public that risk reduction measures are in place and not just planned on paper. Long term license extensions should be based on completion of risk reduction, not on promises of making plans to do so.

It would be insufficient to write that plans have been made to make plans to do the CNSC prescribed items as stipulated by the CNSC Action Items (reference 3). New Brunswick Power must demonstrate that they independently have quantified the risk and taken concrete measures consistent with the safety culture[3] expected of them. The attached list of technical questions (page 21) is a good starting point and I will be happy to provide further technical assistance on each of them. Anything less is an abrogation of trust and duty by both CNSC members and the utility.

Those who understand principles of reactor safety and licensing will also tell you that this reactor, like most reactor of that vintage worldwide, were not designed with severe accidents within their design basis. Many of us who have worked on severe accident issues know now that the Pt. Lepreau CANDU reactor, as it was prior to Fukushima accident in March 2011, will fare rather poorly in the low probability event of a station blackout initiated severe core damage accident similar to that befell 4 reactor units at Fukushima just 6 years ago. We know also that off-site consequences at Pt. Lepreau reactor of a sustained and unmitigated loss of power event, however caused, will be at least as bad as, if not many times worse than Fukushima for a number of reasons. The list is dominated by extremely high potential for large amounts of hydrogen production, weak containment with layout that promotes high local concentrations of hydrogen and poor mixing. Therefore the issue of the high risk (low probability multiplied by very high consequences) from severe accidents is important not only for the utility but also the regulator acting in interest of public safety.

In my discussion below, I will give details of two examples where CNSC has dropped the ball and is treating the severe accident mitigation issue as a paper exercise, rather than as a serious issue requiring multi-faceted response. I also give a number of examples of how there has been practically no evaluation of severe accident related risk from continued operation of Pt. Lepreau reactor and include a large

[3] "Safety culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance." - International Nuclear Safety Advisory Group of the International Atomic Energy Agency (IAEA) (1991), Safety Culture (p. 4)

number of pointers on what needs to be included in the risk evaluation. I will also affirm that risk reduction needs to be undertaken prior to any licence extension and provide a number of engineering solutions that can be implemented to reduce risk.

We have known this for over a decade and have raised the concerns about enhanced severe accident related vulnerabilities that may cause an earlier containment breech/failure internally and in technical forums. The expectation was that appropriate measures would be implemented to reduce the vulnerabilities. What we saw instead was a lot of talk (e.g. Fukushima Action Items) but no concerted efforts. Almost all Action Items involved multi-year paper plans to make work plans. A number of Action Items like Passive Auto-catalytic recombiners (PARS) were tick marked 'closed' irresponsibly in 2013 even while the industry had not done anything to deserve the accolades. Interventions by public were irresponsibly brushed aside.

Many of my colleagues who have given their professional life to the CANDU industry will cringe at the knowledge of the current holders of the baton ignoring for over a decade, the warnings about the vulnerability of the designs with inadequate over pressure protection and propensity to produce copious amounts of flammable Deuterium gas and unfathomable off-site consequences. As men and women of professional integrity, they would want to shield the public from un-necessary risk and not produce 'good news' reports like the March 2015 CNSC fiction – "An Update on the Study of Consequences of a Hypothetical Severe Nuclear Accident and Effectiveness of Mitigation Measures" that has no relevance to ANY severe accident in <u>any</u> CANDU reactor as the source term it uses is just a number picked out from thin air[4]. If that is the extent of technical competence at CNSC in looking at severe accidents and their consequences, the Commission members need to be alarmed. Public needs to be forewarned. New methods of technical discourse developed. Requests to deny licence extensions made.

Canada is perhaps the only jurisdiction where the regulator is totally ineffective, yet loudest in its pronouncements of being a 'watchdog' (a word many detest along with 'lapdog') and the industry brazenly does as little as they can get away with (case in point is their support for the new CNSC 'study' on severe accident consequences – I will summarize the audacious industry input at my presentation and in a supplementary publication).

Those who understand CANDU design, risk sensitivities and the list of vulnerabilities and design fixes I have compiled, agree that a structured approach to fixing the design and implementing effective preventative and mitigating measures, along with serious attempts at training the operators is within our capabilities. In Canada we have adequate technical resources to meet the challenge, only if the upper management at the regulatory bodies and the utilities can provide the necessary leadership or get out of the way of the technical personnel. New Brunswick Power shareholders are ill served by management

[4] The CNSC author's response to picking a target release rather than a predicted release is " *Detailed aspects of severe accident progression and CANDU designs were not part of the scope of the study. A generic large release at the safety goal limit was assumed, reflective of the radionuclide mix in the Darlington reactor units. As a sensitivity analysis, the source term was increased by a factor of 4 to represent a multi-unit accident (e.g., 4 units at Darlington)*." This is not a justification for an irresponsible act of denying the public the emergency preparedness measures it deserves and expects from CNSC staff hired to provide the necessary protection. Release from 4 units into a common containment can be 100 times higher, maybe 1000 times higher or just 2 times higher – but the factor must be determined analytically and with reason. Shutting down the plants until this can be done properly would be a more honorable option. Picking a number out of thin air is irresponsible.

collusion with regulator and should require the management to work instead in their long term interest that is best served by making the reactor safer, not just cheaper to operate.

We cannot pretend anymore that severe accidents occur only in other jurisdictions or that our reactor are somehow superior. PHWR is a different technology but it is dangerously delusional to think that CANDUs represent a superior technology as far as severe accidents are concerned. We can only make a collective decision to accept any level of risk but the risk must be properly quantified. This has not been done for Pt. Lepreau reactor. The PSA numbers presented in the New Brunswick Power submission mean nothing as the a comprehensive evaluation of accident progression and consequences is still incomplete. A number of commonly accepted targets on releases (< 100 TBq of Cs-137), containment failure (none for 24 hours) cannot be met and so not discussed by New Brunswick Power. We cannot accept the risk from continued operation of Pt. Lepreau reactor without its quantification by New Brunswick Power and verification by independent experts.

We can brace the populace for consequences or we can work together to reduce the risk.

As a nuclear safety engineer who first in Canada started a systematic integrated evaluation of severe accidents in CANDU reactor in 1988 when some very technically progressive and visionary leaders (Dr. Alan Brown who headed Nuclear Safety Department and his legendary boss Bill Morrison) at erstwhile Ontario Hydro decided, without any prompting by and in spite of open skepticism by the regulators (any evaluations would be speculative – one CNSC Director wrote in his great wisdom), to start evaluating progression of and consequences of severe core damage accidents. I was engaged to analytically integrate the understanding of DNGS reactor under degraded cooling conditions and develop an integrated computer code that modelled response of all major systems to severe accident phenomena in one package. After 5 years of effort I developed a code, which did such evaluations albeit with multiple limitations, and that code – MAAP-CANDU is still used by the industry. The code contains about 50% of material, mostly irrelevant for CANDU accident progression evaluations, from an EPRI code for LWRs called MAAP. I last worked on that code in 1993 but have continued to work on severe accident issues uninterrupted to the date. After years of frustrating wait to see further innovations and development in the MAAP-CANDU methodology to better predict accident progression and consequences, I have a new, significantly advanced severe accident code ROSHNI that I now use to calculate CANDU severe accident progression and consequences. In addition to having the support of actual calculations, my observations are based on the 25 years of severe accident progression and design evaluation experience so acquired.

After over 25 years of working on the topic of severe accidents, I understand now that the CANDU reactor, especially at Pt. Lepreau need serious upgrades to reduce risk from severe accidents and that our understanding for DNGS units in 1993 was primitive and the MAAP-CANDU (now under a new name MAAP5-CANDU although there were no MAAP1-CANDU, MAAP2-CANDU or MAAP3-CANDU) computer code is incomplete and devoid of any serious improvements in CANDU related modelling in the 22 years since its first release.

I have openly shared that knowledge with the industry and seen the circus around the Fukushima Action Items at CNSC degenerate into a farcical charade culminating in CNSC publishing and as of March 10th 2015 republishing without much change, a study on consequences of a severe accident at CANDU reactor with an impossibly small source term (100 TBq of Cs-137 out of a total of ~70-100,000 TBq in one unit) totally devoid of any supporting analysis on how an accident would actually progress in a CANDU reactor (authors at CNSC perhaps did not know how that was to be done). How come not one commission member ever clued into the study having no merit, being of dubious quality and dangerous for emergency planning purposes given that the source term was fictitious and represented a wishful target? Although I must say that a couple of Commissioners continue to ask some of the right questions and give me hope.

'HYDROGEN' ISSUE

This issue should have been addressed 20 years ago for design basis accidents. The oxidation potential of feeders as significant sources of flammable Deuterium / hydrogen gas was never addressed. Thus the hydrogen mitigation measures designed for under 100 kg of H_2 based solely on partial oxidation of Zircaloy sheaths would never be sufficient for the 'hydrogen' that can be generated by oxidation of carbon steel feeders by steam for LOCA+LOECC scenarios as well as severe core damage accidents.

Commissioners should look first at the design based accident analysis submissions by New Brunswick Power and ask the simple question of why extensive fuel heatup under LOCA + LOECC scenarios is predicted as anticipated but never is the thermo-chemical behaviour of end fittings and feeders analyzed.

My analysis shows that carbon steel feeders produce enough flammable deuterium gas for a sustained LOCA+LOECC scenario lasting many hours to make the Zircaloy source deuterium look inconsequential. Also, please ask why the whole safety report never acknowledges difference between deuterium (D_2) production and hydrogen (H_2) production in a reactor that is cooled and moderated by D_2O. While you are at it, also ask why with a factor of 2 differences in transport and combustion properties, is the lighter hydrogen assumed to be same as deuterium in almost all Pt. Lepreau and other CANDU submissions. Last time such a question was raised publically by a Commission member the response from a staff member was totally wrong when it was asserted that no differences exist between 2 gases. Ignorance is such a blissful state of mind.

For severe accidents, a comprehensive deuterium gas source term has never been determined as well. The severe accident computer codes in use (e.g. MAAP-CANDU) have no consideration of heavy water. They use light water properties and only consider H_2 production, not D_2 production just as the ability of PARS to mitigate it. After all PARS are first designed and tested for lighter hydrogen, not heavier deuterium. Do not let them tell you as in a previous public meeting that the two gases are the same in combustion and recombination. They are not. At least a hundred scientific papers attest to that. Deuterium would recombine at least 41% slower and burn quite differently. At a previous CNSC public meeting a CNSC staffer quite smugly and with a straight face mis-informed, hopefully only in ignorance, the commission about the gases being of identical behaviour.

Pt. Lepreau safety report will confirm to you that for larger breaks fuel bundles as well as the feeders are hotter earlier and longer as ECC fails to inject (see Figure 2). These will produce more combustible deuterium. The small (65 kg, if I recall correctly) source term 'hydrogen' for LOCA+LOECC in the safety reports is amusingly wrong. New Brunswick Power should amend estimates of that 'design basis' risk before being granted a licence extension. They should also provide a 'hydrogen' mitigation system that does not cause explosions beyond 6% hydrogen concentration as the current AECL PARS do. AECL has done experiments showing explosions caused by PARS and this was made public at last year's CANSAS conference organized by KAERI at CNSC. I am sure the good engineers at AECL can come up with better PARS (alternate designs already available) or the industry as a whole can come up with a better hydrogen mitigation option than the current PARS that are so poorly suited for CANDU reactor spewing large concentrations of 'hydrogen' into the relatively small and congested reactor vault.

For severe accidents, the estimates of accident progression and hence deuterium production cannot be adequately undertaken by the computer codes currently available to the Canadian industry. There are far too many errors and omissions in the code MAAP-CANDU that they use now. These have been presented to the industry many times; last about a year ago at CNSC. None have been fixed.

Installation of Passive Autocatalytic recombiners (PARS) has become an acceptable hydrogen mitigation system for severe accident because of their passive action, relatively well understood phenomenology, start-up at low hydrogen concentrations, efficiency under both beyond-design-basis and design-basis accident conditions, and implementation that does not constrain normal operation.

Yet, there are three issues that must be considered:

1. The PARS units should be sufficient in number and placement to avoid a hydrogen burn (limit hydrogen concentration to less than ~4%). Tests have shown that at any concentration greater than 5%, these units with a wash coat layer of the catalyst exude flames. There are other designs of catalytic plates that do not have this problem as by limiting the recombination rate the maximum substrate temperature is limited to below the auto-ignition temperature of hydrogen (Figure 5). At 6% hydrogen concentration they cause explosions. With such performance characteristics, no PARS are better than these PARS if the hydrogen concentration cannot be guaranteed to be kept well below 4%.
2. The PARS units should be qualified (sized and tested) for the actual flammable gas (deuterium in CANDUs) and not just for simple hydrogen. Data show that processes that dominate recombination by a catalyst maybe slower by a factor of up to $\sqrt{2}$ for Deuterium (reference 4). None of the installed units were tested for Deuterium. They were tested for common, lighter Hydrogen. CANDU severe accidents result in production of Deuterium first and predominantly so. CNSC staff do not know that as evident from a previous response[5] from them to an intervener.

[5] A response from CNSC to a question regarding Deuterium vs. Hydrogen in an email states "*While there has not been to our knowledge any demonstrated issue associated with deuterium versus hydrogen in the PARS, we are of the view that it would be at most a minimal concern given that the scenario where the PARS is needed assumes a severe accident where the heavy water coolant has been lost and is being replaced with emergency cooling water (which is light water).*" What an interesting (and patently wrong) understanding of when and which flammable gases are produced in a CANDU severe accident. Again it would be funny if it was not painful to realize that certain guardians of our nuclear safety know so little about severe accidents in reactors they are paid to regulate and that they are still allowed to hold their jobs. The email was copied by the CNSC author to the highest CNSC senior management. I wonder if the CNSC management (1) laughed silly as I did; or (2) smirked in knowledge that another intervener was smugly silenced with arbitrary answers; or (3) could not tell the difference between Deuterium and Hydrogen gases as well; or (4) were some of the original authors of this amazing revelation for which they would be laughed out of any high school chemistry class discussing the accident progression.

3. PARS units should not cause a containment failure by the heat of recombination reaction or by the fires potentially caused by the high temperature gases exiting the PARS units. The recombination kinetics for hydrogen is;

$$H_2 + 1/2O_2 = H_2O + 240 \text{ kJ/mole of } H_2$$

A 1 kg/hr removal of hydrogen by PARS is, from the above, equivalent to ~33 kW introduction of heat into the containment. An addition rate of about 10 MW heat can be anticipated for removal of hydrogen produced in a severe core damage accident when the correct number of AECL PARS units (~75 in a CANDU 6 building) are installed. This energy addition is enough to fail the containment by overpressure or potentially cause fires if the PARS are operated in high H_2/D_2 concentrations. If recombined with oxygen in a recombiner, only the hydrogen from steam oxidation of Zircaloy in a CANDU 6 reactor will produce over 225 GJ of energy (equivalent to 110 FPS, 3 hours of decay power at 1%). PARS units at a Pt. Lepreau reactor, if properly sized and populated, will produce about 25% more per reactor unit.

The issue of recombiners requires a serious re-evaluation but this must wait until a more complete source term for deuterium gas has been established for Pt. Lepreau reactor. Given that at present their analyses do not include feeder oxidation, any 'hydrogen' source term New Brunswick Power have is likely incomplete. This is an important safety concern and no license extension should be granted unless the issue is properly addressed.

PHTS OVER-PRESSURE PROTECTION ISSUE

None of the over pressure protection systems in the heat transport system, moderator or the shield tank are sufficient to remove decay heat when other means of heat removal are not available following an accident that may lead to severe core damage. Of primary concern is the over-pressure protection in the heat transport system.

After about 15 years of review of the issue of inadequacy of relief capacity of the over pressure protection safety relief valves, CNSC has now accepted the Canadian nuclear industry position that the steam relief capacity does not have to be sufficient to remove the thermal load (decay heat) and an uncontrolled rupture of the reactor pressure boundary is an acceptable outcome. After insisting erroneously for 10 years that the safety relief valves were properly sized for decay heat removal, it is claimed now that the rupture will most likely occur in a fuel channel once the boilers dryout and the relief becomes the sole heat sink. If the uncontrolled rupture were, however to occur in the boiler tubes, the resulting containment bypass can have catastrophic consequences and needs to be reviewed further now.

Pt. Lepreau CANDU over pressure protection on the main heat transport system (HTS) is atypical of pressurized water reactor. (the fact that the design is atypical is not the issue but that the over-pressure mitigation capability of the implemented design is inadequate upon a loss of heat sinks). Instead of being a direct and unobstructed relief path as required by the ASME code, section III, NB-7141 (b) - it is composed of two sets of valves in series (Figure 7), separated by a small low pressure vessel called the bleed condenser. The first set of valves are typically called Liquid Relief Valves (LRVs) and the second set of valves are called Safety Relief Valves (SRVs), although both sets are designed in CANDUs for a certain liquid relief with a small steam relief capacity, typically also not certified. Under conditions of boiler heat sink termination, these valves must pass enough steam to match that produced by decay heat, in order to avoid an over pressure.

This is an uncommon arrangement that can work if both sets of valves open when required and adequately relieve the excess energy thus maintaining the pressure in the HTS at levels that are safe. Canadian AECB regulatory document R-77 defines 'safe' as 10% overpressure for events that are frequent and 20% for rare events. In no case is any over pressure protection system allowed by ASME Boiler & Pressure Vessel (BPV) code to permit a failure of the pressure boundary. Strict rules exist for ensuring, by pre-installation testing, that the valves would function as required under extreme conditions. NRC even insists on periodic certified steam relief capacity testing of the installed safety relief valves, something that CNSC apparently does not.

The design relief capacity of the over-pressure protection SRVs at Pt. Lepreau is ~2 kg/s of steam at ~10 MPa per valve. Both sets of valves are essentially specified for liquid relief, typically based on a D_2O bleed closed with D_2O feed full strength in. Steam relief capacities are improperly specified as very small values, with perhaps the expectations that the design basis does not include passage of steam. Compare the 3 to 4 kg/s steam relief capacities of the two SRVs to a reference value of ~20 kg/s as the decay heat equivalent for a Pt. Lepreau reactor at the time of boiler dryout under a station blackout scenario.

The design value of the steam relief is inadequate just by inspection. It was easily shown by application of a simple ASME equation on the actual valve geometries (tested flow area of about 35 mm^2 in steam) that the SRVs can never discharge enough steam (Figure 10) to avoid an overpressure. It was also shown by some AECL testing at Wylie Labs & valve spring analysis that the valves cannot open fully under steam conditions (lift of about 1mm out of a total possible lift of 4mm) and thus are only able to relieve less steam than needed. A proper over pressure protection will not be available when required. This can result in an uncontrolled rupture of the pressure boundary.

So a serious safety problem arises if the safety relief valves cannot relieve enough steam or if one or more of them fail to actuate when required to do so. Good designs provide redundancy and adequacy. In case of a station blackout scenario (loss of all AC power) the derived engineering requirements on the overpressure protection system are exactly the same for all reactor worldwide – **remove excess energy by steam discharge equivalent to decay heat by actuating passively and reliably and avoid an over-pressure**. These requirements are easy to quantify and understand.

Decay heat at boiler dryout is typically about 1% and for a Pt. Lepreau reactor that is about 25 MW or 25 kg/s of steam equivalent. For a larger PWR that is about 30 MW equivalent to about 30 kg/s of steam roughly. The US PWRs typically have 5 SRVs with an ability to remove up to 250 kg/s of steam resulting in an ability to maintain the pressure in the system at the set point of the safety relief valves (Figure 8), while the CANDU steam relief capacity from 2 SRVs is capped at 4 kg/s will result in an uncontrolled rupture (Figure 9). It is not that the US PWRs need to relieve 250 kg/s. They would never need to relieve any more than 30 kg/s steam after a SBO but the redundancy and adequacy of steam relief is result of the good engineering practices in design and safety margins. The difference in relief capacities of 6000% with CANDUs is alarmingly high with the difference in core thermal power relatively small, ~30%.

The subject valves in all CANDUs replaced properly designed valves in 1996 when the industry panicked after the relief valves chattered and stuck open at Pickering and caused an unprecedented ECC actuation.

Again, the safety concern is as follows. If the SRVs cannot relieve the heat load when required and a resulting overpressure causes the vulnerable boiler tubes to fail then the release of activity through the open Main Steam Safety Valves (MSSVs) will cause a containment bypass and an undesirable exposure of public to activity contained in the steam. If fuel failures follow, the resulting exposures can be catastrophic. If the accident happens at Pt. Lepreau, parts of Nova Scotia and Maine along with much of New Brunswick will suffer greatly and immediately. The issue therefore is not frivolous but the response of the industry has certainly been so. The valves cost $38k each.

The SRVs are spring loaded valves whose claimable capacity to relieve a certain flow rate of liquid and certain specified flow rate of steam is required by ASME code to be certified by tests. CNSC has not understood this simple requirement or required the licensees to produce results of such tests.

ASME Boiler and Pressure Vessel Code, section III, NB-7000 requires that SRV fluid (steam or liquid) relief capacity be certified by tests and only tests. From information made available by the licensees to CNSC, it is apparent that none of these replacement valves for any of the CANDU reactor were most likely tested properly for any service and were definitely never certified for steam relief (an examination of the test data indicates that even liquid relief capacity tests did not meet the 5% scatter rule). A small number of tests for liquid relief for Bruce/CANDU 6 type valves at Wylie labs did not fully conform to the ASME testing requirements either. However, the design capacity of 1.5 to 2 kg/s for steam discharge were indicated by sample tests performed by AECL on Pt. Lepreau like SRVs at Wylie Labs. It should be noted that Wylie Labs is not an accredited facility as required by ASME and the CNSC granted an exemption that was un-necessary. The manufacturer does not have any capabilities to test valves in steam.

The following is a summary of the SRV test requirements that should be all followed by CANDU licensees:

1. *The actual safety relief valves must be tested individually in steam at representative conditions in a certified facility. Tests are mandatory and cannot be substituted by a computer models unless verified by test data for the same geometry of valves.*
2. *Installation geometry must be replicated in tests.*
3. *Three to four valves are to be tested (number depends upon the method used to certify relief capacity). Three discharge tests per valve are required.*
4. *Test data on Opening Pressure or the Set Pressure (pressure at which the valves open to sustain a discharge) must fall within 3% of the design value.*
5. *Rated discharge capacity must be attained within 110% of the set pressure.*
6. *Inlet pressure losses on valves as installed be no more than 3% (non-mandatory)*
7. *Any valves that give a relief discharge more than 5% from the average must be rejected.*
8. *Effect of uncertainties in measurement should be considered.*
9. *Only 90% of the average tested relief capacity is used as certified relief capacity.*
10. *Maximum possible steam discharge can be pre calculated using Napier equations and their corrections for superheat and pressure. A coefficient of discharge equal to the ratio of the actual flow to the maximum flow is developed and used.*
11. *Extrapolation or proration to a pressure higher than the pressure at which the relief capacity has been certified is permissible by the ratio of pressures. So at a pressure greater by 20% over the certification pressure, the relief capacity can be claimed to be greater by only 20%.*
12. *Extrapolation to other fluids is according to Section XI of the ASME code. Steam service valves should always be tested in steam.*

Safety Relief valves are required in all pressure vessels when there is a mismatch between heat generation and heat removal. In a Station Blackout Scenario in any nuclear reactor including CANDUs, that occurs when the boilers run dry. At that time, in absence of another heat sink the fuel decay heat must be removed by the SRVs to avoid an over pressure. If the SRVs are properly sized they would relieve the decay heat load as equivalent amount of steam and maintain the system pressure at about 10% above the operating pressure. In a CANDU reactor the decay heat at boiler dryout may be about 1% of the total original thermal heat production. In a Pt. Lepreau reactor that is about 25 MW or about 25 kg/s of steam equivalent. Adequacy of the SRVs has been demonstrated in all reactor except operating CANDUs. The 250 kg/s of relief capacity at a PWR does not mean that the actual relief is 250 kg/s. it just means that the relief will balance production of steam.

If the safety relief valves cannot relieve decay heat energy by steam relief, as is the case in CANDU reactor where the total SRV steam relief capacity is about 4 kg/s at opening pressure against about 20 kg/s of internal steam production, system pressure will rise, steam discharge rise and if inadequate will cause the pressure to rise uncontrollably such that some component will eventually rupture. ASME BPV codes are formulated to avoid this outcome and it is an ASME requirement for Class 1 components that SRVs be properly sized and tested. This includes testing of the actual valves to certify whatever fluid (liquid and/or vapour) relief capacity needs to be credited. In a Pt. Lepreau reactor a certified steam relief capacity of at least 25kg/s (from one valve if the usual single failure is accounted for, otherwise from 2 valves) will insure that the energy relief will be sufficient to balance energy production when boilers run dry. A larger relief capacity as in all LWRs will not cause a larger overall relief. The relief will never average more than production.

It is clear that the subject valves, replacing a properly designed valves in 1996, are ill designed for ALL CANDU reactor and their designer specified steam relief capacity of ~1.5 to 2 kg/s of steam is just not sufficient to remove energy production at the time when they are required to work. The subject Pt. Lepreau SRVs were actually designed for **liquid** relief of about 27 kg/s and a steam relief of 2 kg/s. Tests showed that these valves lift fully under liquid relief conditions but lift only partially (20%) under steam relief conditions (thrust force by steam on valve seat is significantly lower than for liquid water). The discharge area is proportional to lift and is significantly smaller for steam. This was confirmed by testing and actually an engineered valve spring feature to meet the design specifications of 1.5 to 2 kg/s of steam discharge capacity. The reactor must enhance the over protection system by installing safety relief valves that preclude a pressure boundary failure. AECL confirmed the inadequacy of the steam relief capacity (Figure 11) in analyses presented in 2011.

The fact that the PHTS over pressure protection by the bleed condenser relief valves is inadequate is well established. What is also well established is that the industry, including New Brunswick Power misinformed about steam relief capacity for 10 years and the CNSC staff assigned to the task were unable to check the facts using a simple equation. It was only 10 years later in 2011 that AECL finally admitted in public that the submissions from the industry on the critical steam relief capacity were wrong and an uncontrolled over pressure induced failure is an inevitable outcome. CNSC has done nothing since then to fix the problem and has now accepted an undesirable outcome of an uncontrolled over-pressurization of the heat transport system and failure. It is claimed now that the fuel channels are the weakest link and would fail, ignoring the fact that it is a terrible outcome (what if an end fitting is ejected and the moderator drains? Etc.) This disregards available evidence on vulnerability of boiler tubes. An attempt was made to discredit the issue using an outside consultant who made no effort to justify the low steam relief capacity but took issue with the language used by the intervener. CNSC has let this important issue fester and considers the issue closed. It will not go away by wishful thinking. Given how this has been handled for 14 years, it just makes them look petty, uncaring, unresponsive and technically challenged. I will be happy to provide further details and failing a clear resolution I am planning on bringing this up in an important international forum this summer.

If CNSC members cannot collectively understand the importance and gravity of this simple technical problem as both a safety issue and an economic issue for the utility, then the whole regulatory regime will have to be publically re examined.

SUMMARY OF PT. LEPREAU SEVERE ACCIDENT PROGRESSION & MITIGATION ISSUES

- Pt. Lepreau reactor did not consider severe accidents in the design process. Unreasonable to expect easy severe accident mitigation.
- Current Pt. Lepreau designs inherently forces a reactor damage even before an ECC loss leading to severe core damage.
- No provisions for manual depressurization after SBO. No super high pressure ECC or makeup intervention / injection.
- Onset of a severe core damage in a CANDU reactor puts activity directly into the containment. 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 and Zircaloy. Recombiners will cause explosions.
- Enhanced potential for energetic interactions with enveloping water
- Pressure relief in ALL relevant reactor systems in inadequate (PHTS, Calandria, Shield Tank, Containment)
- Containment bypass from reactivity device failure a likely outcome after a severe core damage
- Calandria vessel cannot contain debris and can fail catastrophically at welds.
- Shield Tank cannot contain pressure upon boiling and can fail. Restoration of cooling after water depletion problematic as flow outlet at the top of vessel.
- Inadequate instrumentation and control.
- Poor equipment survivability
- 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, especially seismic events.
- Need to reconsider malevolent actions and sabotage.

QUESTIONS THAT COMMISSION MEMBERS MUST ASK NEW BRUNSWICK POWER TO PROVIDE ANSWERS TO

Any licence renewal should be subject of satisfactory resolution of the following set of questions as adjudicated by an independent panel of experts (and not the sham COG group assembled after Bruce relicensing hearings - see letter to CNSC president after March 8th 2017 hearings on industry disposition of these issues, included following the list of unresolved issues).

A. WHAT ARE THE SEQUENCE OF EVENTS AND CONSEQUENCES OF A SEVERE CORE DAMAGE ACCIDENT LIKE THAT AT FUKUSHIMA IN WHICH ONE OR ALL CURRENTLY LICENSED AND OPERATING UNITS ARE AFFECTED BY A LOSS OF AC POWER. GIVEN THAT THE UTILITY SUBMISSIONS ARE MISSING THE NECESSARY INFORMATION, CAN THE UTILITY PROVIDE INFORMATION ON ANALYSES PERFORMED TO DERIVE REACTOR CONDITIONS AS A FUNCTION OF TIME, SOURCE TERM TRANSIENTS AND THE CONSEQUENCES THEREOF. WHAT NEW MEASURES ARE IN PLACE NOW FOUR YEAR AFTER FUKUSHIMA TO DEMONSTRATE THAT THE UTILITY CONSIDERS SEVERE ACCIDENTS SERIOUSLY AND THAT CONCRETE STEPS (NOT PLANS TO MAKE PLANS AS REQUIRED BY THE CNSC FUKUSHIMA ACTION ITEMS) HAVE BEEN TAKEN TO :

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
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 incore 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 a100% 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 with consideration of full core inventory of fission products released over the first day and consideration of hydrogen explosions within the CFV vessel.
30. Consider improvements to pressure suppression system in reactor building
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.

The list of design and operational enhancements must complement a plan for operator training and emergency preparedness.

The above issues were examined by COG during 2015-2016 and it is unfortunate that Pt. Lepreau was party to the untruths that were propagated in that review.

My views on a meeting organized to discuss the COG and CNSC staff response to the above issues are in the letter to CNSC president Michael Binder below. I emphasize that New Brunswick population is under grave risk from continued operation of this reactor unless a more honest assessment of the underlying severe accident related issues is undertaken. It is alarming that the NB Power management has sponsored such an irresponsible COG managed response to some very genuine concerns raised in interest of public safety. Should CNSC and NB Power continue to conspire to overlook these concerns, they will be jointly responsible for the consequences as determined by a Japanese court in March 2017 when it declared that both the regulator and the utility were in complicity that caused the Fukushima tsunami to progress to a bad nuclear accident.

21 March 2017

Michael Binder, President
Canadian Nuclear Safety Commission
280 Slater Street,
P.O. Box 1046, Station B
Ottawa, ON
K1P 5S9

Ref: Abysmal Severe accident mitigation Capabilities of operating CANDU reactors and the CNSC review meeting on 8th March 2017 on my recommendations to rectify; Your reaction to Canadian equivalence of a Japanese regulator complicity with industry being recognized by a Japanese court as a contributing factor to Fukushima disaster.

Dear Dr. Binder:

It is quite possible that you have come to believe what you wrote to me about the CNSC being a 'fiercely independent regulator'. Most independent observers of the CNSC do not think that way. Many of your own staff do not think that way. They tell me, and we all notice, that the powerful utilities are typically able to manipulate CNSC management almost any way they like. If the implications of you being wrong were not huge for my country, I would not waste these precious years trying to convince you otherwise.

It is understandable then that you take exception to my insinuating that CNSC is as complicit with the industry it regulates as was the Japanese regulator found to have been with its industry; the latter ended up on the wrong side of history and the court judgment[6] last week. My hope is that before you retire you will have one sober reassessment of the damage that such a delusion may cost this country.

I am willing to meet you in any forum - private or public - and challenge that questionable belief with facts that are apparently so uncomfortable for you. What really matters is that the risk to my people is still high and the CNSC is busy congratulating itself on what a great regulator it is while paying lip service to power reactor safety and putting under the rug the glaring examples of design weaknesses that many have pointed out for years. CNSC has been only too eager to give multi-year license extensions covering unprecedented long periods without ever making serious reactor upgrades a condition of their license. Right now with the way it conducted a 'public hearing' review of my findings of the CANDU design flaws related to severe accident mitigation, it has further exposed a new low in its safety culture. In addition to irresponsibly denying that serious reactor design issues exist, the meeting turned into a parade of consultants unqualified to talk about severe core damage accidents calling for an end to public discussion of the serious issues that plague the CANDU reactors that CNSC is trusted to regulate.

[6] http://www.japantimes.co.jp/news/2017/03/17/national/crime-legal/first-government-tepco-found-liable-fukushima-disaster/

It is unfortunate that the charade of an 'independent' regulator will likely be lifted only when the government agrees to a truly independent inquiry into Canadian nuclear reactor regulatory practices including those related to severe accidents. I invite you to work together to ensure that it happens well before an accident maims my country and forces an inevitable judicial inquiry. An investigation is required also into the various decisions that the Commission regularly rubber stamps for the industry and the agreements that the senior staff enter into with the utilities to bypass it's own regulations and water down new requirements for reactors that are not only obsolete but also very, very poorly able to mitigate a severe accident. There should also be an inquiry into the haste with which the CNSC staff approves submissions by the industry, however flawed, incomplete or irrelevant to the issue at hand.

Perhaps there is a fair chance that you, not being a nuclear engineer, are just ill informed and poorly counseled by your senior staff in this field. What that means is that a half dozen senior employees are setting the tone, sullying up CNSC's image and exposing Canadians to unwarranted risk. That is a scary prospect but quite plausible. The other problem of course is that most of the past Commission members have required a lot of educating and have been unable to critically question the technical aspects in many instances.

The staff sponsored reports we discussed at March 8th 2017 meeting cited paper exercises of Fukushima Action Item solutions and inflated claims of superior regulatory processes we know have failed in the past and will cost Japan hundreds of billions of dollars and a century of work to rectify. Have they learnt nothing from TMI, Chernobyl, Fukushima? To many independent observers, a similar complicity exists in Ottawa to accept faulty submissions from an intransigent industry no longer interested in improved safety for public good sake and able to take any corrective actions only kicking and screaming. Examples of the complicity are many, least of which is the manner in which the CNSC dispositioned some very serious issues I raised. I will refer to a handful of them later in this letter.

I am disappointed and even sad; but not surprised - given the history of my interactions with CNSC over 15 years - to see that public safety interests are again not being well served by the processes undertaken to look at issues that have immense public safety implications. To many, it seems that a few CNSC management personnel have once again lost all objectivity. As a result, the whole CNSC seems have turned its back on the legal responsibility it has and succumbed to the temptation of constant public self aggrandization. Meanwhile the Commission seems to be in a firm utility capture. Some wonder if the CNSC funding formula or the lack of related technical expertise is to blame. CNSC staff have acquired thinly veiled paid endorsements of their past inactions using specific external consultants (*some retired from the U.S NRC whose approved designs have had a PWR and a BWR severe accident at TMI and Fukushima respectively!!!*) with no visible CANDU severe accident related technical expertise to share, no mandate to be fair; or any demonstrated independence of review or maturity of views.

As far the base COG review report of the issues I raised at Bruce hearings in 2015 is concerned, I never for a single day, after my initial meeting with them in July 2015, expected them to be objective or even attempt to rock the utility boat they ride; or challenge the interests they represent. Their team mandate to deny everything that is wrong with their obsolete reactors was made clear to me right from the first time I met with them in July 2015 in COG offices. Their review demonstrated that they were not interested in facts or data or even basic engineering sense. It was all expected from an industry mouthpiece and my little attempt to even teach them of the difference between Deuterium gas that CANDUs produce after an accident and the Hydrogen gas they don't, so miserably failed that I gave up on things slightly more complicated such as oxidation of steel, relief valve sizing, off-site radiation measurements, pressurizer location, containment failures, hydrogen explosions, etc. And they missed the point in more complicated issues rather miserably and just denied it all rather robotically and smugly. But it is the actions of certain CNSC managers to assemble external consultants to sing their praises and condemn my interventions without explaining properly any technical basis of opposing them, at the chosen time and format, that particularly confound me.

I completely reject the process and the conclusions the staff sponsored and totally biased and largely unqualified consultants had the audacity to present in defiance of the public safety you should have keep foremost in our discussions, and did not. I suggest, like I did at Darlington hearings, that a fully independent, technically competent commission of inquiry be set up to look into these matters. We can now explore our options on how that may come about; and the decision that the CNSC takes on the March 8th proceedings will well define the path that we chose to challenge these impediments to public safety.

It is interesting that the CNSC chose for meeting a time (March 8th) when only one commission member was present. The response of that lone member to my submissions as an 'outlier' and 'intemperate' was likely prompted by similarly audacious and suggestive presentations by your consultants seemingly trained to parrot repeated words of praise and condemnation for CNSC staff and this intervener respectively.

The so called review process started at Bruce relicensing hearings was faulted by design and a useless exercise from the start. It did buy the industry an additional 2 years of pretending to investigate additional risk reduction measures to finally adapt none. Then the CNSC staff chose to fund with precious public money some selected, sole sourced, compliant, friendly, comforting voices that agreed to parrot the untenable position of CANDU infallibility CNSC have cornered themselves into flaunting; CNSC chose to not talk to me over the 2 years of this review that should have taken a more competent group no more than 2 weeks. Meanwhile, CNSC senior management seemingly instructed its technical staff to not communicate with me ever (one asked me to not email him anything); and chose to spend un-necessary couple hundred thousand dollars on the utility side that decided to pretend that their 50 year old design reactors were perfect for the current millennium. Then these hired guns have the audacity to not only blindly set aside these carefully crafted concerns they barely understood but also suggest that interventions like mine be shut out early !!!. How dare they come into my country and teach us about democratic interventions into matters of critical national importance and how dare the staff even ask them to talk down to us in that manner. These are from the same group of people whose ignorance of severe accident issues allowed 2 severe accidents in reactors designed in their country. I, for one, am trying my best to avoid one in my country in spite of CNSC senior staff resistance to reason and ignorance of bare engineering facts that pertain to severe accidents.

In the end, the industry must be happy that for a mere $150k or so in absolutely useless but friendly consultants' reports presented by staff, the industry was saved some half a billion dollars in suggested design improvements. This team of geniuses should get the CNSC / OPG medal of obedience and suitably knighted. Now they can ask for anything. I wonder what is their next project. Pickering life extension for another 10 years?; relicensing of Douglas Point?; proving that radiation exposure is good for living things?

I see that technical arguments meant nothing to CNSC or COG and that sham processes like the paper plan exercises of thoughtlessly generated and hurriedly closed Fukushima Action Items meant more. The staff paid big monies for meager reports to their retired colleagues from NRC who have no demonstrable expertise in CANDU severe accident mitigation or any successes in ensuring that there never was a reactor melt down in any of the PWR or BWR designs they so gloriously steered. I note also that none of the presenters gnawed at the hands that feed them and parroted in unison the same story. "*He is wrong. You are right. Let us shut him up. Close the books."*

I was also not surprised to read the scolding opinions and denunciations of my work from an ex-employee of a competing overseas reactor design company Westinghouse. I have long suspected that they have no interest in success of CANDU designs, no understanding of it's severe accident vulnerabilities, nor any expertise in CANDU remediation. The reports that CNSC so adoringly presented as gospel were a sham exercise in a technical debate and doctored (by providing to him inflated flow areas for valves) to get favourable response from that otherwise well meaning consultant. An inquiry will likely find the relationships and modes of engagement of consultants disturbing. I will not comment on the report by the Canadian consultant as it contained personal attacks and judgments that may require an external competent legal overview. What worried me more was that CNSC even entertained his suggestion that 8 km of cheap, low chromium, low carbon steel feeder piping just downstream of melting fuel sheaths cannot oxidize in superheated steam inside or the air outside and produce combustible Deuterium that may cause spectacular containment explosions and irreparable damage to Ontario.

An interesting outcome of the March 8th meeting was that CNSC taught us all a new meaning of the word 'Independent Review'. *Use a different font, and slightly altered words to repeat the same condemnation. Lie through your teeth but remain in perfect harmony. Strength in numbers creates an 'outlier'. Forget science; just make noise. Any noise. Invoke superior regulatory processes.*

Having spent 30 years working on the severe accident issues at all four Canadian irradiated nuclear fuel configurations (power reactor, fuelling machine, spent fuel pool, research reactors); having done extensive design reviews and developed a half a dozen analytical codes the CANDU industry uses worldwide, I have genuine concerns that I raised at Bruce and Darlington relicensing hearings in good faith and in interest of public good and advancement of our CANDU technology to meet the current public expectations of risk. I was disappointed in the CNSC regulatory senior staff taking my suggestions, that included issues that were long crying for resolution in the open, as a personal affront; and then seemingly conspired with the industry to openly ignore the evidence I put before them. I cannot wait to discuss the evidence of critical vulnerabilities in CANDU reactor ability to mitigate severe accidents in front of an independent judicial tribunal that I hope you will soon participate in. I will also present evidence of how some amateurish arguments to the contrary were manufactured in complicity and on purpose. Obfuscation, blurring facts and peddling falsehoods are considered contrary to our professional mandate and a dangerous activity in our line of work.

Now that the province of Ontario is considering spending over $15 billion to refurbish the Darlington reactors during a lengthy shutdown of the plant, we have a once-in-a-lifetime chance to correct a number of hitherto overlooked design weaknesses that could contribute to severe core damage and unacceptable offsite releases of radioactivity following an accident as simple as a station blackout. CNSC is squandering that opportunity by being illogically, unashamedly and openly biased towards an industry that has no interest in accepting that the reactors they operate are tired and obsolete. On the flip side, there is far too much money to be made by companies that are eager to rebuild those old machines but know not how to design new safer ones or fix the glaring faults in the old ones. I wanted to ensure that the public did not suffer because of that inherent conflict within the industry. While I have no regrets that I have paid a huge personal price for that, I am not sure if you feel a similar duty to public safety.

Darlington reactors were designed before any meltdowns had occurred in commercial reactors. Three Mile Island (1979) , Chernobyl (1985) and Fukushima (2011) that have subsequently undergone severe unanticipated accidents going far beyond "design basis" occurrences for a loss of 5 reactor units in just under 15,000 reactor-years of power reactor operation and an anticipated integrated damages of better part of a trillion dollars, an amount that far exceeds Canada's risk appetite. Reactor designs of that bygone time period have proven unable to deal effectively with such severe events and would not be built today. I have sent you a number of ASME papers on CANDU reactor vulnerabilities for you to read.

I am surprised that your staff and their chosen consultants seemed to argue that our forefathers who designed these CANDU reactors over 50 years ago using slide rules had the foresight or information to make them fool proof, future proof and error proof. I respect what these pioneering engineers and scientists were able to create with the knowledge and resources they had then; but I know that they would not approve of turning into blasphemy, any attempts to incorporate new knowledge into a technology that was operating inherently in unchartered territory and had presented numerous unanticipated material failures and operational difficulties.

The new low in the safety culture evident at the March 8th 2017 hearings, spearheaded by CNSC has ruined any intelligent conversations and the unspoken directive from all powerful utilities for these obsolete designs with numerous severe accident related issues seems to be - *nothing much needs to be done to improve them.* A newly coined term of *'holistic'* approach to severe accident mitigation from the technically challenged folks at Bruce Power would do the trick. Maybe Bruce Power should copyright that slogan before the Japanese colleagues at TEPCO realize it was stolen from them and crawl dazed and angry out of the radioactive ruins such attitudes created at Fukushima and a thousand square kilometers around. I am so scared that this 'holistic' approach that nobody in the whole world uses as an excuse from meeting all challenges head-on, will get a lot of my fellow citizens killed.

There is no question that any of the currently operating Canadian power reactors, designed 40-50 years ago, will not be built today in any jurisdiction with a responsive, accountable, law abiding, rule based, robust regulatory regime because they do not meet the current expectations of risk profiles against severe accidents. Even the large fission reactors that were designed just10-20 years ago will not be built today in any country, except perhaps in Canada where the regulator now will license anything the industry pays it to examine; agree to almost any violation / relaxation of rules and norms as article 7 of the Nuclear Safety & Control Act lets it do, blinded by its own propaganda of inherent nuclear safety in the obsolete reactor designs it so poorly regulates.

There is also no question that the CANDU multi-unit reactors at Darlington and Bruce sport some of weakest and leakiest containments that will also trap large quantities of explosive Deuterium potentially produced by large amounts of Zircaloy and tens of kilometers of cheap, low chromium, carbon steel piping. A number of critical equipment like boilers and reactivity deck are <u>outside</u> the containment; something that is unheard of in other PWRs. They have no pressure vessels to isolate from containment any overheating core materials. So the fission products released from overheating fuel will be ejected directly into these leaky containment structures built like rectangular industrial buildings and not classical cylindrical-spherical pressure retaining geometries of nuclear reactor containments around the world. In absence of a heat sink, these designs cannot remove decay heat from any of the enveloping water volumes that by turn become heat sinks - Heat Transport System, Moderator System or the Shield Tank - without bursting them or rupturing a few disks to create a large path for fission products to eject out. There are no means to directly depressurize the HTS or to add high pressure emergency coolant to it. They do not have any meaningful Deuterium mitigation systems or instrumentation to follow accident progression. There are no severe accident simulators and the table-top/paper SAMG exercises based on obsolete codes and rosy assumptions are meaningless. The actual list of design weaknesses is significantly higher and many have been extensively discussed in the multiple papers and presentations I have submitted to the regulator and the industry.

Opportunities for design improvements are abundant but unfortunately mostly ignored. Both accident progression and consequence assessments by the utilities and regulator are usually presented in a distorted positive light in defiance of engineered realities and public safety. Recall the blatantly irresponsible report published by CNSC wherein the off-site releases of Cs-137 after a severe accident were claimed, without any analysis whatsoever, to be limited to 100 TBq; or the report by CNSC staff that claimed that the Darlington allowed 5 hours for the boilers to be replenished; or the statements by CNSC management staff at public meeting that no evacuations were necessary for 24 hours. All these were summarily wrong and misleading and in another industry would be subject to censure. In fact, the information on source terms and timing given to Ontario Fire Marshal's office for emergency preparedness was so distorted that measures they would develop to shelter / evacuate people would not only be ineffective, but also counter-productive.

There are many glaring examples of CANDU reactor inability to honorably mitigate a sustained loss of heat sinks; an accident often also called a Station Blackout Scenario that is the simplest to understand and caused by any one of a hundred different instigators. Here are a few for Darlington NGS:

Over-pressure protection systems in all relevant reactor systems (PHTS, Calandria, Shield Tank, and Containment) are inadequate for decay heat, let alone for other anticipated severe accident loads. Early passive heat removal by steam generators after a station blackout can be compromised by primary coolant removal into a large pressurizer located well below the pump bowl (This alone would ring LOUD alarm bells in any other jurisdiction but there has been no response from CNSC). There are no emergency means of high pressure water addition to the steam generators or the heat transport system which not only has an inadequate steam relief capacity for over pressure protection such that an early containment bypass (read un-attenuated and early contamination of public) by steam generator tube ruptures is a distinct possibility, but also lacks a method of manual depressurization for early accident mitigation. In absence of a retaining LWR like pressure vessel, the reactor cores would release fission products without attenuation into the box like containments. The containments at Darlington and Bruce are at 48% per day leak rate at design pressure very leaky by any measure (a PWR containment design pressure leakage is less than 0.1%/day - <u>480 times less</u> than in CANDU multi-unit stations) and at less than 1 bar design pressure, structurally weakest of all operating reactor containments. The reactor buildings around each individual reactor core are inverted cup like traps for combustible gases. A large number of safety significant components like the steam generators, pumps and the reactivity control devices are all outside the containment envelope. The production of combustible Deuterium gas from over ten km of carbon steel piping and over 50 tons of Zircaloy can be extremely high making the installed numbers and types of PARS not only inadequate but as early ignition sources also dangerous. Improvements after Fukushima are perfunctory and the analytical methods in support of severe accident management guidelines are outdated and incomplete. A lax and uninformed regulatory regime blindly supporting an intransigent industry resisting basic design enhancements has further exasperated, like it did in Japan, the severe accident related risk from continued operation of these reactors.

These conclusions are based on thirty years of working on severe accident related issues at CANDU reactors, conducting extensive design reviews and developing computer codes and analytical methods for severe accident progression and consequence assessments. It was hoped that open discussions by professional engineers would foster change in name of public safety. It is now feared that nothing will change unless an accident occurs or CNSC is overhauled.

All this from me, a nuclear engineer, a person with over 30 years of experience in actual analysis of CANDU reactor accidents who is strongly pro nuclear but will not stand for the newly developing culture of Alternative Facts to justify continuing to operate these obsolete designs. I am also disturbed by the chirps the regulator puts out in the constant barrage of self congratulations and smugness. We in Canada are being left behind technologically and expose our unsuspecting population to unreasonable risk because of the regulatory complicity and attitudes exhibited and the industry favouring decisions we all know are preordained despite the charade of public consultations.

The companies and consultants chosen by CNSC to bulldoze any opposition to rebuilding the same old reactors in the 1960s technology are like designers of Lada cars advising us to rebuild in Oshawa the infamous Ford Pinto from my dad's time. If the airbags do not fit; just denounce the idea of airbags. If the car can burn from rear end or side collisions, put in an escape ladder or an ejection seat as standard equipment. Borrow more design ideas from Lada. But rebuild the Pinto that they understand and not the new Chevy Impala they don't. They would rather run to Lada favouring friends from the beerhouse they know can be bought, rather than Impala favouring outsider who cannot be. Consumer ignorance is such a blessing sometimes and the politicians are hostage to the myth of clean and cheap energy; not to mention the jobs the industry creates - forgetting that it is the same populace who will be irradiated.

At the March 8th charade, I did not respond to the personal attacks on me (even when a Commission member said I was an outlier and intemperate) as public safety is more important than personal denunciations that any hothead can pen. It is from extensive knowledge of the design of the three types of CANDUs we use; from their accident response specific design analysis, from 25 years of developing and using severe accident related computer codes, that I volunteered an initial list of 34 concerns that I felt must be soberly dispositioned before tens of billions of dollars are poured into the refurbishment of the now obsolete designs at Bruce or Darlington reactors. In support, I have for years freely provided results of highly technical self funded analyses and paid dearly with industry's condemnation of my blowing the whistle. The response of the industry has been summarized by the consultants and COG team hired to blindly, complicitly and in perfect harmony defend the status quo. I was first surprised that they saw no technical merit in my recommendations, but then I saw no technical expertise or honesty behind their meaningless words either and understood their terms of reference. Their reports will please the pay masters as now there be no interruption in the freeloading from the provincial coffers by the companies and interests these people seem to favour; and the unfortunate steamrolled refurbishment of these tired old designs will continue. This at a time when tens of reactors just south of us are being retired for mostly economic or safety reasons in spite of significantly superior resistance of many to severe accidents.

I had proposed analysis of all the severe accident hazards and design enhancement opportunities and suggested examining them individually because the uncertainties in accident propagation and accident management successes are large. Remember the GE BWRs at Fukushima that your NRC friends nurtured for years and did nothing meaningful about. After losing 5 reactors in less than 15000 reactor years of operation and a trillion dollar loss to the countries and a loss of thousands of square km of God's pristine land, I do not understand the smug suggestion that our CANDU reactors remain as obsolete as some of theirs.

I was looking to have the industry commit to actual risk reduction instead of making plans to make further plans as the Fukushima Action Items did for CNSC and drowned all issues in endless paper trail with the compliant CNSC staff issuing closure of all items, mostly without any completion of real work on most fronts. Investigations will reveal that the industry only paid lip service to developing an understanding of the severe accident challenges and CNSC staff obediently complied by 'closing' the so called Action Items with a full knowledge that severe accident related source terms of energy, fission products, Deuterium had neither been calculated, nor mitigated by the measures proposed (such as the scant few PARS).

One of my aims was to create a consensus on safer reactors NOW and a good technical basis for the next generation of reactors including the refurbished ones; an opportunity for my industry maimed by privatization and sale to foreign interests, to grow again; for my alma mater AECL to become a technological force again. But the bean counters never saw this as a national opportunity but an interference in their feast, cushy jobs and responded defensively. My main aim of course was to protect my people from continued operation of these sickly reactors from a severe core damage that has a thousand instigators (not just a tsunami) in the atmosphere of the unholy alliance being forged by this industry and its regulator.

Instead, a newly coined and dangerous phrase 'holistic approach' to severe accidents was spit out of Bruce Power and the industry surrogate COG. During the hoopla of Fukushima improvements, CNSC quietly decided that even the age old design basis accidents like Large LOCA, LOCA+LOECC did not need dedicated mitigation. If I may use the language my physician wife would - that approach rejects dedicated radiation therapy to the tumors that plague the industry body and insists on fancy pain killers, magical salt baths and mysterious herbal teas instead. I will be tempted to add to that - if the rich patient dies, someone there will make money from the fancy funeral instead.

Let me again recall a conclusion of The National Diet of Japan Nuclear Accident Independent Investigation Commission Report:

> *The TEPCO Fukushima Nuclear Power Plant accident was the result of collusion between the government, the regulators and TEPCO, and the lack of governance by said parties. They effectively betrayed the nation's right to be safe from nuclear accidents. Therefore, we conclude*

> *that the accident was clearly "manmade." We believe that the root causes were the organizational and regulatory systems that supported faulty rationales for decisions and actions, rather than issues relating to the competency of any specific individual.*

Since public safety must be of paramount importance, such concerns cannot be dismissed by simply denying the problems or by treating them as a public relations concern, or by pretending that technical evidence of grave safety shortcomings can be treated as an academic debate between experts — a kind of "he said, they said" type of fruitless interchange. Like the one we are having now.

Let me patiently outline - as just one example of an unresolved safety problem at Darlington & Bruce - what would happen in the case of a total station blackout without immediate operator intervention, such as happened at Fukushima. Because of the location of the pressurizer below the level of the boilers, in a relatively short time it will be impossible to restore cooling to the core no matter how much water operators add to the steam generators and how. The pressurizer would swallow most of the water from within the boiler tubes and no thermosyphoning can be maintained. This a serious design error at Bruce and Darlington, not to be found in other nuclear reactors anywhere, but it can be corrected by any surviving good engineers during refurbishment. It is important to understand that inadequate cooling of the core is the precursor to a meltdown.

Let me as bring up another very important issue — the accelerated buildup of deuterium gas due to the rapid oxidation of low-quality carbon steel in the feeder pipes, In this context it is important to say that hydrogen gas explosions occurred in 4 units at Fukushima, at Chernobyl, likely at TMI, and at the NRX reactor in 1952 during the partial core melt at Chalk River. Within an hour of a loss of coolant in a CANDU fuel channel, the fuel reaches very high temperatures and the downstream feeder pipes can become extremely hot in the next 2 hours and will oxidize from both inside and from outside once feeder cabinet is blown, adding more explosive deuterium gas than from zirconium in the core by the end of the day. This design weakness can be addressed during refurbishment by using better quality materials for the feeders and rethinking the deuterium gas mitigation strategy currently in place. Of course the PARS added so thoughtlessly to that containment will cause an explosion well before the 24 hours the Fire marshal has been given to evacuate people.

There is a lot of talk about DCRVs and the external CNSC consultants have had the audacity to teach us how to cut off debate based on the issue remaining unresolved for 15 years. Let me tell you that the DCRVs were properly designed when the plants were built; they were changed midstream in 1996-1997 to wrong size valves. I must also tell you that the mistake was rectified in the new ACR reactors because it was not left to any paper pushers to design that reactor.

A serious loss of reactor integrity can occur even without a severe accident and contamination of the regions around the reactor can also occur even without the accident progressing to a severe accident caused by the inadequate overpressure protection capability in the primary heat transport systems of all CANDU reactors. A loss of heat sinks is a documented design basis accident. Should the accident progress to a loss of core cooling and a severe core damage, the potential containment bypass will cause totally avoidable early exposures to the population. Again the defensive posturing and technical inaccuracies in the industry response put the population at extraordinary risk. Just imagine how Pickering BCRVs - designed AND tested for 40 grams/s of steam can relieve over 20000/s grams as has been irresponsibly outlined in the COG report. This design error can be rectified during refurbishment.

It is necessary for the CANDU nuclear establishment to treat these and a host of other such design problems soberly and objectively. We must learn to put aside the almost automatic knee-jerk defensiveness on the part of the industry and of the regulator who have failed to address or even indentify these problems in the past. Safety has got to be of paramount concern, seeing how devastating a severe nuclear accident can be not only to the industry but to Canadian society and the Canadian economy. My critics in COG and CNSC, and the consultants who have been paid tens of thousands of dollars by CNSC to deny that these deign weaknesses even exist, are simply repeating, over and over, words that have been written before, without providing any new scientific evidence or any detailed technical analysis to back up their judgment — to the effect that my concerns need not be considered further.

Many people no longer employed by the Canadian industry CNSC so ineffectually regulates, have a lot to contribute, share and reveal but I was told privately that the CNSC staff, for example, will not talk to me directly while this current CNSC president is still around. I suggest that this is contrary to public interests as is advertizing all the way from the Ontario Fire Marshal to the Parliament that our reactors can cause no off-site consequences to fear and publically stating in hearings that no evacuations are necessary for 24 hours. I humbly implore that this is about the last chance for retiring these reactors or abandoning this dangerous attitude before we go down a road of no return. As I said in the brief oral presentation I was allowed, we cannot afford a Fukushima near highway 1 or highway 21 and certainly not near Highway 401.

I remember the staff presenting bizarre industry support positions in authoritative tones at relicensing hearings and know that they have been trained now to immediately succumb to any position demanded by the power industry that largely funds the CNSC. Or to the 'positions' that senior management has chosen to support irrespective of where the facts lie. Examples are the 100 TBq release ceiling for a severe accident that Patsy Thompson said was highly conservative and her surrogate McAllister said was orders of magnitude higher than possible; or the now famous 'feeders won't get warm' declaration by CNSC's genius director Gerry Frappier at the Bruce hearings.

In spite of the childish video on the CNSC website about a station blackout scenario, we all know from analyses now that opposite of the picture presented there is true. The operators will sit in dark of behavior of a reactor they just cannot cool because in absence of electric power, the low lying pressurizer irreversibly swallowed all the primary coolant from within the boiler tubes in a few hours. An improbable 'intermittent bouyancy induced flow (IBIF)' solution for the 480 fuel channels burping out steam to be condensed in the boilers was suggested by the industry at the March 8th meeting but it is easy to see that the boilers cannot be a heat sink without liquid water inside the boiler tubes and that the void formation in the channels will foremost pressurize the reactor to failure. Because the odd shaped leaky reactor building will trap copious amounts of hydrogen produced in the first day by steel in piping and zircaloy in core and because there will be wide spread irradiation of public before the 5 hours the irresponsible and misplaced CNSC reports claim are available to re-establish heat sinks.

Meanwhile an unsuspecting public and even our parliament is fed a dangerous and patently untrue position on consequences of a severe accident in Canada and is kept in dark of the true severe accident related vulnerabilities of these reactors. Any person discussing the risks is condemned and marginalized and shouted down in an obscene display of regulatory power. Not a single external idea is found acceptable or of any merit. I can only say that the consultants have served their masters well but forgot public safety in the process. I will in my next opportunity discuss each of the 34 issues and summarize where the process you chose of blanket denial and negativity missed out.

When the right opportunity is presented, I will ask you to disregard the fairy-tale video on severe accident capabilities of CANDU reactors that CNSC has put up on their web-site. I will explain to you how a simple sustained loss of AC power at Darlington will lay to waste a few hundred square kilometers of this pristine Ontario land even today after their obligatory certification of compliance with the paper tigers of the Fukushima Action Items drawn up in the spirit of seem to be doing something instead of actually doing something meaningful. I will explain the dangers of dependence upon and inadequacies of the SAMGs that include shiny pumper trucks, brightly colored additional diesel generators and an inadequate and poorly designed hydrogen mitigation system and a poorly thought out filtered venting - part of the strange, delusional and dangerous 'holistic' approach that some geniuses at Bruce and COG have suggested as a catch-all cure. All this for a reactor with an obsolete 50 year old design and unprecedented severe accident related challenges no other reactor has come close to endangering public safety with, anywhere in the world.

I have documented for public record and conveyed to you that there are things the industry can do to reduce the likelihood of bankrupting the country but refuse to do them perhaps because it exposes the design flaws that would make sale and refurbishment of these reactors impossible to justify. I will tell you that the recommendations of this review are faulty to the core; undertaken with a singular aim of condemning any outside interference into the existing unholy alliances and assembled in complicity to protect the status quo. Citizens of Ontario will pay through the nose for refurbishment of these reactors re-creating an obsolete design that should be retired gracefully or revitalized before it hurts them financially or worse. For centuries these edifices will line highways of Ontario with future generations wondering why our generations were so careless and what gave them the right.

I recommend that you, yourself take a calm look at the CNSC staff safety culture that you have allowed to develop in certain parts of your organization. The country cannot afford a Fukushima and you must not allow the charade and delusions of a 'fiercely independent regulator' ruin the country one day. Let us meet and talk in person for a couple of hours and I will try to convince you that you have had bad advice from a small group of senior CNSC managers.

I also promise you that my reaching out to you again is only one step; and as a nuclear engineer dedicated to the concept of our Canadian ability to design and operate safer reactors, I will not abandon this fight for risk reduction from these obsolete reactors for my people. My hope is that a new CNSC which properly nourishes the expertise of its own staff and respects scientific principles, will itself decide that these tired old design reactors will either be shut down or fixed properly for improved mitigation of severe accident vulnerabilities. Otherwise we should invest in more houses of worship to pray in; unless some event shuts all the reactors down earlier.

Regards,

Sunil Nijhawan, Ph.D, P.Eng.

B. IF THE LICENCE RENEWAL COVERS REFURBISHMENT OF ANY UNITS AT PT. LEPREAU SITE, THE FIRST QUESTION THAT NEEDS TO BE ANSWERED RELATED TO SEVERE ACCIDENT PREVENTION, MITIGATION AND CONTROL CAPABILITIES IS:

What specific standards have been set for severe accident related capabilities for new reactor at design stage and whether a gap report has been prepared or is required to be prepared for the reactor capabilities that would be instilled in the reactor units upon refurbishment.

All questions raised for operating reactor (see A above) also apply to any units in refurbishment plans. No licence renewal should be granted unless satisfactory resolution has been agreed upon at a public technical forum. It is hoped that mature and detailed design requirements and realistic risk targets will be developed by a competent authority for a new generation of Canadian nuclear reactor.

C. CNSC MEMBERS SHOULD LOOK FOR AND PROVIDE TO PUBLIC FOR REVIEW REPORTS ADDRESSING THE FOLLOWING FUNDAMENTAL QUESTIONS ABOUT RELICENSING

1. Does the aging plant still meet the original licensing basis using the acceptance criteria employed by regulators last time the plant was licensed
2. Has any new information changed the understanding of previously employed acceptance criteria within the original licensing basis
3. Does compliance with original licensing basis mean that risk from the original licensing basis is acceptable today
4. Has there been any relaxation of original licensing basis along the way
5. Has an independent, off-shore review of the licensing basis and its compliance been undertaken
6. Will the plant be licensable today in Canada and in other jurisdictions
7. Does/should the public have different expectations of risk today
8. Is it fair that plant be required to meet different public expectations
9. Should risk from accidents previously not considered in licensing basis be evaluated and has it been properly evaluated and acceptable today
10. Is the regulatory regime independent, impartial, competent, effective & relevant

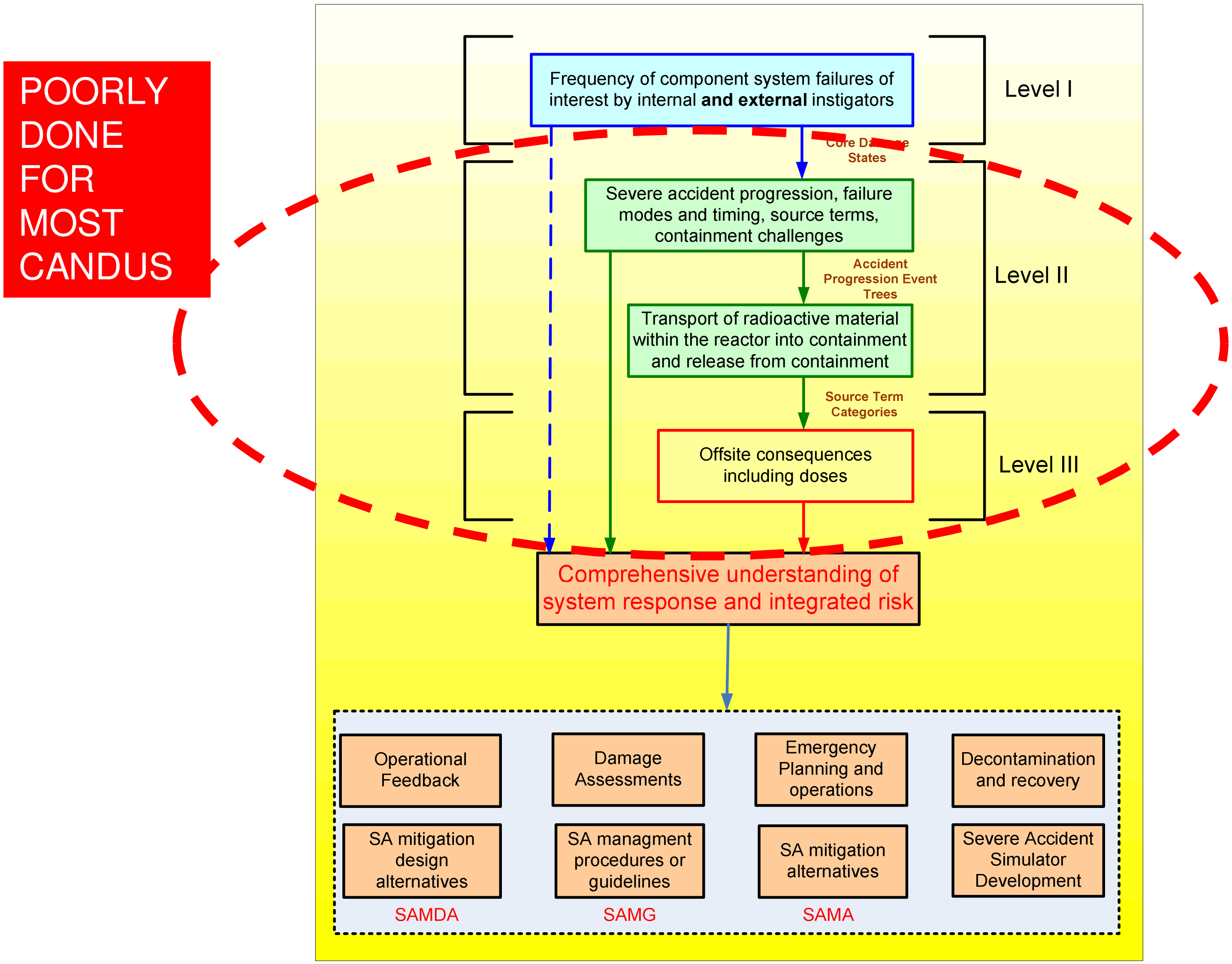
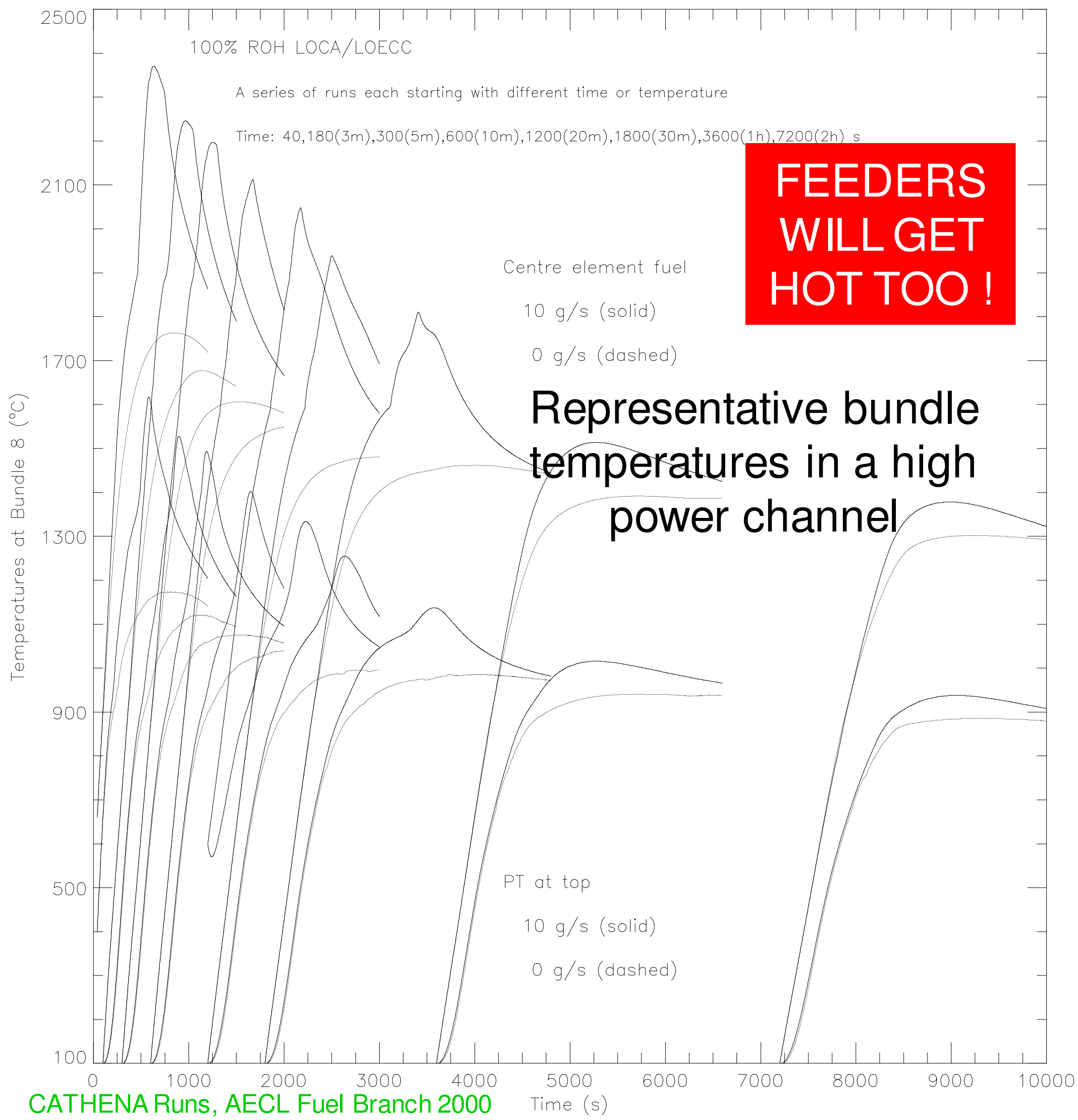


Figure 1 : RISK EVALUATION AND RISK REDUCTION PROCESSES



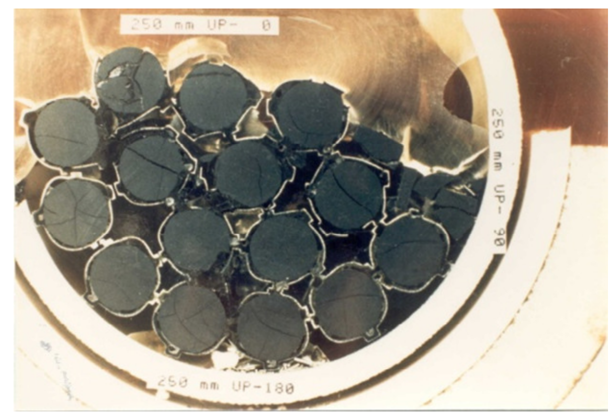


Figure 2 : Example of LOCA + LOECI fuel temperatures as a function of onset of fuel dryout

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									3187	3299	3378	3378	3298	3187								
B						2862	3421	3919	4206	4394	4437	4437	4394	4205	3917	3419	2860					
C					3194	3808	4437	4908	5215	5363	5333	5333	5362	5213	4906	4434	3805	3191				
D				3319	4014	4716	5316	5727	5971	6059	5970	5969	6058	5969	5725	5313	4713	4010	3314			
E			3188	4012	4748	5390	5901	6216	6356	6387	6278	6278	6386	6355	6213	5899	5386	4744	4006	3179		
F			3868	4689	5320	5857	6241	6437	6369	6350	6285	6284	6350	6367	6435	6239	5853	5315	4683	3859		
G		3497	4410	5224	5693	6111	6362	6479	6415	6397	6400	6400	6396	6414	6478	6360	6108	5689	5219	4404	3491	
H		3983	4925	5665	6002	6303	6469	6536	6460	6441	6456	6456	6440	6459	6535	6467	6301	5999	5662	4921	3984	
J	3213	4317	5302	5977	6192	6329	6447	6485	6450	6425	6414	6414	6425	6449	6484	6447	6328	6191	5977	5303	4322	3207
K	3434	4611	5584	6207	6301	6362	6463	6475	6432	6383	6320	6320	6383	6432	6475	6463	6362	6301	6209	5587	4620	3432
L	3577	4763	5739	6365	6493	6515	6526	6495	6427	6346	6228	6228	6346	6427	6495	6527	6516	6495	6368	5744	4775	3578
M	3564	4768	5762	6413	6572	6593	6575	6521	6440	6350	6221	6220	6350	6441	6522	6576	6595	6574	6417	5768	4782	3568
N	3394	4610	5627	6323	6549	6608	6592	6542	6467	6395	6295	6296	6395	6468	6543	6593	6609	6552	6327	5364	4624	3399
O	3184	4334	5365	6113	6467	6596	6585	6557	6495	6458	6444	6444	6549	6495	6558	6586	6598	6469	6116	5369	4345	3186
P		3975	4937	5713	6100	6409	6539	6584	6499	6485	6513	6514	6485	6500	6584	6540	6410	6102	5715	4939	3981	
Q		3470	4368	5173	5639	6070	6343	6482	6439	6447	6481	6480	6447	6439	6483	6343	6071	5640	5173	4367	3471	
R			3762	4534	5096	5656	6129	6390	6372	6402	6392	6393	6402	6273	6390	6130	5656	5096	4534	3761		
S			3045	3818	4479	5156	5784	6182	6385	6459	6372	6372	6459	6385	6182	5784	5156	4478	3817	3044		
T				3073	3756	4485	5155	5640	5944	6076	6003	6003	6076	5944	5640	5155	4485	3756	3072			
U					3073	3756	4485	5155	5640	5403	5384	5384	5403	5206	4832	4288	3596	2926				
V						2548	3187	3761	4132	4384	4463	4463	4384	4132	3761	3187	2518					
W									2961	3174	3300	3300	3174	2961								

Figure 3: Channel Power ranges and feeders affect subsequent behaviour

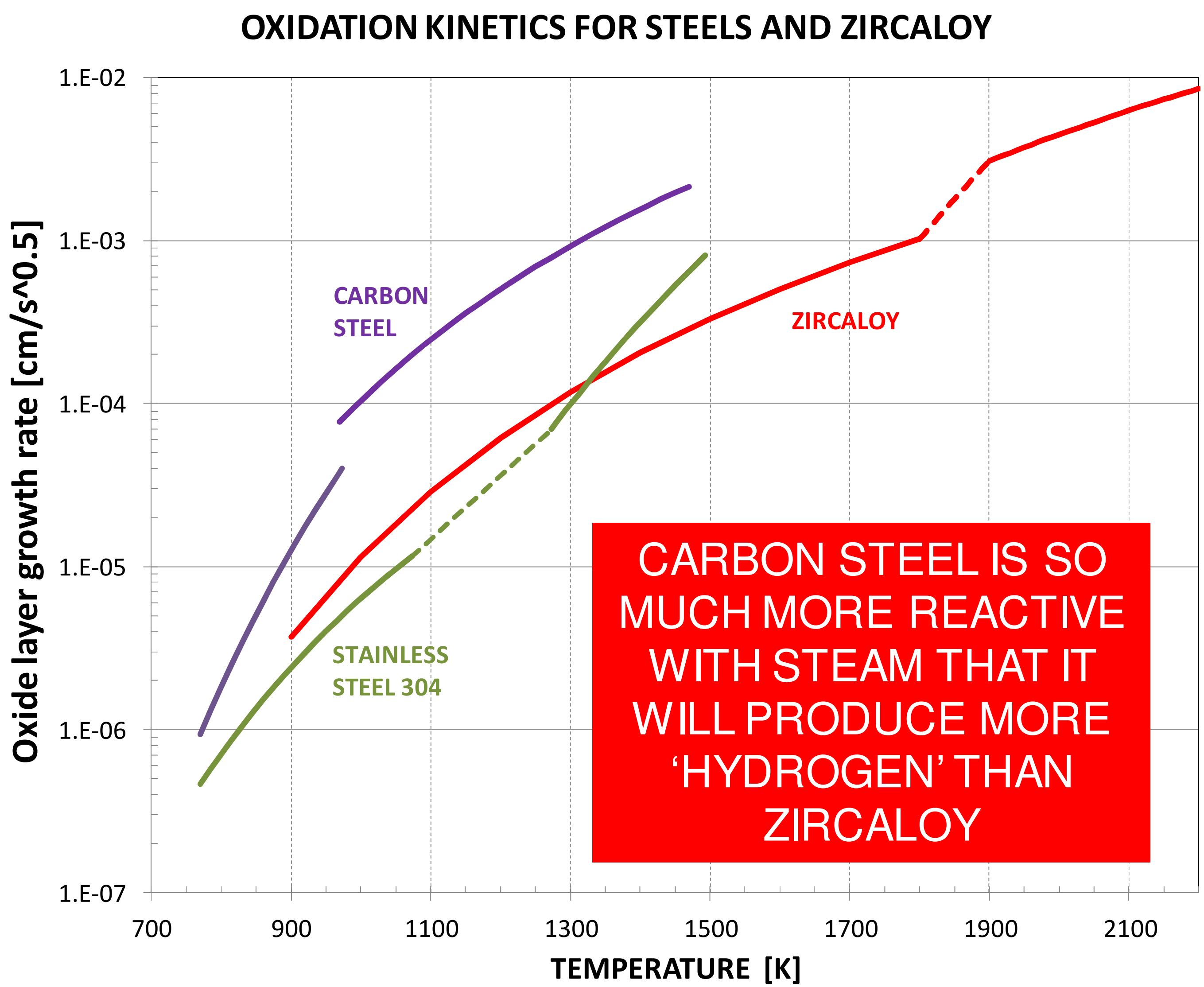


Figure 4 : Oxidation kinetics of different core materials

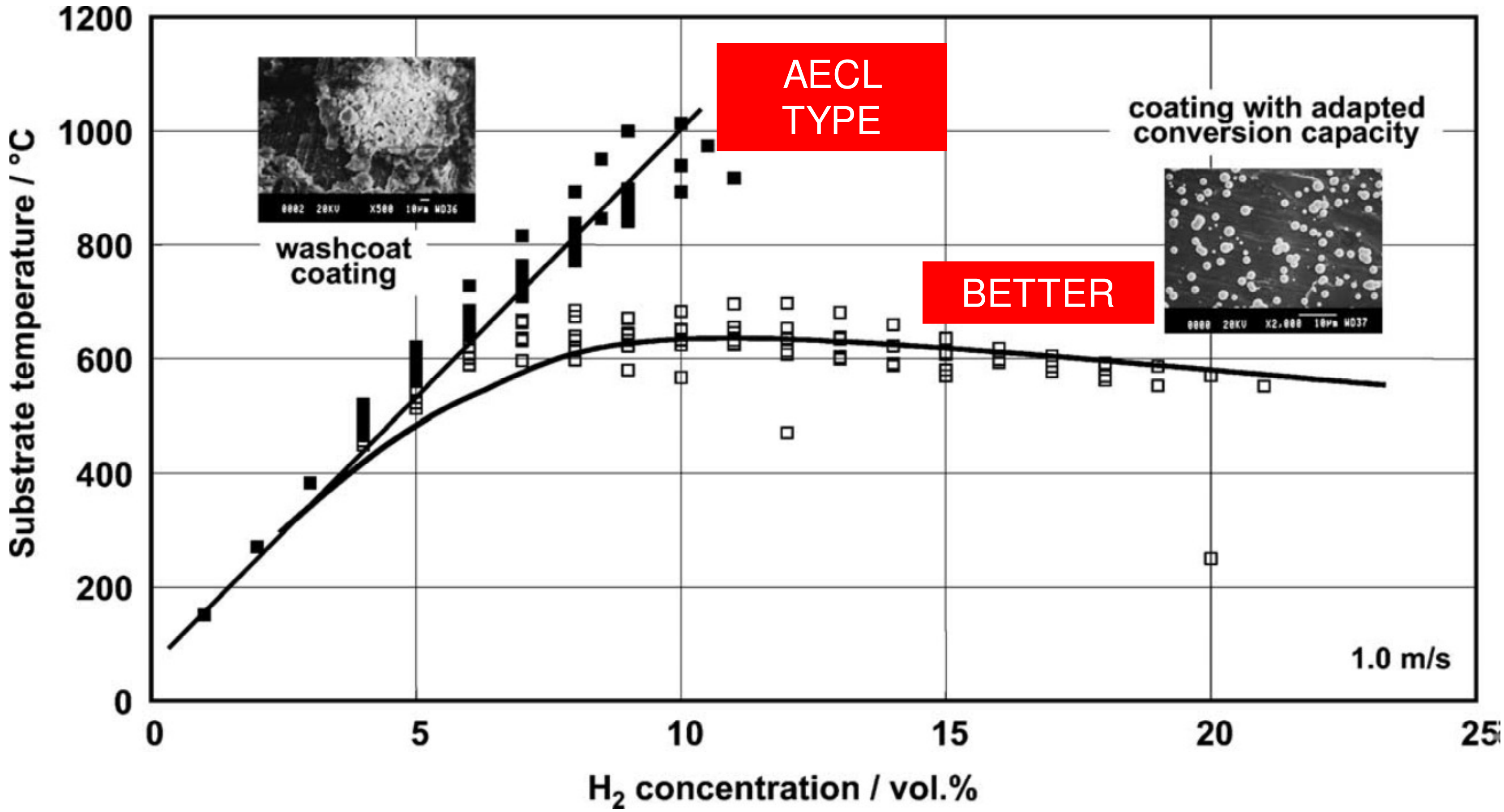


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Figure 5: Typical PARS exit temperatures

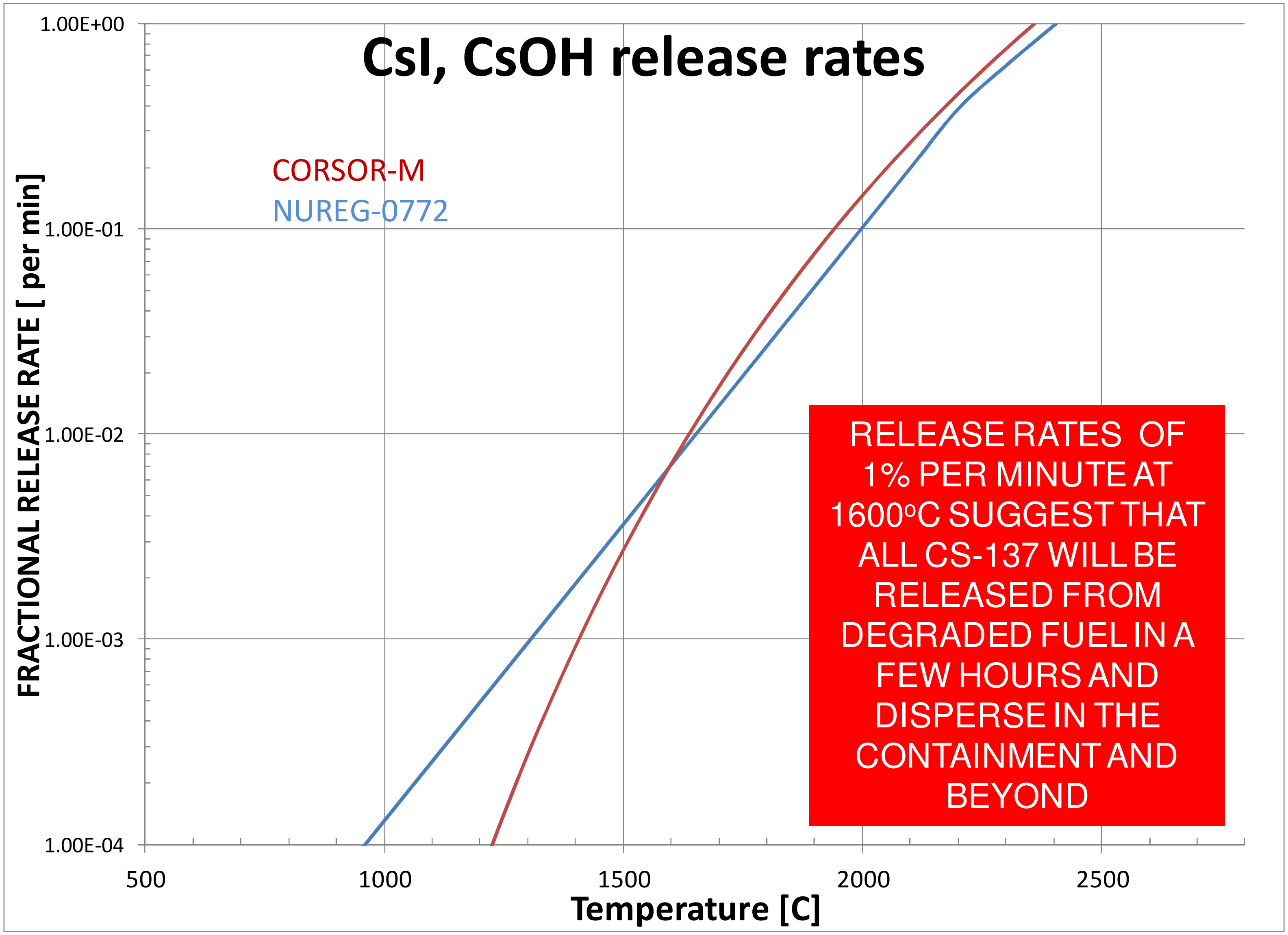


Figure 6: example of fission product release rates

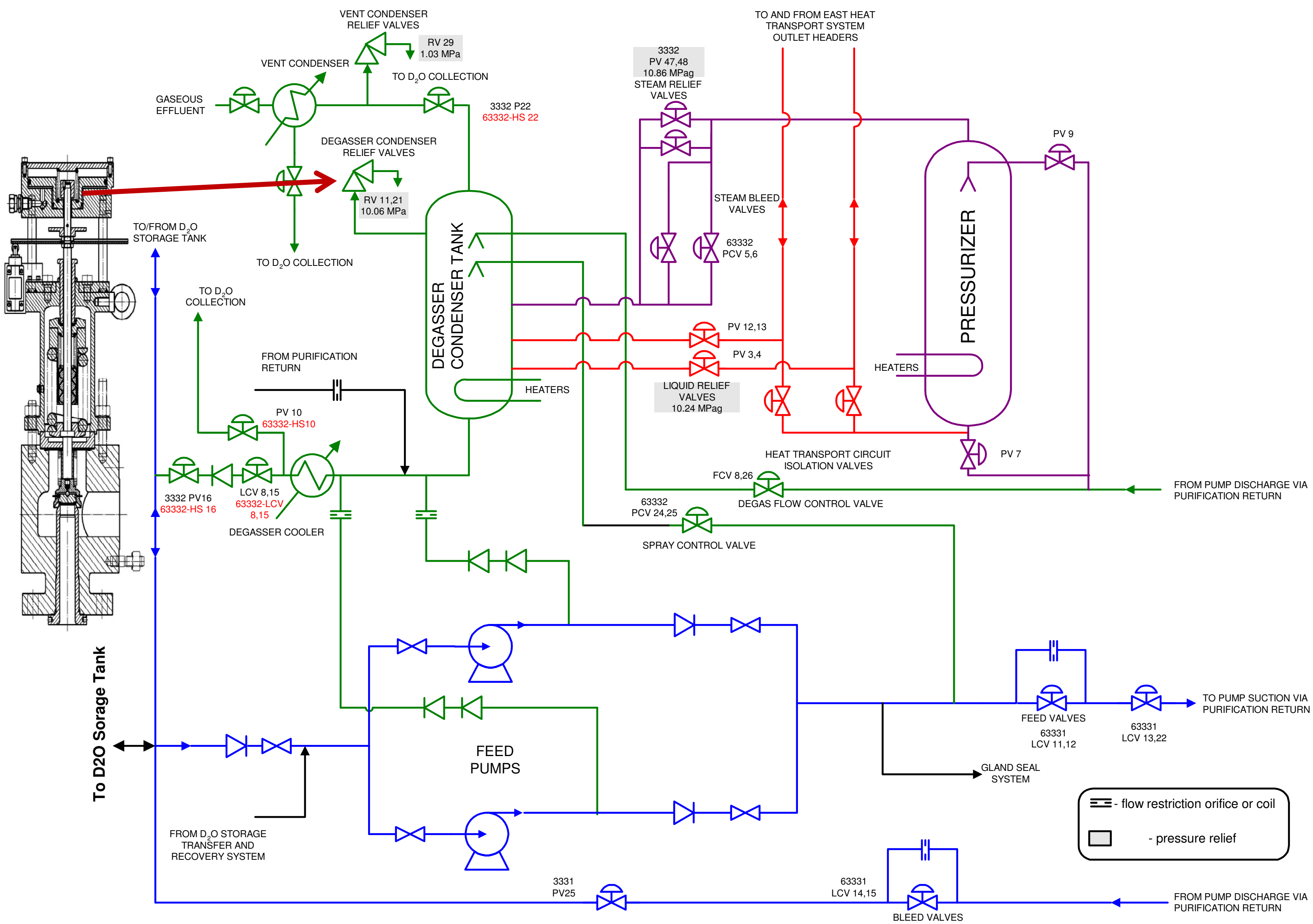


Figure 7: PT. LEPREAU CANDU HTS over pressure protection

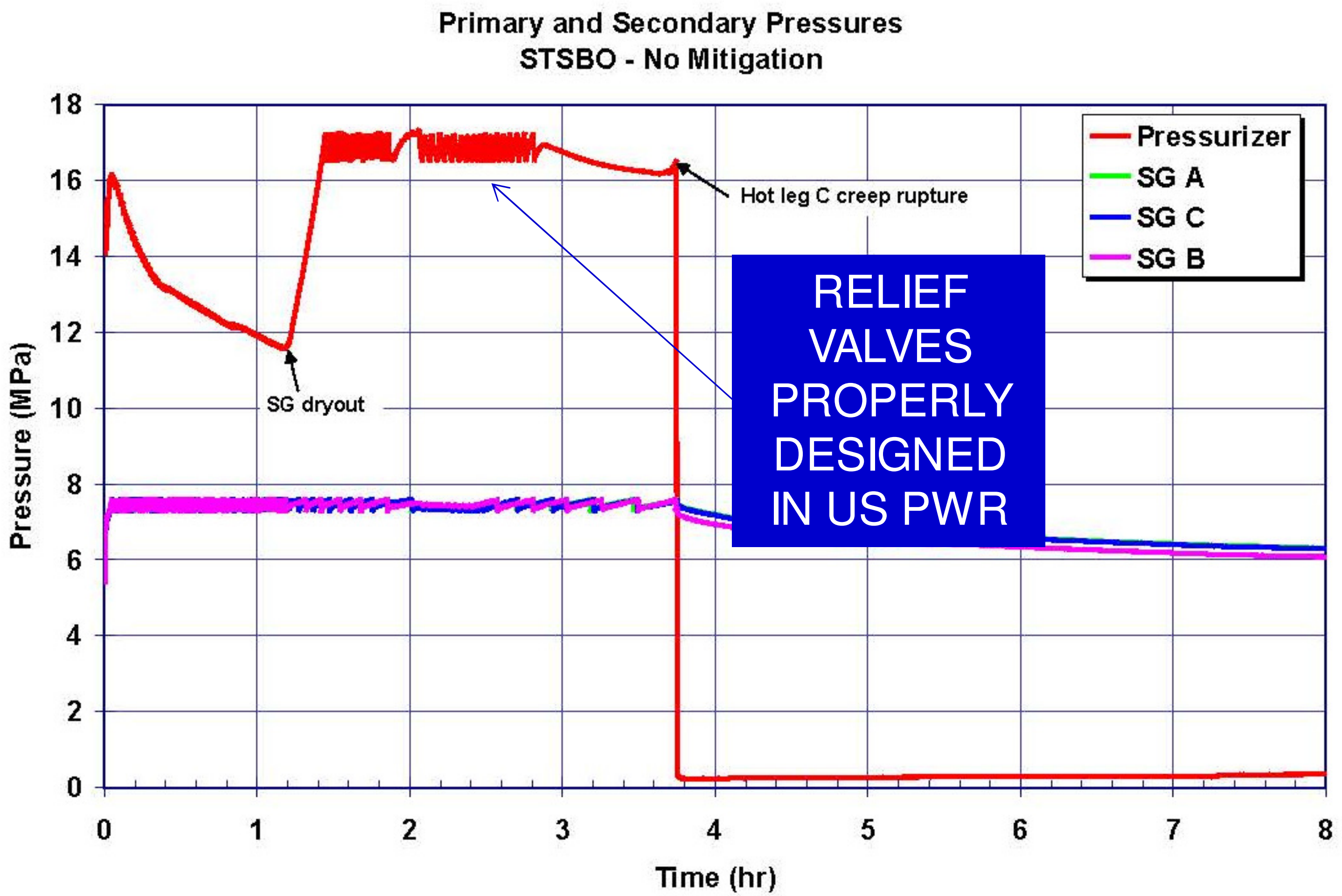
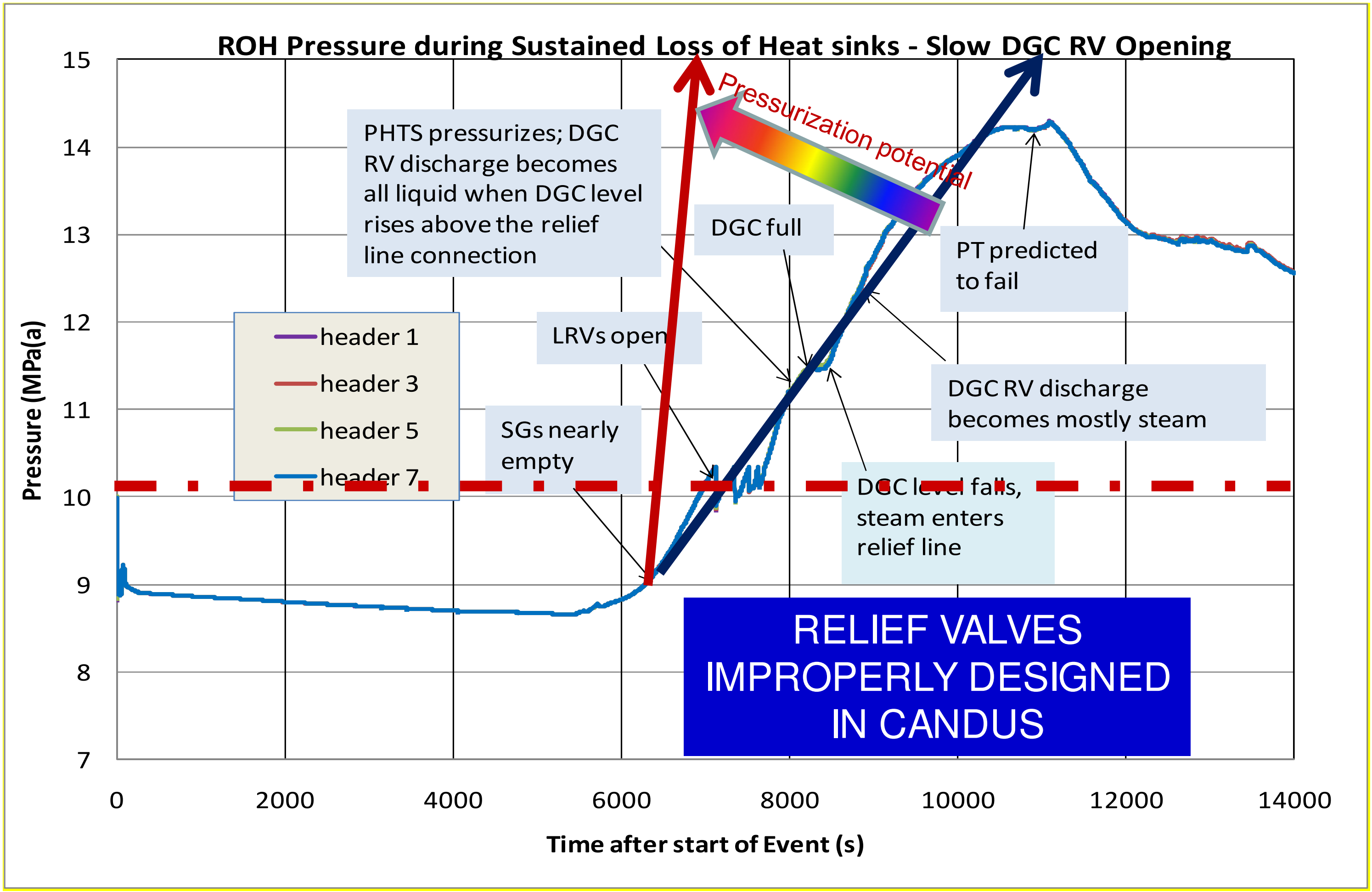


Figure 5-28 Unmitigated STSBO primary and secondary pressures history

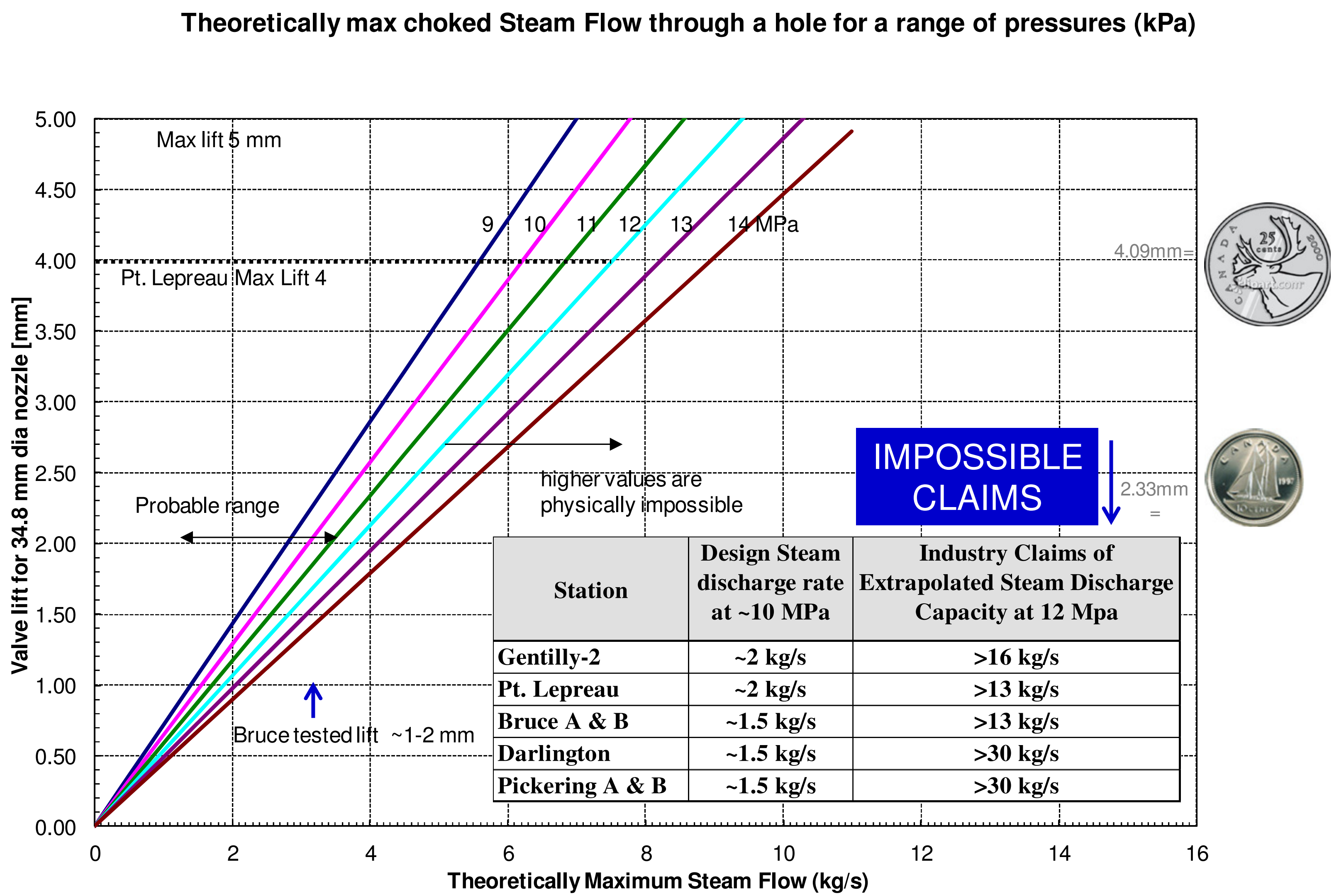
Source – NUREG/CR 7110

Figure 8 : Example of PHTS response to a properly designed relief valve



Source : AECL 2011

Figure 9 : A typical CANDU response to a loss of heat sinks - uncontrolled over pressurization due to improperly designed valves with inadequate steam relief capacity (AECL calculations with my arrows, dark blue notation)



Station	Design Steam discharge rate at ~10 MPa	Industry Claims of Extrapolated Steam Discharge Capacity at 12 Mpa
Gentilly-2	~2 kg/s	>16 kg/s
Pt. Lepreau	~2 kg/s	>13 kg/s
Bruce A & B	~1.5 kg/s	>13 kg/s
Darlington	~1.5 kg/s	>30 kg/s
Pickering A & B	~1.5 kg/s	>30 kg/s

Figure 10: Sample calculations to demonstrate that Pt. Lepreau safety relief valves with 1mm lift cannot relieve enough decay heat steam (~20 kg/s) to avoid an uncontrolled rupture

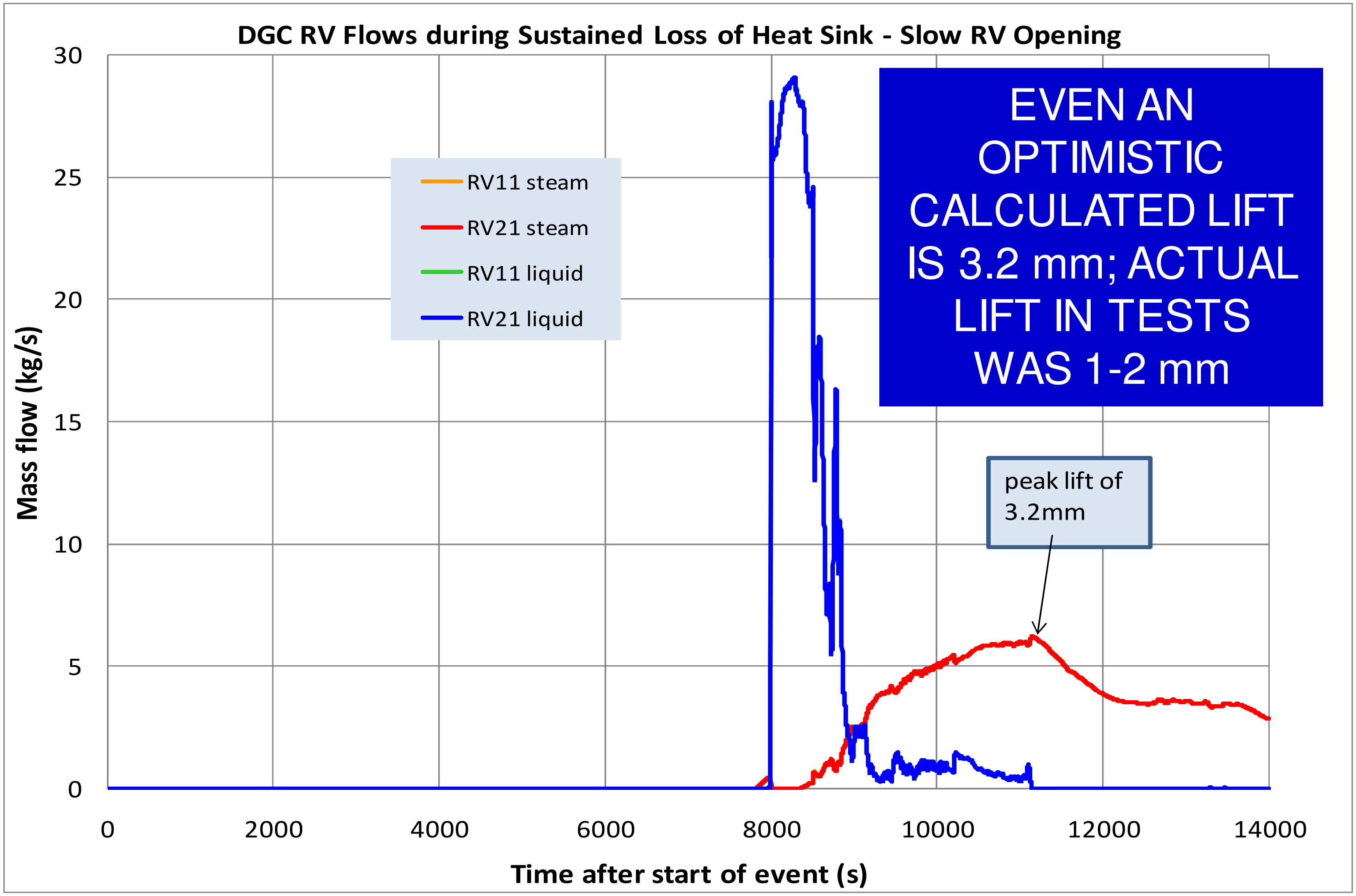


Figure 11 : AECL calculations confirming that the steam relief capacity of the Pt. Lepreau type safety relief valves at <10 kg/s is inadequate and will cause uncontrolled ruptures

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