Enough is Enough

a report submitted by the

Canadian Coalition for Nuclear Responsibility

to the

Canadian Nuclear Safety Commission

regarding OPG's requested

authorization for the

Proposed Pickering B Refurbishment

Dr. Gordon Edwards, Ph.D. and Dr. Sunil Nijhawan, Ph.D.

August 12 2026

RECOMMENDATIONS

Recommendation 1: The Canadian Coalition for Nuclear Responsibility regards it as an unreasonable risk to expose the City of Toronto and environs to the unnecessary threat of a devastating nuclear disaster, however low the estimated probability might be. CCNR therefore recommends that CNSC refuse to grant the licence requested by OPG to refurbish and continue to operate four nuclear power reactors (units 5 to 8) at the Pickering Nuclear Generating Station. This recommendation is reinforced by the following facts:

(1) There are more than 1.1 million people living within 20 kilometres of Pickering NGS, and twice as many living within a 30-kilometre radius;

(2) Every homeowner's insurance policy has a "nuclear exclusion clause" which voids coverage in case of radioactive contamination from a nuclear accident'

(3) As recently as 2016, OPG confirmed its earlier decision that refurbishment of the Pickering B reactors even at a lower cost ($10.7 billion) is unjustified

(4) As of 2010, OPG was fully prepared to see its Pickering Nuclear Generating Station completely shut down by 2030;

(5) The current estimated cost of refurbishing Pickering B is prohibitively expensive compared to other energy options, both nuclear and non-nuclear.

Recommendation 2: If CNSC decides to approve the planned refurbishment of the Pickering B reactors, that it be made mandatory for safety-related items that do not meet current safety standards – such as the existing Degasser Condenser Relief Valves – be replaced with new items that are fully compliant with all modern safety requirements,

Recommendation 3: If CNSC licences OPG to refurbish units 5 to 8, a thorough analysis be done of the potential for hydrogen generation under the most adverse accident circumstances, and the number and positioning of recombiners be planned so as to minimize the likelihood of a massively destructive hydrogen gas explosion. In carrying out such analyses, CCNR urges that special attention be given to the behavioural characteristics of deuterium as opposed to protium.

Recommendation 4: For the benefit of emergency planners, that CNSC provide a series of possible source terms for cesium-137 releases, following a severe nuclear accident, with 100 terabecquerels being the smallest entry. This will encourage first responders to envision and plan for a spectrum of different responses depending upon the severity of the release. Moreover, that CNSC staff publish proportionality documents indicating in each case the anticipated magnitudes of release of other radionuclides of public health or environmental importance.

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The Canadian Coalition for Nuclear Responsibility

The Canadian Coalition for Nuclear Responsibility (CCNR) is a federally incorporated not-for profit organization, dedicated to education and research on a wide range of nuclear issues, both civilian and military – including non-nuclear alternatives – especially those pertaining to Canada.

CCNR has been actively intervening in reactor licensing decisions for more than five decades. In the early years, CCNR provided expert testimony on nuclear reactor safety, and cross-examined nuclear experts from Ontario Hydro, Atomic Energy of Canada Limited, and the Atomic Energy Control Board over a period of three months, during public hearings of the Royal Commission on Electric Power Planning (1976-77-78).

In its 1978 report on Nuclear Energy in Ontario entitled A Race Against Time, the Commission found that Dr. Edwards' estimate of the probability of a core meltdown in a CANDU reactor – based on published industry reports – was “more credible” than estimates given by nuclear industry experts.

Dr. Edwards, as president of CCNR, presented five times to the Select Committee on Ontario Hydro Affairs (1979-1980) (always by invitation) on nuclear reactor safety and other related issues. A couple of years earlier, in 1976, at the Cluff Lake Board of Inquiry into Uranium Mining in Saskatchewan, Dr. Edwards cross-examined Dr. Norman Rasmussen, head of the Nuclear Engineering department of MIT and principal author of the fourteen-volume US NRC Reactor Safety Study in 1974.

Over the years, CCNR has actively participated in public hearings related to Pickering A & B, Bruce A & B, Gentilly-2, Point Lepreau and the NRU reactor at Chalk River. In more recent times, CCNR provided expert testimony to the Joint Review Panel for the Darlington New Build project back in 2010-2011. CCNR has subsequently given expert testimony to CNSC on three separate occasions – 2022, 2023, 2024 – after the BWRX-300 reactor design was chosen by OPG as their reactor of choice for DNNP.

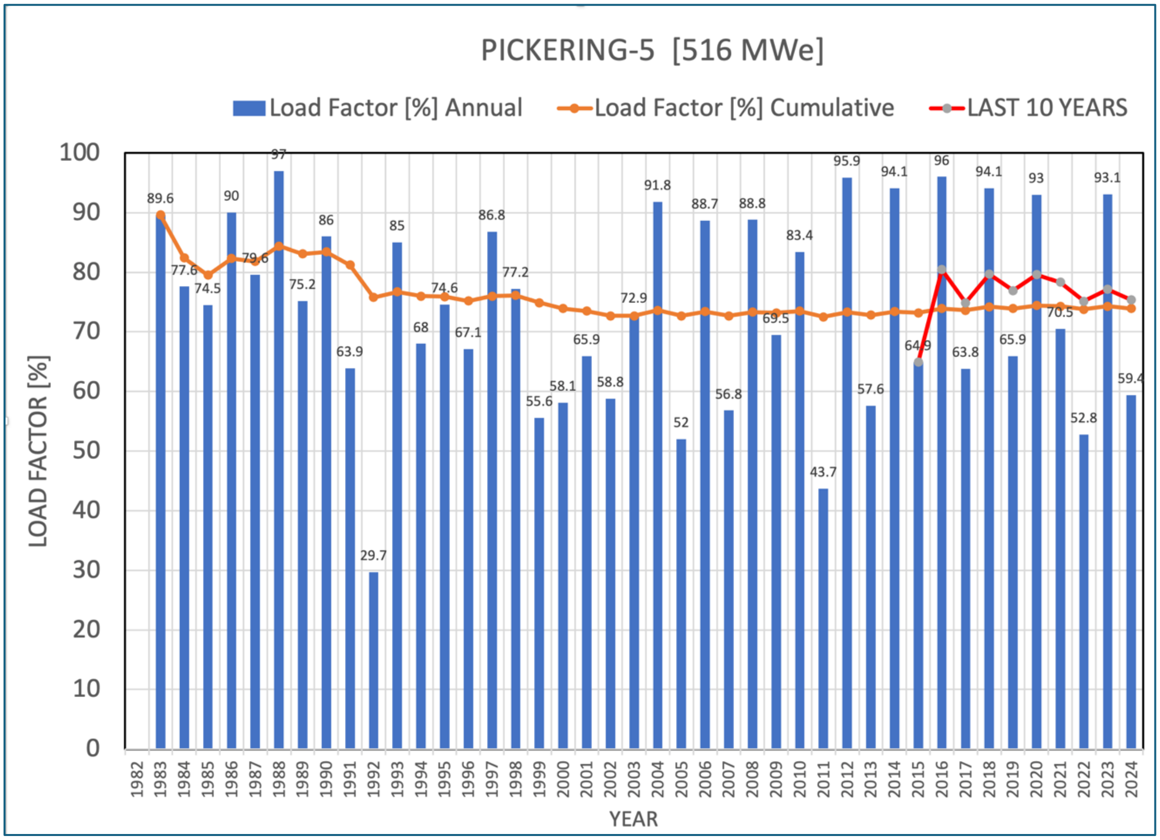
The Pickering Licence Application

OPG has requested a 10-year extension for the Pickering Nuclear Generating Station Power Reactor Operating Licence (PROL), incorporating the Pickering Waste Management Facility in that same licence application. In addition, OPG is requesting permission to refurbish the four Pickering B reactors, units 5 to 8, and continue to operate them, while proceeding with the initial decommissioning stage for the four shut-down Pickering A reactors, units 1 to 4. This would result in a “storage with surveillance” provision for managing the radioactive waste resulting from all these activities, presumably on the existing site.

OPG submitted a request in June 2025 to refurbish and continue to operate units 5 to 8. The refurbishment would include major component replacements (fuel channels, feeder pipes and steam generators) along with other enhancements, including a new deep water intake, a new containment filtered venting system, and new high wind hazard protections.

The total cost of this refurbishment has been estimated at $26.8 billion. That’s more than twice the cost of refurbishing the four Darlington reactors, with a budget of $12.8 billion. Moreover, the Darlington reactors are considerably more powerful than the Pickering reactors, with an electrical capacity of 3512 megawatts for Darlington versus 2086 for Pickering. So the Darlington refurbishment has cost about $3,644 per kilowatt, while the Pickering refurbishment is expected to cost $12,847 per kilowatt. That’s about three and a half times more expensive in terms of a cost/benefit ratio.

Pickering units 5 to 8 are very old reactors, designed over fifty years ago. They have not been performing well in recent years (only 75 to 80% capacity factors) for reasons that have no obvious connection to any of the major core components that OPG intends to replace. It is therefore unlikely that refurbishment will significantly improve their performance. Taking into account less than 80 percent capacity factors, the Pickering B refurbishment project is well over 4 times more expensive (per kilowatt of power) than in the Darlington case.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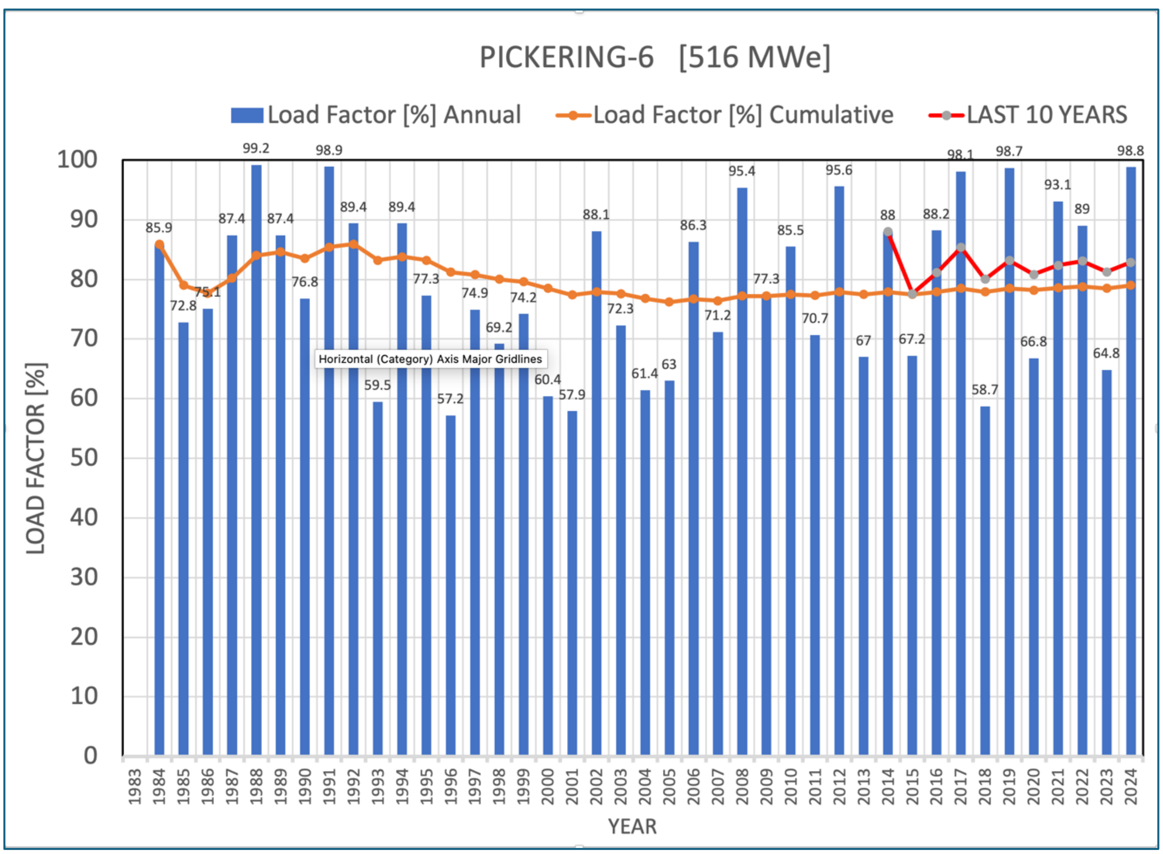
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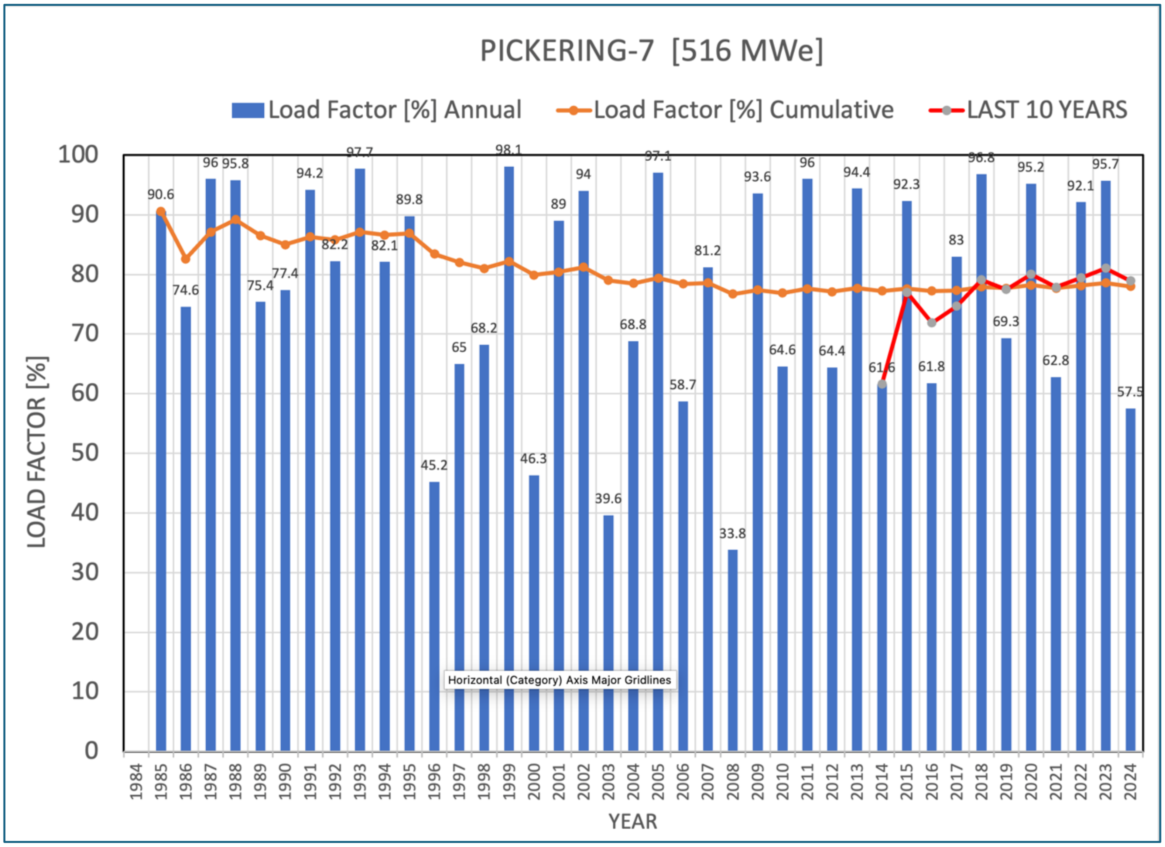
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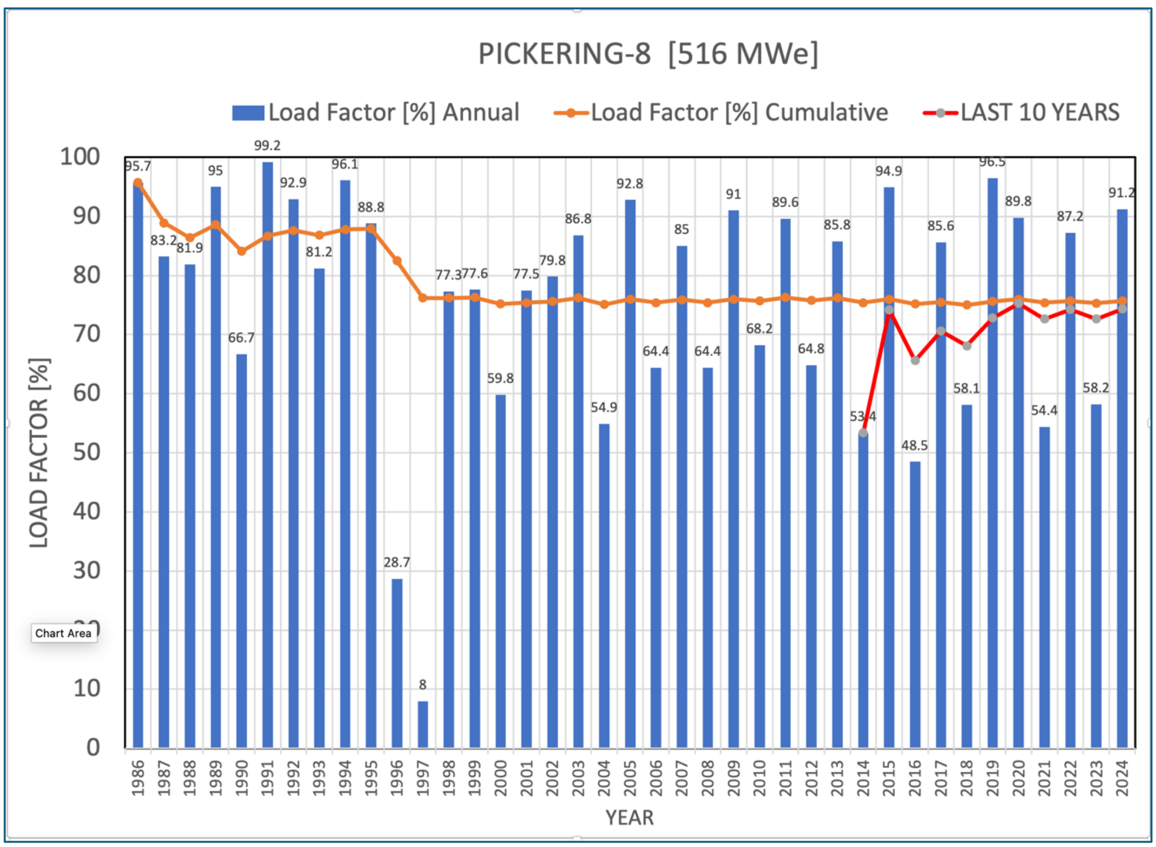
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Chart Area

It is unclear how OPG derives its confidence that the refurbished Pickering B reactors will perform adequately for forty years after refurbishment, when previous attempts to revitalize the four Pickering A reactors near the turn of the century were so unsatisfactory. Those costly efforts resulted in two of the Pickering A units being permanently shuttered and the other two managing to add only another two decades of performance.

In S. Korea, the only CANDU refurbishment so far has been the Wolsung-1 reactor. That unit was permanently shut down in December 2019 after having been refurbished less than ten years earlier, from 2009 to 2011 – despite a complete replacement of all fuel channels and feeder pipes. New Brunswick Power spent between $2.4 and $3.3 billion to refurbish its Point Lepreau reactor from 2008 to 2012, only to face erratic operation in recent years and lengthy outages. So far in 2026 Point Lepreau has been offline for more than 4 months, since April 10. The previous year, 2025, the plant experienced a five month outage, and the year before that, a seven-month outage. Evidently, a refurbished CANDU reactor is not at all equivalent to a newly built reactor. Hydro Quebec cancelled its proposed refurbishment of the Gentilly-2 CANDU reactor, choosing to shut it down permanently in December 2012.

The four Darlington reactors were refurbished over ten years, from October 2016 to March 2026, "within budget and ahead of schedule" according to OPG. However, two of the four refurbished reactors went offline in June 2026, with one of those outages expected to last 8 or 9 months. And major outages at Darlington are already being planned for 2029. As at Point Lepreau, most of these post-refurbishment outages are related to overlooked problems experienced in the non-nuclear "conventional" parts of the plant, but it serves to illustrate the point that a refurbished reactor is not a new reactor. Like an old car, fixing one thing often leads to other things going wrong.

CCNR is concerned that the safety of refurbished plants may be crippled by undetected oversights during the refurbishment process, only to be discovered during an emergency! The safety culture within the nuclear community is of the utmost importance in this regard.

Safety Culture and Communication in the late 1990s

On December 31, 1997, all the operating Pickering A and Bruce A reactors were suddenly shut down by Ontario Hydro for safety reasons. This action was quite dramatic. It meant the immediate overnight closure of two four-unit operating CANDU power stations. (One of the Bruce A reactors, unit 2, was already shut down for other reasons at the time.)

Such a dramatic turnabout – shutting down over one-third of the province's fleet of twenty nuclear power reactors – was unprecedented in the history of nuclear power up to that time. It remained unparalleled until August 6, 2011, when the German government shut down eight of its seventeen power reactors at one stroke in the wake of the Fukushima triple meltdown in March of that year. The German action was just the opening salvo of a planned nuclear phaseout in that country – a phaseout that was concluded in 2023 with the final shutdown of the last German nuclear power reactor.

The circumstances surrounding Ontario's shutdown of multiple reactors was of a different nature. It was the result of a shocking 64-page "report card" (henceforth called the Andognini report) released on July 21 1997 assessing the safety of Ontario's nuclear power reactors. The report was based on an investigation by a six-person SWAT team of nuclear experts hand-picked by Carl Andognini, himself an American nuclear engineer.
See www.ccnr.org/hydro_report.html

Andognini was hired as Ontario Hydro's Chief Nuclear Officer in 1996 to address "long-standing management, process and equipment problems in Ontario Hydro Nuclear (OHN) plants." [from the Executive Summary of the Andognini Report]. He quickly formed a Nuclear Performance Advisory Group to conduct a three-month Independent Integrated Performance Assessment of Ontario Hydro Nuclear.

The Andognini report charged that "…all of the operating stations (Pickering, Bruce and Darlington) are 'minimally acceptable' [in terms of safety]. The rank 'minimally acceptable' is consistent with the lower ranks that the Institute of Nuclear Power Operators (INPO) would issue and still permit the plant to operate if in America."

The report cited "excessive human error, degraded equipment, and reactor control room employees who receive just half the training hours of their U.S. counterparts. There are workers too terrified to send bad news up the ladder, and senior managers who do not know what subordinates are doing. Deficiencies, say the authors, exist in virtually all performance areas. The problems are well-known, are long-standing and have gone unaddressed." *from Ontario Hydro Meltdown, Macleans, March 17 2003*

The obvious question is, where was the nuclear regulator during all this commotion? Why did Ontario Hydro management have to hire independent investigators to find out about the safety of their own fleet of 20 nuclear power reactors? Shouldn't the regulator have been "on the job"? Or at least communicating with OPG management about safety problems?

And how does the ranking "minimally acceptable" relate – if at all – to the concept of "unreasonable risk", which CNSC is legally mandated to prevent since its inception in the year 2000? Has there ever been a situation in which CNSC has declared a given risk to be "unreasonable"? Indeed, who at CNSC actually decides whether a given risk is reasonable or unreasonable, and according to what criteria? Does the term "unreasonable risk" – as used in the Nuclear Safety and Control Act – have any legal or scientific definition?

Back then, the regulator was the Atomic Energy Control Board (AECB), still operating under the 1946 Atomic Energy Control Act. AECB is the precursor to the Canadian Nuclear Safety Commission (CNSC). OPG only came into existence in April 1999, as part of the provincial government's plan to break up and restructure Ontario Hydro into separate successor corporations. These were major organizational changes in the nuclear field.

During this period it was made very clear to the public that failure to communicate clearly was a fundamental difficulty in the nuclear field. William Farlinger, Chairman of Ontario Hydro, told a press conference that the problems in Hydro's nuclear division had been percolating for a decade, and that the divisional culture, which he likened to "some sort of special nuclear cult," was at the heart of the problem. But he remained steadfast that the people at the top did not have a clue. That drew frustrated objections from Pickering regional councillor Maurice Brenner. "We've been telling you that for a number of years," he said of the management problem. "Over and over again, sir."

Ontario Hydro Meltdown, Macleans, March 17 2003

In a later interview, Farlinger, a former accountant, said that when he arrived at Ontario Hydro in November, 1995, "I kept hearing, 'The nuclear company does this. The nuclear company does that.' I often said, 'You people talk about this as if it's a different company.' " The nuclear division, according to Farlinger, felt it should not be able to be penetrated by the rest of the corporation. "The issue of culture," echoed Andognini, "is that nuclear tries to keep everything confined within it. [They] won't tell anybody anything. That's not just the Canadian culture, it's the nuclear culture."

Ontario Hydro Meltdown, Macleans, March 17 2003

The Andognini report focussed special attention on this 'nuclear culture'. "There are pronounced cultural problems that need to be addressed. The malaise is deep and wide and continues to worsen. It will not be quickly or easily corrected…. This culture is the result of long-standing managerial tolerance for cumbersome and ill-defined work processes that make it virtually impossible to get work done at the shop floor level and in the engineering office…. It's acceptable to cut corners. It is not acceptable to make waves. Those who have made waves have been fired or sidelined. The messenger with bad news will be told to fix the problem." *Ontario Hydro Meltdown, Macleans, March 17 2003*

One of the striking revelations of the Andognini report was the many thousands of unresolved safety-related maintenance problems, in a very long list that just seemed to keep getting longer every year.

> *"Long standing management, process and equipment problems in Ontario Hydro Nuclear plants are well known but have not been aggressively resolved. As a result, the overall performance of OHN is well below the level of performance typically achieved by the best nuclear utilities. Immediate attention is needed to improve performance so that the value of OHN's assets does not depreciate beyond recovery." Andognini report, Executive Summary*

Pushing the Boundaries

OPG's current request for permission from the powers that be to refurbish Pickering B at such exorbitant expense is surprising, given the plant's checkered history. OPG firmly announced in 2010 that it was not going to refurbish the Pickering B reactors at all, but just to run them as long as permissible. This was after the corporation had experienced embarrassing cost overruns and delays in trying to refurbish and restart the four Pickering A reactors following a lengthy shutdown of those reactors in 1997 (for safety reasons that were not specific to Pickering, but endemic to the entire fleet of Ontario's nuclear reactors.)

In 1999, the newly created OPG estimated that the four Pickering A reactors, units 1 to 4, could be spruced up and restarted at a cost of $1.1 billion; that is, on average, $275 million per reactor. But the restart of Pickering A unit 4 alone ended up costing $1.25 billion, greatly exceeding its estimated cost, and two years behind schedule. A review panel, presided over by former federal Minister of Natural Resources Jake Epp, investigated.

The panel released an official report in December 2003 criticizing OPG for severe mismanagement, flawed project assumptions, and massive cost overruns during the restart of the Pickering A reactor. The panel strongly advised against restarting the remaining laid-up units without a rigorous business plan and independent verification.

In March 2004, a Commission of Inquiry headed by former Deputy Prime Minister John Manley urged the reactivation of one more reactor – Pickering A, unit 1 – in hopes that that safety upgrade and restart could be accomplished at lower cost – say, $500 million. The other restarts would follow. In September 2005 unit 1 returned to the grid at a cost of about $1 billion. Too much. That settled it: units 2 and 3 were never going to be restarted. The job of relaunching just two of the four reactors had cost twice what the entire restart operation of all 4 reactors was expected to cost. Units 2 and 3 were never to be returned to service.

The Customer is Always Right

In all of these discussions about restarting the shut-down reactors, nuclear safety was seldom if ever mentioned. It was not even revealed publicly that since 1977, all of the Pickering A reactors had been unwittingly operating out of compliance with the AECB's safety requirements, because none of those reactors had two fully independent, fully effective, fast shutdown systems as mandated.

This requirement hearkened back to 1952, when the world's first major nuclear accident occurred at Chalk River Ontario: the NRX research reactor suffered a partial meltdown. Because of a sudden unexpected power rise in the reactor core, the supervisor ordered an emergency shutdown – accomplished by rapidly inserting neutron absorbing "control rods" into the core of the reactor. Well, it didn't work. Some of the rods didn't descend at all, some got stuck partway. As a result the reactor blew itself apart, radioactivity was released to the environment, and a partial meltdown ensued.

Future US president Jimmy Carter was among those who participated in the radioactive cleanup following the NRX accident. He was a nuclear engineer in the US navy under Admiral Hyman Rickover at that time. So he was better prepared than most when the partial meltdown at Three Mile Island happened during his presidency in 1979.

Everyone knew that if a large power reactor were to self-destruct the way the small NRX research reactor had done, the consequences would be disastrous – especially with Pickering so close to Toronto. So from the get-go, every Canadian power reactor had to have two fully independent shutdown systems. In case one fails, the other will do the job.

The designers of Pickering A utilized two shutdown systems: (1) control rods, and (2) moderator dump. For technical reasons, *w*ithout the heavy water moderator surrounding the fuel in a CANDU reactor, no nuclear chain reaction is possible. So if you can just make the moderator disappear, the chain reaction screeches to a halt: disaster averted. How do you make the moderator go away? Simple. Just dump it into a large tank under the reactor.

All the newer Canadian reactors had abandoned the moderator dump, opting instead for a very fast-acting "liquid injection" shutdown system. Nozzles inside the reactor core spray a neutron-absorbing fluid – gadolinium nitrate – into the moderator water, stopping the flow of neutrons and shutting down the reaction. But the Pickering A reactors had the moderator dump built-in, and no "liquid injection" system. (Technically, it is called "poison injection" because of the effect it has on the neutrons in the reactor core!)

Then came the 1986 Chernobyl nuclear meltdown in the Ukraine. A shocking event. By the early 1990s, Ontario Hydro engineers, working with the Atomic Energy Control Board staff, concluded that the moderator dump could not avert a disaster in all cases because it is too slow. It takes more than ten seconds for the moderator to drain out of the core. The chain reaction has to be stopped in less than two seconds or the core will be destroyed, making massive leaks of radioactivity into the environment inevitable.

AECB wanted Ontario Hydro to make these reactors compliant with current safety regulations, by installing a poison injection fast shutdown system. Ontario Hydro balked. Rather than refusing to licence these old reactors for safety reasons, AECB knuckled under by accepting a cheaper fix involving the installation of an additional bank of shutoff rods.

When the Pickering A units were poised to be restarted, the newly created CNSC uncritically accepted AECB's compromise to give the Pickering A reactors a "free pass". From CCNR's perspective, CNSC began its mandate with an act of capitulation.

Indeed, the newly-created Canadian Nuclear Safety Commission seemed to be putting the city of Toronto and environs at added risk, unnecessarily so, to satisfy the wishes of the newly-created Ontario Power Generation. Neither OPG nor CNSC sees it that way, but the point is that neither the Canadian public nor their elected representatives were consulted. The decision was made by the nuclear regulatory people and their colleagues in the industry. Just over a decade later, the Japanese Diet's 2012 report on the root causes of the Fukushima Daiichi triple meltdown laid the blame on "collusion" between the government, the regulators, and the utility. It's called "regulatory capture."

The fact that Pickering A units 1 and 4 were operating without benefit of two fully independent fast shutdown systems was so little known that even Dr. Binder, as President of the CNSC, did not know it. This shocking fact is revealed in the transcript of a CNSC hearing devoted to the relicensing of the Pickering Nuclear Generating Station in 2018:

> **THE PRESIDENT:** Okay. Let me try to explain what I understand from Staff. Pickering A was first built with one shutdown, Pickering B – and the rest – were with two. When the instruction was to improve, they put in an alternative means to a fast shutdown, which is the gadolinium and all that stuff. Is that my understanding?
> **DR. EDWARDS:** No, that's not right.
> **THE PRESIDENT:** The alternative stuff is as good or is as functional as having two shutdowns. Am I understanding correctly?
> **MR. FRAPPIER:** Gerry Frappier, for the record. Not quite. So Pickering A does not have a gadolinium system.... Perhaps I could ask Dr. Al Omar to explain exactly where it's centred around the shutdown system for Pickering A?
> **DR. OMAR:** Al Omar, for the record. Again, Pickering A originally has two systems to assist the shutdown of the reactor. One of them was slow – or judged slow – which is the dump tank.... So we have two shutdown systems now for Pickering A. The only difference is that the extended or enhanced shutdown system, we'll call it shutdown system 2, does not have gadolinium injection, but we added absorber rods, but it satisfied the design function, safety and regulatory requirement as shutdown system 2.
> **THE PRESIDENT:** Okay, thank you. Dr. Edwards, anything else you want to ask?
> **DR. EDWARDS:** Well, it's definitely cutting corners, there's no question about it. I mean, it's an old reactor. You would not require the Bruce reactors, the Darlington reactors, the overseas reactors, you wouldn't require them to have two fast shutdown systems if one was enough. If you solved the problem for Pickering A so that you have a fast shutdown system based on rods, one that's totally adequate ... then why go to the extra expense of having two shutdown systems in every other CANDU reactor?

All the hoopla about correcting an ingrown nuclear culture and fixing the industry's failure to acknowledge problems – as laid out in the Andognini report – had apparently fallen on deaf ears. To many observers, the nuclear industry seemed to be operating in a world of its own, like a state within a state.

Damn the Torpedoes

Between 2006 and 2009, OPG completed an assessment of the scope, cost and duration of a project to refurbish the four Pickering B units in order to extend their life by 30 years. The cost estimate at that time was $10.7 billion. The economic assessment was unfavourable and so, early in 2010, the OPG Board announced its decision not to proceed (see Annex B) with the Pickering B Refurbishment. (As it happens, Units 1 and 4 of Pickering A were shut down for good in 2024, so both of those reactors only managed to gain another 20 years of life after being "fixed up" – not 30, and certainly not 40, as is now predicted for units 5 to 8.)

Running the Pickering B reactors until 2020 would take OPG and CNSC into uncharted territory. Without replacing the fuel channels – acknowledged by OPG to be the "major life-limiting components" of any CANDU reactor – the existing embrittled fuel channels would have to be operated well beyond their specified design lifetime of 210,000 Equivalent Full-Power Hours (equivalent to 24 years of operation at full power).

In 2010, CNSC granted a three-year licence for Pickering NGS. That period of operation would take all four reactors to the brink of the 210,000 EFPH limit. Thus began the adventure of extending the operating licence of the Pickering B reactors well beyond the limiting value of 210,000 Equivalent Full-Power Hours that had previously been regarded as a final benchmark, indicating the end-of-life for these reactors.

In 2013, CNSC granted Pickering a five year licence extension to 2018, allowing units 1 to 5 to operate up to 247,000 EFPH. A further ten-year extension of the operating licence from

2018 to 2028 was subsequently granted, extending this crucial limit to 295,000 EFPH, a target that was later updated to 305,000 EFPH.

The current understanding from CNSC is that the Pickering B reactors will all be shut down either by December 31 2026 or when the pressure tubes reach 305,000 equivalent full-power hours, whichever comes first. Accordingly, all four reactors will be shut down by December 2026 and will remain shut down in anticipation of the four-reactor refurbishment of units 5 to 8 at an expected cost of $26.8 billion. That will require a new operating licence – and that is what OPG is hoping to get from CNSC by the end of 2026.

Pressure Tube Anxiety

Why are pressure tubes the "major life-limiting components" of a CANDU reactor? It's because they are needed to keep the nuclear fuel from overheating and melting. If that happens, radioactive materials are released and can be disseminated into air or water.

The core of a Pickering B reactor has 380 horizontal pressure tubes. Each pressure tube holds 12 fuel bundles, end-to-end. During operation, heavy water coolant flows through the pressure tubes, exiting the core at a temperature above 300 degrees C. The coolant is under so much pressure that it is not allowed to boil: hence the term "pressure tube".

The coolant carries the intense heat produced by the fissioning fuel to one of the 12 boilers, located outside the core, where steam is produced. Just as in an oil-fired or coal-fired plant, steam is then used to turn the blades of a turbine to generate electricity.

If coolant is lost due to a pipe break or for some other reason, we have a Loss of Coolant Accident or LOCA. This is an emergency and must be dealt with immediately. Failure to keep the fuel cooled can lead to a major nuclear accident with long-term consequences. For this reason, it is vital that the pressure tubes are kept in top shape and are not allowed to become unserviceable. The longer a pressure tube is in the core while nuclear fission is taking place, the more brittle it becomes and thus more prone to break or rupture. Public safety as well as corporate well-being depends on not allowing the pressure tubes to become too brittle. This can best be done by limiting the length of time they spend in the core. Eventually, they have to be replaced with brand new pressure tubes, for safety's sake.

The longer the old pressure tubes are left in the core without refurbishment, the greater the risk of tube failure and LOCA. For that reason, CCNR has consistently argued against the policy of prolonging the life of these aging components beyond the design life of 210,000 equivalent full-power hours, calling it a "gamble" that is not worth taking, given the enormous stakes in terms of public health an d safety as well as environmental sustainability. See www.ccnr.org/CCNR_CNSC_Pickering_2014.pdf .

In January 2013, just one month after shutting down the Gentilly-2 nuclear reactor rather than refurbishing it, the CEO of Hydro Quebec Thierry Vandal testified as follows before a Committee of the Quebec National Assembly:

> ...I can tell you that Hydro-Québec's management in no way would have considered to go beyond 210,000 hours even if it had been possible according to the design. I would no more operate Gentilly-2 beyond 210,000 hours than I would climb onto an airplane that does not have its permits and that does not meet the standards. So, it is out of question to put anyone, i.e. us, the workers, the public, and the company, in a situation of risk in the nuclear realm.

Here is the original French text:

> ...je peux vous dire que d'aucune façon est-ce que la gestion d'Hydro-Québec n'aurait envisagé d'aller au-delà de 210 000 heures même s'il avait été possible, le design. Je n'opérerais pas plus Gentilly-2 au-delà de 210 000 heures que je monterais dans un avion qui n'a pas ses permis puis qui ne respecte pas les normes. Alors, il ne serait pas question de placer qui que ce soit dans une situation, nous, travailleurs, la population, l'entreprise, la société, dans une situation de risque dans le domaine nucléaire.

It is refreshing, and rare, to see a utility executive putting the public interest, largely conceived, above narrowly construed industrial or corporate advantages to the utility.

The Reason for Nuclear Regulation

According to the law, the sole responsibilities of the Canadian Nuclear Safety Commission in its capacity as a regulator of nuclear energy, nuclear substances, prescribed equipment and prescribed information, are the following:

(1) preventing unreasonable risk to the health and safety of people and the environment;

(2) preventing unreasonable risk to national security;

(3) ensuring that Canada's international obligations in the nuclear field are met, and

(4) disseminating objective scientific information to the public about nuclear risks.

Every operating nuclear power reactor contains an enormous inventory of highly dangerous radioactive materials, most of them never found in nature before 1940. The unmitigated release of these materials into the environment by whatever means imaginable would be an unprecedented disaster in the Canadian context. On the domestic front, this unavoidable risk is one of the strongest reasons why we have a Canadian Nuclear Safety Commission – to ensure, insofar as it is possible, that such a possibility never actually materializes.

As stated by both the International Atomic Energy Agency (IAEA) and the International Commission for Radiological Protection (ICRP), any non-zero risk of preventable radiation exposure requires prior justification.

What is the nature of the risk? To quote from "A Race Against Time: Interim Report on Nuclear Power" published by the Ontario Royal Commission on Electric Power Planning:

> By definition, a major reactor accident would lead to the severe overheating, and subsequent melting, of the nuclear fuel, which would give rise to a substantial quantity of radioactive material escaping, after breaching several formidable barriers, into the environment....
>
> It is assumed that if a substantial quantity of radioactivity were to be released to the atmosphere, the radioactivity would collect in a "cloud" and would be carried down wind. *[continued next page]*

> The closer to the reactor building … the greater the probability of an individual's being exposed to intense radiation. At distances of two or three kilometres, depending on wind velocity, the cloud would begin to disperse (the dispersal zone could extend to distances of several hundred kilometres) and radioactive materials would be deposited on the ground….
>
> Apart from the direct radiation to which individuals might be exposed in consequence of the released radioactivity, there would also be a threat to the public in the immediate vicinity of the affected nuclear power station, from radioactively contaminated food and water.
>
> The emergency measures which would be necessary in the event of such a major accident would vary with the circumstances. Immediate evacuation of people living in the vicinity, and down wind, of the station might be necessary.
>
> Furthermore, the long term hazards, mentioned previously, would necessitate the isolation of contaminated food and water, and local decontamination procedures and the evacuation of people from heavily contaminated areas might be essential.
>
> When we talk about the safety of a nuclear reactor, we are referring essentially to how effectively the fantastic amount of radioactivity contained in the reactor core can be prevented from escaping into the ground and atmosphere in the event of major malfunctions.
>
> Clearly, if a major release of this accumulated radioactivity occurred, the consequences would be extremely serious….
>
> During normal operation, not only is a great deal of radioactivity created in the reactor core but also a great deal of thermal energy. If the shutdown system fails to operate in response to a fuel temperature rise, caused by a major rupture in the primary coolant circuit, a rapid escalation of heat and temperature would occur.
>
> The purpose of the ECCS [Emergency Core Cooling System] is to remove the heat from the core as rapidly as possible. If, however, both primary coolant and emergency coolant fail there would probably be partial or complete melting of the reactor core.
>
> An uncontained complete core meltdown would almost certainly give rise to a large release of radioactivity…. This would only occur, however, in the very unlikely event of the containment system – both reactor building and vacuum building – being breached…. *A Race Against Time, 1978, pp. 73-78*

While such a catastrophic nuclear accident is acknowledged to be extremely unlikely, it is by no means impossible. Scientifically speaking, probabilities have no predictive value. The probability of an event is related to the frequency of occurrence and has nothing to say about the time of occurrence. Besides, a seemingly low probability number may not be as comforting as many might think. As stated in the Royal Commission Report cited above:

> The Rasmussen Report *[The Reactor Safety Study, a fourteen-volume report published in 1974 by the U.S. Nuclear Regulatory Commission]* has concluded that the probability of an uncontained meltdown in a light water (U.S.) reactor is 1 in 20,000 per reactor per year (it has been suggested, moreover, that this figure could be out by a factor of "5 either way").
>
> Assuming, for the sake of argument, that within the next forty years Canada will have100 operating reactors, the probability of a core meltdown might be in the order of 1 in 40 years, if the most pessimistic estimate of probability is assumed.
>
> *A Race Against Time, pages 78-79*

Recommendation 1: The Canadian Coalition for Nuclear Responsibility regards it as an unreasonable risk to expose the City of Toronto and environs to the unnecessary threat of a devastating nuclear disaster, however low the estimated probability might be. CCNR therefore recommends that CNSC refuse to grant the licence requested by OPG to refurbish and continue to operate four nuclear power reactors (units 5 to 8) at the Pickering Nuclear Generating Station. This recommendation is reinforced by the following facts:

- **(1) There are more than 1.1 million people living within 20 kilometres of Pickering NGS, and twice as many living within a 30-kilometre radius;**
- **(2) Every homeowner's insurance policy has a "nuclear exclusion clause" which voids coverage in case of radioactive contamination from a nuclear accident'**
- **(3) As recently as 2016, OPG confirmed its earlier decision that refurbishment of the Pickering B reactors even at a lower cost ($10.7 billion) is unjustified**
- **(4) As of 2010, OPG was fully prepared to see its Pickering Nuclear Generating Station completely shut down by 2030;**
- **(5) The current estimated cost of refurbishing Pickering B is prohibitively expensive compared to other energy options, both nuclear and non-nuclear.**

The nuclear industry employs many safety systems in an effort to prevent major releases of radioactivity. However, accidents do happen and society has to be prepared to respond in an effective way to protect men, women and children from excessive radiation exposures, and to monitor food, water, soil and property for radioactive contamination. Financial compensation through the Nuclear Liability and Compensation Act may also be in order.

Preventing Ruptures of the Primary Heat Transport System

Based on analytic work done by Dr. Sunil Nijhawan, CCNR believes that the pressure relief valves in the primary heat transport system of most CANDU reactors, including the Pickering B reactors, are too small to do the job in the event of a serious reactor accident involving boiler dryout (e.g. through loss of feedwater). Any reactor accident that causes sustained overpressurization of the Heat Transport System will lead to a rupture if the plant is unable to relieve that pressure. Such a rupture may make it impossible to remove the radioactive decay heat rapidly enough to prevent severe core damage, resulting in massive radioactive releases into the reactor building – through the ruptured pipes – and eventually into the environment. This poses an unreasonable risk; but it can be fixed.

The existing valves in the Pickering B reactors are too small to do this job. They are unequal to the task of keeping the pressure in the heat transport system to no more than 10 to 20 percent above the design pressure, as stipulated by the American Society of Mechanical Engineers (ASME) recommendations. This is such a serious shortcoming that the Pickering B reactors should not be allowed to operate at all until the necessary large-sized pressure relief valves are installed. Yet there is no indication that such an operation is intended by OPG during the planned refurbishment.

To be specific, it is well known that immediately after shutdown, the radioactive "decay heat" generated by the irradiated nuclear fuel in the core of the reactor is about 7 percent of full power heat – that is, close to 7 percent of 1744 megawatts of heat from a Pickering B core operating at full tilt, which is 122 megawatts of heat. After two hours, the decay heat has dropped to about 20 megawatts of heat.

However, in the absence of other heat sinks (e.g. in the case of boiler dryout), the existing valves in the heat transport system of the Pickering B reactor – known as the "Degasser Condenser Relief Valves" (DCRVs) – are only able to remove about 4 megawatts of that 20 megawatts of heat that is being produced by radioactive decay, even two hours into the accident situation. Those valves are simply too small to do the job of removing all that surplus heat. This of course guarantees that there will be overheating of the fuel, overpressure of the heat transport system, and a pipe rupture somewhere.

So where will the system rupture? Overpressure rupture in the steam generators coupled with severe core damage would lead to uncontrolled atmospheric releases of radioactivity, for the feedwater that passes through those same steam generators will carry the fission products and other radionuclides from the damaged core to a place outside the containment boundary of the reactor, bypassing all normal filtering or retardation mechanisms.

Alternatively, if the ruptures were to take place in the core itself, due to bursting pressure tubes, the sudden pressurization of the calandria could have a devastating impact. The calandria vessel and the reactor vault are not designed to withstand such violent overpressurization.

Official Denial and Possible Cover-up

These concerns about pressure relief valves were raised by Dr. Gordon Edwards on behalf of CCNR at a CNSC hearing in Saint John, New Brunswick, on May 11, 2022. It was in the context of relicensing the Point Lepreau Nuclear Reactor Operating Licence. Both the licensee and senior CNSC staff stated definitively, on the record, that these valves (called degasser condenser valves) had been tested over and over again and that they met all ASME safety standards. They indicated that any uncertainty on this matter had been fully resolved years ago. Excerpts from the hearing transcript are found in Annex D.

However, one of the CNSC Commissioners, M. Bérubé, asked to speak to a CNSC subject matter specialist on the matter, Mr. Nitheanandan, who told a somewhat different story.

Mr. Nitheanandan said (in part) “There was a COG Report that we wrote, COG JP4534, in the year 2002 and according to that all these concerns were resolved. But lately, several papers were published in international journals and then we started to look at those claims. So we brought in about six of our experts and the author of the paper [Dr. Nijhawan]. We have completed approximately four different meetings over a period of about six to seven months and we are very close to writing the report. At the moment we don't see any safety concerns per se, but there [is] some credibility to some of the claims that [need] extensive review. We are at that stage and we are writing the report.”

At that point in the proceedings, Dr. Edwards asked the Commission members to make a point of reading the internal report referred to by Mr. Nitheanandan and deciding for themselves the truth of the matter. CCNR has subsequently obtained a copy of the CNSC Working Group’s Briefing Note on Degasser Condenser Relief Capacity for CANDU Reactors, dated June 17, 2022. It is attached to this submission as Annex C.

The final paragraph of the Working Group’s Briefing Note states, in part,

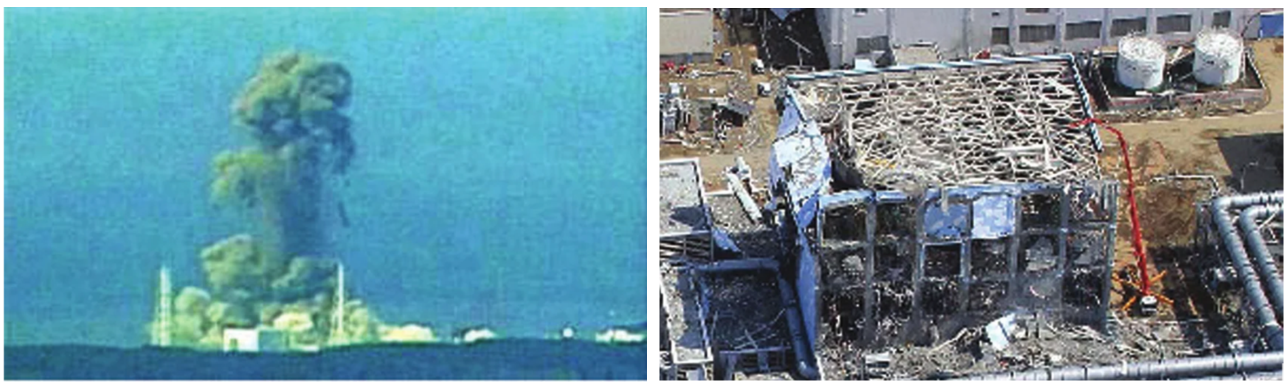
> After significant debate and analysis, the Working Group concludes that the relief capacity of the degasser condenser is inadequate for beyond design basis accidents such as a prolonged loss of all AC powers leading to a sustained loss of all heat sinks. It does not meet the current ASME and CSA's stated design requirements. A comparison with a similar Light Water Reactor system also points to the inadequacy for beyond design basis accidents....

Earlier in the same Briefing Note, the Working Group notes that old reactors (such as the Pickering B reactors) do not have to comply with modern standards: “the current design requirements are not required to be met for an existing old design. However, for a new design such as the ACR-700 design....a design change for the component ... is justifiable.” [ACR is an acronym for the “Advanced CANDU reactor”, which has never yet been built.]

Recommendation 2: If CNSC decides to approve the planned refurbishment of the Pickering B reactors, that it be made mandatory for safety-related items that do not meet current safety standards – such as the existing Degasser Condenser Relief Valves – be replaced with new items that are fully compliant with all modern safety requirements,

Preventing Deuterium Gas Explosions

Historically, major reactor accidents – especially those involving meltdowns or partial meltdowns – have been accompanied by one or more hydrogen gas explosions. During the partial meltdown of the NRX reactor at Chalk River in 1952 there was a series of such explosions. Explosions were also reported during the Three Mile Island accident in Harrisburg, Pennsylvania, in 1979. And television audiences were astounded to see the powerful explosions that took place during the Fukushima Daiichi triple meltdown in 2011.



A photo from video footage taken March 14, 2011, shows a hydrogen explosion at the Fukushima Daiichi nuclear power complex at the moment of the event and afterward. (Reuters/NTV photo via Reuters TV.)

In a very hot environment, at least several hundreds of degrees C, steam reacts with metal to produce metallic oxide and release hydrogen gas. In the Canadian context, the steam is "heavy" steam made from "heavy" water, and the hydrogen gas that is released is "deuterium" – that is "heavy" hydrogen (twice as heavy as normal hydrogen). All hydrogen is explosive, however, and the fundamental problem is the same – preventing explosions.

The steam-metal reaction is exothermic (heat-producing) and it accelerates as the temperature climbs. Hydrogen (or deuterium) is a very combustible gas, and when it mixes with air it quickly becomes explosive. Explosive mixtures range from 4% to 75% hydrogen gas (or deuterium gas) by volume in air. The most violent mixture occurs near the 30 percent mark. And it takes very little energy to detonate it.

One of the worst design basis nuclear accidents postulates a LOCA (loss of coolant accident) – possibly due to a pipe break – coupled with a loss of emergency core coolant (LOECC) – possibly caused by a failure to inject emergency coolant into the fuel channels. In such an event the nuclear fission process is immediately halted. There are two independent fast shutdown systems, each of which can terminate the chain reaction in less than two seconds. That prevents a runaway nuclear accident where the core would self-destruct. Stopping the nuclear chain reaction immediately is essential.

Nevertheless, the intense radioactivity in the fuel itself cannot be shut off. It continues to generate a great deal of heat called "radioactive decay heat". Unless this decay heat can be removed as fast as it is being produced, the temperature will climb and the zirconium metal cladding of the fuel will react with steam to produce more and more deuterium gas. Although there is not sufficient heavy water to cool the fuel, pockets of heavy water trapped in the primary heat transport system quickly turn to steam and feed the steam-metal reaction. That produces more and more deuterium gas and adds even more heat, accelerating the heating of the fuel which approaches ever closer to the melting point.

Preventing a massive deuterium gas explosion is evidently very important as it can do great damage to the containment system, creating pathways by which radioactive gases, vapours, aerosols and particulates can escape into the outside environment. Accordingly, nuclear reactor designers install "hydrogen recombiners" at many locations inside the reactor building. These recombiners turn hydrogen/deuterium gas back into water vapour by combining the hydrogen /deuterium molecules with oxygen molecules to make H_2O or D_2O molecules – ordinary water, or heavy water. Deuterium moves a little more sluggishly than ordinary hydrogen, making the danger greater because the recombination can be slowed down or delayed in certain cases.

If the amount of deuterium gas is seriously underestimated, or if the recombiners are poorly positioned, there may be a buildup of deuterium gas inside the reactor building that can become an explosive mixture. A powerful gas explosion can breach the containment and provide a pathway for the escape of radioactive materials into the outside environment.

Various published papers have pointed out that deuterium gas generation in a CANDU reactor may be far greater than anticipated due to the fact that (1) CANDU reactors contain far more zirconium metal than most other reactors, and (2) each fuel channel is attached to a steel “feeder pipe” which also generates deuterium gas when exposed to very hot heavy-water steam.

There is no indication in the current Pickering licence application that OPG intends to greatly increase the number of hydrogen recombiners, or to test their performance with deuterium gas rather than ordinary hydrogen gas, or to optimize their placement so as to prevent dire off-site consequences in the event of a severe nuclear accident.

It should be noted that almost all engineering analyses of the hydrogen gas problem do not pay enough attention to the quite different characteristics of deuterium (“heavy hydrogen”) as distinct from those of normal hydrogen “light hydrogen”, technically called “protium”). See Annex A on this point.

Recommendation 3: If CNSC licences OPG to refurbish units 5 to 8, a thorough analysis be done of the potential for hydrogen generation under the most adverse accident circumstances, and the number and positioning of recombiners be planned so as to minimize the likelihood of a massively destructive hydrogen gas explosion. In carrying out such analyses, CCNR urges that special attention be given to the behavioural characteristics of deuterium as opposed to protium.

One final word: A violent hydrogen or deuterium gas explosion can seriously impair the containment of a nuclear reactor thereby establishing a direct and unimpeded pathway to the outside environment. Unlike American-style light water reactors that have a very strong metal containment structure, the Canadian CANDU containment is very leaky and very weak (unable to withstand high pressure).

The Importance of the Radioactive "Source Term"

Simply put, the "source term" is the amount of radioactive material that is released to the environment in the event of a nuclear accident. Proper emergency planning requires having accurate information about the source term in order to understand the hazards and the precautions that must be taken. To be prepared, you need to know the facts.

How much toxic material are we dealing with, and how bad is it? Is evacuation required? Do first responders need to wear protective equipment? How does one treat radioactively contaminated objects or persons? What kind of radiation detectors are needed and how are they best used? What do we do about contaminated food and water? How large an area will be affected, and for how long?

It is very difficult to answer these questions satisfactorily. A severe nuclear accident is an event where a nuclear reactor suffers extensive core damage and partial or complete melting of its radioactive fuel, releasing many different varieties of radioactive materials into the environment as gases, vapours, aerosols and particulates. They all behave differently.

To quote from the 1978 Royal Commission Report "A Race Against Time" once again:

> All operating nuclear reactors accumulate in their cores ... a large quantity of radioactive material. For the most part this is made up of fission products, many of which are short lived and usually very radioactive, and the actinides (e.g. plutonium-239) which are very long lived and highly toxic substances.
>
> The major health and environmental threat would be due to the escape of the fission products to the atmosphere. The most important of these are caesium, ruthenium, tellurium and the fission gases, iodine, krypton and xenon....
>
> It is generally agreed that the greatest threat to health in the event of a major reactor accident is the considerable quantity of the radio-isotope iodine-131 ... which would be released to the atmosphere.
>
> It is well known that iodine-131, after ingestion or inhalation, concentrates in the thyroid gland and may cause, after a latency period, thyroid cancer. The threat to children in such circumstances is particularly serious because the iodine-131 could be ingested in the form of contaminated milk....

> Because of its half-life of about 8 days, iodine-131 remains highly radioactive for a few weeks. Subsequently, the major contributor to the radiation field is caesium-134 with a half-life of two years. The radioactivity arising from this isotope would persist for many years.
>
> Apart from the direct radiation to which individuals might be exposed in consequence of the released radioactivity, there would also be a threat to the public in the immediate vicinity of the affected nuclear power station, from radioactively contaminated food and water.
>
> *A Race Against Time, pp.93-94*

These radioactive materials all have difference pathways through the environment and through the human body. One of them, called cesium-137, is often used as a kind of standard indicator of how much radioactivity has been released. Once one knows how much cesium-137 is given off, planners can estimate how much of all the other varieties of radioactive materials have also been released, using proportionality as a guide.

Cesium-137 has a relatively long half-life of 30 years, so it will be around in the environment for several centuries. It is easy to measure with radiation monitoring equipment because it gives off a distinctive and very powerful beam of gamma radiation.

Cesium-137 is also biologically important. It is stored in the soft tissues when inhaled or ingested, sometimes making meat from animals unfit for human consumption. For twenty years after the Chernobyl meltdown of 1986, sheep farmers in Wales and Northern England had to have their meat regularly tested for radioactive cesium contamination – the toxic fallout from a nuclear accident over 2000 kilometres away. To this day, hunters killing wild boars in Germany, Belgium, and the Czech Republic are warned not to eat the meat because of high radioactive cesium contamination levels from the Chernobyl disaster.

Every CANDU fuel bundle contains all of these radioactive materials in similar predictable proportions. During a severe nuclear accident, the contents of damaged fuel bundles are released into the core of the reactor, then into the containment building, and finally into the outside environment. How many fuel bundles are damaged, and how badly? How effective is the containment building in preventing these radioactive materials to escape?

The Politics of Denial

Nuclear proponents are often unwilling to talk about such severe accidents. Many of them seem to believe that their nuclear reactors are well built, well maintained, and well protected, and are therefore “safe enough”. No one need worry about nuclear disasters.

After the 1979 partial meltdown at the Three Mile Island nuclear reactor in Harrisburg, Pennsylvania, President Jimmy Carter ordered a special Presidential Commission of Inquiry into the root causes of the accident. Here is an excerpt from that report:

> Wherever we looked, we found problems with the human beings who operate the plant, with the management that runs the key organization, and with the agency that is charged with assuring the safety of nuclear power plants. In the testimony we received, one word occurred over and over again. That word is "mindset."
>
> After many years of operation of nuclear power plants, with no evidence that any member of the general public has been hurt, the belief that nuclear power plants are sufficiently safe grew into a conviction....
>
> The Commission is convinced that this attitude must be changed to one that says nuclear power is by its very nature potentially dangerous, and, therefore, one must continually question whether the safeguards already in place are sufficient to prevent major accidents. A comprehensive system is required in which equipment and human beings are treated with equal importance.

Here in Canada, attached to this report as Annex E is a fascinating example of how well-educated, highly motivated professionals in the nuclear field can simply refuse to answer the most straightforward questions related to severe nuclear accidents. It seems as if they simply cannot bear to regard these questions as meaningful, because the premises are so much at odds with their beliefs about the near-absolute safety of nuclear power.

CNSC President Dr. Michael Binder asked his staff at a December 2012 public hearing to explain how radioactive materials can possibly escape from a CANDU nuclear reactor and end up in the outside environment. Disregard the probabilities, he said, just tell me how it can happen. The ensuing hodge-podge of evasions and mumbo-jumbo responses would be hilarious if it were not so serious in indicating the widespread presence of a very pervasive mindset of denial – disturbing in people who are charged with protecting the public against the very risks that they seem unwilling to admit are genuine. The exchanges reported in Annex E are taken directly and verbatim from the hearing transcript of Dec 3-6, 2013.

Because Dr. Binder could not obtain a meaningful answer to his question, he ordered the staff to prepare a report for public consumption on the consequences of a severe nuclear accident in a CANDU reactor.

CNSC Study on Severe Nuclear Accidents

Acknowledging that an accident can happen at any time at any of Ontario's nuclear plants, including those at Pickering, the CNSC published a report in 2015 entitled "Study of Consequences of a Hypothetical Severe Nuclear Accident and Effectiveness of Mitigation Measures". The report postulates a severe nuclear accident at the Darlington site giving off a radioactive release into the environment that would include 100 terabecquerels of cesium-137.

This postulated release of radioactive cesium-137 is 800 times less than the amount of cesium-137 actually released from Chernobyl. It represents the cesium-137 content of only 9 CANDU fuel bundles (out of a total of 4560 bundles in each Pickering B reactor). As such, it is a very low estimate of how much cesium-137 could be released. Is this very low number just another example of denial? Refusing to be blunt about how bad it could be?

Below is a graph from a scientific research paper authored by four staff members from Canadian Nuclear Laboratories (CNL) and one from the Canadian Nuclear Safety Commission (CNSC). The graph indicates a radioactive release into the environment following a postulated CANDU nuclear accident – an unmitigated total station blackout.

The cesium-137 component (indicated in bright green) in this postulated release is at least 80 times greater than that proposed in the 2015 CNSC report. On the vertical scale, 10^{16} becquerels is equal to 10,000 terabecquerels; that is 100 times greater than the 100 terabecquerels indicated in the CNSC report cited above. Why the discrepancy?

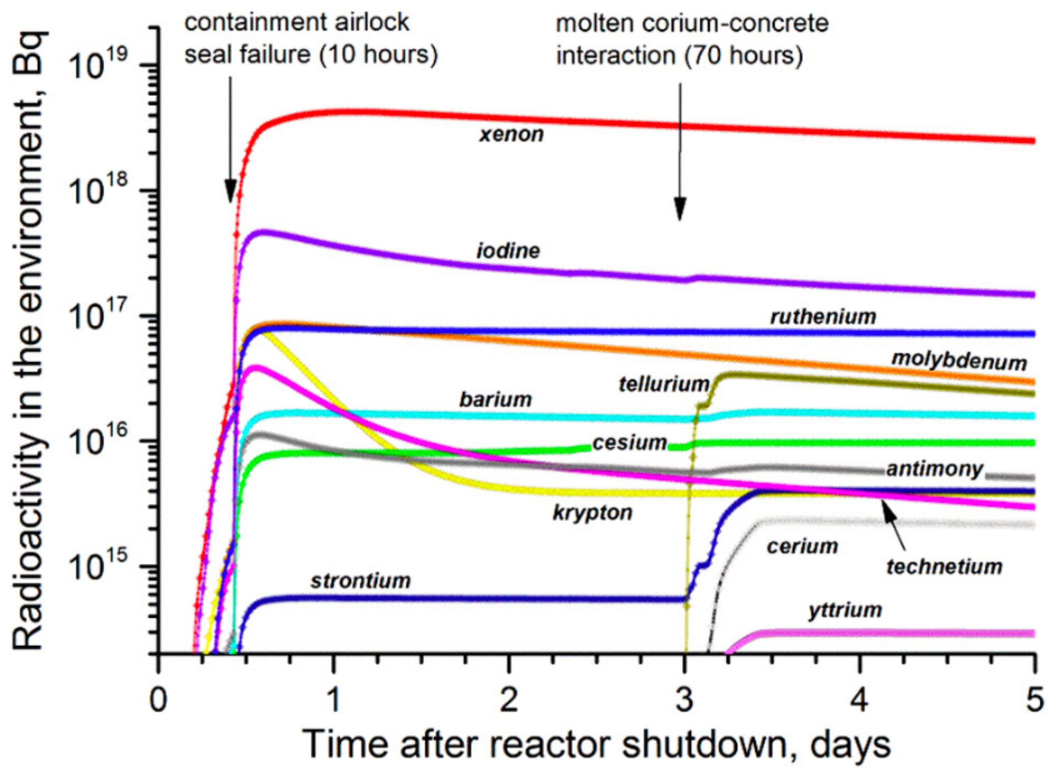


Fig. 4. Radioactivity in the environment from different elements, for an unmitigated SBO severe accident.

From “Severe accident consequence mitigation by filtered containment venting at Canadian nuclear power plants”, Annals of Nuclear Energy, August 2016 [SBO = Station Blackout – complete loss of power]

We at CCNR believe the CNSC postulated release of 100 terabecquerels of cesium-137 is unrealistically low. While there may be a severe accident scenario in which the release of cesium-137 could be this low, CNSC provides no such scenario and we know of no such example. In fact, the 2015 CNSC report makes it quite clear that this particular amount (100 terabecquerels) is the smallest possible release of cesium-137 that can still be categorized as “a large release”. Conservatism dictates that, in the interest of public safety, one should heighten rather than downplay the hazard to ensure adequate response.

The CNSC report asserts that no large release of cesium-137 can occur without “severe core damage”, described in CANDU literature as a “collapse of the core” or a “core disassembly” at temperatures near 1300^{O}C or more. Since radioactivity cannot be shut off, heat continues to be generated even after fission is arrested. Overheating derives from a failure to remove radioactive decay heat fast enough. In the CANDU literature, this is thought to cause progressive collapse of many fuel channels inside the core, leaving damaged fuel bundles in a heap near the bottom of the calandria.

At these elevated temperatures the “rupture discs” at the top of the calandria will burst to relieve the steam pressure, creating a hole for other gases and vapours to exit the core, and expelling moderator water as well. A large fraction (perhaps 20 to 40 percent) of the 50,000 terabecquerels of cesium-137 stored in the irradiated fuel bundles in the core will vaporize and escape out through the top of the reactor vessel (through the hole created by the destruction of the rupture disc).

Apparently, judging from the CNSC report, both CNSC and OPG believe that engineered systems can prevent all but 100 terabecquerels of that 10 to 20 thousand terabecquerels of air-borne cesium-137 from reaching the environment. We are asked to believe that tens of thousands of terabecquerels of cesium-137 will escape from the core of the reactor but never see the light of day. The detailed calculations required to justify this controversial belief , if they exist, have so far not been made public by CNSC – nor by OPG.

Recommendation 4: For the benefit of emergency planners, that CNSC provide a series of possible source terms for cesium-137 releases, following a severe nuclear accident, with 100 terabecquerels being the smallest entry. This will encourage first responders to envision and plan for a spectrum of different responses depending upon the severity of the release. Moreover, that CNSC staff publish proportionality documents indicating in each case the anticipated magnitudes of release of other radionuclides of public health or environmental importance.

Annex A

Sunil Nijhawan
August 6 2026

Information on differences between Deuterium and Hydrogen recombination rates with hydrogen recombiners

Deuterium and protium cannot simply be assumed to have identical performance in a hydrogen recombiner. The chemical pathway is the same, but both reaction kinetics and mass transport differ.

The most important finding is that I could not locate a publicly available, full-scale passive autocatalytic recombiner test that directly compares otherwise identical H_2–air and D_2–air experiments. That is precisely the comparison needed for a CANDU qualification.

Expected differences

Property	H_2	D_2	Likely effect on a recombiner
Molecular mass	2.016	4.028	D_2 moves and diffuses more slowly
Approximate relative molecular velocity	1.00	$1/\sqrt{2} = 0.707$	Fewer D_2–surface encounters per unit time
Gas diffusivity in air	Higher	Roughly 25–30% lower	Lower transport rate to catalyst
Pt/Pd surface reaction	Faster in many regimes	Normally subject to a kinetic isotope effect	Potentially slower cold start and surface reaction
Heat released per mole	Slightly lower	D_2O formation is slightly more exothermic	Some compensation through catalyst heating and buoyancy
Product	H_2O	D_2O	D_2O adsorbs, condenses and desorbs differently

The overall PAR rate can be represented schematically as:

$$\frac{1}{k_{\text{overall}}} = \frac{1}{k_{\text{mass transfer}}} + \frac{1}{k_{\text{surface}}}$$

Deuterium can affect both terms:

1. $k_{\text{mass transfer}}$ is lower because D_2 diffusion is slower.
2. k_{surface} may be lower because breaking D–D bonds and transferring D atoms on platinum or palladium exhibits a primary kinetic isotope effect.

Experiments on platinum confirm that H_2 and D_2 oxidation are distinguishable surface-chemical processes—not universally identical ones. The measured isotope effect depends strongly on catalyst temperature, surface coverage, oxygen concentration and which elementary step controls the reaction. A classic platinum study explicitly investigated both H_2 and D_2 oxidation and found isotope-dependent surface chemistry, although it was conducted at low pressure rather than in a PAR geometry.

What this could mean quantitatively

Two limiting cases are useful:

- If the catalyst is surface-kinetics-controlled, the D_2 recombination rate might be appreciably below the H_2 rate; a factor such as 0.5–0.8 would not be surprising, but it must be measured for the actual catalyst.
- If the hot recombiner is predominantly mass-transfer-controlled, a first estimate is:

$$\frac{\dot{n}_{D_2}}{\dot{n}_{H_2}} \approx \frac{D_{D_2-air}}{D_{H_2-air}} \sim 0.7 - 0.8$$

This is not a qualified design factor. It merely shows that an assumption of unity has no obvious physical basis.

There is an important units trap. AECL characterizes capacity in "kilograms hydrogen per hour." If the same molar quantity of D_2 is removed, its reported mass is approximately twice that of H_2. A recombiner could therefore appear to have a higher **kg/h** D_2 capacity while actually removing fewer gas molecules per hour. Containment safety calculations should compare:

- mol/s or kmol/s;
- removal of combustible-gas volume fraction;
- oxygen consumption; and
- time to remain below the applicable flammability or detonation criterion.

What the publicly available AECL qualification shows—and does not show

The published AECL qualification paper says that:

- the test facility provided controlled additions of **hydrogen**, steam and inert gases;
- capacity was reported in kilograms of hydrogen per hour;
- self-start was demonstrated at approximately 1% hydrogen;
- performance tests covered humidity, temperature, hydrogen concentration and ageing.

However, the paper does not identify a paired D_2 qualification series or provide an H_2-versus-D_2 correction factor. AECL/CNS paper, "LOCA Hydrogen Mitigation in CANDU Stations"

CNL separately confirms that related catalyst formulations are used to recombine deuterium and tritium in heavy-water upgrading and isotope-processing applications. That demonstrates that the catalyst can recombine D_2; it does **not** demonstrate identical full-device removal rates, self-start thresholds or thermal behaviour in H_2 and D_2. CNL recombiner-catalyst paper

My technical conclusion

The statement that "research has shown no difference between H_2 and D_2 behaviour" is too broad and, as a scientific proposition, incorrect. A more defensible statement would be:

"Platinum- or palladium-based recombiners can oxidize both H_2 and D_2, but full-device performance may differ because of kinetic isotope effects, diffusion, adsorption and thermal–hydraulic feedback. Applicability of H_2 qualification data to D_2 service requires experimental demonstration or a justified conservative correction."

For CANDU severe-accident credit, I would ask the utility or regulator for:

1. full-scale paired H_2/D_2 capacity curves versus mole fraction;
2. cold-start and self-extinction concentrations for D_2;
3. results under steam/D_2O vapour and condensing conditions;
4. catalyst-temperature comparisons;
5. the D_2 correction used in MAAP-CANDU or containment analysis;
6. detector response and calibration for D_2 rather than H_2; and
7. whether stated recombiner capacity is compared on a mass, molar or combustible-volume basis.

Without those data, assigning the H_2 performance correlation directly to D_2 with a factor of 1.0 is an unsupported assumption—not an experimentally established equivalence.

Annex B

Ontario Power Generation

PICKERING CONTINUED OPERATIONS PROJECT CLOSURE REPORT
July 2016

Pages 1 to 5 only (of 37 pages)

Pickering Continued Operations Project Closure Report

JULY 2016

2016-07-08

Order Number: N/A
Other Reference Number:

Internal Use Only
General Business

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1.0 EXECUTIVE SUMMARY

This report documents the execution and completion of a project over the period 2011 to 2014 to continue operations of the Pickering B reactor units for a period of 5 years beyond their nominal design life. The project was undertaken by Ontario Power Generation and was completed at a cost of $223M.

During 2006 to 2009, Ontario Power Generation (OPG) completed an assessment of the scope, cost and duration of a project to refurbish the Pickering B units in order to extend these units life by 30 years. The accompanying economic assessment was unfavourable and as a result, early in 2010, the OPG Board announced their decision to not proceed with the Pickering B Refurbishment choosing instead to operate the Pickering reactors until the end of 2020. Subsequent to this decision, a plan to continue operations of the Pickering B Units for a period of 5 years was developed and this plan was used to produce the Continued Operations Business Case Summary (BCS) **[R2]**. The BCS showed that there was substantial value to the Ontario electricity system in continuing to operate the Pickering B units beyond their nominal design life and up to the point in time where continuing further would require replacement of the reactor's major life limiting component – the fuel channels. In May of 2010, OPG submitted the first Continued Operations Program amounts to the Ontario Energy Board (OEB) as part of the rate submission for the 2011 and 2012 rate years, indicating its intention to complete the program during the 2011-2014 timeframe.

The departmental Project Management Model was used to leverage the normal functions in Engineering, Work Management, and Maintenance to get the CO Project work done. Towards the end of 2011 it became clear that this approach was not yielding the required rate of work completion with the result that a dedicated Project Management team was formed in November 2011. Subsequently, OPG was able to complete the inspection, maintenance and regulatory work necessary to support the decision to operate to the end of 2020.

The CO project provided the resources to develop and execute the Pickering 5-8 Continued Operations Plan (COP) which provided the framework to ensure the regulatory life extension actions were implemented to support the technical basis for continued operation of Units 5 to 8 beyond 210k Effective Full Power hours (EFPH).

The success of this approach was evident in the issuance by the CNSC of a 5 year operating licence to the station in September of 2013. Subsequently, early in 2015, sufficient evidence was provided to the CNSC as a result of the life extension project work, to extend the target service life of the Pickering 5-8 pressure tubes to at least 247k EFPH.

2.0 PROJECT OBJECTIVE & PURPOSE

The project objectives were twofold:

- To provide the resources to allow development and execution of the Pickering 5-8 Continued Operations Plan (COP). The COP provided the framework to ensure the necessary regulatory life extension actions were implemented to support continued operation of Units 5 to 8 beyond 210,000 Effective Full Power hours (EFPH) to 247,000 EFPH, in the absence of a refurbishment program.
- To execute a "value added" inspection and maintenance program to ensure that the Pickering B units operate reliably during the life extension period.

3.0 PROJECT OVERVIEW

3.1 Development

During 2006 to 2009, Ontario Power Generation (OPG) completed an assessment of the scope, cost and duration of a project to refurbish the Pickering B units in order to extend these units life by 30 years. A significant output from this assessment was the Pickering B Integrated Safety Report (ISR) NK30-REP-03680-0400585 R003 **[R1]** which identified the requirements for a 30 year life extension to the Pickering B Units. An integral component of the ISR was the supporting Component Condition Assessments (CCAs). These CCAs established the baseline condition of systems, structures and components in order to help assess the costs of a refurbishment for 30 additional years of plant life. The accompanying economic assessment developed for the life extension was unfavourable and as a result, early in 2010, the OPG Board announced their decision to not proceed with the Pickering B Refurbishment choosing instead to operate the Pickering reactors until the end of 2020.

As the 30 year economic assessment was winding down, development work on a plan to continue operations of the Pickering B Units for a period of 5 years was underway. This plan and analysis work was used to produce the Continued Operations Business Case Summary (BCS) NK30-BCS-00000-00001 **[R2]**. This BCS was approved on February 17, 2010 describing a Continued Operations program to support operation of PNGS-B units to 2020 with a preliminary cost estimate of $191M. The BCS showed that there was substantial value to the Ontario electricity system in continuing to operate the Pickering B units beyond their nominal design life and up to the point in time where continuing further would require replacement of the reactor's major life limiting component – the fuel channels. This assessment included the incremental work and the associated additional costs required to execute selected and well defined additional maintenance to improve the material condition of the plant, coupled with in-service inspections and analysis necessary to ensure fitness for service (FFS) of the plant's major components.

Several program elements were defined:

(a) Life Cycle Management (LCM) requirements for major components which identifies the additional inspection and maintenance activities to confirm component FFS in the life extension period. These activities include (but are not limited to):

- Increased fuel channel Spacer Location and Relocation (SLAR) maintenance,
- Increased pressure tube and gap inspections,
- Feeder inspections/replacements,
- Boiler tube inspections and enhanced water lancing.

(b) The incremental reliability work identified in the 2010 CCAs to achieve the target generation reliability performance for the period 2015 to 2020.

(c) Completion of assessments, analysis and commitments to support the nuclear regulatory basis for incremental operation of Pickering B units from 210k EFPH to the new target value of 247k EFPH (approximately five additional calendar years).

Accordingly, OPG decided to further develop the CO work program and allocated funding for this program for the 2011-2014 timeframe. In May of 2010, OPG submitted the first Continued Operations Program amounts to the Ontario Energy Board (OEB) as part of the rate submission for the 2011 and 2012 rate years.

3.2 Scoping Phase

The initial development of the CO Project Scope occurred during 2010 using a scoping process as described in the Pickering B CO Scoping Guide (Feb 2010) NK30-GUID-01S12-00001**[R4]** and the CO Scope Challenge Guide (Dec 2010) NK30-GUID-01512-00002 **[R3]**

The Continued Operations Scope Report NK30-REP-01512-00001 **[R5]** was issued in April 2011 to document the selection of scope. In summary, 991 recommendations received scope challenge preliminary approval. These recommendations totalled to a value of:

Table 1 CO Project Scope Estimate - April 2011 ($M)

Scope Element	Base Funded	Incremental CO Funded	Total
Non–major components	131	46	177
Major components LCMP	113	120	233
Fuel Channel Life Mgt Project	-	13	13
TOTAL ($M)	**244**	**179**	**423**

Annex C

Briefing Note to Mike Rinker

on Degasser Condenser Relief

Capacity for CANDU Reactors

June 17, 2022

BRIEFING NOTE TO Mike Rinker DAA DG

Briefing Note on Degasser Condenser Relief Capacity for CANDU Reactors

June 17, 2022

ISSUE OR OBJECTIVE

Do the overpressure protection devices installed on the degasser condensers of operating CANDU reactors have adequate discharge capacity, location, size, and number? Sufficient discharge capacity is required to limit overpressure when a mismatch between effective thermal load into the primary fluid power and heat removal capacity occurs during postulated accidents. This briefing note aims at providing a technical basis to answer the above question.

BACKGROUND

This question is not new, as at least twice in the past, intervenors had raised the issue, and the matter was closed. The industry and the CNSC conducted two previous reviews of the relief valve capacity [1, 2, 3]. Several papers have continued to raise the issue in conference proceedings. These papers disclosed new evidence overlooked in earlier investigations. One example is using a CFD analysis to justify air tests to obtain a steam discharge. The study performed no mesh density independence analysis [3]. The investigation provided neither a rigorous validation of the model as per the requirement [4] nor an uncertainty analysis using CFD best practice guidelines [5]. An independent review of the disposition of the intervenor's issues was conducted [6]. This study referred to another study conducted by an independent ASME Code specialists who concluded that the licensees are not required to conform to the overpressure relief requirements of ASME Boiler and Pressure Vessel Code Section III during a BDBA [7]. In that report, the ASME specialist stated, "This evaluation did not consider the points made in the documents with reference to the options used in the design of the overpressure protection system, the accuracy and completeness of the calculations and tests and whether or not the existing systems are adequate." A working group within the Directorate of Assessment and Analysis started to review the latest evidence as part of an email exchange between the President and the author of the conference papers. The author of the conference papers, who was hired as a consultant, provided additional analysis in Appendix A.

<u>What accidents are likely to cause overpressure?</u>

The overpressure of the Primary Heat Transport System may occur due to a class of accidents linked to a loss of boilers as heat sinks. Examples leading to loss of heat sink are:

- Loss of inlet feedwater flow to one or more steam generators
- Pipe break in a steam line
- Pump failure and pump seizure
- Loss of Class IV power
- Loss of Class IV power resulting in a partial drain of primary inventory into the below header pressurizers at Darlington and Bruce
- Primary system LOCA with and without reactor trip

AECL believes that sustained loss of heat sink belongs to Beyond Design Basis Accidents [2], and therefore the current design requirements are not required to be met for an existing old design. However, for a new design such as the ACR-700 design, the relief valves open when the pressure in the bleed condenser reaches the value of 12.27 MPa(g), with a relieving capacity on each valve bounded by the sustain loss of all heat sinks event, to give a relief capacity of 21.3 kg/s for steam [8]. Given the importance of a robust system to public safety, a design change for the component as in ACR-700 is justifiable.

Systems available to limit overpressure

There are two systems available to limit the Primary Heat Transport System (PHTS) pressure. The first system is the two shutdown systems acting alone can terminate neutron chain reaction and prevent failure of the PHTS due to overpressure, provided the heat sink capabilities are fully poised and available to remove the decay heat. The second system consists of a pressurizer, a train of valves such as the liquid relief valves and the safety relief valves on the degasser condenser. The latter system is the only pressure relief system when the heat sink function of the reactor is lost.

Design Requirements for pressure relief

There are three design requirements for pressure relief.

1. The design requirement of the pressure relief valves is to protect the PHTS by relieving the coolant volume swells during various disturbances in power to arrest pressure escalation in the PHTS. The nuclear class in CSA Standard CAN3-N285.0-95, Clause 7.4.2 and Section III of the ASME Boiler and Pressure Vessel Code BPVC III 1 NB-2021 provide the overpressure protection design requirements. From the Qinshan Overpower Protection Report [9] the requirements are that under all operating conditions, the relief valves must open when the pressure in the degasser condenser reaches the value of 10.16 MPa(a), irrespective of the pressure signal to the actuator. The relieving capacity of each valve is 26.66 kg/s at 268°C of liquid. The sizing of the degasser condenser relief valves is based on the premise that the PHT System is experiencing a dual accident: loss of inventory control and loss of heat sink. The reactor outlet header is at 110% of the PHT relief valve set pressure, i.e., at 11.37 MPa(a), and the degasser condenser is at 110% of the relief valve set pressure, i.e., at 11.17 MPa(a).
2. The pressure relief valves shall be close to the significant source of overpressure anticipated to arise within the system under the conditions summarized in the Overpressure Protection Report (NB-7200).
3. The back pressure that may exist or develop shall not reduce the relieving capacity of the relieving device(s) below the level required to protect the system.

How have other jurisdictions designed pressure relief?

The Surry Nuclear Power Station located in southeastern Virginia is a 2,587 MW_{th} PWR, an equivalent thermal power to a typical CANDU at Bruce or Darlington, has three safety relief valves and 2 Pilot Operated Relief Valves with a steam relief capacity of 112 kg/s and 53 kg/s, respectively. At Georgia Power's Vogtle (3,625.6 MW_{th}), the three safety relief valves can discharge at 17 MPa of 159 kg/s with an additional 53 kg/s from the 2 PORVs. The single valve liquid relief capacity satisfies pressurization challenges due to liquid swell at full power. These plants have redundancy and a testing program to assure functionality. The Design Requirements for PWR reactors state that they should have sufficient capacity to preclude actuation of safety valves during normal operational transients, when the reactor is operating at the licensed core thermal power level [10]. They also require that the valves should be tested and inspected.

<u>How well does CANDU pressure relief system meet the requirement?</u>

1. The safety relief valves downstream of the Degasser Condenser have a specified steam relief capacity of 4 kg/s. When tested, the actual relief capacity was lower than the specified relief capacity. For example, when tested in 2001 by OPG at Wylie Laboratories under the manufacturer Bopp & Reuther's supervision, the Pickering relief valve's combined steam relief capacity was 120 g/s for its two relief valves [11, 12]. The power of a single Darlington unit is 2776 MW_{th}, and the Bruce unit is 2832 MW_{th}, which is not too far from the Surry Nuclear Power plant. For Qinshan, the specified relieving capacity is of each valve is 26.66 kg/s at 268°C and 10.16 MPa(a).
2. The pressure relief valves in the operating stations in Canada are not close to the source of overpressure expected. As Figure 1 shows, they are several meters away from the headers.

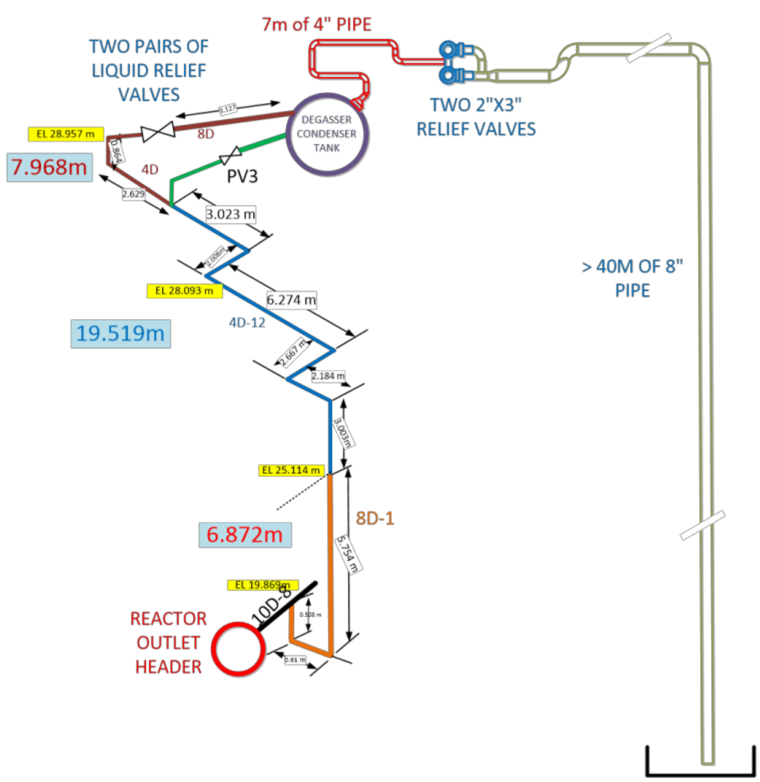

Figure 1: Distances Between Header, the Liquid Relief Valves, the Degasser Condenser Safety Relief Valves and the Downstream Piping to Collection Tanks

3. The potential backpressure from the >40 m long piping downstream of the relief valves will significantly increase pressure. The pressure required to open the valves will exceed the pressure needed for the design.

CONCLUSION

After significant debate and analysis, the Working Group concludes that the relief capacity of the degasser condenser is inadequate for beyond design basis accidents such as a prolonged loss of all ac powers leading to a sustained loss of all heat sinks. It does not meet the current ASME and CSA's stated design requirements. A comparison with a similar Light Water Reactor system also points to the inadequacy for beyond design basis accidents. Post-Fukushima enhancements, including Emergency Mitigating Equipment and Containment Filtered Venting System, which have been added in Canadian nuclear power plants, provide countermeasures to stop the progression of severe accidents and minimize their radiological consequences to the public.

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Approved by: name and title, no signature

Date: 2022-06-17

Annex D

CNSC Denial of the insufficiency of the pressure relief valves (degasser) in the CANDU heat transport system

From the May 10 2022 CNSC Transcript - starting on page 104

MEMBER BERUBE: Well, I would like to thank the Passamaquoddy for that heartfelt presentation, a very deep dive into the meaning of what the land means to you and what you need to do to preserve it….

I am looking at the written submission, and that is document H2.244A, and there is an assertion here with regard to safety and the assertion here is that:

"It has recently come to our attention that there are detrimentally inadequate pressure relief valves installed in the primary heat transport system --valves that are NOT in accordance with ASME recommendations."

I am going to give at this point NB Power the opportunity to speak to that and then I want to hear from CNSC as to their perspective on this assertion.

MR. NOUWENS: Thank you for the question. Jason Nouwens, for the record.

I will turn this over to Jennifer Lennox, who is our Director of Engineering and also the first female Chief Nuclear Engineer in Canada.

Jennifer, if you could provide some high-level comments and turn it over to Derek Mullin if you want for more details. Jennifer, over to you, please.

MS. LENNOX: Jennifer Lennox, for the record. I am the Engineering Director and the Chief Nuclear Engineer at Point Lepreau.

So the valves you are referring to are degasser condenser relief valves in question. I just want to, before I turn it over to Derek Mullin, speak further on this. I just wanted to comment that these have been tested and do meet all safety requirements as per our standards.So I will turn it over to Derek.

MR. MULLIN: Thank you. Derek Mullin, for the record. This issue regarding degasser condenser relief valves was one of 18 issues that were raised in the past. All of these issues were addressed under a CANDU Owners Group joint project and was presented to the Commission during a March 8th meeting in 2017 and again at 2017 licence renewal. Those issues and the technical issues surrounding these were all reviewed by an expert, an independent expert who confirmed the findings of industry on this.

There are no safety issues with these degasser condenser relief valves. They were appropriately sized. They underwent testing at Wyle Labs as a part of a demonstration that **they meet ASME testing requirements**. That was all part of the design. There is no issue with these. Thank you.

MEMBER BERUBE: CNSC, please comment.

DR. VIKTOROV: Alex Viktorov, for the record.

Indeed, the issue of pressure relief valves is a longstanding question we received from one of the intervenors. It has been examined in great depth, both by staff, industry and independent experts. **It has been confirmed again and again that the current valves are adequate and meet the requirements**, so CNSC staff is convinced that there are no safety issues and one of our specialists will be able to provide additional details should you be interested.

MEMBER BERUBE: I think we need to explore this a little deeper just to meet the needs of the Passamaquoddy. They are still concerned about this, so maybe we should use your expert and drill a little deeper.

DR. VIKTOROV: Nithe, if you are online,please step in.

MR. NITHEANANDAN: My name is Nithe, Nitheanandan. I am the Director for the Reactor Behaviour Division of the CNSC.

As New Brunswick Power said, we have been looking at this for a number of years. There was a COG Report that we wrote, COG JP4534, in the year 2002 and according to that all these concerns were resolved, but lately, several papers were published in the international journals and then we started to look at those claims.

So we brought in about six of our experts and the author of the paper and then we have completed approximately four different meetings over a period of about six to seven months and we are very close to writing the report.

At the moment we don't see any safety concerns per se, but there are some credibility to some of the claims that needs extensive review. **We are at that stage and we are writing the report and that will produce a joint report between the author and the CNSC experts.**

From the May 11 2022 Transcript - starting on page 104

DR. EDWARDS: … The second matter has to do with the inadequacy of the existing overpressure relief valves in the Lepreau reactor. These valves are needed to prevent rupturing of the primary cooling system in the unlikely event of a prolonged overpressurization of the heat transport system.

As you know, the tremendous heat generated in the core of the reactor must be dumped somewhere, and the main heat sinks are the boilers or steam generators,which are constantly cooled by feed water that flows past the thousands of little pipes inside each boiler. If for some reason this heat sink is not available, especially for a prolonged period, the temperature and the pressure in the heat transport system will rise and something will rupture unless that pressure can be released quickly enough. This is true even if the reactor is instantly shut down, because the radioactive decay heat continues long after shutdown and even after two hours can cause violent ruptures if the valves are not big enough to relieve the pressure.

This is discussed in CMD 22-228A, and it is based on a detailed analysis by Dr. Sunil Nijhawan, which was published in 2012, ten years ago, in the proceedings of the 20th International Conference on Nuclear Engineering.

My colleagues and I have studied this very closely and we are convinced that he is correct in his reasoning that these valves that are now in place in the Lepreau reactor cannot possibly relieve the pressure when the heat sinks are lost. Yesterday I was pleased to learn that CNSC staff is apparently now examining this situation much more closely, as directed, I believe, by Madam Velshi. And I am confident that if the Commissioners request an in-camera briefing on this matter, they can quickly satisfy themselves that this is both a legitimate and an urgent safety concern.

Although severe accidents are indeed very rare, there are several examples of loss of heat sinks in operating nuclear reactors, including the Three Mile Island partial meltdown of 1979 and half a dozen others which didn't lead to such consequences.

CCNR considers that it would be unconscionable for CNSC to license a plant like Lepreau for operation without ensuring the properly sized relief valves are installed. Point Lepreau is currently shut down for a couple of months and now is the time when the proper valves should be installed.

CCNR is convinced it would also be unconscionable for CNSC staff to withhold this information from the Commissioners, in which case they would be voting on an operating licence without being made aware of potentially catastrophic failures caused by inadequate engineering systems in the unlikely case that a loss of heat sinks occurs.

[hiatus]

I believe that it's been a long time. It's taken more than ten years for people to take the overpressure problem seriously. If the primary heat transport system were to rupture due to overpressurization, it is likely the case that the aging steam generators would prove to be the weakest link in the system. Those pipes would probably rupture first.

During the refurbishment of the Point Lepreau reactor, NB Power chose not to replace the old boilers, unlike Bruce Power which replaced all the boilers in its CANDU reactors during its refurbishment in Ontario. Perhaps this choice on the part of NB Power was due to cost considerations, but there is no doubt that the tube bundles inside those old steam generators are not in prime condition and probably many of them have already been plugged because of leaks that have occurred already.

This is unfortunate, because if there is core damage due to overheating, combined with ruptures in the boilers, there could be large unfiltered releases of radioactivity directly into the atmosphere, bypassing the usual filtering mechanisms. Radioactive contamination that has accumulated in the boiler tubes themselves would be blown out of the plant into the environment, but more significantly there would be a clear pathway created by which fission products, actinides and activation products could easily escape even from the core of the reactor into the outside environment.

Yesterday, by the way, the question was raised incidentally as to whether or not there is any plutonium in the boiler pipes. I would refer the Commissioners to a CNSC staff document of many years ago, CMD 10H.19B, which has a table showing the radioactive contents of two old boilers that were removed during refurbishment at Bruce. These boilers were from two different CANDU reactors, Bruce Units 1 and 2. The mass of radioactive material inside these boilers was mostly plutonium: 88.9 percent in the Unit 1 boiler and 93 percent in the Unit 2 boiler. The actual mass of plutonium is only 3 to 3.5 grams in these two boilers. However, that small amount of plutonium is enough to give 4.2 million atomic workers the maximum permissible body burden of plutonium, which is set at 0.7 micrograms per person. That plutonium would be dispersed in the worst possible form, as a very fine respirable dust, if it escaped into the atmosphere.

Annex E

Failure of CCNR staff to provide a factual answer to a fundamental safety question posed by CNSC President Dr. Michael Binder

Excerpted from CNSC hearing transcripts from December 3-6 2012

This annex includes only the text pertaining to question 1. If one wishes to read about the other questions please see www.ccnr.org/CNSC_CCNR_Supp_2013.pdf

Supplementary Submission from the Canadian Coalition for Nuclear Responsibility

To: CNSC Commissioners
From: Gordon Edwards, Ph.D., President,
Canadian Coalition for Nuclear Responsibility (CCNR)
Date: January 13, 2013

Re: ***Failure to provide factual answers to questions posed by the Chairman during the December 3-6 Hearings on the EA for OPG's proposed refurbishment and continued operation of the Darlington reactors.***

(1) This memo is a Supplementary Submission from the Canadian Coalition for Nuclear Responsibility (CCNR) in connection with the December 3-6 Public Hearings on the OPG Environmental Assessment of the Proposed Refurbishment and Continued Operation of the Darlington nuclear power reactors.

(2) Due to the failure of the proponent and the CNSC Staff to provide factual answers to three specific questions posed by the Chairman during the hearings – questions of crucial importance for assessing the potential environmental impacts resulting from realistic accident scenarios – CCNR considers it unacceptable for the panel to approve the EA for the proposed refurbishment and continued operation of the Darlington reactors based on the record of the public hearings. A full EA panel review will be required to fill in the gaps in information needed to adequately assess the potential impacts on human health and the environment associated with the proposed refurbishment of the Darlington reactors and their continued operation, particularly in light of the post-Fukushima emphasis on preventing large radioactive releases. The three questions posed by the Chairman that were not factually answered by either the proponent or the CNSC Staff are:

QUESTION 1: What is a scenario that will result in a large radioactive release?
QUESTION 2: Are the shutdown systems of a CANDU reactor sometimes unavailable?
QUESTION 3: Is the use of low-void-reactivity fuel safer than using the current fuel?

(3) The Chairman states on the record that the primary concern of the CNSC is to prevent large, potentially catastrophic, releases of radioactive materials into the environment, regardless of the calculated probability of such an event. *"Even if the plant is ruined, even if the plant is melted, as long as you can prevent [an] extreme release of radiation – that has always been our concern. Post Fukushima that's all we've been doing,"* he said. *"I don't care what [probability] level it is, negative 7, negative 8, negative 10."* [Transcript, December 5] Here, the phrase *"negative 7"* refers to one chance in 10 million, *"negative 8"* refers to one chance in 100 million, and *"negative 10"* refers to one chance in ten billion, per reactor per year.

(4) In this statement, the Chairman makes it emphatically clear that the prevention of an *"extreme release of radiation"* is a matter of paramount importance, no matter what the calculated probability – or improbability – of such an event may be.

QUESTION 1: What is a scenario that will result in a large radioactive release?

(5) Here is Chairman Binder's question: *"What I want to know is, what is a scenario that actually will result in a release? I'm still not there. Because they, according to the improvements that I've been hearing about, with all the offsite facility, with all the mobile facility, with all of that, I just don't see a scenario where you're going to give a release . . ."*

(6) A minute or two later, the Chairman repeats his question with the words, *"So that's what I'd be interested in reading, would this scenario actually cause a large release?"* The scenario under discussion is one of those itemized in the OPG document entitled 'Darlington NGS Risk Assessment: Summary Report'. That report confirms a large release of radioactivity for this scenario, with an estimated probability of 5 times 10 to the negative six – that is, five chances in a million per reactor per year.

(7) The proponent Ontario Power Generation (OPG) does not answer the Chairman's question in spite of the Chairman's having stated the question twice. OPG does not even try to address the question, despite its critical importance in assessing the environmental impacts of postulated accident scenarios. Here are the two statements by OPG spokespersons immediately following the Chairman's question:

> **MR. TREMBLAY (OPG):** Pierre Tremblay for the record, you know, I guess what I would say is the issue here is, you know, that whether it's an analysed scenario or not an analysed scenario, things could happen, and that means not only do you have to have this mitigating equipment, but you need to be prepared and the video pointed this out very well, to act and to respond to an event should it occur.
>
> So it may be to that point. I'll ask Laurie Swami to talk a little bit about that aspect of it because really, that's the natural transition to this discussion.
>
> **MS. SWAMI (OPG):** Laurie Swami for the record. I understand all of the discussion we've had from an EA process. I agree with Dr. Thompson's comments with respect to the planning tool, looking at possible mitigation, so if we take that one step further, one of the mitigations that's available is to implement the emergency response plans. And we've talked at great length already about the 5.7 millisieverts at the 1 kilometre level and we've talked about the sheltering that we've – that has been fully assessed in this environmental assessment. In addition to that, we have assessed what the evacuation would look like should there be one in the Darlington area and you'll see that in our technical support documents on human health.
>
> We've actually discussed that. We've included in our assessment the evacuation time estimate studies very similar to the work that we did on Pickering when we looked at the nuclear accidents. So all of that work while I understand that it's not as – not in the same way that the sheltering was because that actually was included in the final environmental assessment report. We do have a discussion of this. We do talk about evacuation and we do talk about the effects that that could have on the community, as well as in the environment. So I think that is covered in our work.
>
> And from an accident analysis itself, I agree also with Mr. Elliott's comments that this is the process that we're in, where we do studies. We look for improvements and we implement those improvements, and that is what we're talking about today.
>
> I hope that answers your question. I'm not sure I was as short and brief as I could have been. *[Transcript, December 5]*

(8) But OPG did not answer the Chairman's question. The Chairman originally asked how a large release of radioactivity could possibly result from the scenario specified in the OPG document. Yet no effort was made by OPG to provide such an explanation or even to describe any such scenario.

(9) CNSC Staff also failed to answer the Chairman's question. CNSC Staff made these two statements very shortly after the two OPG statements quoted above:

> **MR. JAMMAL (CNSC Staff):** Mr. President, if you allow me, sir, I want to try to take an attempt at this without the numbers or anything. The study that is out is identifying a release. ***We agree that there is a release***. *[EMPHASIS ADDED]*
>
> Now, as part of the design change for this refurbishment, the improvement that's taking place, that is what we are talking about right now. So with the improvements that's going to be installed as part of the refurbishment, or even before the refurbishment is going to be done, that's the release that's going to – ***the added mitigation measures is going to protect and provide mitigation for that release***. *[EMPHASIS ADDED]*
>
> So I'm trying to give it without going through the numbers. So in summary, this was done without any improvements. Now, once you put the improvements in place, that is going to be the release. I'll pass it on to Mr. Frappier if he's got anything else to add.
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> **MR. FRAPPIER (CNSC Staff):** Perhaps just real quickly, Gerry Frappier, Director General of Assessment and Analysis.
>
> ***So the report that Mr. Stensil is making reference to, we are very, very aware of. We've gone through it in detail.*** In fact, it is an outcome of the process that CNSC has put in place with the requirement of S-294 and the creation of a detailed probabilistic safety assessment. *[EMPHASIS ADDED]*
>
> The reason for that – and as was mentioned by OPG, the demonstration that it works – is exactly what we're talking about here, ***a release category that perhaps was possible, although very, very, very unlikely,*** through this process, has now been shown to be something that OPG has now put a design fix in place. *[EMPHASIS ADDED]*
>
> So whereas when that report was released that Mr. Stensil is making reference to, OPG had that as ***one of the potential release paths***. *[EMPHASIS ADDED]*
>
> With the design changes that are going to be put in place as part of this enhancement to the facility, ***that is no longer considered a possibility and it is no longer one that has to be considered***. *[EMPHASIS ADDED]*

(10) These responses by CNSC Staff fail to describe any scenario or any sequence of events that will result in the large release of radioactivity described in the OPG document. But that is what the Chairman was asking for. He wanted details about *"the potential release path"*. He was seeking a description of the mechanism or mechanisms by which such large releases might occur. Can they happen when ventilation dampers become jammed in the open position, as they did during the 1952 NRX reactor accident at Chalk River? Might they happen when personnel airlock doors are not properly sealed, as has often happened in OPG reactors? Could they happen when one, several, or many of the hundreds of penetrations of the containment wall are not airtight? Will they happen if there is a failure of negative pressure in the vacuum building? Or might they happen as a result of other causes not listed here?

(11) By failing to answer or even address the Chairman's direct question, both OPG and CNSC Staff have in effect prevented the Chairman and the other Commissioners – as well as the intervenors in this public hearing – from gaining an understanding of the mechanisms by which large releases of radioactivity are possible. Without that understanding, Commissioners cannot objectively judge the claims of OPG and the CNSC Staff that such releases are "*no longer considered a possibility"* and can therefore be excluded from further consideration.

(12) So let's take a look at the OPG Report that CCNR Staff is *"very, very aware of"* and has *"gone through in detail."* It is the Darlington 2012 Probabilistic Safety Analysis (PSA) Summary Report, and as Mr. Webster of OPG noted during the hearings, the document *"is already on the [OPG] website"* at **http://tinyurl.com/awloq2x**.

(13) Here's what the OPG Report say about large releases of radioactivity. On page 100, the top line of Table 16 of the Report addresses the largest radioactivity release category that is considered possible by OPG. It is coded as D-RC1, presumably meaning "Darlington Release Category 1". Table 10 of the same Report tells us what the D-RC1 code represents: it refers to *"a release of airborne fission products from the containment to the environment large enough to require prolonged population relocation."* Evidently such a large radioactive release would play a major role in judging the acceptability of OPG's EA for the refurbishment and continued operation of the Darlington reactors.

(14) The Chairman's earlier remarks referencing the Fukushima disaster underscores the importance of acknowledging and preventing a D-RC1 type release: *"Even if the plant is ruined, even if the plant is melted, as long as you can prevent [an] extreme release of radiation – that has always been our concern. Post Fukushima that's all we've been doing."*

(15) In Table 16, OPG states that its "baseline analysis" shows a very large radioactive release – category D-RC1 – is predicted to occur at Darlington with a frequency of about 5 times in a million reactor-years. This frequency may sound low, but it is in fact 5 times higher than OPG's *"safety target"* of no more than once in a million reactor-years for a large radioactive release (as laid out in Table 11). Moreover, this predicted frequency is only marginally better than OPG's absolute *"safety limit"* of 10 times in a million reactor-years for a large radioactive release (also in Table 11). Far from being unthinkable, a large radioactivity release is therefore not only possible but also probable. While the calculated probability is low, it is not zero – and it is actually higher than the best current safety practices would demand.

(16) Table 16 also states that using an *"enhanced model"* – a different mathematical treatment utilizing different assumptions – the predicted frequency of occurrence of a large D-RC1 radioactive release can be reduced to about 8 times in 10 million years – and that's without making any *"safety opportunity improvements"* (SOIs). In other words, the actual safety of the plant is completely unaltered; the only thing that has changed is how the arithmetic is carried out by OPG's analysts. If SOIs (safety

improvements) are carried out, and the enhanced mathematical model is also used, then the predicted frequency of a large D-RC1 radioactive release can be reduced further to about 5 times in 100 million reactor-years.

(17) But why was none of this communicated to the Commissioners in response to the Chairman's question regarding large radioactive releases? Both OPG and CNSC Staff indicated that they are well aware of the document and its findings. Table 16 clearly states that a large radioactive release from Darlington – one that would "*require prolonged population relocation*" – is a possibility regardless of safety improvements. What changes is only the calculated probability of that event, and the Chairman made it clear in advance that this does not diminish the importance of the issue: *"I don't care what [probability] level it is, negative 7, negative 8, negative 10"* he said.

(18) This failure to communicate goes beyond the question of the professionalism of the OPG and CNSC experts at the hearings. It touches on the single most important issue regarding public health and the other social and environmental impacts of nuclear power reactors. ***CCNR considers it unacceptable for CNSC to approve the EA for the proposed refurbishment and continued operation of the Darlington reactors given the failure of all the professionals involved to thoroughly inform the Commission on the one issue that the Chairman himself has identified as being of paramount importance.***

[For more, see www.ccnr.org/CNSC_CCNR_Supp_2013.pdf]