Submission to the CNSC by the Canadian Coalition for Nuclear Responsibility

Regarding the Darlington New Nuclear Project (DNNP)
and the documents provided in support of
siting one of four BWRX-300 reactors
at the existing Darlington NGS site

EIS Environmental Impact Statement
PPE Plant Parameter Envelope

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Part 1. Much Ado About Siting.

by Gordon Edwards, Ph.D.

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1.1 The original Darlington New Nuclear Project (DNNP-1)

Forty-six years ago, the Ontario government announced its decision to build four new CANDU nuclear reactors at the Darlington site. CANDU reactors are pressurized heavy water reactors. Heavy water is used both as coolant (to cool the fuel) and moderator (to slow down the neutrons).

That event marked the end of an era of rapid nuclear power growth in North America. After that date, from 1978 to 2008, the nuclear industry on this continent endured a three-decades-long drought in domestic reactor sales.

It seemed the drought might be ending fifteen years ago when, in March 2008, Infrastructure Ontario issued a competitive Request for Proposal (RFP) for a new nuclear power station in Ontario. Four vendors were invited to participate in the RFP process: AECL (the ACR-1000), Areva (the EPR), Westinghouse (the AP1000), and GE-Hitachi (ESBWR – Economic Simplified Boiling Water Reactor). These are all water-cooled reactors.

GE-Hitachi chose not to participate in the RFP process. Its reactor, the ESBWR, was the only Boiling Water Reactor design in the mix. The three vendors that remained in competition were all offering pressurized water reactors (PWRs using light water as coolant and moderator, or PHWRs using heavy water for those two functions).

In 2009, Ontario Power Generation (OPG) produced an Environmental Impact Statement (EIS) for the Darlington New Nuclear Project (DNNP). **The utility decided that the new reactors, if approved, would be co-located with four existing CANDU reactors that were already on the Darlington site.** Along with the EIS, the utility produced a Plant Parameter Envelope (PPE) document. **No choice of reactor model had yet been made**.

The Darlington New Build of 2008-2009 would have constituted the first order for new power reactors in North America since 1977, had it come to pass. But it didn't. The project underwent a full Environmental Assessment (EA) review in 2011, and Ontario Power Generation (OPG) even received a licence from the Canadian Nuclear Safety Commission (CNSC) to prepare the Darlington site for the new reactors.

Then, in 2014, the Ontario government abruptly cancelled the order for the first two of the four new reactors. Queen's Park balked at the exorbitant price tag, rumoured to be in the ballpark of $14 billion per unit. In the wake of that decision, none of the planned new reactors found their way into OPG's long-term energy plan.

However, OPG insisted that the DNNP New Build project was deferred, not cancelled. The utility ensured that the CNSC licence permitting it to prepare the Darlington site to accommodate new reactors would remain in force until 2022. Then, in 2020, OPG saw to it that the site preparation licence was extended even beyond 2022.

1.2 The current Darlington New Nuclear Project (DNNP-2)

Today OPG wants to use that 12-year old licence to prepare the Darlington site for a smaller reactor that was never under consideration in the first go-around. It is a previously unbuilt General-Electric-Hitachi (GEH) Boiling Water reactor design, the BWRX-300, touted as one member of a gang of "Small Modular Nuclear Reactors" (SMRs or SMNRs).

OPG must now persuade CNSC that the old site preparation licence is still valid, despite altered circumstances, and can be used for this new, unforeseen purpose. To do this, OPG has dusted off two documents that were written in support of the original Darlington New Build Project conceived 15 years ago, involving three completely different reactor designs.

Those documents are:

(1) OPG's 2009 Environmental Impact Statement (EIS-2) report for the original Darlington New Build Project; updated version October 2022.

(2) OPG's Plant Parameter Envelope (PPE-2) report; updated October 2022 (revision #5).

OPG has modified these two pre-Fukushima documents by adding some data relevant to the BWRX-300, without describing the reactor design in any meaningful detail. In the modified PPE, for example, the description of the BWRX-300 reactor design is limited to just three-quarters of a page and one diagram– the very last two pages of PPE-2.

The 2011 EA Report noted that, following a request for OPG to consider other reactor designs, "a revised version of the plant parameter envelope was submitted by OPG on November 30, 2010. OPG noted that a similar **assessment was not performed for a boiling water reactor as insufficient information was available** to allow OPG to do so."

CCNR also finds insufficient information available in the aforementioned documents for our reviewers to do a meaningful analysis bearing on the site preparation licence for the boiling water reactor BWRX-300. Indeed, it appears to us that this entire exercise may be merely a formality – a prelude before CNSC grants OPG a licence to construct, which seems to be taken by all players as a forgone conclusion.

Just one day after Canada's Infrastructure Bank gave OPG a $970-million "low-interest loan" to develop the BWRX-300 at Darlington, the Minister of Natural Resources Canada boasted to a Washington audience that it would soon be Canada's first commercial SMNR.

Coincidentally, the Minister of Natural Resources (NRCan) is designated as the "responsible minister" in the Canadian Nuclear Safety and Control Act. That's the law establishing CNSC as an agency of the crown, whose mandate is to protect the health and safety of Canadians and the environment from unreasonable radiation exposures, and to disseminate objective scientific information on nuclear matters.

Although the International Atomic Energy Agency (IAEA) has urged that nuclear regulators not be linked to government agencies that promote the nuclear industry, that sensible suggestion does not seem to have been implemented in Canada.

CNSC president Rumina Velshi has publicly lauded the speed at which the BWRX-300 licensing is proceeding, saying that Canada will be the first western country to approve an SMNR built for the grid. She has stated publicly that the CNSC is there to protect people against radiation, not against progress.

CNSC has not yet approved the reactor. However, OPG held a ground-breaking ceremony at Darlington in December 2022. So the licence to construct seems to be a foregone conclusion – to NRCan, to CNSC, and to OPG. In 2017, CNSC freely admitted that from the year of the agency's inception, in 2000, it has never refused to grant a licence for any major nuclear facility.

Government, regulator and industry are already on board. So what is the intended purpose of this review?

On page 5 of the PPE-2 we read: "The concept of a PPE was developed in the United States for use in the Early Site Permit (ESP) process to resolve siting and environmental issues at a particular site **before a reactor design has been chosen**."

However, we have now arrived at the point where a reactor design has been chosen. So the PPE-2 document is actually moot and irrelevant– filled as it is with extraneous information about the three original candidate reactors that have since gone by the wayside. Adding sparse numerical data about the BWRX-300 – data supplied by the vendor, without any detailed design information to allow others to verify or to challenge those data, hardly constitutes a meaningful review process.

Continuing from page 5 of PPE-2: "The PPE concept is also consistent with the Canadian Nuclear Safety Commission (CNSC) statement in Revision 1 of the CNSC Information Document INFO-0756 [R-12]; 'An application for a Licence to Prepare Site does not require detailed information or determination of reactor design; however, **high level design information is required for the environmental assessment that precedes the licensing** decision for a Licence to Prepare Site.'"

It is crystal clear that "high level design information" about the BWRX-300 reactor has never been made available to the public, nor to the Joint Review Panel that reviewed the original EIS and produced the 2011 EA Report. OPG just wants the site approval.

That information vacuum and accompanying pressure to accept the sleight-of-hand of replacing one reactor for three others, inspired the title of this report – Much Ado about Siting.

According to CCNR, both documents – the PPE-2 and the EIS-2 – cannot be considered satisfactory surrogates for the real thing: an actual honest-to-goodness environmental impact assessment of the BWRX-300 reactor itself, sited at Darlington or elsewhere.

The present report, Much Ado About Siting, is based on the professional services of Dr. Gordon Edwards and Dr. Sunil Nijhawan. The report is a critical commentary by the Canadian Coalition for Nuclear Responsibility (CCNR) on the use of the afore-mentioned documents as the basis for a decision-making procedure regarding the siting of up to four new BWRX-300 reactors very close to the four existing co-located CANDU reactors.

Part 1, by Dr. Edwards, deals with the siting question directly, while Part 2, by Dr. Nijhawan, deals with the OPG surrogate documents, especially PPE-2.

In the next two sections it will be shown that the construction of the first of these new BWRX-300 reactors (1 of 4) is intended to take place well within the exclusion boundary of the existing Darlington reactors. CCNR believes that this should not be allowed.

Recommendations:

1. CCNR's main contention is that the present procedure lacks validity given the realities of the post-Fukushima world and the paucity of information provided about the BWRX-300 boiling water reactor – a type of reactor that was never considered in the original EIS.

2. Drawing on the lessons of Fukushima regarding the special vulnerabilities of co-located reactors, CCNR urges that construction of any new reactors within the exclusion zone of the existing DNGS four-reactor complex must be ruled out as against the public interest.

3. In keeping with the CNSC regulatory practice as outlined in PPE-2, OPG should be required to prepare a new environmental impact statement with high level design information about the BWRX-300.

4. The EIS for the BWRX-300 must provide a sufficiently detailed description of the plant's design to allow for independent verification of numerical values that are assigned to various parameters such as source terms. It should not be accepted as a foregone conclusion that the Darlington site is necessarily suitable as compared with other sites.

1.3 Radioactive Emissions from Darlington New Build

Let's consider one of the numbers missing from PPE-2, the total atmospheric release of radioactive noble gases (last entries in tables 4.1 and 4.2). We know boiling water reactors tend to release more radioactive gases into the atmosphere than pressurized water reactors.

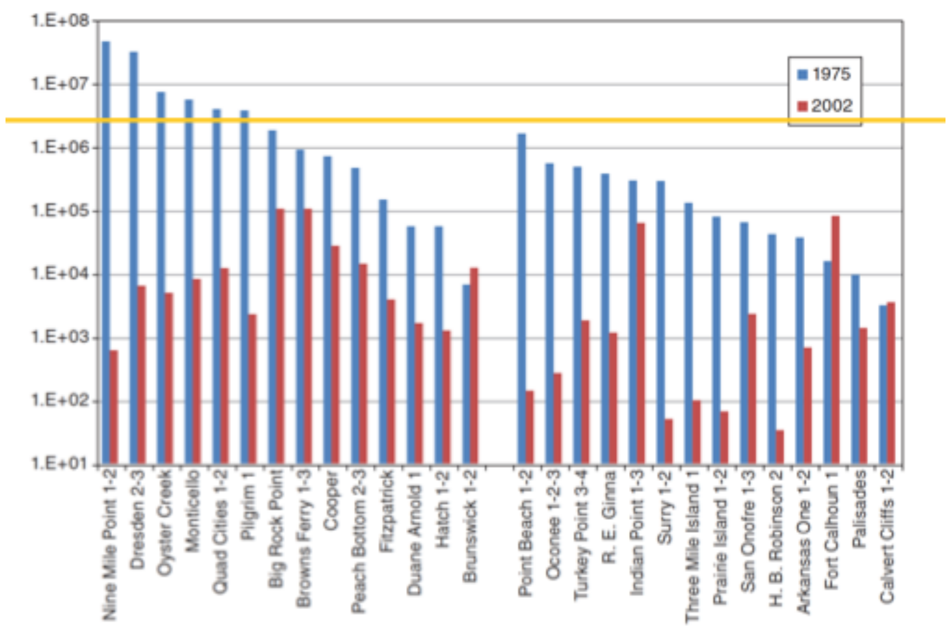


FIGURE 2.5 Comparison of atmospheric releases of noble gases for selected BWRs (left) and PWRs (right) in the United States. The units on the vertical scale are in gigabecquerels (GBq = 0.03 Ci). SOURCE: Data from the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR).

It is troubling that OPG would omit listing the total noble gas emissions in both tables of PPE-2. After all, the BWRX-300 is the only boiling water reactor ever considered in the context of this pre-licensing process.

But even if the appropriate numbers had been given, it would not be enough. You cannot judge the environmental impact of radioactive noble gas emissions just by the number of becquerels released each year. These gases are considerably heavier than air. They have to be released at a great altitude to minimize the gamma dose ("sky shine") to people and animals on the ground below.

But, the BWRX-300 reactor is underground and the building does not reach as high (35 m) as any of the other reactors previously considered in the PPE (typically around 48 m). So the possibility of a near-ground release cannot be excluded. That would be problematic.

There are a great many other considerations surrounding the important topic of radioactive releases. Dr. Frank Greening discussed many such aspects authoritatively in the original DNNP EA hearings of 2011. With his permission, Dr. Greening's original work on this subject is attached as Annex D: "Radioactive Emissions from Darlington New Build." His work should be considered as an integral part of this report.

Dr. Greening's work was originally submitted by le Mouvement Vert Mauricie, along with other reports by Dr. Gordon Edwards and Dr. Michel Duguay, in the original DNNP EA Hearings of 2011. The entire MVM submission is found at www.ccnr.org/MVM_final.pdf

1.4 Fulfilling the JRP conditions

The EIS-2 and PPE-2 documents have been modified by OPG in an effort to include some aspects of the newly chosen design, the GE-Hitachi Boiling Water reactor called BWRX-300. However, very little information about the actual reactor design is given.

Numbers are provided by the vendor without any clear evidence of how they were derived. These numbers are used by OPG to bolster its contention that the GEH BWRX-300 reactor, although never an object of scrutiny during the 2011 EA review, is nevertheless within the scope of that review and therefore acceptable.

As noted earlier, an EA review of the EIS was carried out in 2011. Public hearings were held before a three-person Joint Review Panel (JPR). Two of the Panel members were drawn from the Environmental Assessment Agency and the third from the Canadian Nuclear Safety Commission (CNSC).

The JPR recommended that the EIS be accepted and the project be approved, subject to a large number of important conditions. Approval is given "provided the mitigation measures proposed and commitments made by OPG during the review, and the JRP's recommendations, are implemented."

Those conditions are reproduced in Annex A as a ten-page document.

A great many of the JRP conditions are very specific to the Darlington site. Not only the licensee OPG, but also the regulator CNSC is required to act. Some of the conditions apply "Prior to Site Preparation", some apply "During Site Preparation", some apply "Prior to Construction", and so forth. Here are some examples:

• "CNSC [shall] require OPG to conduct a comprehensive soils characterization program." [Rec. 2];

• "CNSC [shall] require OPG to develop a follow-up and adaptive management program for air contaminants [and] must require OPG to develop an action plan acceptable to Health Canada for days when there are air quality or smog alerts." [Rec. 8]

• "CNSC [shall] require OPG to undertake a detailed site geotechnical investigation prior to commencing site preparation activities." [Rec. 10]

• "CNSC [shall] require OPG to perform a thorough evaluation of site layout opportunities before site preparation activities begin, in order to minimize the overall effects on the terrestrial and aquatic environments and maximize the opportunity for quality terrestrial habitat rehabilitation."

Recommendations:

5. CNSC shall ensure that all of the conditions laid down by the JRP are fully implemented before a construction licence is considered.

6. CNSC shall require OPG to publish, in tabular form, all measures taken to implement each applicable JRP condition and subcondition, with links to appropriate documents detailing how the implementation was carried out. CNSC staff shall certify that the implementations have been satisfactorily realized or that they must be redone.

A particularly important condition is the one dealing with geotechnical aspects of the site:

Recommendation # 38 (Section 5.9):
The Panel recommends that the Canadian Nuclear Safety Commission require that the geotechnical and seismic hazard elements of the detailed site geotechnical investigation to be performed by OPG include, but not be limited to:

Prior to site preparation:

- demonstration that there are no undesirable subsurface conditions at the Project site. The overall site liquefaction potential shall be assessed with the site investigation data; and

· confirmation of the absence of paleoseismologic features at the site and, if present, further assessment to reduce the overall uncertainty in the seismic hazard assessment during the design of the Project must be conducted.

During site preparation and/or prior to construction:

· verification and confirmation of the absence of surface faulting in the overburden and bedrock at the site.

Prior to construction:

· verification of the stability of the cut slopes and dyke slopes under both static and dynamic loads with site/Project-specific data during the design of the cut slopes and dykes or before their construction;
· assessment of potential liquefaction of the northeast waste stockpile by using the data obtained from the pile itself upon completion of site preparation;
· measurement of the shear strength of the overburden materials and the dynamic properties of both overburden and sedimentary rocks to confirm the site conditions and to perform soil-structure interaction analysis if necessary;
· assessment of the potential settlement in the quaternary deposits due to the groundwater drawdown caused by future St. Marys Cement quarry activities; and
· assessment of the effect of the potential settlement on buried infrastructures in the deposits during the design of these infrastructures.

OPG contends that BWRX-300 should be accepted as an acceptable surrogate for the three reactor designs that were indeed considered by the Joint Review Panel (JPR), and that PPE-2 and EIS-2 be accepted as acceptable surrogates for the original EIS-1 and EIS-2.

The Canadian Coalition for Nuclear Responsibility does not share this view, as already indicated. Reasons for the CCNR position will be laid out in the following sections.

1.4 Infiltrating the Exclusion Zone

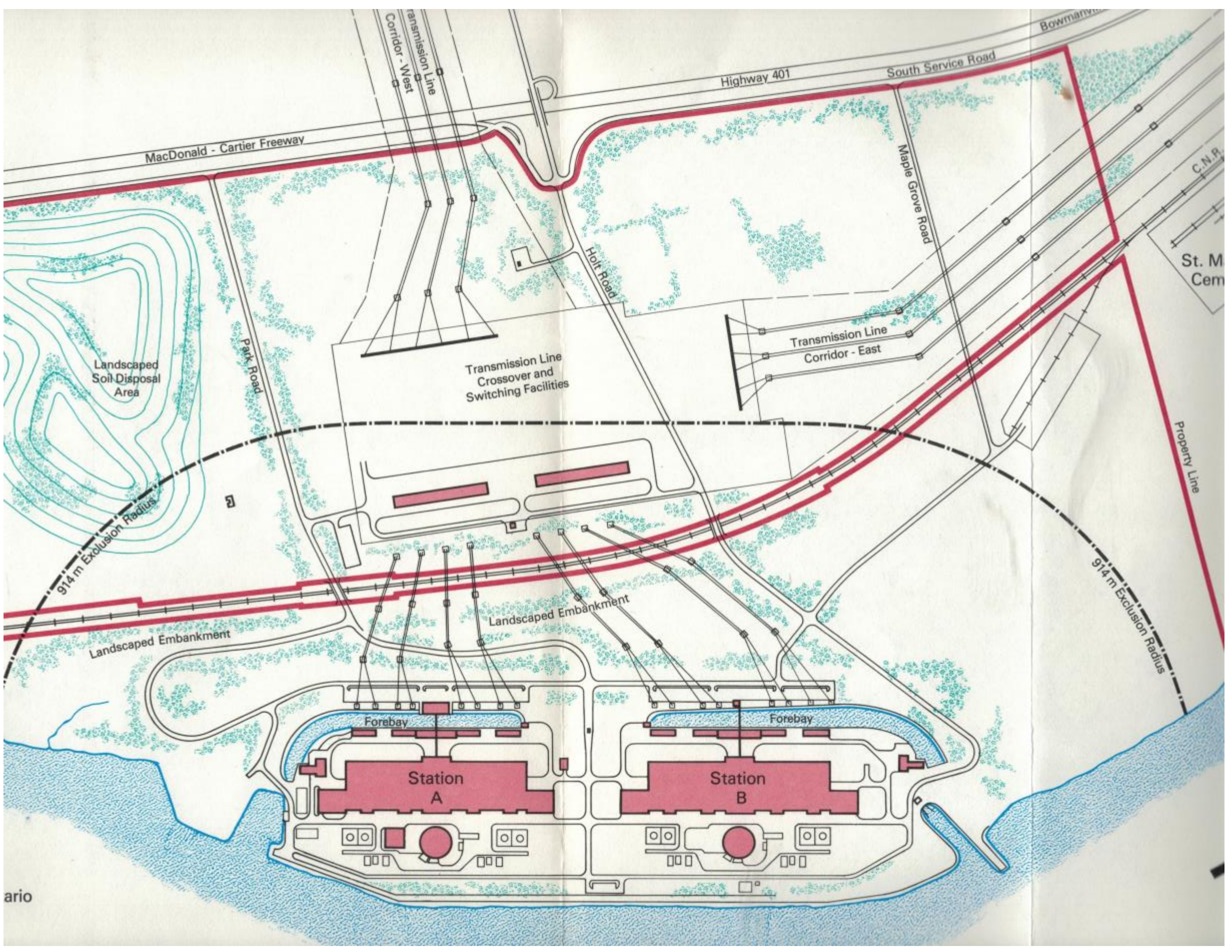
To maintain that the BWRX-300 has essentially been approved "in absentia" by the Joint Review Panel's Environmental Assessment Report of 2011, is unacceptable given
(1) the lack of detailed consideration of the idiosyncrasies of the new reactor choice
(2) the proximity of the Darlington site to Lake Ontario, and
(2) the lessons of Fukushima, which were not available to OPG, the CNSC, the JPR or the Canadian public at the time when the original EIS, PPE and EA report were drawn up.

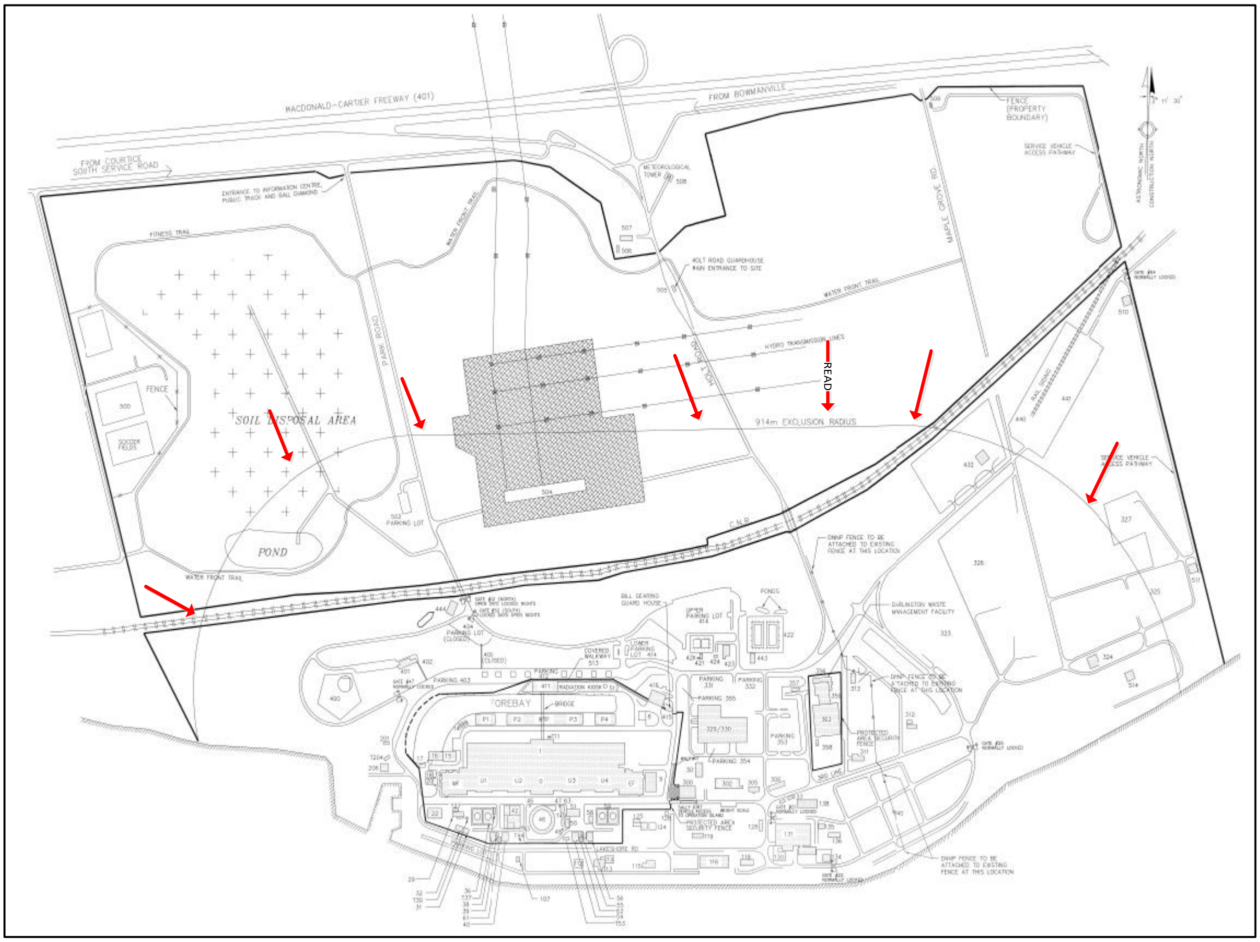
As an example, consider the implications of having the major working portions of a nuclear reactor situated in an underwater chamber, subjected to hydrostatic pressure from all sides. That could be the BWRX-300, if built on the Darlington site. Unlike any of the other three reactor designs considered in the 2009 EIS or the 2011 EA, the BWRX-300 will extend 38 metres underground and well as 35 metres above ground.

Imagine two ten-storey buildings. Each will be about 128 feet tall, or in metric units, 36 metres. Now imagine a ten-storey building turned upside down, going down into the ground, with another ten-storey building going up. Imagine the underground portion to be holding the heart of a 300 megawatt nuclear power reactor. That's the BWRX-300. It is an unusual picture, made more unusual because the underground portion will be in water,

Due to the proximity of Darlington to Lake Ontario, any excavation 38 metres downwards will fill with water very quickly and almost totally, so it will have to be constantly pumped out (dewatered) during construction. Unless dewatering is made permanent – and EIS-2 says it will not be – the hydrostatic pressure on the outside walls of the finished reactor building will be in the range of 300 kilopascals (kPa) at 35 metres depth. That's 6000 pounds (3 tons) per square foot. Yet there is no detailed discussion of the possible implications of such an unprecedented situation in either of the two updated documents, EIS-2 or PPE-2, except for one brief paragraph in EIS-2.

Then there's the geometry. Until the government of Ontario nixed the original DNNP project nine years ago, it was assumed that DNGS A & B (8 large reactors total) would sit side by side. The exclusion zone was designed to accommodate all eight reactors.





In these two images, the dotted lines indicate the boundary of the exclusion zone. The first image is from 1978, the second is from the 2012 Darlington Safety Report.

The DNGS exclusion zone was subsequently redrawn, without commentary, taking in a much smaller area.

Some of the space previously allocated to DNGS B has now been reassigned for the storage of nuclear waste.

The two pictures on the next page are both from 2022. The first image is from OPG's documentation supplied for the recent Waste Management Licence extension hearings, the other one is from current DNNP documentation.

Attachment 5 CD# 00044-CORR-00531-01153

Figure 1: Darlington Site

Note: The blue dotted line in this figure is the DNGS exclusion zone.

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NEDO-33951 REVISION 1
NON-PROPRIETARY INFORMATION

Figure 2.1.1-2: Darlington Nuclear Generation Station and Darlington New Nuclear Project Lands

In each of these two diagrams, green circles represent the boundary of the exclusion zone.

In the aftermath of the Fukushima triple meltdown of 2011, we have a better understanding of the dangers of co-locating reactors. It is perhaps a blessing in disguise that DNGS B never got built. Be that as it may, OPG and CNSC are now considering up to 4 new reactors of the BWRX-300 variety to fit into this rather crowded space, with spent fuel dry storage facilities now occupying some of the space originally intended for DGNS B.

It appears that the first BWRX-300 will be right inside the redrawn DNGS exclusion zone, Its own exclusion zone (circle below, radius 350 m) largely overlaps the one from DNGS.

NEDO-33950 REVISION 2
NON-PROPRIETARY INFORMATION

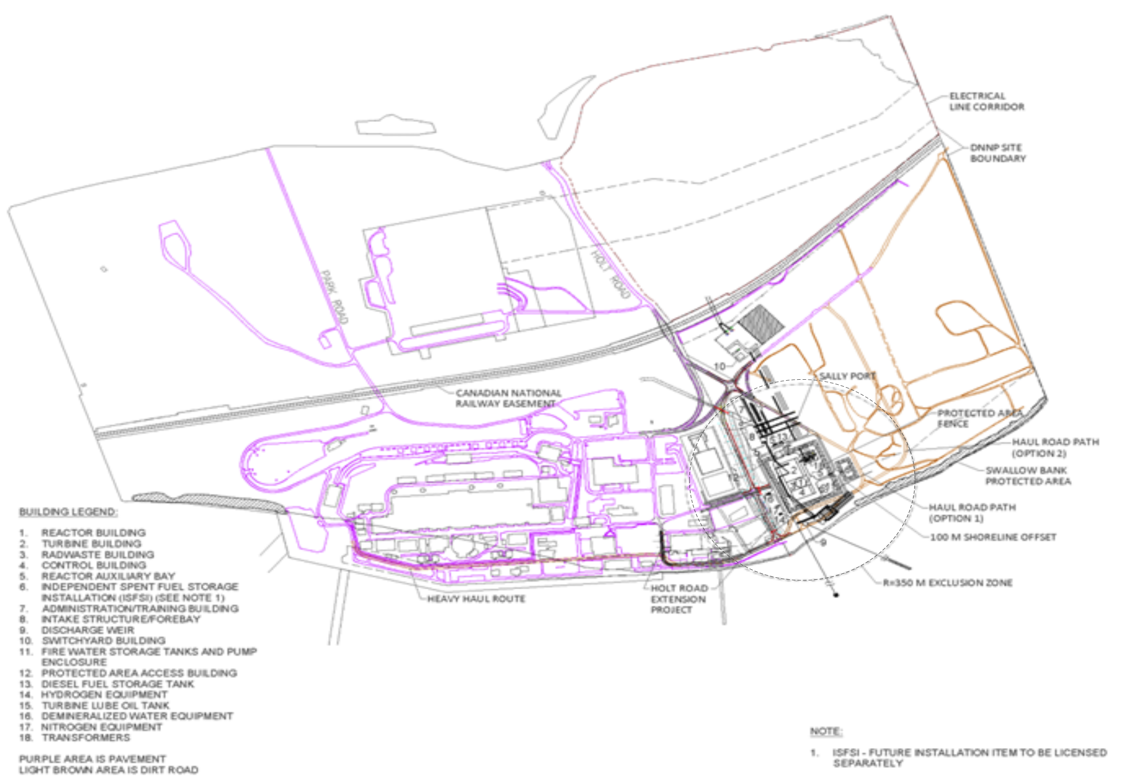

Figure A1.1-2: Darlington Nuclear Site (DNNP Proximity to DNGS)

In light of the lessons we have learned from Fukushima, CCNR believes it is unacceptable to have a new reactor built inside the exclusion zone of an existing reactor. In the event of a severe accident at one or more of the existing Darlington reactors, the entire construction crew of 1,000 to 2,000 workers could receive radiation exposures greater than 25 rems (250 mSv) within two hours. There is no reason to expose the workers to such a risk. They are not even classified as radiation workers.

The exposure of 25 rems in two hours, mentioned in the previous paragraph, is based on the precise definition of an exclusion zone given by the U.S. Nuclear Regulatory Commission (NRC). That definition is explained in the next section. Judging by the rather cavalier way in which the Darlington exclusion zone has been drawn and redrawn, and how the much smaller exclusion zone for the BWRX-300 has been drawn as a perfect circle, CCNR is convinced that CNSC is not doing its job by requiring OPG to define meaningful science-based exclusion zones using quantitative criteria and a detailed analysis of potential radiation exposures.

1.5 Defining the Exclusion Zone

CNSC has signed a Memorandum of Understanding (MOU) with NRC to harmonize regulations, and the two agencies are working together on BWRX-300 licensing matters. It is therefore appropriate to expect consistency between the two bodies in the definition of nuclear reactor exclusion zones.

U.S. Nuclear Regulatory Commission (NRC) document 10 CFR 100.11 details how to determine exclusion zones around nuclear power plants. The document is reproduced in Annex B.

According to 10 CFR 100.11, the applicant must begin by assuming a significant fission product release from the core of the reactor. "The fission product release assumed for these calculations should be based upon a major accident . . . that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products." See Annex C of this report.

Document 100.11 makes special mention of sites with "multiple reactor facilities" such as Darlington. Again, see Annex B. "If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area . . . shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously." The document discusses other factors that might be brought to bear so as to reduce this requirement to some degree. However, any reduction would have to be justified to the satisfaction of the Commission.

Once the fission product release from the core has been established, the applicant must then proceed to calculate how much escapes into the atmosphere by using "the expected demonstrable leak rate from the containment". The meteorological conditions pertinent to the specific site shall then be used to derive an exclusion zone "of such size that an individual located at any position on its boundary would not receive a total radiation dose to the whole body in excess of 25 rem [250 millisieverts] or a total radiation dose in excess of 300 rem [3 sieverts] to the thyroid from [radioactive] iodine exposure."

Recommendations:

7. That OPG be required by CNSC to derive science-based exclusion zones for both Darlington NGS and for the proposed BWRX-300 reactor according to the criteria laid out in U.S. NRC document 100.11.

8. That no new reactor be allowed by CNSC to be built within the exclusion zone of any other existing reactor.

Lest CNSC or OPG staff or any other party mistakenly think that these criteria make it acceptable for ordinary construction workers to work within the exclusion zone of an existing operating reactor, the NRC offers the following clarification:

> "The whole body dose of 25 rem referred to above corresponds numerically to the **once in a lifetime accidental or emergency dose for radiation workers** which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute **acceptable limits for emergency doses to the public under accident conditions**. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation." *[Footnote #2, US NRC 100.11]*

Just to be perfectly clear, NRC states that these calculated doses (25 rem whole body, 300 rem to the thyroid) are **NOT "acceptable limits for emergency doses to the public under accident conditions".** That implies that people who are not radiation workers should not be working in the exclusion zone of an operating nuclear reactor.

The mandate of the CNSC is to protect people against radiation exposure. There is nothing in the mandate of the CNSC having to do with progress. The question is, will CNSC live up to its real mandate? Or will it pursue a fictitious mandate of its own making?

CCNR believes that if CNSC allows OPG to site the BWRX-300 reactor within the exclusion zone of the Darlington Nuclear Generating Station, it will be acting in dereliction of its duty as defined under the Nuclear Safety and Control Act.

1.6 The relevance of the Fukushima accident

The original 2009 EIS and PPE documents were written for a Darlington New Nuclear Project that never came to pass. Those documents were conceived in complete ignorance of the triple meltdown that was about to take place at the Fukushima Daiichi nuclear complex in March 2011. As a result, the two reports do not incorporate any of the lessons learned from the Fukushima disaster – lessons which go far beyond the merely technical.

The reactors that melted down in Japan were Boiling Water Reactors (BWRs) of an early design (circa 1960) supplied by General Electric (GE), Toshiba and Hitachi. They were early precursors of the GE-Hitachi BWRX-300 reactor now under consideration by OPG.

There were six such BWRs co-located at the Fukushima Daiichi site. Unit 1 was rated at 461 megawatts of electrical power (MWe) (half again as large as the BWRX-300) while units 2 to 5 were rated at 780 MWe each. Unit 6 was the largest, rated at 1100 MWe –two and a half times the power of unit 1 and almost 4 times that of BWRX-300.

On March 11, 2011, a powerful 9.1 magnitude earthquake offshore led to the safe shutdown of all these reactors. But within 30 minutes a gigantic tsunami struck, disabling the backup electrical generators and causing a prolonged total station blackout. Without power to run the pumps, there is no way to remove the intense radioactive decay heat from the spent fuel inside the core. In units 1, 2, 3, the fuel began to melt, releasing radioactivity.

Radioactive gases mingled with superheated steam and explosive hydrogen inside the reactor containment vessel. The gases were vented in order to to relieve the pressure that was rapidly building up inside. Once released, the hydrogen gas exploded, punching holes in the outer containment building and spreading radioactive contamination over a vast area. 120,000 people living nearby were evacuated in 2011. Twelve years later, 30,000 of those evacuees are still unable to go home.

An important lesson from Fukushima is that mathematical probability calculations do not protect people from catastrophic events. Before 2011, few in the nuclear industry would have believed that a simultaneous triple meltdown was a credible event. Yet that's what happened. There was a "common cause" for all three meltdowns.

Lessons from Fukushima

1. Simultaneous nuclear disasters can occur at a multi-unit nuclear power plant due to a "common cause" that cannot be predicted accurately ahead of time.

2. For emergency planning one must "expect the unexpected" by postulating a possible radioactive release that may be regarded as having a vanishingly small probability.

Units 1, 2, and 3 are the ones that melted down. Unit 4 was defueled at the time of the disaster, but its outer containment structure (not the reactor containment vessel) was blown apart by one of three violent hydrogen gas explosions. No one knows the exact cause of the unit 4 explosion to this day. The blast blew off the roof of the building and exposed the spent fuel pool to the open air, situated as it was several stories above ground level.

Planes and helicopters were used as water bombers, to douse the spent fuel pool of unit 4. This was done to prevent extensive fuel damage caused by inadequate cooling. If the fuel in the pool had been uncovered by water, overheating could have released far more radioactivity into the atmosphere than had already been released from the 3 reactors that were melting down. Unlike the core which is situated inside a sturdy containment vessel, the pool had no containment at all. Had the uncovered spent fuel become exposed to the open air, a raging zirconium fire could have been ignited amongst the overheated fuel assemblies, leading to unparalleled radioactive releases.

Fukushima has taught us that spent fuel pools are particularly vulnerable to large radioactive releases under certain extreme conditions. Even raging metallic fires are possible when the fuel is not fully covered with water. Even years later, when the risk of overheating has subsided, spent fuel remains intensely radioactive and deadly when dispersed – whether that happens years or decades, or indeed even centuries after removal from the reactor core.

A typical dry storage container for Pickering used fuel weighs 60 tons when empty, and 70 tons when fully loaded. The reason why the dry storage containers designed to hold spent fuel are so much heavier (six times heavier) than the inventory of used nuclear fuel they contain inside, is for one reason only: shielding. Without massive shielding, the penetrating radiation would not be abated and the external risk would be prohibitive

Cooling is another concern. For the last 12 years, hundreds of tonnes of water have been used each day to cool the melted nuclear fuel from the stricken reactors. The water becomes contaminated with fission products flushed out of damaged fuel. Not all radionuclides can be filtered from the water; some, like tritium, can't be removed at all, others remain in residual amounts. More than a million tonnes of radioactive water is currently stored in over 1000 steel tanks.

Despite objections from China, Korea and local fishers, Japan plans to begin dumping that huge inventory of contaminated water into the Pacific Ocean very soon this year. The Pacific Ocean is at least 30,000 times larger in volume than the Great Lakes. It is daunting to think what would happen if such an enormous amount of radioactive water had to be discharged into the Great Lakes basin, the source of drinking water to 40 million people.

Nuclear proponents and supporters say that, on the whole, nuclear power is acceptably safe. But no insurance company in the western world believes that the risk of a nuclear accident is acceptable on actuarial grounds. Every homeowner's insurance policy, without fail, contains a nuclear exclusion clause that voids all coverage in the event of radioactive contamination of property or persons due to a nuclear accident.

1.7 Lessons learned from Fukushima applied to BWRX-300

There are so many lessons to be learned. We now know that co-located reactors may be vulnerable to "common cause" events that can trigger severe core damage in several units at once. It doesn't have to be an earthquake or a tsunami.

It could be a fire that disables all the pumps and electrical controls for example. That nearly happened at the Brown's Ferry nuclear plant in Alabama in 1975. The risk of losing complete control of a nuclear reactor in this way is exacerbated by the continued use of flammable insulating material in nuclear power plant electrical systems – materials that are so flammable they can turn a small fire into a raging inferno.

There is no information in the DNNP documentation about the vulnerability of BWRX-300 to electrical fires. Nor is there any information about the electrical insulation material used in that plant, or about its ability to feed a fire once a fire has started. There is also no information about duplication of wiring systems within the BWRX-300 layout, or the degree of separation between those duplicated wires so that the chances of one fire eliminating all electrical circuits vital for safety by burning up all the wires at once, even the duplicated ones, is minimized.

Fukushima shows us that station blackouts can be especially challenging. Radioactivity cannot be shut off. Therefore effective cooling of spent fuel is essential long after the reactor is shut down.

At Fukushima we also witnessed how much damage hydrogen gas explosions can do. We see how important it is not to underestimate the amount of hydrogen or miscalculate the risk of detonation. A severe nuclear accident always gives rise to hydrogen gas formation in a water-cooled reactor, because hot metals will react with hot steam, stealing the oxygen atoms out of the water molecules and releasing the hydrogen gas into the air.

In Annex C of this report, entitled "Unmet Challenges to Successfully Mitigating Severe Accidents in Multi-Unit CANDU Reactors", Dr. Sunil Nijhawan goes through a litany of examples of how things can go wrong in a multi-unit plant like the Darlington Nuclear Generating Station. Among other things, he discusses the frequent miscalculation of the amount of hydrogen gas buildup in a damaged CANDU reactor core, and the subsequent risk of explosion, which increases the potential radioactive releases from the plant and which serves to increase the area of the exclusion zone – assuming we use the scientific approach laid out by the US Nuclear regulatory Commission, as spelled out in Annex B of this report, instead of the OPG and CNSC practice if simply drawing perfect circles of an arbitrary radius and calling it an exclusion zone.

This entire discussion of CANDU safety would be beside the point and would have no bearing in the siting of the BWRX-300 reactor, were it not for the fact that OPG wants to put the new reactor smack dab inside the exclusion zone of the Darlington multi-unit nuclear power plant.

Of course, when the four CANDU reactors were first built, they were all built within the exclusion zones of each other. However, during the construction period (which began at Darlington in 1981 and finished in 1993) most of the work was done when none of the reactors were operating, The first unit startup was in 1990, so there was less than 3 years of working in the shadow of an operating reactor.

But this was long before the Fukushima experience. We now know better. Fukushima taught us to treat nuclear reactor disasters with respect and not dismiss them as inconsequential because they are unlikely. Knowing what we know now, it would be wrong to allow thousands of workers to labour within the exclusion zones of operating nuclear reactors. Those day are gone.

If the currently chosen site for the BWRX-300 were adopted – and OPG is diligently working on that site right now, even as we speak – the workers would be labouring not only within the exclusion zone of a 3500 megawatt nuclear power complex – one of the largest in North America – but also within a stone's throw of spent fuel in dry storage casks stored in warehouses quite close to the construction site.

The amount of radioactive material inside these spent fuel facilities equals or exceeds the amount inside the cores of the four reactors, because the waste warehouses will accommodate years and years of used fuel bundles that have been accumulating for a long time. A disaster that liberated the radioactive poisons from those containers would constitute a grave threat. Yet OPG and CNSC do not bother to even include them as a "blip" in their risk perception radar, for they do not ascribe any exclusion zone to the spent fuel itself. Only to the reactors.

The lessons of Fukushima are not limited to the physical domain. The breach of trust, the sense of betrayal, can be felt so deeply that it amounts to a rending of the social fabric. In Japan, the greatest sorrow was not related only to the nuclear mishap, enormous as that grief was, but to the fact that people felt they had been lied to by people they trusted. Scientists had repeatedly assured them that nuclear power is safe, safe, safe, and they were stunned and shocked to learn that this was a complete falsehood. A betrayal. How can one learn to trust such people ever again?

What caused the Fukushima nuclear catastrophe? Most people blame the tsunami. The Commission of Investigation in Japan concluded otherwise. In its report to the National Diet, the Commission found that the root cause was a lack of good governance.

The accident "was the result of collusion between the government, the regulators and TEPCO [the nuclear company], and the lack of governance by said parties. They effectively betrayed the nation's right to be safe from nuclear accidents. Therefore, we conclude that the accident was clearly 'man-made.' We believe that the root causes were the organizational and regulatory systems that supported faulty rationales for decisions and actions…" *[Executive Summary of the Commission report to the National Diet of Japan]*

The Commission chairman wrote: "What must be admitted — very painfully — is that this was a disaster 'made in Japan.' Its fundamental causes are to be found in the ingrained conventions of Japanese culture: our reflexive obedience; our reluctance to question authority; our devotion to 'sticking with the program'; our groupism; and our insularity... Nuclear power became an unstoppable force, immune to scrutiny by civil society. Its regulation was entrusted to the same government bureaucracy responsible for its promotion."

Canada has not heeded these warnings. After Justin Trudeau was elected in 2015, his government did away with environmental assessments for any new reactors below a certain size, thus eliminating – or at least sharply limiting – scrutiny by civil society. This leaves all decision-making in the hands of the Canadian Nuclear Safety Commission (CNSC). CNSC was previously identified by an Expert Review Panel (reporting to the Minister of Environment) as an agency that's already widely regarded as a captured regulator.

The CNSC, mandated to protect the public and the environment, reportedly lobbied government to abolish full impact assessments for most "small modular nuclear reactors" (SMNRs). The government of Canada complied. That's why there is no full impact assessment for the BWRX-300 reactor today. And that's why the regulator has cobbled together this charade of allowing OPG to spruce up its PPE and rewrite its EIS of 15 years ago so as to pretend that the public is not being deprived of a genuine opportunity to speak up on behalf of the public interest.

Apparently the Canadian Nuclear Safety Commission feels that it has a more mature appreciation of the public interest than most. In the Globe and Mail, journalist Shawn McCarthy wrote: "The CNSC encourages the government to exempt small modular reactors from the list of designated projects that would receive a full [environmental assessment] panel review, and warns that lengthy regulatory delays could kill a promising industry" Who knew that an "independent regulator" would be so dedicated to the well-being of the industry it is mandated to regulate? Who knew that regulatory delays would be so galling to the regulator? Could it be because CNSC receives most of its operating budget from the licensees? Or has the CNSC adopted a higher purpose, more appealing than the one parliament deigned to give to it?

During the 17 days of Environmental Assessment hearings, held from March 21 to April 8, 2011, many intervenors raised the Fukushima accident in their testimony to the Joint Review Panel. In their EA Report, the JRP mentioned the Fukushima accident 19 times. Here are some examples:

"Participants explained that they felt that the OPG safety analysis was probabilistic and not deterministic or realistic enough. They felt that worst-case beyond design basis accidents were not fully considered, despite the fact that nuclear accidents can and do happen, such as at Three-Mile Island (1979), Chernobyl (1986) and Fukushima Daiichi (2011). Participants noted that accidents could be caused by a combination of factors, including human error, severe weather, equipment failure and improper design. Participants felt that even if the probability of an accident is low, the consequences would be unacceptable should one occur."

"The Panel … notes that the Long-Term Energy Plan and Supply Mix Directive were developed before the Fukushima Daiichi nuclear accident. Since this accident, more concerns have been raised about nuclear power generation globally…. The Panel wishes to acknowledge the desire expressed by many participants for a re-examination of the Ontario energy alignment."

The people of Ontario, indeed the people of Canada and the world, deserve to have an independent and thorough Environmental Assessment of this new, untested reactor, the BWRX-300, especially as it is intended to be built within the exclusion zone of a very large nuclear power complex, not far from major rail line and highway linking Toronto to Montreal, and within a relatively short distance (as the crow flies) from one of Canada's largest cities and most important manufacturing centres.

The Canadian Coalition for Nuclear Responsibility is confident that an independent environmental impact review would conclude that the proposed siting of this proposed reactor is quite simply wrong.

PART 2: A REVIEW OF THE ENVIRONMENTAL IMPACT STATEMENT AND PLANT PARAMETER ENVELOPE FOR ONTARIO POWER GENERATION'S DARLINGTON NEW NUCLEAR PROJECT.

Canadian Coalition for Nuclear Responsibility

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Summary of findings and review

Siting a new reactor within the exclusion area boundary (EAB)of an operating reactor that has significant and unresolved vulnerabilities to severe core damage is contrary to safety principles, regulatory requirements as well as requisite stakeholder obligations to worker security and public safety. Even the currently accepted US NRC inspired 1000 yard (914m) EAB on the existing nuclear installations will not meet the U.S 10CFR100.11 requirements, and public expectations to consider source term from a severe accident core melt. Therefore, the indicated EAB for BWRX-300 of 350m is incredulous as is its full inclusion within the 40 year old EAB for the Darlington plant and within the EAB for the dry storage shed structures housing > 130 reactor core loads of spent fuel in concrete casks with > 70% of volatile, high dose sensitive fission product species still active. The CNSC needs to exercise due diligence in requiring OPG to resolve existing critical safety issues with Darlington CANDU severe accident mitigation before an event causes all reactors on site to have to be abandoned. Mere siting of BWRx-300 next to sheds containing hundreds of reactor years of spent fuel and 50 odd meter separation of its switchyard from an operating public railway line is another step in the Ontario Power Generation sleep walk towards an impending disaster at Darlington. GE-H should reconsider this siting plan for their own corporate interests. Their design is neither small, nor so modular as a first of kind construction and requires a comprehensive Environmental Assessment to protect Canadian public interests, previous actions in this regard notwithstanding.

The Plant Parameter Envelope document sent for review is grossly incomplete; has seen no substantive OPG additions in 12 years; and the US 10CFR52 process of site qualification it mimics is of no relevance today as a vendor GE-Hitachi and their BWRX-300 design has already been selected by OPG. Once a new site for the reactor is identified, a detailed reactor design information binder will help qualify that site for the chosen design. There has to be an independent technical review of vendor GE-Hitachi claims of enhanced safety in their BWRX-300 design with access to their actual safety reports with detailed analytical assumptions, code descriptions and accident simulation results with numerical information they have been unable to reveal so far.

Contents

1. REVIEW OF PROPOSED SITING OF OPG'S NEW NUCLEAR PLANT

In the U.S, a principal policy objective of 10 CFR Part 52 in the development of a Plant Parameter Envelope (PPE) process reflects the NRC objective to decouple siting from design for early resolution of safety and environmental issues. The driving force behind this process is that siting issues are to be resolved with the same finality as was done under 10 CFR part 50 for a specific design. This means that siting decisions must meet the regulatory requirements for any of the designs considered within the Plant Parameter approach allowed for some under 10CFR52. Canada has signed onto that process in its CNSC memorandum of understanding with the US NRC.

Siting criteria for new builds the world over now consider core melt down as clearly stated, for example for the US in 10CFR100.11 (see a listing in Appendix 2) . As an example, see the exclusion area boundary for ESBWR at North Anna for Dominion Energy below. It extends upto 1.7 km which is about double of the 0.914 km widely accepted before Fukushima and was developed on consideration of the whole range of releases, including likely from a severe core damage accident.

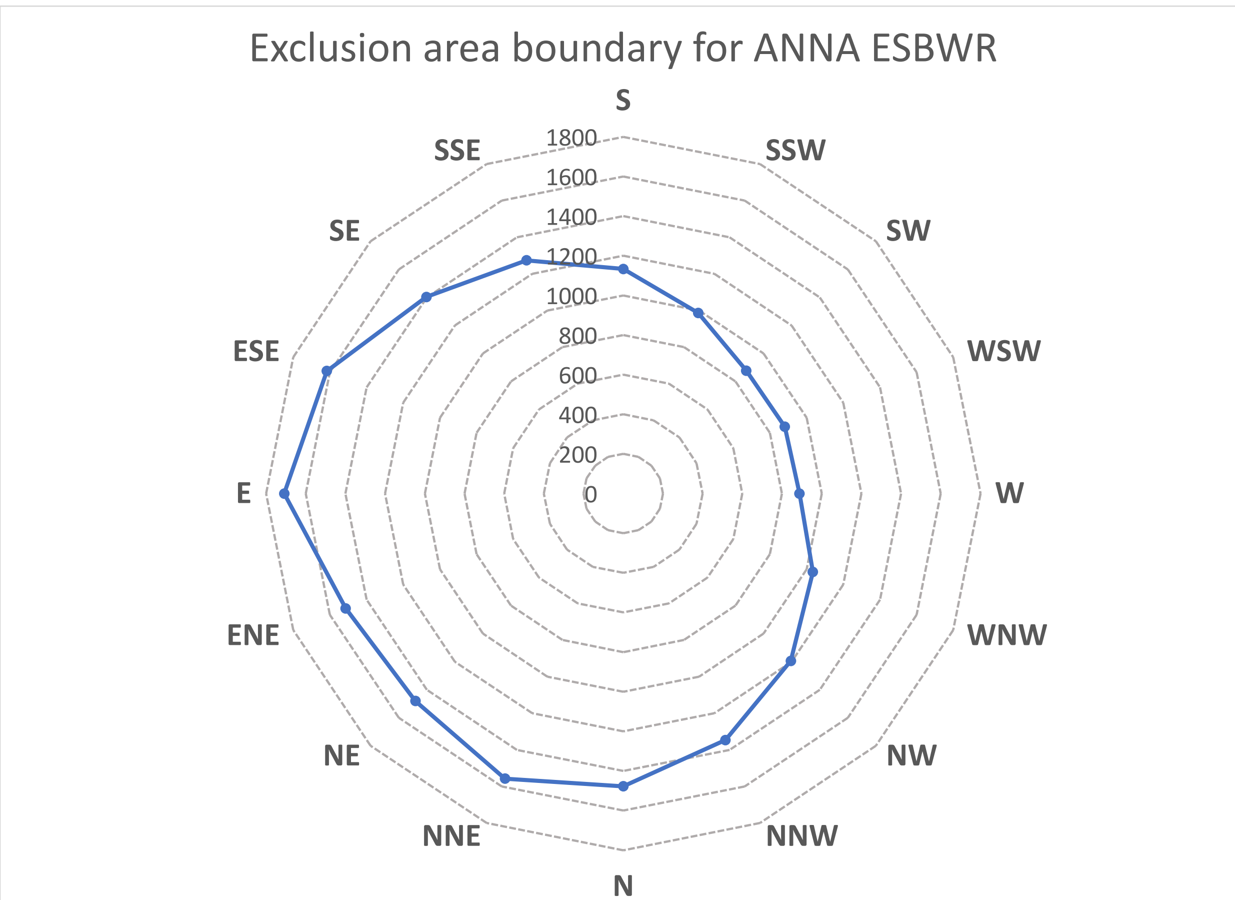


Figure 1 : Exclusion Area Boundary for a modern Nuclear Power Plant is double of what was for older power plants pre Chernobyl.

1.1 OPG's new nuclear plant siting proposal at existing lands they own for DNGS

OPG proposed in 2007 that they plan to build a new nuclear station within the lands they own in Bowmansville, part of which were used for DNGS-A. Several reactor designs were considered, and a PPE process was undertaken. The exact location of the new station was not revealed initially but a site preparation license and funding was approved over 10 years ago.

It seems now that a site preparation license was issued to OPG for their new build without consideration of a number of issues, some of which significantly affect their MOU for harmonization of licensing issues with NRC that only occurred a decade later.

1.2 DNGS exclusion area boundary

The original exclusion area boundary (EAB) for Darlington reactors was designated long before TMI accident in 1979 and was based without an understanding of nuclear safety principles and proper risk assessments that the 3 severe core damage accidents in 15000 reactor years of experience have taught us. We now look at severe core damage as credible and responsible regulatory bodies have enshrined their implications in reactor designs, siting, safety assessments and emergency planning.

The Darlington 1000 yard (914m) wide exclusion zone taken from end of the reactor building footprints was a copy of what the US NRC had for their older reactors and the Canadian utilities were able to show to AECB then that it was sufficient for their benign operational release estimates compared to the so called Derived Emission Limits from operational releases and from extensive and conservatively stylized 'design basis' accidents they did consider. The original 1978 Exclusion area boundary was based on the plan of building two identical 4-unit CANDU stations DNGS-A and DNGS-B. The planned Station B was slowly abandoned and the designated space used for spent fuel dry storage that was not envisioned in the original design of the two reactor station complex, as a permanent storage solution for fast accumulating after ~10 years in the spent fuel pool wet storage was never found. AECL and OPG scrambled then to design dry storage solutions.

As Darlington B idea was abandoned over the years, the space that was to be used for that 'B' station was used for an extensive field of dry storage industrial buildings. The exclusion zone remained the same, (Figure 3) as was illustrated in the Darlington 2012 safety report and perhaps even later.

The existing exclusion zone included spent fuel cask storage buildings that contain about 30 years' worth of spent CANDU nuclear fuel with about 2000 concrete casks, each containing 384 spent fuel bundles (6.15% of a full core charge of fuel bundles in each cask). These large buildings now contain about full 120 reactor loads of spent dry fuel. While the decay heat has waned over the period to about 4kW per cask, there is still over 70% of the original volatile inventory of long lived fission product species. There are significant safety and security issues associated with this accumulating dry fuel casks and the safety and security problem is no different than with spent fuel from other reactor types in other countries. With such large quantities of the highly volatile and of amongst many other fission products, of high dose conversion factor like element - Cs-137, the exclusion zone should have been even bigger than for an operating station, but the existing EAB of 1978 and 2012 in Figure 2 and Figure 3 was deemed acceptable over this period.

1.3 Redrawing of EAB for Darlington Nuclear Generating Station and its Dry Fuel Storage complex

Then suddenly for us who follow developments in the nuclear industry, and inexplicably, in the 2022 application for an extension to operating license for the dry storage facility the exclusion zone for Darlington station was redrawn as in **Figure 4**. The 2023 public edition of the BWRX safety report, with area demarked for its operations in yellow mimicked the same diminished exclusion zone for Darlington CANDU reactors as in Figure 5. No exclusion area was added for the large dry spent fuel cask storage structures.

It is proposed now by OPG that this yellow area that overlaps even the diminished EAB for DNGS be set aside for BWRX-300.

1.4 Disturbing information about the siting of first BWRX

As details emerged from what we typically saw as only highly redacted versions of BWRX-300 documents, and some rough sketches of BWRX-300 buildings were seen, it became apparent that there were these interesting characteristics of that site:

1. An active CN rail line bisected the lands designated now for the proposed new build (Figure 5)
2. The siting of the first unit was smack in the middle of the exclusion area for only the Darlington station that we had known for over 40 years (Figure 2 and Figure 3).
3. The siting of the first BWRX unit was next to the dry storage buildings and included parts of even the now reduced 914m EAB around DNGS-A.
4. The switchyard for the BWRX unit was a meter 58m from the active CN railway line.

1.5 EAB for first unit of BWRX

The newly released redacted publicly available safety report for BWRX claimed a exclusion area boundary radius of 350m. That seems to be an arbitrary number as no source term for any accident within the design basis or beyond design basis was provided in the PPE. Given our experience analysing reactor accidents at various pedigrees of nuclear power plants, it the following cam up quickly:

1. BWRX vendor seems to claim immunity from severe accidents.
2. The regulatory expectations currently respected world over are ignored.
3. Siting next to a huge field of spent fuel casks with enormous amount of fission products, safety and security concerns, seems odd.
4. Siting within the exclusion boundary of an operating station seems to have been done with real estate considerations only; not for safety of and from a nuclear reactor..
5. There was no consideration of risk profile of the two neighbourhood nuclear installations - the 4 reactor units of DNGS-A whose vulnerabilities we had examined for years; and the expansive spent fuel field in metal buildings not much stronger than a Costco warehouse.

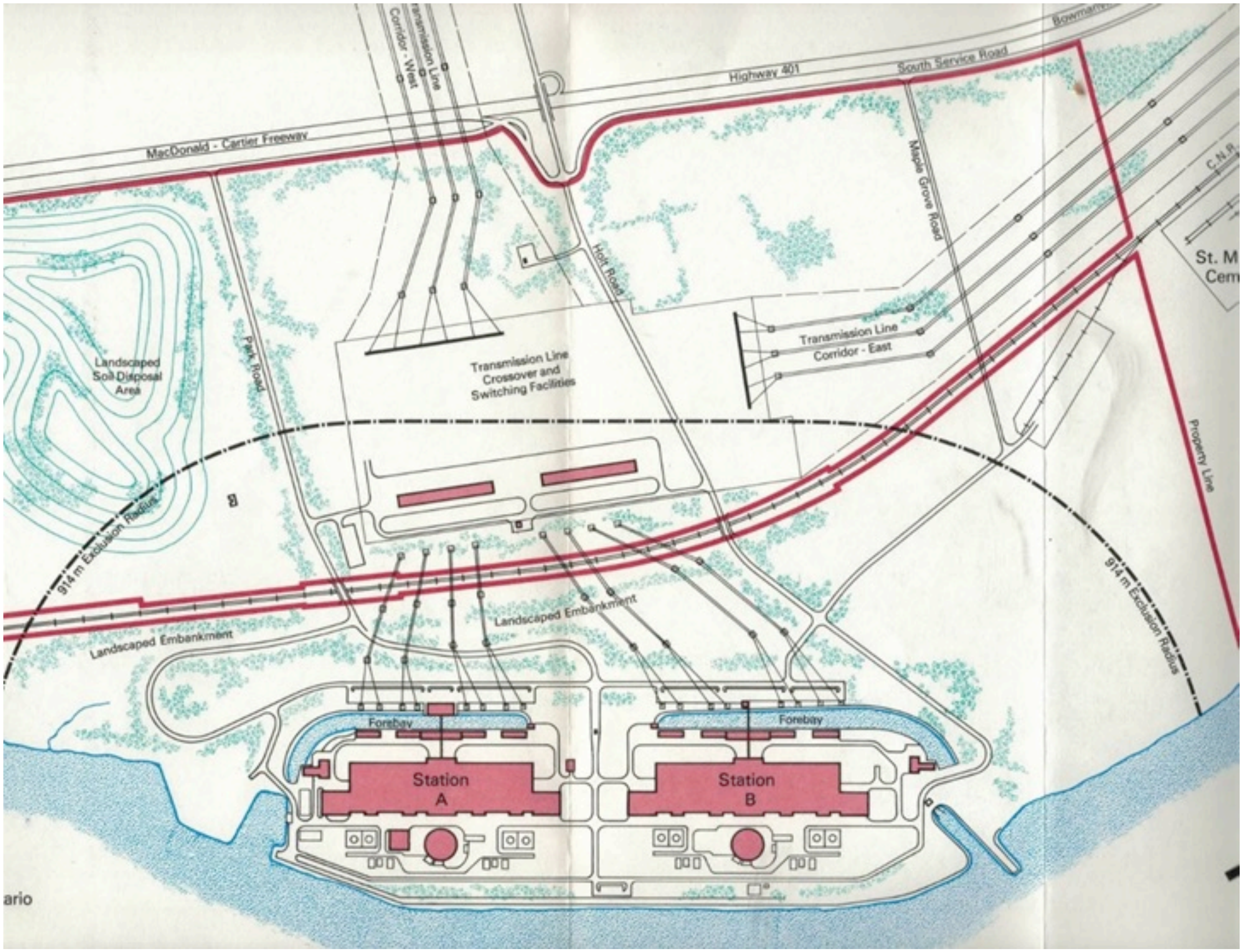


Figure 2 : The 1978 Exclusion area boundary based on two 4 unit stations. Station B space now taken by dry storage.

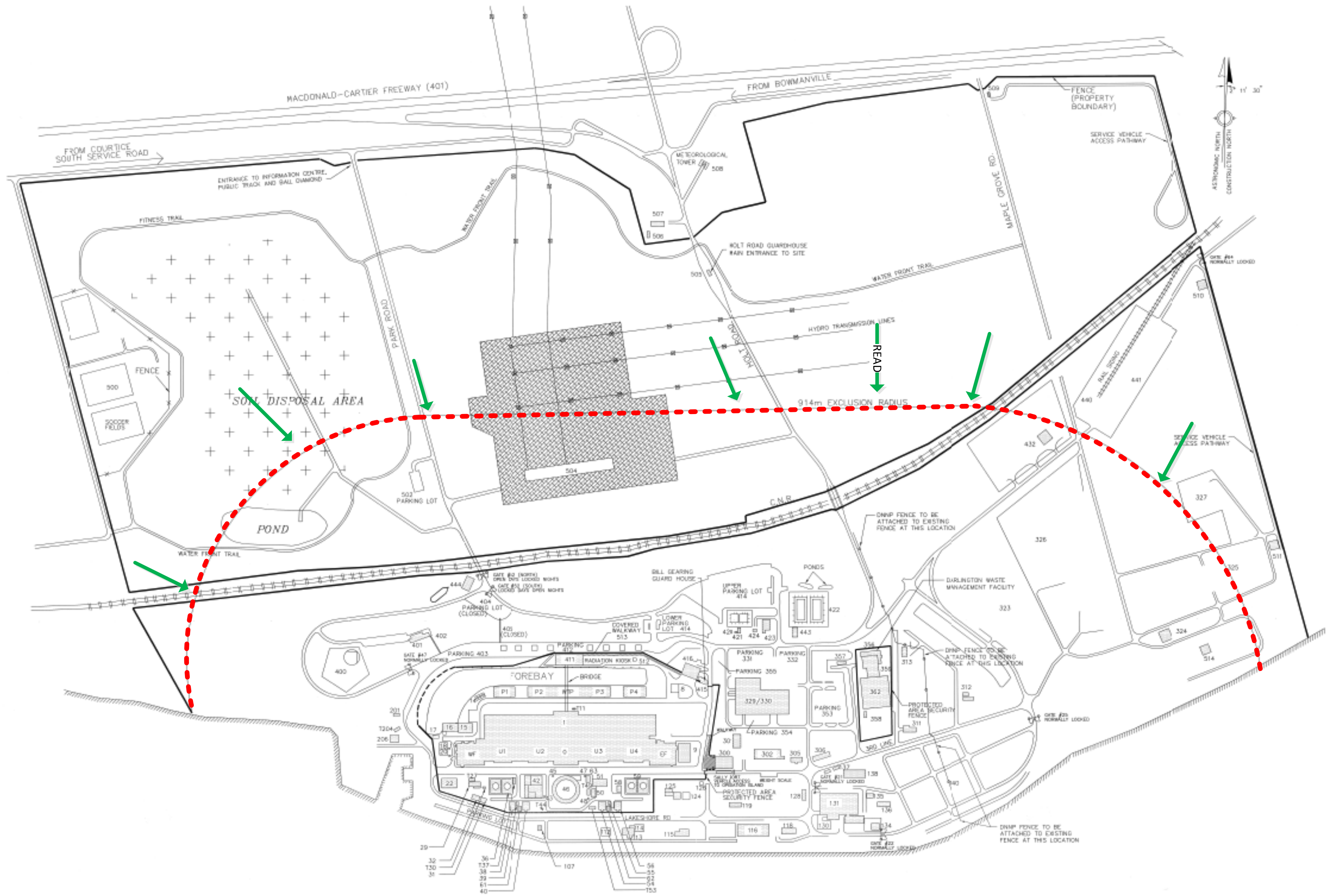


Figure 3 : DNGS Exclusion zone remained the same for at least 34 years as again so represented by OPG in 2012 Darlington Safety Report.

Figure 4: DNGS exclusion Area Boundary in 2022 Application for Waste Management Facility License Extension

Figure 5 **Redefined DNGS Exclusion Area Boundary from NEDO-33952**

Figure 6 : DNNP stated Exclusion Area Boundary of 350m!

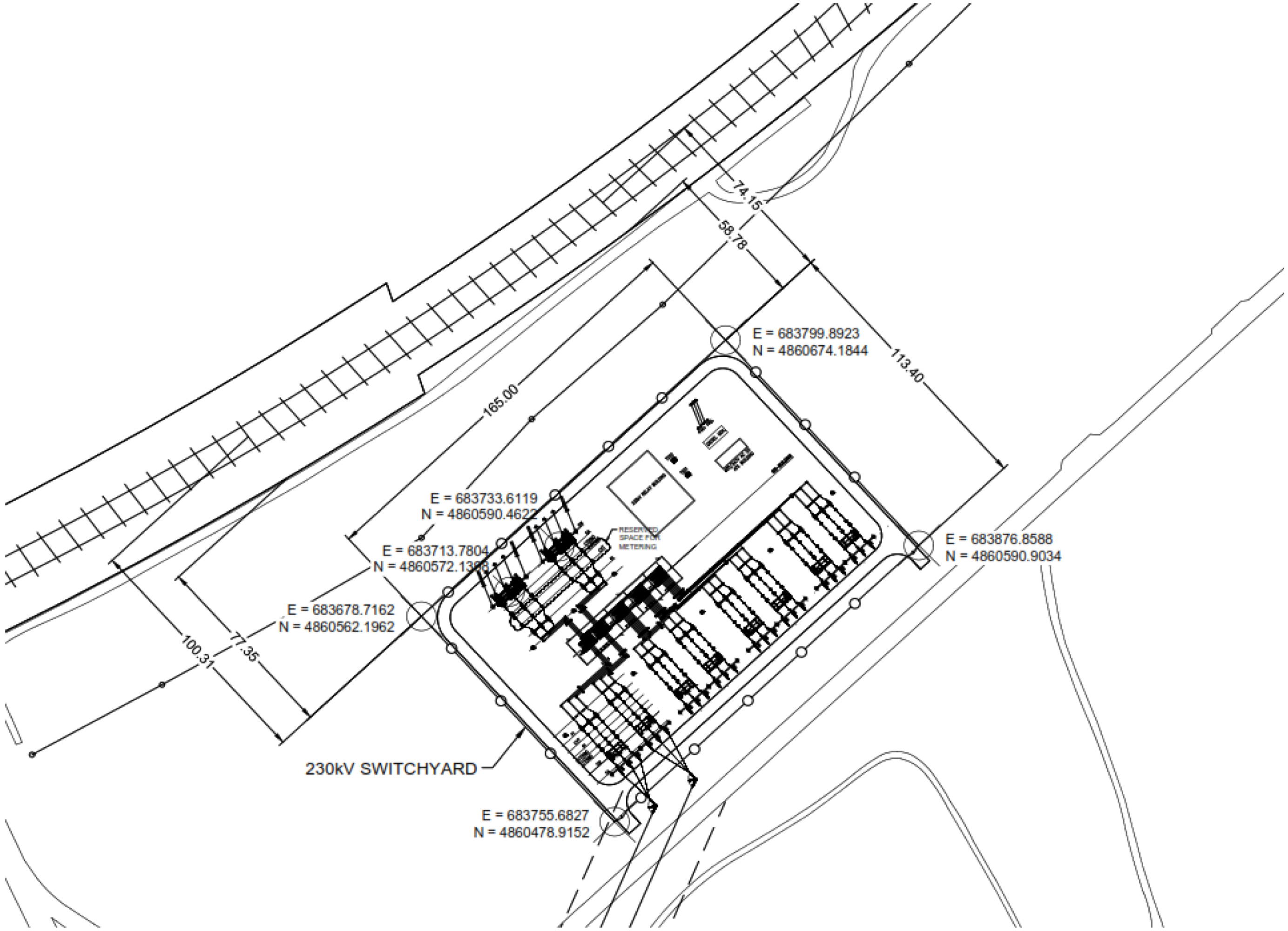


Figure A1.4-2: DNNP Switchyard Site Plan

Figure 7: BWRX- switchyard is 58 meters from railway line. My house is at a greater distance from a line and shakes when trains go by relatively slowly.

2. Review of the siting proposal

After careful consideration and mindful of the minor cost associated with abandoning the proposed site for BWRX compared to the trillion dollar and hundreds of sq. km land loss consequence similar to what Fukushima is going to cost the Japanese society, we have strong and compelling reasons to recommend that the chosen site for BWRX be abandoned in favour of one that actually meets the nation's long stated requirements for safety of and security from a new build nuclear plant. Nobody should be allowed to manufacture alternate facts, especially for preordained commercial decisions that can have such profound effect on the nation's destiny. Not a wayward federal regulatory body or a misguided provincial utility, for sure.

2.1 Rationale for the recommendation that the present DNGS site is wrong for the purpose and dangerous.

As our understanding of reactor accidents and their unfathomable consequences has become clearer, both the regulatory requirements, reactor designs and their containments became sturdier. Even ESBWR, the so called safest incarnation of a BWR has an exclusion zone of over 1700m at North Anna for Dominion Enetrgy.

Then came TMI and the world woke up to a new reality and the US laws were toughened to include a requirement that the dose to a person at the boundary of the exclusion area after a core meltdown must be considered (10.CFR. 100.11). In Canada we kept dreaming up excuses and never did a comprehensive severe accident progression and consequence assessment. Exclusion zones remained the same. When an idiot doing experiments 1m 1985 with low power operation at Chernobyl blew the roof off after a reactivity induced power excursion there, the Canadian submissions to Hare Commission in Ontario boasted of a Pickering containment that would just crack to relieve the pressure and then just closeup even if a reactivity induced accident did happen at Pickering A that had only a single shutdown system. None of us ever saw a wakeup call in the slow pressurization induced, year 2001 blowing up of a 150 million dollar quarter scale containment in integrity experiment at Sandia that created this image:

Figure 8 : Containments will not crack and reclose; they will burst. Results of SANDIA joint Japanese-U.S tests.

Off-site consequences of a containment failure for various fractions of fission product inventories show that the only hope of meeting the regulatory limits is to have a containment that is robust or a reactor design that is low risk. Here in Figure 10is a simple parametric analysis of doses at various distances from the reactor from releases of just Nobles gases, Iodine-131 and Tritium. Longer term releases from a severe accident of Cs-137 etc. not included.

Locating a new nuclear reactor within the now suddenly and artificially narrowed exclusion area boundary of an existing, operating reactor that has long been identified to have an enhanced risk profile and dozens of unresolved design vulnerabilities is not in any one's interest. For GE-H, the existing Darlington reactors can incapacitate their new build any day. This is not an anti-nuclear activist's warning. This is a scream of warning from a Canadian nuclear engineer who loves his country and has examined Darlington design for 40 years, contributed substantially to it's original licensing in 1980s and since then pointed out reactor design vulnerabilities that if left unaddressed will likely cause damage to this nation. Locating BWRX-300 practically next to Darlington station is not a sane decision.

As a Canadian nuclear industry professional I have spent over 40 years analyzing nuclear reactor designs, accident initiators, accident progression and accident consequences that define risk from reactor accidents and documenting the vulnerabilities of our current fleet of CANDU reactors. To me, as it to others with similar background, the proposal to site the BWRX-300 within the exclusion zone of an operating four unit Darlington station (and practically abutting industrial sheds containing over 130 reactor loads of dry spent fuel storage in 2000 unmonitored concrete casks) is an inconceivable act of recklessness by both the vendor GE-H and the host utility OPG. Just the idea that an application to that effect is forthcoming may just be on GE-Hitachi's part, simply in ignorance of the perils in the decision. To me, the CNSC continues to go down the path that the Japanese regulator led the Japanese nation to in its subservient collusion with the Japanese electric utilities – Fukushima. I present information in plain language on why I consider this decision to have far reaching negative and history altering public safety, economic and environmental consequences. I am happy to present detailed technical arguments on Darlington as well and challenge the CNSC and OPG staff to an honest discussion.

My hope is that saner heads will prevail and if nothing, corporate self-interests and a hard cold look at the enhanced economic risk such a path entails – (like in losing the use of the new reactor even before it can go online) - will dissuade going any further with such an irresponsible decision. Ontario has a huge land mass; there likely are a dozen good sites close to the transmission corridors we already have for a safe new nuclear station.

It is irrelevant that the CNSC just this last week sent out a missive stating that they approve of the BWRX-300 design. For some unknown reason they have occupied the mantle of knowing better than us all, while having knowingly pushed such dangerously wrong and technically impossible and socially irresponsible positions on progression of accident (reference 1) and on consequences to environment of severe accidents in the existing reactors (reference 2) and done nothing for 22 years on simple to understand but extremely critical to public safety issues like what an ASME code compliant reactor primary coolant overpressure protection should look like and how the ones at Canadian CANDUs violate those basic tenants of proper engineering. That is either an extreme level of callous disregard for public safety or gross incompetence but certainly contrary to their legislated public duty, irresponsible and shocking (reference 3). This specific design error can cause a pressure boundary rupture that can lead to a core damage accident of potentially inordinate off-site consequences. Therefore, my submission is directed to those in the board room of GE-H and OPG asking them to not go down that path for their own corporate self preservation without addressing the issues that the host reactor presents. Look for another site and then we can have an honest technical review of the merits of BWRX-300 and its environmental impact. I also hope that a CNSC Commissioners may stumble upon this report one day and begin to ask the requisite questions.

I reserve expressing my opinion on BWRX-300 risk profile at this time because that is irrelevant to the task at hand and at this juncture.

The regulatory body staff whose lethargy, 'safety culture' and inactions I amply dissect in this submission, has nothing to lose in this process as their organization's financial interests are in billing for their services to any new project they can find, under their cost recovery arrangement. AECL already spent a billion dollars of public money on reactor designs that the CNSC 'approved' or found no impediments to the licensing of, but got nowhere because others saw deficiencies in those designs that CNSC conveniently overlooked.

It is not private funds that are driving this project. These are public funds. This country certainly cannot afford the negative consequences associated with the proposed site for BWRX–300 as well as the process being followed in likely granting OPG a construction license as soon as possible, bypassing the processes of the past by now using this Plant Parameter Envelope (PPE) process that has no meaning since a reactor design was decided upon years ago (see my comments in next section 3 on page 22 on the Darlington site PPE report).

I begin by pointing out that that the lethargy and intransigence that the CNSC staff have exhibited in rectifying known errors and inadequacies in the operating CANDU reactors at Darlington post Fukushima, contribute strongly to my premise that BWRX-300 cannot be built at the proposed location. I have detailed the underlying issues on the risk profile of the 'host' power plant in the technical paper attached as Appendix 1 and will use some examples in this section to demonstrate why the risk to environment and public is unacceptably high of putting in upto 4200 construction workers on the site for 4-5 years. This first begs the question – why we need permission for that many workers there if this site is really for a small, modular reactor?.

An approval of siting of the any new nuclear power plants within the exclusion area boundary of another operating nuclear station is contrary to modern safety principles and understandings that govern licensing and operation of such facilities. See for example see – *US Code of Fedetral regulations 10 CFR 11 Determination of exclusion area, low population zone, and population center distance*. While the CANDU design in building sister units was acceptable and met the knowledge base, safety culture and regulatory expectations of 40-50 years ago but was found later to be severely wanting in its provision of a safe, low risk, operating envelope today, it cannot host a new reactor next to it without transferring its inherent risk to the new addition and to the construction workers.

Siting a BWRX-300 within the boundaries of a multi-unit CANDU station cannot be permissible until the so glaring vulnerabilities in the CANDU station's operation are resolved completely. Such a siting is also in direct conflict with public expectations of minimal risk as we no longer describe severe core damage accidents as 'hypothetical' and better understand certain specifics of design of our operating reactors that we recognize today as vulnerabilities that, if not mitigated properly and honestly, can cause severe damage to this nation.

We have summarized what we understand today as errors in design of Darlington CANDU station and shortcomings in improvements made over the years in a number of papers and reports. A number of these errors/vulnerabilities can come into play not just in a severe core damage accident but in a design basis, higher probability accident – like a simple rupture of a feedwater inlet pipe due to wall thinning, something that has occurred at a dozen power plants already, and following an uncontrolled over-pressurization of the primary coolant circuit due to improperly sized safety relief valves, cause a rupture of boilers tubes. Resulting release of radioactivity into the atmosphere will incapacitate and render useless a new reactor being built within its exclusion area boundary and affect health and well-being of workers on site. Another example is the mindless placement of the pressurizers in all units at Darlington.

One of the reports that details the perils of the design errors that we have identified is reproduced in the appendices.. We have also included in that report the blame that the regulatory body CNSC must be assigned in promoting some false narratives and accepting some very bizarre and questionable submissions from CANDU utilities post Fukushima as well.

Darlington station with four interconnected reactor units was designed in mid 1970s. After living through three severe core damage accidents at TMI, Chernobyl and Fukushima since that time, we have had an opportunity to subject our designs to intense scrutiny using increasingly sophisticated analytical methods. Our investigations were supported by research in various allied sciences, technologies and industries world-over. Today we do not look at our Darlington reactors designed in the era of rotary phones and slide rules as 'state of the art' as many decisions were made by designers who were handicapped by what was at their disposal. Today we can do better.

When we look back at the conclusion drawn by investigation commissions into root causes of Fukushima, we know that the Japanese regulatory bodies played a pivotal role in causing that trillion dollar fiasco. They were too eager to accept anything that the utilities wanted. In Canada we are following the same path. We have so far not learnt the Fukushima lessons, but must do better. <u>The new BWRX-300 reactor that is being proposed may very well be one that best minimizes risk, but the Darlington site is the wrong place to build it on many counts</u>. In addition, the process of ramming its acceptance through, not least of which is the utter lack of any numerical information under guise of being 'proprietary' is looking very suspect. Just go and look at the volumes of information that was made public when new Canadian reactor designs like CANDU-1250, CANDU 950, ACR-1000, ACR-700 and CANDU-3 were conceptualized, designed, revealed and marketed or how much information on our existing reactors is in public domain.

Today as we have an improved understanding of risks associated with operation of nuclear power plants of any pedigree, it is apparent that siting the BWRX or any other new reactor next to the exiting Darlington units and within its exclusion boundary is also not in the economic interest of the utility or the vendor. There are many very obvious reasons. I will try to allude to a few here and will be happy to brief you all once again on the long outstanding issue of the design vulnerabilities of the design of operating reactor units at Darlington which I have summarized in Appendix 1. There is a great chance of the Darlington A reactors causing accidental releases which will definitely jeopardize construction, commissioning and operation of the proposed BWRX-300 reactor(s).

I reserve for now any judgment on veracity of claims of near absolute safety claims by GE-H of its ESBWR and BWRX-300 as that discussion is moot, as I present to you the dangers and risk within the original exclusion area boundary of DNGS ( ) which in its 1978 safety report was drawn at the north

American standard 1000 yard (814m) distance from the reactor boundaries and envisioned another identical station with its four 800 MW CANDU units.

As nuclear engineers our generation has lived through 3 severe core damage accidents that hit pretty close to home for me - at Three Mile Island in 1979 (a mere hundred miles from where I was finishing my PhD), Chernobyl in 1985 (in the country where I got my first masters degree) and Fukushima in 2011 (in a country where my God daughters Sumiko and Noriko live). Economic damage to these three countries from these 3 severe core damage accidents is in trillions of dollars and the cleanup and retributions are still ongoing. I specialize in evaluation of consequences of severe core damage accidents and have developed a half dozen computer codes to that effect. Industry uses my codes but manipulates the results that make no sense to me but presents a rosy picture of the reactors they operate.

2.2 Range of Consequences of a loss of fuel cooling

Adverse health consequences of fission product releases entering the public domain with or without a core damage accident at the subject nuclear station are not debatable. Just look in Figure 9 at the definition of various dose magnitudes, our AECB/CNSC definitions of dose classes, and how they sit with respect to probabilities of a resulting fatality. Then, please look at Figure 10.

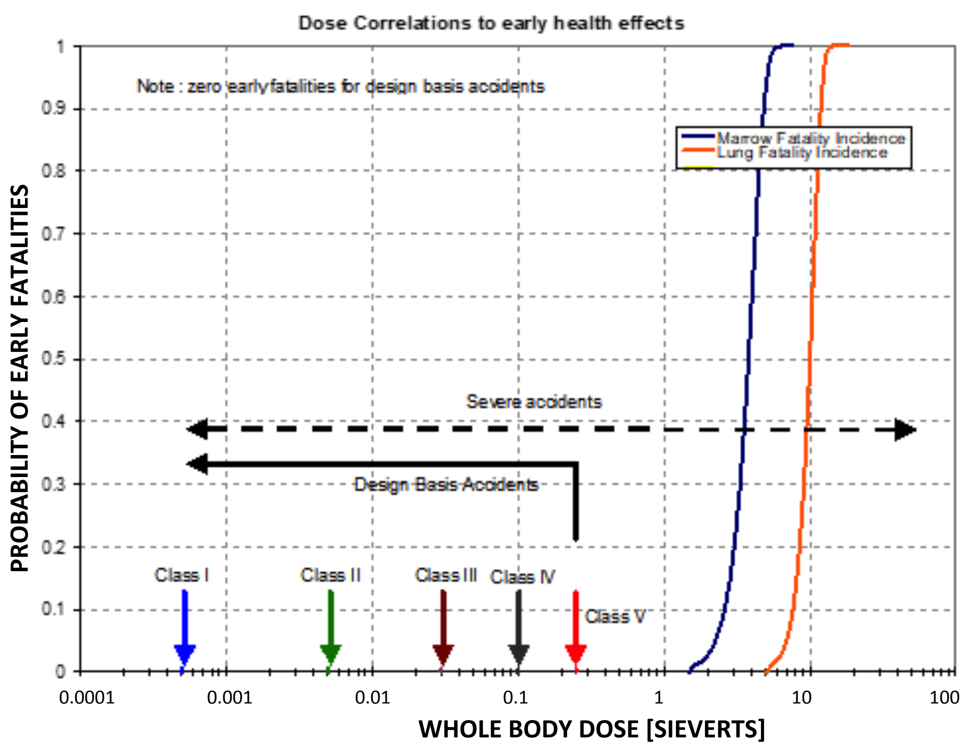


Figure 9: Canadian dose classes and probability of an earlier fatality

Here is a simple exercise. Even loss of primary fluid inventory with maximum permissible operational levels of Tritium, I-131 and Noble gases into the atmosphere can cause high doses 1.0 km away and just short of what can cause immediate fatalities. Such an event is possible if an unmitigated overpressure would cause some boiler tubes to rupture and the primary fluid is discharged into the atmosphere. At Darlington the overpressure protection – on of the most elementary things to design at a power plant - is a at least 25 times smaller than required. At Pickering it is 1000 times too small. CNSC staff know it and have done nothing to ask the utilities, who themselves were told about it a dozen times, to fix it.

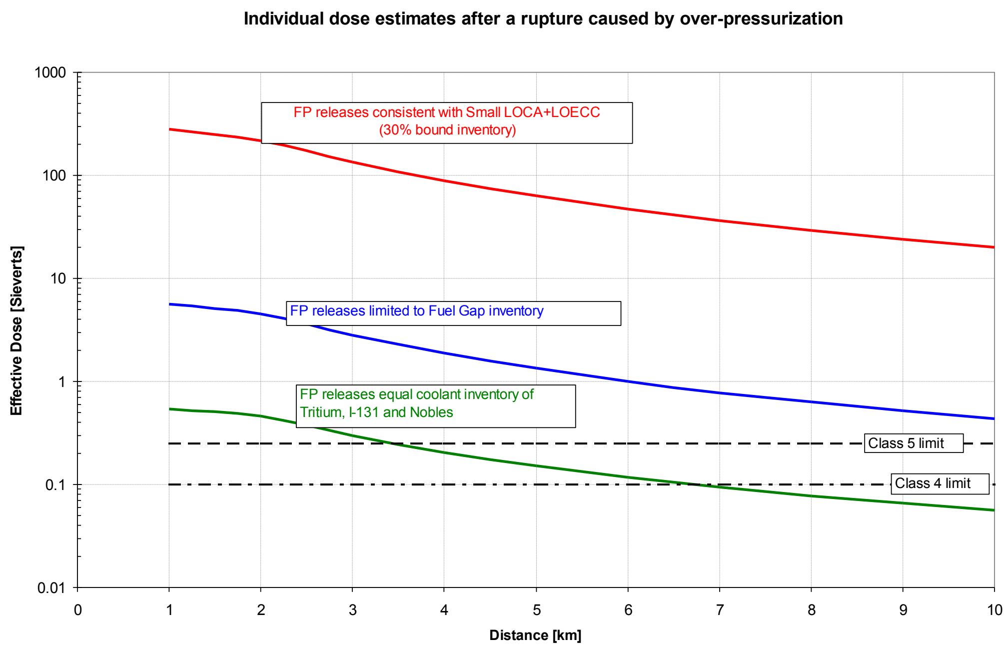


Figure 10 : Sample dose calculations using PEAR

Then look at doses from just 30% of bound inventory along with Iodines and Noble gases and look at the probability of fatalities at distances upto 10 km. A design basis accident, an unmitigated LOCA+LOECC will not only under certain circumstances causes such fission product releases but will also cause explosive amounts of hydrogen (mostly Deuterium really) accumulation in Darlington containment. Utility analysts do not include that in safety reports because the accident consequences are calculated only for an hour and they conveniently ignored hydrogen production by oxidation of our carbon steel feeders. I quantified that source of the highly combustible gas and demonstrated by extensive analyses that oxidation of feeders whose surface area for oxidation by steam alone is over $2000m^2$, is more energetic and faster and starts at a lower temperature than for Zircaloy. Thus, feeders become major sources of explosive hydrogen. We have been asking the industry to accept that reality for ages but the Trumpian handling of truth at the management level has been a big roadblock. No effective hydrogen mitigation systems have been out into place and we still continue to rebuild the Darlingtom reactors with low Chromium carbon steel feeders. So the hydrogen source term has not been reduced and the reactor will create a LOCA just because on a loss of heat sinks, the inadequate in their steam relief capacity – relief valves will let the overpressure lead to a LOCA and on a loss of ECC to inject into the over pressurized cooling circuit, the hydrogen generated by iron feeders and Zircaloy will cause the reactor containment to blow up.

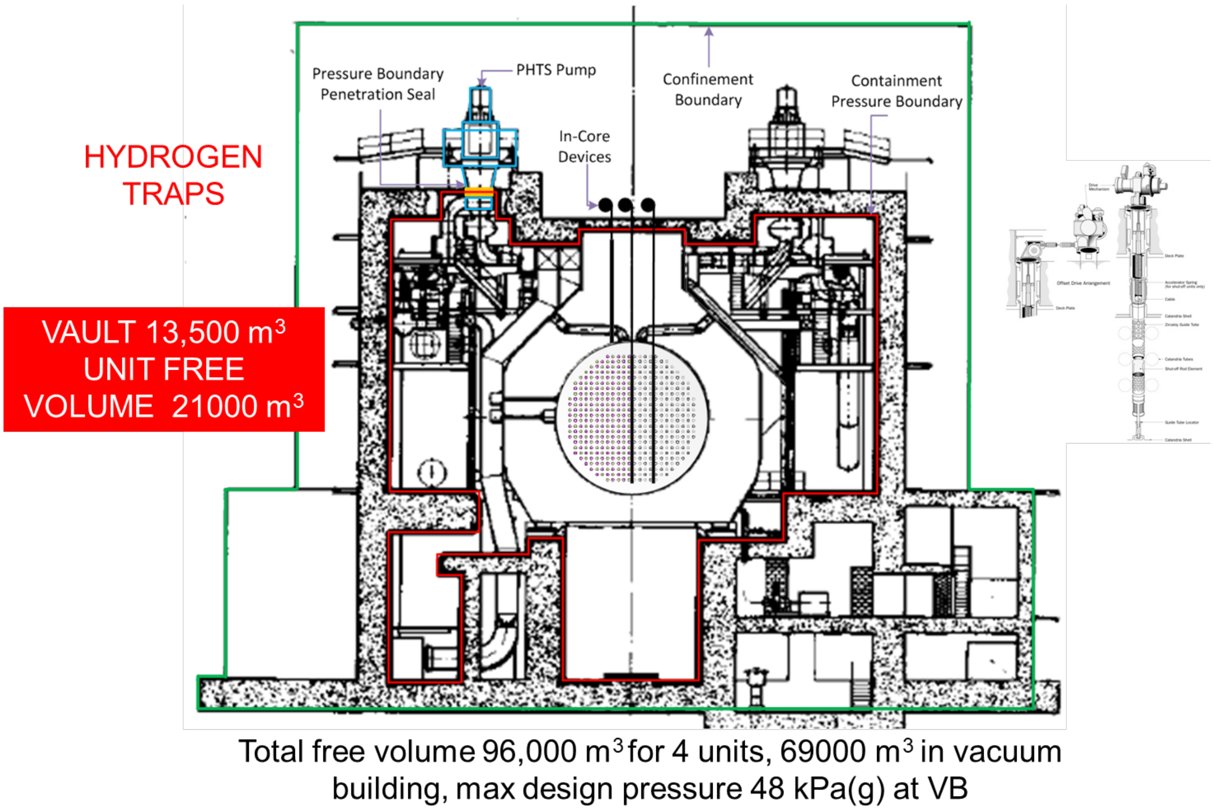


Figure 11: Darlington reactor vault allows no release path for any hydrogen produced in a core damage accident

Explosion of hydrogen in a Darlington containment in early stages of a severe core damage accident is of a great probability because there is little pathway for it's escape from the congested top of the inverted-cup shaped reactor vault where it will accumulate and escapes to, in an accident which we fondly call Limited Core Damage Accident a LOCA with a Loss of ECC. In a severe core damage accident whose hydrogen production will be 10 times more than the 65 kmole that the industry has claimed. No matter what the total amount of hydrogen produced, an accumulation above 4% by volume in air will start a fire and above 8% an explosion. That means a Fukushima like ending. We also point out that Darlington does not have a containment to speak of, once the vacuum building has spent itself. Top of the reactivity decks is an industrial building roof.

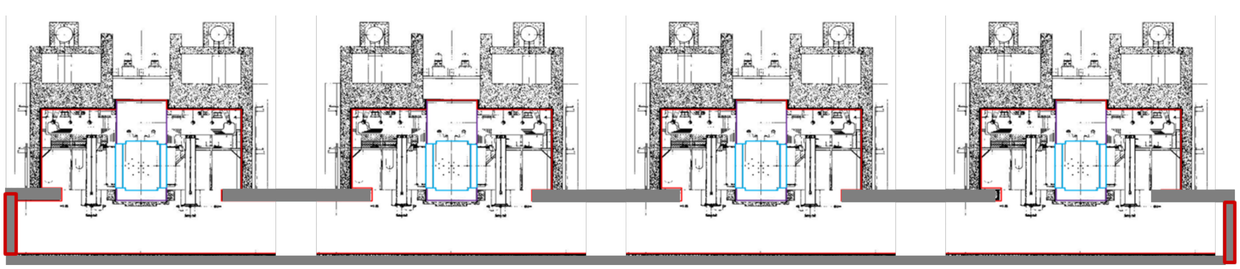


Figure 12 : 4 reactor units interconnected with common duct

This is illustration but a small number of vulnerabilities that can affect the environ around the existing Darlington units. Spending a couple billion dollars each to put in 4 more units there would be fool hardy, unless of course worker and public safety was of no concern in the rush to put in the first of a kind reactors on our soil. Attached. presentation slides and paper include a summary of some of the Darlington station vulnerabilities .

2.3 EXAMPLES OF DARLINGTON MULTI UNIT SEVERE ACCIDENT PROGRESSION & MITIGATION CHALLENGES

The following is a partial list for illustration Please read the whole paper in APPENDIX 1)

2.3.1 Weak and leaky Containment

- Low containment design pressure (<0.9 bar) and high design leakage at design pressure(48% per day)
- Reactivity devices, steam generators, pumps and other equipment critical for long term heat removal are outside the containment and located under an industrial building .
- Containment bypass from over-pressure and thermal creep induced steam generator tube ruptures and from reactivity device failure a likely outcome after a severe core damage.
- Reactor vaults shaped and arranged to be highly likely traps for combustible gases.

2.3.2 Poor Reactor Overpressure Protection Design in a number of systems

- Safety relief valves not directly on the main cooling circuit (ASME section III , NB-7141 (b) requires a direct and unobstructed relief path) and require another pair of downstream valves to open. All valves designed for liquid relief.
- Only two safety relief valves (called 50% capacity valves but the 'capacity' is misrepresented) - contravenes single failure criteria
- Undersized over pressure protection with steam relief capacity of the 2 safety relief valves by a factor of upto 10 - contravenes common sense - relief capacity must exceed anticipated loads, which will always exceed decay heat.
- Inadequate primary cooling circuit relief inherently forces reactor damage by uncontrolled over-pressurization even before an ECC is given a chance to avoid severe core damage. An uncontrolled relief through a rupture in pressure boundary is an unacceptable outcome.
- Accelerated depletion of moderator inventory due to expulsion through pressurized Calandria rupture disks upon channel voiding and fuel heatup to cause moderator boiling.

2.3.3 Poor Pressure and inventory control

- No provisions for direct manual depressurization of the Primary Heat transport system.
- Pressurizer located well below the core can drain water from primary coolant system upon cooling upon loss of power and inhibit thermos-syphoning flows.

• No systems for high pressure ECC or any emergency measures for high pressure primary makeup intervention / injection.

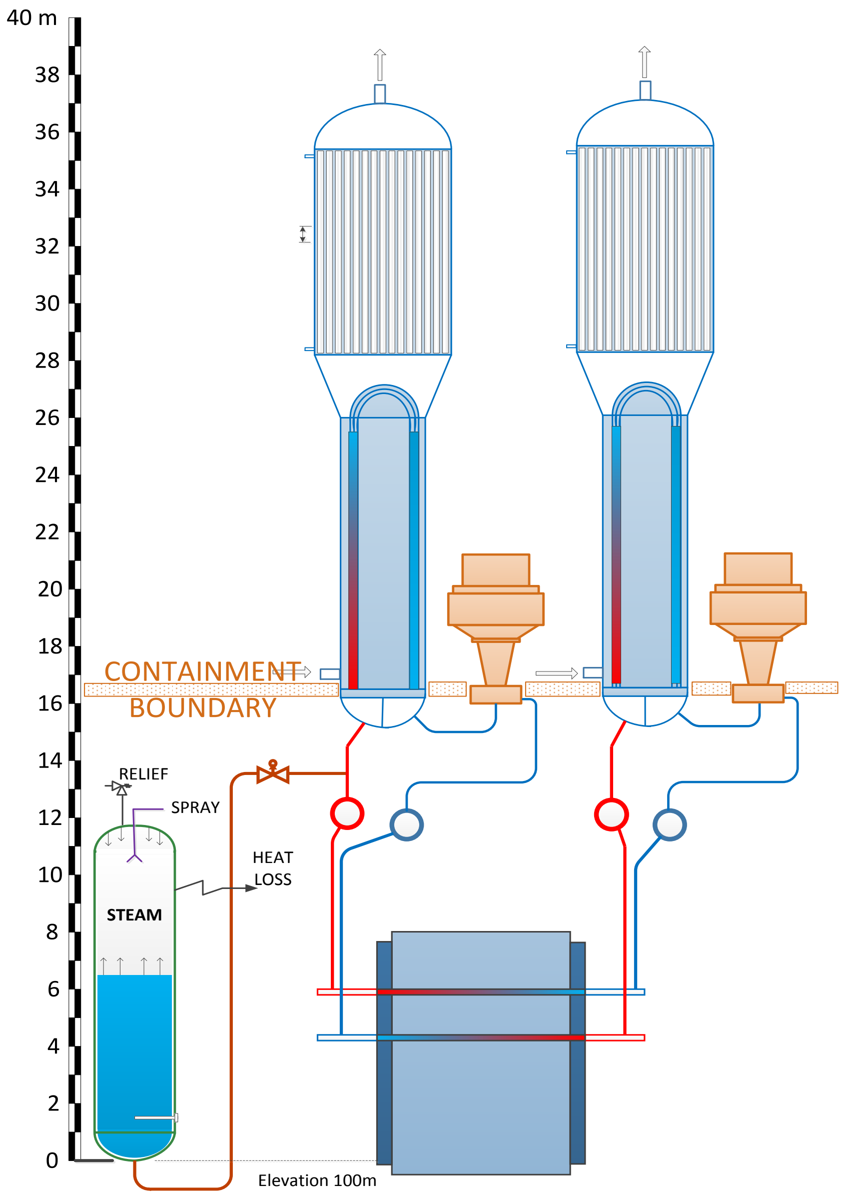


Figure 13 : Lower than core placement of pressurizer that will drain boiler tube inventory at Darlington/Bruce reactors in SBO

The strange choice of pressurizer location below the boilers and reactor will cause draining into it of primary coolant from boiler tubes in a SBO to an extent that boilers will become useless as heat sinks and no amount of emergency measures to add water to boilers will restore cooling to the reactor core unless the primary cooling system was replenished as well, something that cannot be done after an SBO in the present design and presently configured SAMGs.

2.3.4 Lack of a pressure vessel causes direct containment contamination

- Onset of severe core damage puts activity directly into the containment. There is no isolation of damaged core and its activity in a closed vessel like in a PWR pressure vessel.

2.3.5 Poor Deuterium Hydrogen mitigation systems

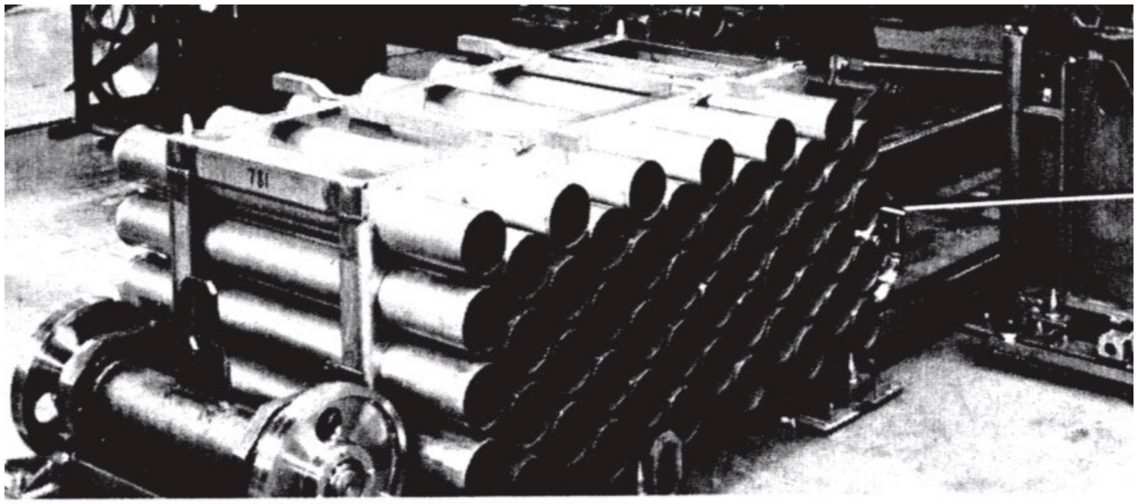
- Significantly higher sources of hydrogen from large amounts of carbon steel and Zircaloy.
- Currently planned hydrogen mitigation systems (igniters + a small number of PARS) inadequate and potentially dangerous. Poor combustible gas mitigation measures. Small number of Autocatalytic Recombiners inadequate for severe accident scenarios and will cause explosions.
  - All hydrogen detection and mitigation systems designed for H_2, not D_2 as required.

2.3.6 Calandria vessel a very unlikely core catcher

- Calandria vessel is designed to hold warm water at a low pressure, It is constructed by welding together a dozen 28mm thick stainless steel plates, bent to 120^o and joined by a thinner annular plate. It is a non nuclear Class 2C vessel not designed to take any pressure pulse from a BLEVE event
- Energetic interactions of disassembling core debris with underlying boiling moderator water in the low pressure Calandria vessel can cause vessel structural failures.
- Calandria vessel failure by weld failures is a likely outcome even before debris melt. There are a number of pipe penetrations at the bottom of the vessel that can fail by thermal interactions with hot debris.
- Should the Calandria vessel fail, interaction of hot debris with Shield Tank water also similarly challenging to integrity of structures holding the reactor vessels connected to the reactivity deck at the containment pressure boundary
- Calandria vessel likely cannot contain melting reactor core debris and can fail catastrophically at welds causing energetic interactions with potential for gross structure failures.

2.3.7 Spent Fuel storage

The spent fuel medium term storage in spent fuel pools is poorly designed and highly susceptible to Zircaloy fires.



2.3.8 Backup Diesel Generators

These are located at the lowest grade elevation in the plant and are no more than 2m above the water line at Darlington and 3m at Bruce. The tunnel carrying the cables is below the water line by about 4m and can get deluged with water. Pickering station has seen its basement level flooded in the past from water swell in the lake. Location of backup diesel generators has been pointed out as the single most critical error at Fukushima; something that has escaped the CNSC despite repeated warnings.

And…

- Inadequate instrumentation and control for severe accidents
- Poor equipment survivability due to poor containment layout
- No dedicated operator training / simulators for severe accidents.
- Severe accident simulation methods are outdated, crude and inadequate.
- No significant design changes implemented. Known problems ignored for decades.
- Current SAMGs are unrealistic and inadequate. Many potentially favourable emergency hookups not implemented.
- Environmental assessments for off-site releases after a severe accidents performed with a source term that represents barely 0.15% of the total core inventory

3. REVIEW OF PLANT PARAMETER ENVELOPE DOCUMENT

CNSC provided a Darlington New Build Plant Parameter Envelope report in February 2023. This, the 5th revision of the subject report (reference 4), was issued by OPG in October 2022 and is almost an exact copy of original Rev. 1 issued in 2009. OPG changed only 6 entries for BWRX-300 in the only substantial revision made to the PPE report in 12 years. Four managers signed off on inclusion of this sparse data that actually disclosed nothing about the reactor except the reactor power, depth and height ofvthe building along with some unreferenced data on normal duty emissions.

Given what the original purpose of a PPE was, issuance of this outdated and incomplete report for public comments or its use as a surrogate for site suitability assessments and pseudo design reviews makes no sense as a specific design has long been chosen for the site. What is amusing is that and there is little numerical information on the chosen BWRX-300 design in this PPE. A mere 75% of a single page is devoted to describing the BWRX-300 design on the last page of the PPE.

It is also of considerable concern that most important lessons learnt from Fukushima are lost both on the proponents of the four designs whose data was never updated for any severe accident related information as well as on the regulatory body who totally ignored it, as has been usually the case, especially in this habitual hurry to push through a preordained decision, any such omissions by the utility whose outdated, incomplete and devoid of any engineering analysis, Plant Parameter Envelope (PPE) was put out for public reviews.

On top of all that, the documents now issued were neither complete nor comprehensive in making a case for acceptability of the chosen site, and much less in using the exclusion area of an operating reactor for siting and almost totally devoid of any technical analysis of data in this fourth new revision put out for consideration. Significant amount of data pertinent to effect on site suitability is plain missing. More than anything else, the location.

Without going into the benefits or drawbacks of abandoning at this stage, the traditional, prescriptive expectations of standard design review plans in Canadian Regulatory Guides for new builds and the difficulty in the public evaluating a new reactor proposal of a reactor design without any detailed safety assessments, environmental impact analyses , socio-economic impacts, or risk assessments, it is very clear that neither the CNSC, nor OPG have taken this process seriously and in ramming substandard documentation through what there should have been due diligence, they are making a mockery of the legislated public participation process. It is also possible that the OPG is out of its depth as an electric utility in evaluating advanced reactor designs and having used external help (Condesco) in creating the original PPE in 2009 and done practically nothing on it for 13 years, finds itself unable to complete even the elementary stages of a new reactor licensing application process. Even the preliminary safety analysis report, stripped of any data that would reveal any meaningful details about the reactor or the accidents considered, reads as totally prepared by the vendor.

We understand that neither NRC, nor CNSC have any regulatory requirements for the process used by an applicant on which information to gather and how to go about deriving it or on how to present that information that would allow the applicant 'to assume certain design parameters for an early site permit (ESP) application when a specific reactor technology has not been selected for a proposed site. A PPE serves in its inclusion of design data as a surrogate for design information of a specific design and bounding site parameters for comparison with its actual site characteristics in an elementary impact assessment on environmental, socio-economic and safety issues.

The report also does not meet the current industry guidelines from NEI issued almost one and a half years earlier on how to collect information from vendors whose designs the early site permit applicant wishes to be bounded by the plant parameter envelope (PPE). It ignores lessons learnt from NRC reviews of past practices by other utilities following the same PPE process, ignores regulatory expectations in the only jurisdiction where such documents have been reviewed as well as the Canadian public expectations for being kept informed of the industry decision process. The latest mid-March 2023 pronouncement by CNSC on the vendor understanding the regulatory process; while acting within a MOU to harmonize the licensing process with NRC whose elementary requirements on consideration of core melt in determining the exclusion zone are totally ignored. There are no grounds, without doing a proper severe core damage progression and consequence assessment on both stations at Darlington, on which BWRX cannot be located a few hundred meters from station boundary, next to a station with little mitigation capabilities to handle a core damage accident and with an exclusion ‘radius’ of only 350m, and across the road from a dry spent fuel storage complex. It would be prudent for CNSC to take a few steps back and do due diligence instead of ramming this process through a hoax of a public consultation process.

We understood that our review was to be conducted to see if the bounding information presented was such that an educated commentary can be made of its veracity and completeness for the purpose and see if the regulatory bodies have the right information to decide that the content, values, rationale are reasonable and sufficient to comply with traditional regulatory expectations and that public participation in the process is meaningful.

It is not apparent to us, however, why CNSC has not asked OPG to do better. Given that CNSC has a MOU with NRC on licensing new designs like the BWRX-300, the DNNP PPE does not venture at all into the suggestions made by NRC in its own reviews of the four sets of submissions by xx,yy,zz,tt utilities or the development of various safety guides like DG-4029 that venture into expectations from new builds.

3.1 SUMMARY OF FINDINGS ON PLANR PARAMETER ENVELOPE

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1. The report has not fully considered the industry guidelines for such a report that were issued by NEI almost one and a half years earlier. It does not address all data suggested by NEI-10-Rev-2. It does not reflect any industry experience, NRC queries or advances in reactor safety expectations that NEI says were reflected in this revision. Some of these questions are quite central to a safe design, Many numbered subject headings that were not broached. These include:
    a. Soil characteristics to bear dynamic loads
    b. Design basis maximum hurricane data (3 second gust speed)
    c. Water intake into condenser and service water and its temperature rise
    d. Water blowdown rate and temperature into the lake upon an accident
    e. Exhaust stack height
    f. Heat rejection rate into the atmosphere upon an accident
    g. Any Items unique to non-water Fire Protection Systems
    h. The design radiological dose consequences due to airborne releases from postulated accidents.
    i. The annual activity, by radionuclide, contained in routine plant liquid effluent streams.
    j. The assumed activity, by radionuclide, contained in accidental liquid radwaste release from postulated tank failure,
    k. The assumed volume of accidental liquid radwaste release.
    l. Detailed information on spent fuel; spent fuel pool
        i. Spent Fuel Pool Capacity - the number of spent fuel assemblies capable of being stored in the spent fuel pool.
        ii. Fuel Bundles Discharged per Refuel Outage - The number of spent fuel assemblies discharged to the spent fuel pool for a typical refuel outage.
        iii. Fuel Cycle Duration
        iv. Fuel Bundles Discharged During Licensed Operation - The total number of spent fuel assemblies discharged during the 40 year operating license life of the plant.
    m. Detailed information on gas turbines
        i. The total generating capacity of the gas turbine generating system.
        ii. The elevation above finished grade of the release point for standby gas turbine exhaust releases.
        iii. The expected combustion products and anticipated quantities released to the environment due to operation of the emergency standby gas-turbine generators. Provide in Table 6.
        iv. The maximum expected sound level produced by operation of gas turbines, measured at 1000 feet from the noise source.
        v. The type of fuel oil required for proper operation of the gas turbines.
    n. The weight of the heaviest SMR component that is expected to be shipped to the site.
    o. Information on Fuel:
        i. 18.1 Maximum Fuel Enrichment
        ii. 18.2 Maximum Average Assembly Burnup
        iii. 18.3 Peak fuel rod exposure at end of life

iv. 18.4 Maximum Average Discharge Batch Burnup
v. 18.5 Maximum Thermal Power
vi. 18.6 Mass of uranium in the reload batch.
vii. 18.7 Clad Material

2. Concept of PPE, the plant and site data that it collects was developed before the Fukushima disaster struck in 2012. That was also long before we all took a good look at the vulnerabilities to severe accidents that our reactors inherited and developed a semblance of accident management guidelines, engineered measures, new systems and coordination mechanisms for emergency planning. Given that the OPG PPE is so wanting in detail and the new reactor design make unsubstantiated claims about their infallibility, there is a need to reflect these topics in that in the PPE data. Both common sense and NEI-10-Rev. 2 guidelines require that severe accident mitigation related information be included and with clarity and detail. It feels like the parties never heard of Fukushima or the conclusions of its investigations into the root causes.

3. An important omission in the PPE and site description is in discussion of why the new build HAS TO BE within an existing station's exclusion boundary, in spite of all the risks such a decision entails.

4. The PPE provided bounding values for 3 reactors tabulated in 2009 for them by Condesco with ZERO additions made through the next 13 years or any feedback from any person or organization.

5. A composite spreadsheet for all vendor data was not created (one column for each design). While bounding values (numerically maximum or minimum of data sent in by the vendors without any accompanying description) were identified, no rationale for comparing the supplied data with diverse origins, meaning or credibility was discussed. There was no discussion of any missing data, consistency check within vendor data set or any discussion of any reasonableness of data or error margins. These are actually explicit requirements and expectations in repeated NRC and NEI documents on the subject. A mere dump of bounding values makes no meaningful contribution to the stated intent.

6. Even simple, typically publicly available information on reactor designs was not made available for the design ultimately chosen under the inexplicable guise of being 'proprietary'. Such blatant cover of 'proprietary' information is inconsistent with the vendor's obligations to people of Canada where the vendor hopes to benefit from a proof of concept with public funds. Reactor data on new Chinese reactor designs is more abundantly available than was made available for BWRX-300. This is not a time machine design or a shoulder carried hypersonic missile design.

7. Public relations propaganda about the chosen reactor design's safety was freely dispensed without giving any numerical information on the reactor design that could be verified by nuclear safety experts working in public interest.

8. The process of arriving at the bounding value is not transparent as the data provided by each of the three vendors that dominate the information scape is not individually tabulated or referred to in a separate summary document for the design. That should have been an easy thing to do and with sufficient volume of information on the actual design, a proper way of verifying if the bounding data values were in context of ANY new design that may show up on the horizon layer, just as the BWRX-300 did, many years after the PPE was issued first. Observations on the specific features of a reactor design from which the bounding value was derived were not made.

9. The data set does not contain any information that would be necessary and be specific to the Darlington site where other operating reactors already exist. This includes data on Derived Emission Limits and actual emission history that would be added to that from new units.
10. Site parameter characteristic data on effect of operations of the existing reactors on operation of the proposed new reactor (and vice versa) was not clearly given.
11. Effect of an accident at one of the operating units on construction, operation or decommissioning of the new reactor was not given..
12. Effect of a severe core damage accident at an operating unit on safety of personnel engaged in construction of the new reactor(s) was not considered. The source term data given to the Emergency Management Organizations by utility running the operating reactors is irresponsibly fraudulent and cannot be used to prepare emergency evacuation or sheltering processes for our fellow citizens working on site.
13. The effect of a limiting severe core damage accident on the plant parameter envelope was not considered..
14. When the new entry into the list of potential reactor designs had a parameter that was outside the enveloping limits defined in the earlier incarnation of the PPE, the envelope was extended without any explanation. For example when the BWRX-300 required to be built onto a depth of 38 meters feet underground and above ground, equivalent to a total structure height of a 25 story building – the PPE was merely re-written to make these parameters acceptable.
15. Source term from normal operation from a number of release points was provided (also recommended by NRC in its review of NEI-10) with certain entries missing. Source term from regular emissions was provided without providing any information on it's nature (continuous or frequency of puffs if any) and what sources it includes, of basis of its derivation, analytical assumptions and tools.
16. No comparison of source terms between 4 reactors without giving any information about the reactors, their vulnerabilities, accident scenarios, release locations, release frequencies is meaningless.
17. NRC expects a clear statement on margin of error on the bounding values chosen. CNSC should too. Certain critical data where margins of error are critically important.
18. Some very important lessons were learnt from Fukushima. There is no mention of any comparison of risk between various designs; especially from BWRX-300 except that claims of eternal and near absolute safety are made.
19. Need for radiation monitoring equipment that would detect and save data on normal operation effluents as well as radiation fields from accidents.
20. Information on fuel procurement
21. Decommissioning responsibility
22. Decommissioning costs

3.2 RECOMMENDATION ON PPE

We propose that the current PPE be not accepted as surrogate to anything and a renewed set of documents be prepared that details the actual data for BWRX-300 and issued for comments to me. It should include enough information on each of the reactor designs that were considered (as a summary design description with pictures and tables and references) and a much broader discussion of the chosen BWRX-300 design.

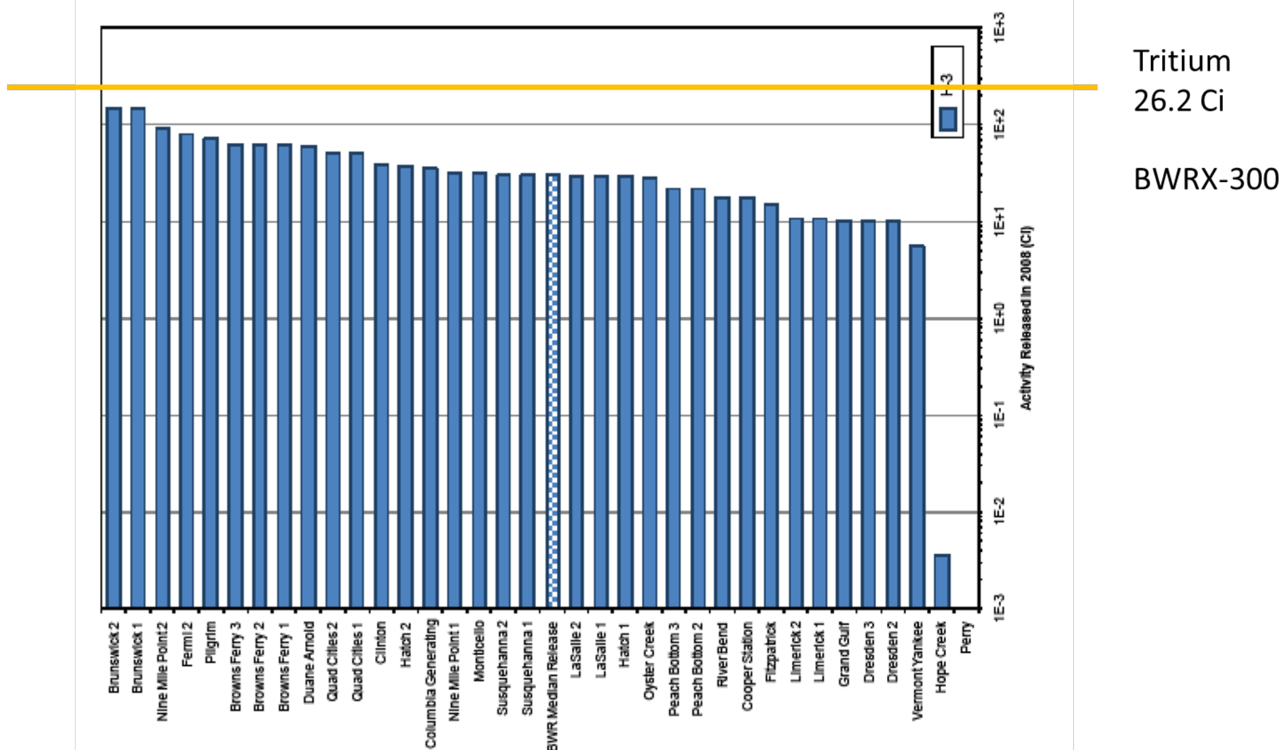

FIGURE 2.4 Tritium (H-3) releases from (A) BWRs and (B) PWRs in 2008.
SOURCE: Daugherty and Conatser (2008).

Figure 14 : Sample results of operational releases of Tritium from US BWRs in 2008 from reference xx and PPE data for BWRX-300,

NOBLE
GASES
BWRX-300
3.67E6 GBq

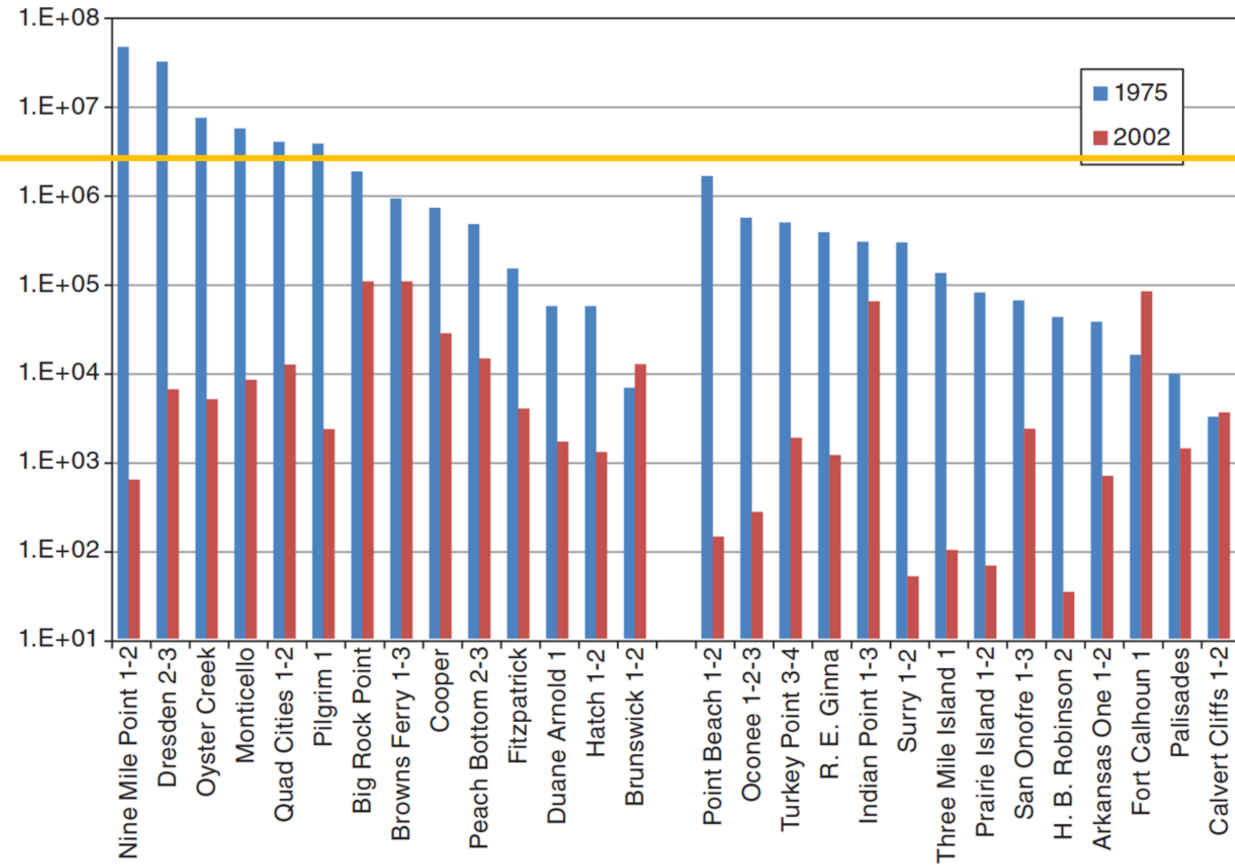

FIGURE 2.5 Comparison of atmospheric releases of noble gases for selected BWRs (left) and PWRs (right) in the United States. The units on the vertical scale are in gigabecquerels (GBq = 0.03 Ci). SOURCE: Data from the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR).

Iodine -131
BWRX-300
1.08e-2 Ci
=600 Sv

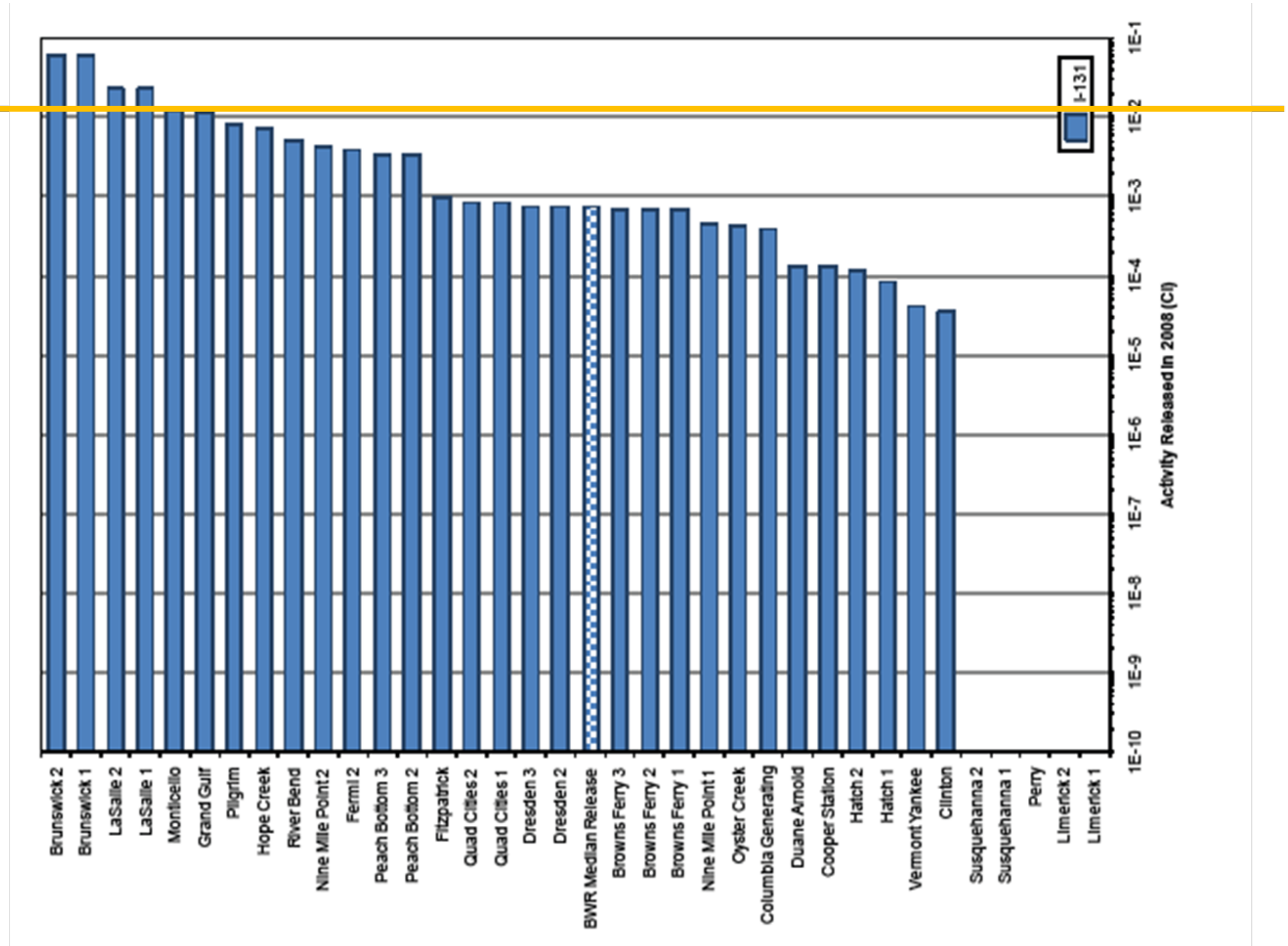

FIGURE 2.2 Iodine-131 releases from (A) BWRs and (B) PWRs in 2008. SOURCE: Daugherty and Conatser (2008).

BWRX-300 no data

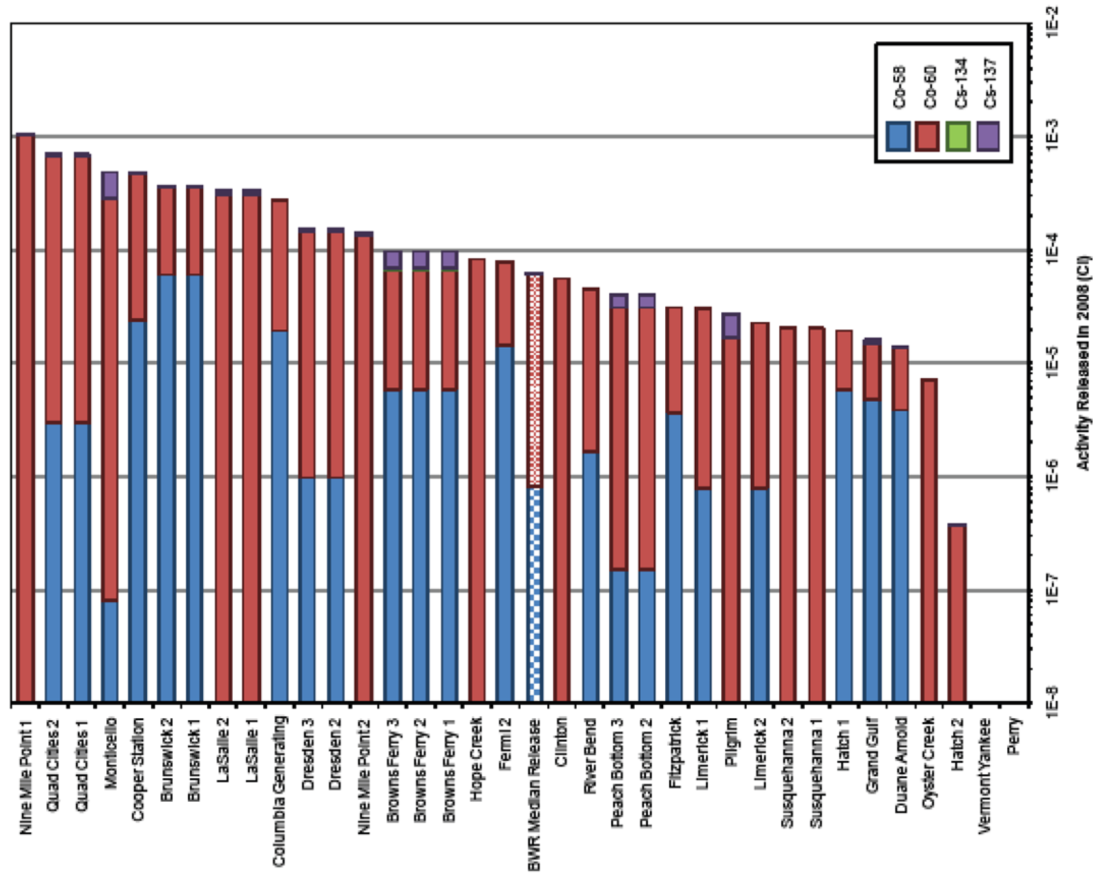

FIGURE 2.3 Particulate releases from (A) BWRs and (B) PWRs in 2008. SOURCE: Daugherty and Conatser (2008).

4. Summary of findings and review

Siting a new reactor within the exclusion area boundary (EAB)of an operating reactor that has significant and unresolved vulnerabilities to severe core damage is contrary to safety principles, regulatory requirements as well as requisite stakeholder obligations to worker security and public safety. Even the currently accepted US NRC inspired 1000 yard (914m) EAB on the existing nuclear installations will not meet the U.S 10CFR100.11 requirements, and public expectations to consider source term from a severe accident core melt. Therefore, the indicated EAB for BWRX-300 of 350m is incredulous as is its full inclusion within the 40 year old EAB for the Darlington plant and within the EAB for the dry storage shed structures housing > 130 reactor core loads of spent fuel in concrete casks with > 70% of volatile, high dose sensitive fission product species still active. The CNSC needs to exercise due diligence in requiring OPG to resolve existing critical safety issues with Darlington CANDU severe accident mitigation before an event causes all reactors on site to have to be abandoned. Mere siting of BWRx-300 next to sheds containing hundreds of reactor years of spent fuel and 50 odd meter separation of its switchyard from an operating public railway line is another step in the Ontario Power Generation sleep walk towards an impending disaster at Darlington. GE-H should reconsider this siting plan for their own corporate interests. Their design is neither small, nor so modular as a first of kind construction and requires a comprehensive Environmental Assessment to protect Canadian public interests, previous actions in this regard notwithstanding.

The Plant Parameter Envelope document sent for review is grossly incomplete; has seen no substantive OPG additions in 12 years; and the US 10CFR52 process of site qualification it mimics is of no relevance today as a vendor GE-Hitachi and their BWRX-300 design has already been selected by OPG. Once a new site for the reactor is identified, a detailed reactor design information binder will help qualify that site for the chosen design. There has to be an independent technical review of vendor GE-Hitachi claims of enhanced safety in their BWRX-300 design with access to their actual safety reports with detailed analytical assumptions, code descriptions and accident simulation results with numerical information they have been unable to reveal so far.

5. 10 CFR 100.11 Determination of exclusion area, low population zone, and population center distance.

(a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release[1] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem[2] or a total radiation dose in excess of 300 rem[2] to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfaction of the Commission the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

Note: For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document 14844, dated March 23, 1962, which contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation.

[27 FR 3509, Apr. 12, 1962, as amended at 31 FR 4670, Mar. 19, 1966; 38 FR 1273, Jan. 11, 1973; 40 FR 8793, Mar. 3, 1975; 40 FR 26527, June 24, 1975; 53 FR 43422, Oct. 27, 1988; 64 FR 48955, Sept. 9, 1999; 67 FR 67101, Nov. 4, 2002]

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

[2] The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

Page Last Reviewed/Updated Tuesday, August 29, 2017

6. REFERENCES

1 . Severe accident progression without operator action, Canadian Nuclear safety Commission, Oct 2015, http://www.nuclearsafety.gc.ca/eng/pdfs/Reports/Severe-AccidentProgression-without-Operator-Action.pdf

2 Study of Consequences of a Hypothetical Severe Nuclear Accident and Effectiveness of Mitigation Measures, CNSC, Sept 2015, http://nuclearsafety.gc.ca /eng/resources/health/ hypothetical-severe-nuclear-accident-study.cfm

3 Regulatory Actions That Hinder Development Of Effective Risk Reduction Measures By The Nuclear Industry For Enhanced Severe Accident Prevention And Mitigation Measures After Fukushima, Sunil Nijhawan, ICONE24-60700, Proceedings of the 2016 24th International Conference on Nuclear Engineering ICONE24, June 26-30, 2016, Charlotte, North Carolina, USA

4 Use of Plant Parameters Envelope to Encompass the Reactor Designs being considered for the Darlington Site, N-REP-01200-10000-R005, Ontario Power Generation, 2022-Oct-4

Joint Review Panel Recommendations (10 pages)

Prior to Site Preparation

Recommendation # 2 (Section 4.5):
The Panel recommends that prior to site preparation, the Canadian Nuclear Safety Commission require OPG to conduct a comprehensive soils characterization program. In particular, the potentially impacted soils in the areas OPG identifies as the spoils disposal area, cement plant area and asphalt storage area must be sampled to identify the nature and extent of potential contamination.

Recommendation # 6 (Section 4.6):
The Panel recommends that prior to site preparation, the Canadian Nuclear Safety Commission require OPG to update its preliminary decommissioning plan for site preparation in accordance with the requirements of Canadian Standards Association Standard N294-09. The OPG preliminary decommissioning plan for site preparation must incorporate the rehabilitation of the site to reflect the existing biodiversity in the event that the Project does not proceed beyond the site preparation phase. OPG shall prepare a detailed preliminary decommissioning plan once a reactor technology is chosen, to be updated as required by the Canadian Nuclear Safety Commission.

Recommendation # 7 (Section 4.6):
The Panel recommends that prior to site preparation, the Canadian Nuclear Safety Commission require that OPG establish a decommissioning financial guarantee to be reviewed as required by the Canadian Nuclear Safety Commission. Regarding the decommissioning financial guarantee for the site preparation stage, the Panel recommends that this financial guarantee contain sufficient funds for the rehabilitation of the site in the event the Project does not proceed beyond the site preparation stage.

Recommendation # 8 (Section 5.1):
The Panel recommends that prior to site preparation, the Canadian Nuclear Safety Commission require OPG to develop a follow-up and adaptive management program for air contaminants such as Acrolein, NO_2, SO_2, SPM, $PM_{2.5}$ and PM_{10}, to the satisfaction of the Canadian Nuclear Safety Commission, Health Canada and Environment Canada. Additionally, the Canadian Nuclear Safety Commission must require OPG to develop an action plan acceptable to Health Canada for days when there are air quality or smog alerts.

Recommendation # 9 (Section 5.1):
The Panel recommends that the Canadian Nuclear Safety Commission, in collaboration with Health Canada, require OPG to develop and implement a detailed acoustic assessment for all scenarios evaluated. The predictions must be shared with potentially affected members of the public. The OPG Nuisance Effects Management Plan must include noise monitoring, a noise complaint response mechanism and best practices for activities that may occur outside of municipal noise curfew hours to reduce annoyance that the public may experience.

Recommendation # 10 (Section 5.2):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to undertake a detailed site geotechnical investigation prior to commencing site preparation activities. The geologic elements of this investigation should include, but not be limited to:
· collecting site-wide information on soil physical properties;
· determining the mechanical and dynamic properties of overburden material across the site;
· mapping of geological structures to improve the understanding of the site geological structure model;
· confirming the lack of karstic features in the local bedrock at the site; and
· confirming the conclusions reached concerning the liquefaction potential in underlying granular materials.

Recommendation # 12 (Section 5.3):
The Panel recommends that before in-water works are initiated, the Canadian Nuclear Safety Commission require OPG to collect water and sediment quality data for any future embayment area that may be formed as a consequence of shoreline modifications in the vicinity of the outlet of Darlington Creek. This data should serve as the reference information for the proponent's post-construction commitment to conduct water and sediment quality monitoring of the embayment area.

Recommendation # 13 (Section 5.3):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to collect and assess water quality data for a comprehensive number of shoreline and offshore locations in the site study area prior to commencing in-water works. This data should be used to establish a reference for follow-up monitoring.

Recommendation # 20 (Section 5.5):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to perform a thorough evaluation of site layout opportunities before site preparation activities begin, in order to minimize the overall effects on the terrestrial and aquatic environments and maximize the opportunity for quality terrestrial habitat rehabilitation.

Recommendation #22 (Section 5.5):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to develop a follow-up program for insects, amphibians and reptiles, and mammal species and communities to ensure that proposed mitigation measures are effective.

Recommendation # 25 (Section 5.5):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to conduct more sampling to confirm the presence of Least Bittern before site preparation activities begin. The Panel recommends that the Canadian Nuclear Safety Commission require OPG to develop and implement a management plan for the species at risk that are known to occur on site. The plan should consider the resilience of some of the species and the possibility of off-site compensation.

Recommendation # 38 (Section 5.9):
The Panel recommends that the Canadian Nuclear Safety Commission require that the geotechnical and seismic hazard elements of the detailed site geotechnical investigation to be performed by OPG include, but not be limited to:

Prior to site preparation:

· demonstration that there are no undesirable subsurface conditions at the Project site. The overall site liquefaction potential shall be assessed with the site investigation data; and
· confirmation of the absence of paleoseismologic features at the site and, if present, further assessment to reduce the overall uncertainty in the seismic hazard assessment during the design of the Project must be conducted.

During site preparation and/or prior to construction:

· verification and confirmation of the absence of surface faulting in the overburden and bedrock at the site.

Prior to construction:

· verification of the stability of the cut slopes and dyke slopes under both static and dynamic loads with site/Project-specific data during the design of the cut slopes and dykes or before their construction;
· assessment of potential liquefaction of the northeast waste stockpile by using the data obtained from the pile itself upon completion of site preparation;
· measurement of the shear strength of the overburden materials and the dynamic properties of both overburden and sedimentary rocks to confirm the site conditions and to perform soil-structure interaction analysis if necessary;
· assessment of the potential settlement in the quaternary deposits due to the groundwater drawdown caused by future St. Marys Cement quarry activities; and
· assessment of the effect of the potential settlement on buried infrastructures in the deposits during the design of these infrastructures.

Prior to operation:

· development and implementation of a monitoring program for the Phase 4 St. Marys Cement blasting operations to confirm that the maximum peak ground velocity at the boundary between the Darlington and St. Marys Cement properties is below the proposed limit of three millimetres per second (mm/s).

Recommendation # 41 (Section 6.1):
The Panel recommends that prior to site preparation, the Canadian Nuclear Safety Commission coordinate discussions with OPG and key stakeholders on the effects of the Project on housing supply and demand, community recreational facilities and programs, services and infrastructure as well as additional measures to help deal with the pressures on these community assets.

Recommendation # 47 (Section 6.7):
The Panel recommends that prior to site preparation, the Canadian Nuclear Safety Commission ensure the OPG Traffic Management Plan addresses the following:
· contingency plans to address the possibility that the assumed road improvements do not occur;
· consideration of the effect of truck traffic associated with excavated material disposal on traffic operations and safety;
· further analysis of queuing potential onto Highway 401; and
· consideration of a wider range of mitigation measures, such as transportation-demand management, transit service provisions and geometric improvements at the Highway 401/Waverley Road interchange.

Recommendation # 48 (Section 6.7):
In consideration of public safety, the Panel recommends that prior to site preparation, the Canadian Nuclear Safety Commission coordinate a committee of federal, provincial and municipal transport authorities to review the need for road development and modifications.

During Site Preparation

Recommendation #5 (Section 4.6):
To avoid any unnecessary environmental damage to the bluff at Raby Head and fish habitat, the Panel recommends that no bluff removal or lake infill occur during the site preparation stage, unless a reactor technology has been selected and there is certainty that the Project will proceed.

Recommendation # 19 (Section 5.4):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to expand the scope of the groundwater monitoring program to monitor transitions in groundwater flows that may arise as a consequence of grade changes during the site preparation and construction phases of the Project. The design of the grade changes should guide the determination of the required monitoring locations, frequency of monitoring and the required duration of the program for the period of transition to stable conditions following the completion of construction and the initial period of operation.

Recommendation # 21 (Section 5.5):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to compensate for the loss of ponds, like-for-like, preferably in the site study area. The Panel also recommends that the Canadian Nuclear Safety Commission require OPG to use best management practices to prevent or minimize the potential runoff of sediment and other contaminants into wildlife habitat associated with Coot's Pond during site preparation and construction phases.

Prior to Construction

Recommendation # 1 (Section 4.5):
The Panel understands that prior to construction, the Canadian Nuclear Safety Commission will determine whether this environmental assessment is applicable to the reactor technology selected by the Government of Ontario for the Project. Nevertheless, if the selected reactor technology is fundamentally different from the specific reactor technologies bounded by the plant parameter envelope, the Panel recommends that a new environmental assessment be conducted.

Recommendation # 3 (Section 4.5):
The Panel recommends that the Canadian Nuclear Safety Commission require that as part of the Application for a Licence to Construct a reactor, OPG must undertake a formal quantitative cost-benefit analysis for cooling tower and once-through condenser cooling water systems, applying the principle of best available technology economically achievable. This analysis must take into account the fact that lake infill should not go beyond the two-metre depth contour and should include cooling tower plume abatement technology.

Recommendation # 14 (Section 5.3):
The Panel recommends that following the selection of a reactor technology for the Project, the Canadian Nuclear Safety Commission require OPG to conduct a detailed assessment of predicted effluent releases from the Project. The assessment should include but not be limited to effluent quantity, concentration, points of release and a description of effluent treatment, including demonstration that the chosen option has been designed to achieve best available treatment technology and techniques economically achievable. The Canadian Nuclear Safety Commission shall also require OPG to conduct a risk assessment on the proposed residual releases to determine whether additional mitigation measures may be necessary.

Recommendation # 16 (Section 5.3):
The Panel recommends that prior to the start of construction, the Canadian Nuclear Safety Commission require the proponent to establish toxicity testing criteria and provide the test methodology and test frequency that will be used to confirm that stormwater discharges from the new nuclear site comply with requirements in the Fisheries Act.

Recommendation # 17 (Section 5.4):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to provide an assessment of the ingress and transport of contaminants in groundwater on site during successive phases of the Project as part of the Application for a Licence to Construct. This assessment shall include consideration of the impact of wet and dry deposition of all contaminants of potential concern and radiological constituents, especially tritium, in gaseous emissions on groundwater quality. OPG shall conduct enhanced groundwater and contaminant transport modelling for the assessment and expand the modelling to cover the effects of future dewatering and expansion activities at the St. Marys Cement quarry on the Project.

Recommendation # 26 (Section 5.5):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to develop a comprehensive assessment of hazardous substance releases and the required management practices for hazardous chemicals on site, in accordance with the Canadian Environmental Protection Act , once a reactor technology has been chosen.

Recommendation # 27 (Section 5.6):
The Panel recommends that prior to any destruction of the Bank Swallow habitat, the Canadian Nuclear Safety Commission require OPG to implement all of its proposed Bank Swallow mitigation options, including:
· the acquisition of off-site nesting habitat;
· the construction of artificial Bank Swallow nest habitat with the capacity to maintain a population which is at least equal to the number of breeding pairs currently supported by the bluff and as close to the original bluff site as possible; and
· the implementation of an adaptive management approach in the Bank Swallow mitigation plan, with the inclusion of a threshold of loss to be established in consultation with all stakeholders before any habitat destruction takes place.

Recommendation # 35 (Section 5.7):
In the event that a once-through condenser cooling system is chosen for the Project, the Panel recommends that prior to operation, the Canadian Nuclear Safety Commission require OPG to include the following in the surface water risk assessment:
· the surface combined thermal and contaminant plume; and
· the physical displacement effect of altered lake currents as a hazardous pulse exposure to fish species whose larvae passively drift through the area, such as lake herring, lake whitefish, emerald shiner and yellow perch.
If the risk assessment result predicts a potential hazard then the Canadian Nuclear Safety Commission shall convene a follow-up monitoring scoping workshop with Environment Canada, Fisheries and Oceans Canada and any other relevant authorities to develop an action plan.

Recommendation # 37 (Section 5.7):
In the event that a once-through condenser cooling system is chosen for the Project, the Panel recommends that prior to construction, the Canadian Nuclear Safety Commission require OPG to determine the total area of permanent aquatic effects from the following, to properly scale mitigation and scope follow-up monitoring:
· the thermal plume + 2_o C above ambient temperature;
· the mixing zone and surface plume contaminants;

· physical displacements from altered lake currents; and
· infill and construction losses and modifications.

Recommendation # 39 (Section 5.9):
The Panel recommends that prior to construction, the Canadian Nuclear Safety Commission require OPG to prepare a contingency plan for the construction, operation and decommissioning Project stages to account for uncertainties associated with flooding and other extreme weather hazards. OPG shall conduct localized climate change modelling to confirm its conclusion of a low impact of climate change. A margin/bound of changes to key parameters, such as intensity of extreme weather events, needs to be established to the satisfaction of the Canadian Nuclear Safety Commission. These parameters can be incorporated into hydrological designs leading up to an application to construct a reactor, as well as measures for flood protection. OPG must also conduct a drought analysis and incorporate any additional required mitigation/design modifications, to the satisfaction of the Canadian Nuclear Safety Commission, as part of a Licence to Construct a reactor.

Recommendation # 40 (Section 5.9):
The Panel recommends that **prior to construction**, the Canadian Nuclear Safety Commission require OPG to:
· establish an adaptive management program for algal hazard to the Project cooling water system intake that includes the setup of thresholds for further actions; and
· factor the algal hazard assessment into a more detailed biological evaluation of moving the intake and diffuser deeper offshore as part of the detailed siting studies and the cost-benefit analysis of the cooling system.

Recommendation # 52 (Section 6.8):
The Panel recommends that **prior to construction**, the Canadian Nuclear Safety Commission require OPG to make provisions for on-site storage of all used fuel for the duration of the Project, in the event that a suitable off-site solution for the long-term management for used fuel waste is not found.

Recommendation # 53 (Section 6.8):
The Panel recommends that **prior to construction**, the Canadian Nuclear Safety Commission require OPG to make provisions for on-site storage of all of low and intermediate-level radioactive waste for the duration of the Project, in the event that a suitable off-site solution for the long-term management for this waste is not approved.

Recommendation # 57 (Section 7.2):
The Panel recommends that **prior to construction**, the Canadian Nuclear Safety Commission require OPG to undertake an assessment of the off-site effects of a severe accident. The assessment should determine if the off-site health and environmental effects considered in this environmental assessment bound the effects that could arise in the case of the selected reactor technology.

Recommendation # 58 (Section 7.2):
The Panel recommends that **prior to construction**, the Canadian Nuclear Safety Commission confirm that dose acceptance criteria specified in RD-337 at the reactor site boundary—in the cases of design basis accidents for the Project's selected reactor technology—will be met.

Recommendation # 63 (Section 8.1):
The Panel recommends that **prior to construction**, the Canadian Nuclear Safety Commission require OPG to evaluate the cumulative effect of a common-cause severe accident involving all of the nuclear reactors in the site study area to determine if further emergency planning measures are required.
During Operation

Recommendation # 15 (Section 5.3):
The Panel recommends that following the start of operation of the reactors, the Canadian Nuclear Safety Commission require OPG to conduct monitoring of ambient water and sediment quality in the receiving waters to ensure that effects from effluent discharges are consistent with predictions made in the environmental impact statement and with those made during the detailed design phase.

Recommendation # 18 (Section 5.4):
The Panel recommends that based on the groundwater and contaminant transport modelling results, the Canadian Nuclear Safety Commission require OPG to expand the Radiological Environmental Monitoring Program. This program shall include relevant residential and private groundwater well quality data in the local study area that are not captured by the current program, especially where the modelling results identify potential critical groups based on current or future potential use of groundwater.

Recommendation # 36 (Section 5.7):
In the event that a once-through condenser cooling system is chosen for the Project the Panel recommends that during operation, the Canadian Nuclear Safety Commission require OPG to undertake adult fish monitoring of largebodied and small-bodied fish to confirm the effectiveness of mitigation measures and verify the predictions of no adverse thermal and physical diffuser jet effects.

Recommendation # 54 (Section 7.1):
The Panel recommends that during operation, the Canadian Nuclear Safety Commission require OPG to implement measures to manage releases from the Project to avoid tritium in drinking water levels exceeding a running annual average of 20 Becquerels per litre at drinking water supply plants in the regional study area.

Recommendation # 61 (Section 8.1):
The Panel recommends that during operation, the Canadian Nuclear Safety Commission require OPG to monitor aquatic habitat and biota for potential cumulative effects from the thermal loading and contaminant plume of the discharge structures of the existing Darlington Nuclear Generating Station and Over the Life of the Project

Recommendation # 4 (Section 4.6):
The Panel recommends that the Canadian Nuclear Safety Commission exercise regulatory oversight to ensure that OPG complies with all municipal and provincial requirements and standards over the life of the Project. This is of particular importance because the conclusions of the Panel are based on the assumption that OPG will follow applicable laws and regulations at all jurisdictional levels.

Recommendation # 11 (Section 5.2):
The Panel recommends that the Canadian Nuclear Safety Commission require OPG to develop and implement a follow-up program for soil quality during all stages of the Project.

Recommendation # 43 (Section 6.2):
The Panel recommends that the Canadian Nuclear Safety Commission engage appropriate stakeholders, including OPG, Emergency Management Ontario, municipal governments and the Government of Ontario to develop a policy for land use around nuclear generating stations.

Recommendation # 56 (Section 7.1):
The Panel recommends that over the life of the Project, the Canadian Nuclear Safety Commission require OPG to conduct ambient air monitoring in the local study area on an ongoing basis to ensure that air quality remains at levels that are not likely to cause adverse effects to human health.

Fisheries and Oceans Canada

Prior to Construction

Recommendation # 30 (Section 5.7):
In the event that a once-through condenser cooling system is chosen for the Project, the Panel recommends that **prior to the construction** of in-water structures, Fisheries and Oceans Canada require OPG to conduct:
· additional impingement sampling at the existing Darlington Nuclear Generating Station to verify the 2007 results and deal with inter-year fish abundance variability and
· additional entrainment sampling at the existing Darlington Nuclear Generating Station to better establish the current conditions. The program should be designed to guard against a detection limit bias by including in the analysis of entrainment losses those fish species whose larvae and eggs are captured in larval tow surveys for the seasonal period of

the year in which they occur. A statistical optimization analysis will be needed to determine if there is a cost-effective entrainment survey design for round whitefish larvae.

Recommendation # 32 (Section 5.7):
In the event that a once-through condenser cooling system is chosen for the Project, the Panel recommends that Fisheries and Oceans Canada require OPG to mitigate the risk of adverse effects from operation, including impingement, entrainment and thermal excursions and plumes, by locating the system intake and diffuser structures in water beyond the nearshore habitat zone. Furthermore, OPG must evaluate other mitigative technologies for the system intake, such as live fish return systems and acoustic deterrents.

During Construction

Recommendation # 31 (Section 5.7):
Irrespective of the condenser cooling system chosen for the Project, the Panel recommends that Fisheries and Oceans Canada not permit OPG to infill beyond the two-metre depth contour in Lake Ontario.

Over the Life of the Project

Recommendation # 28 (Section 5.7):
The Panel recommends that Fisheries and Oceans Canada require OPG to continue conducting adult fish community surveys in the site study area and reference locations on an ongoing basis. These surveys shall be used to confirm that the results of 2009 gillnetting and 1998 shoreline electrofishing reported by OPG, and the additional data collected in 2010 and 2011, are representative of existing conditions, taking into account natural year-to-year variability.
Specific attention should be paid to baseline gillnetting monitoring in spring to verify the findings on fish spatial distribution and relatively high native fish species abundance in the embayment area, such as white sucker and round whitefish. The shoreline electrofishing habitat use study is needed to establish the contemporary baseline for later use to test for effects of lake infill armouring, if employed, and the effectiveness of mitigation.

Recommendation # 29 (Section 5.7):
The Panel recommends that Fisheries and Oceans Canada require OPG to continue the research element of the proposed Round Whitefish Action Plan for the specific purpose of better defining the baseline condition, including the population structure, genome and geographic distribution of the round whitefish population as a basis from which to develop testable predictions of effects, including cumulative effects.

Recommendation # 33 (Section 5.7):
The Panel recommends that Fisheries and Oceans Canada require OPG to conduct an impingement and entrainment follow-up program at the existing Darlington Nuclear Generating Station and the Project site to confirm the prediction of adverse effects, including cumulative effects, and the effectiveness of mitigation. For future entrainment sampling for round whitefish, a statistical probability analysis will be needed to determine if unbiased and precise sample results can be produced.

Transport Canada

Prior to Construction

Recommendation # 49 (Section 6.7):
The Panel recommends that **prior to construction**, Transport Canada ensure that OPG undertake additional quantitative analysis, including collision frequencies and rail crossing exposure indices, and monitor the potential effects and need for mitigation associated with the Project.

Recommendation # 50 (Section 6.7):
The Panel recommends that **prior to construction**, Transport Canada require OPG to conduct a risk assessment, jointly with Canadian National Railway, that includes: · an assessment of the risks associated with a derailment or other rail incident that could affect the Project;

· an analysis of the risks associated with a security threat, such as a bomb being placed on a train running on the tracks that bisect the Project;
· a comparative evaluation of the effectiveness of various mitigation measures or combination of measures (e.g., blast wall, retaining wall, recessed tracks, berm and railway speed restrictions within the vicinity of the site);
· a determination of the design criteria necessary to ensure the effectiveness of these measures (e.g., the appropriate height, strength, material and design of a blast wall); and
· a critical analysis to confirm that these measures, when properly designed and implemented, would be sufficient to provide protection to the Project site in the event of a derailment at full speed or other adverse event.

Recommendation # 51 (Section 6.7):
In the event that a once-through condenser cooling system is chosen for the Project, the Panel recommends that **prior to construction,** Transport Canada work with OPG to develop a follow-up program to verify the accuracy of the prediction of no significant adverse effects to boating safety from the establishment of an increased prohibitive zone. OPG must also develop an adaptive management program, if required, to mitigate potential effects to small watercraft.

Environment Canada

Prior to Site Preparation

Recommendation # 62 (Section 8.1):
The Panel recommends that **prior to site preparation**, Environment Canada evaluate the need for additional air quality monitoring stations in the local study area to monitor cumulative effects on air quality.

During Site Preparation

Recommendation # 24 (Section 5.5):
The Panel recommends that **during the site preparation stage**, Environment Canada shall ensure that OPG not undertake habitat destruction or disruption between the period of May 1 and July 31 of any year to minimize effects to breeding migratory birds.

Prior to Construction

Recommendation # 34 (Section 5.7):
In the event that a once-through condenser cooling system is chosen for the Project, the Panel recommends that **prior to construction**, Environment Canada ensure that enhanced resolution thermal plume modelling is conducted by OPG, taking into account possible future climate change effects. Fisheries and Oceans Canada shall ensure that the results of the modelling are incorporated into the design of the outfall diffuser and the evaluation of alternative locations for the placement of the intake and the diffuser of the proposed condenser cooling water system.

During Operation

Recommendation # 23 (Section 5.5):
The Panel recommends that Environment Canada collaborate with OPG to develop and implement a follow-up program to confirm the effectiveness of OPG's proposed mitigation measures for bird communities should natural draft cooling towers be chosen for the condenser cooling system.

Health Canada

Over the Life of the Project

Recommendation # 55 (Section 7.1):
The Panel recommends that Health Canada and the Canadian Nuclear Safety Commission continue to participate in international studies seeking to identify long-term health effects of low-level radiation exposures, and to identify if there is a need for revision of limits specified in the Radiation Protection Regulations.

The Canadian Environmental Assessment Agency

General

Recommendation # 64 (Section 8.1):
The Panel recommends that the Canadian Environmental Assessment Agency revise the Canadian Environmental Assessment Agency Cumulative Effects Practitioner's Guide to specifically include a consideration of accident and malfunction scenarios.

The Government of Canada

Prior to Construction

Recommendation # 60 (Section 7.3):
The Panel recommends that **prior to construction**, the Government of Canada review the adequacy of the provisions for nuclear liability insurance. This review must include information from OPG and the Region of Durham regarding the likely economic effects of a severe accident at the Darlington Nuclear site where there is a requirement for relocation, restriction of use and remediation of a sector of the regional study area.

Recommendation # 66 (Section 8.5):
The Panel recommends that the Government of Canada update the Nuclear Liability and Compensation Act or its equivalent to reflect the consequences of a nuclear accident. The revisions must address damage from any ionizing radiation and from any initiating event and should be aligned with the polluter pays principle. The revised Nuclear Liability and Compensation Act, or its equivalent, must be in force before the Project can proceed to the construction phase.

Over the Life of the Project

Recommendation # 65 (Section 8.5):
The Panel recommends that the Government of Canada make it a priority to invest in developing solutions for long-term management of used nuclear fuel, including storage, disposal, reprocessing and re-use.

General

Recommendation # 67 (Section 8.5):
The Panel recommends that the Government of Canada provide clear and practical direction on the application of sustainability assessment in environmental assessments for future nuclear projects.

The Government of Ontario

Over the Life of the Project

Recommendation # 44 (Section 6.2):
The Panel recommends that the Government of Ontario take appropriate measures to prevent sensitive and residential development within three kilometres of the site boundary.

Recommendation # 46 (Section 6.3):
Given that a severe accident may have consequences beyond the three and 10-kilometre zones evaluated by OPG, the Panel recommends that the Government of Ontario, on an ongoing basis, review the emergency planning zones and the emergency preparedness and response measures, as defined in the Provincial Nuclear Emergency Response Plan (PNERP), to protect human health and safety.

The Municipality of Clarington

Over the Life of the Project

Recommendation # 45 (Section 6.2):
The Panel recommends that the Municipality of Clarington prevent, for the lifetime of the nuclear facility, the establishment of sensitive public facilities such as school, hospitals and residences for vulnerable clienteles within the three kilometre zone around the site boundary.

Recommendation # 59 (Section 7.3):
The Panel recommends that the Municipality of Clarington manage development in the vicinity of the Project site to ensure that there is no deterioration in the capacity to evacuate members of the public for the protection of human health and safety.

Ontario Power Generation

Over the Life of the Project

Recommendation # 42 (Section 6.1):
The Panel recommends that on an ongoing basis, OPG pursue its strategy to ensure that Aboriginal students can benefit from the permanent job opportunities that will be available during the lifetime of the Project. In this regard, OPG should collaborate with various secondary and post-secondary education institutions as well as Aboriginal groups to ensure that such programs would be successful.

Annex B: Determination of exclusion zone per US NRC 100.11 (2 pages)

§ 100.11 Determination of exclusion area, low population zone, and population center distance.

(a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release[1] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem[2] or a total radiation dose in excess of 300 rem[2] to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from

simultaneous releases. The applicant would be expected to justify to the satisfaction of the Commission the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

Note:

For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document 14844, dated March 23, 1962, which contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation.

[27 FR 3509, Apr. 12, 1962, as amended at 31 FR 4670, Mar. 19, 1966; 38 FR 1273, Jan. 11, 1973; 40 FR 8793, Mar. 3, 1975; 40 FR 26527, June 24, 1975; 53 FR 43422, Oct. 27, 1988; 64 FR 48955, Sept. 9, 1999; 67 FR 67101, Nov. 4, 2002]

Footnotes - 100.11

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

[2] The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

P R O L E T   I N C

UNMET CHALLENGES TO SUCCESSFULLY MITIGATING SEVERE ACCIDENTS IN MULTI UNIT CANDU REACTORS

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1. SUMMARY

One sees eerie similarities here in Canada to the cozy relationship between regulator and utilities in 'pre-Fukushima' Japan. Such ties are hardly limited to Canada though. The chronic degradation of real commitments to continued improvements in reactor safety systems and a decline in overall safety culture that discourages critical design reviews and wilfully ignores well justified, safety critical hardware upgrades, has created alarming conditions that are likely inching us towards another nuclear disaster. Operating CANDU reactors are now close to being obsolete but have barely seen any substantive severe accident-related risk reduction upgrades nine years after Fukushima, hoopla in Canada around some minor improvements and premature closure of even otherwise sparse and what were really weak regulatory *'Fukushima Action Items'* notwithstanding.

With a number of common barriers to fission product releases to environment missing or weak, one would expect the regulator to be extra vigilant in promoting prevention and encouraging delays in onset of core damage. On the contrary, it has only made matters worse by its collusion & obfuscation as long summarized in [1] and even denying the additional burden of age-related degradations as in long operating licenses 50% longer than design life at Pickering [2]. Whether the regulatory actions are out of ignorance, inability or intent is debatable but equally disturbing.

The multi-unit CANDU stations sport some of the weakest and leakiest containments in the world. With no reactor pressure vessel to isolate the overheating channel and debris, these leaky containments will directly see un-attenuated fission products releases from the fuel. They will trap combustible D_2 gas in interconnected from below inverted cup like crowded reactors vaults to an increased gas explosion potential. The reactor units have high steam and air oxidation potential on both sides of over 10 km of low carbon steel feeder piping with over 1800 m^2 hot surface areas exposed for each of internal steam and external air oxidation and copious amounts of core Zircaloy (> 50,000 kg, twice of that in a BWR of similar power).

Combustible gas detection and mitigation systems are designed for Hydrogen (H_2) instead of Deuterium (D_2) gas in these D_2O cooled and D_2O moderated PHWRs. The pressure relief systems in primary cooling and moderating systems are dangerously inadequate, resulting likely in pressure boundary ruptures and early containment bypass, accelerated onset of core damage and vessel failures. Backup diesel generators are located low and close to water as in Fukushima. Spent fuel pools are overcrowded with horizontally stacked fuel bundles akin fish in fish-baskets. Yet, the emphasis has shifted to passing wishful thinking of low off-site releases [3] and convenient half-truths of an early core collapse terminating further core degradation and releases into containment as facts and ignoring [4] known design vulnerabilities that amplify risk actively denying [5] even the basic science on high temperature oxidation of carbon steel [6].

Even more dangerous are the unsubstantiated claims being made of near impossibility of off-site releases of long-lived species from these multi-unit reactors by utility management [7] without nay a challenge by the regulators. The life management issues of ageing, elongating, thinning, hydriding, embrittling and deforming CANDU Pressure Tubes is yet to be resolved but these obsolete reactors keep getting ever longer license extensions (e.g. for 10 more years, over 50% beyond original Pickering pressure tube design life - ignoring their own data [8]that suggests that safe operation cannot be guaranteed due to elongation. There are loud, ambiguous references to compliance with un-named IAEA documents and standards. No IAEA document has yet identified or discussed the PHWR design vulnerabilities that may lead to disastrous outcomes and this paper is repeating in forums akin ICONE for the n^{th} time. Of equally great harm to risk reduction are the IAEA team of experts missions (Integrated Regulatory Review Service (IRRS) follow-up missions - for example [9]that issue oversight certifications / seals of approval to the Canadian regulator CNSC without anything resembling a technical evaluation of CANDU design elements that contribute to risk.

Many critical vulnerabilities and proposed engineering fixes that can be undertaken to overcome also been highlighted routinely [10] but are groundlessly rejected as in [11] which begs for an international impartial scrutiny in ingrained obdurate industry intransigence against changes and investments into substantive safety improvements and risk reduction. Emergency preparedness by civil authorities has been illogically conditioned for the smallest possible 'Large Release' source term (of e.g.100 TBq of Cs-137) and available response time for mitigation measures have been exaggerated baselessly. Both acts are irresponsible and dangerous to public and first responder safety. A number of early mitigation measures to externally replenish boiler inventory (a measure common to all PWRs) will not work due to an unusually low, below core, placement of pressurizer that will gradually gravity drain much primary coolant from boiler tubes. So the most important emergency measure to restore core cooling by reflooding boilers to induce natural circulation flows will go to waste. Operators will never know why the core never cooled.

Inability of the utilities to accept responsibility for reactor upgrades and inability of the regulatory management to act independently are the signs of impending implosions in our nuclear industry. It is likely because the regulatory body CNSC is critically dependent upon the licensees financially in a 'cost recovery' plan. Not likely, but perhaps if we get lucky, an impending disaster can be avoided by a return to the first principles, and not mere slogans, of 'safety first'. Right now, an unmitigated station blackout in a CANDU multi-unit station will make the Fukushima disaster look like a walk in the park.

2. INEXPERT REGULATORS AND DESIGN OBSCELENCE

The long ignored severe accident-related design deficiencies, inability to safely, successfully withstand a simple accident such as a station blackout for a reasonable amount of time are amongst many unmet challenges that multi-unit CANDU reactors pose to public safety and very directly to the utility corporate health as well. It is not just that the reactors are now obsolete and were not designed with severe accidents in the design basis so as to make severe accident management predictable and severe accident consequences manageable; it is also that the utilities will do only the minimum they are required to do and that the regulatory body is also neither independent nor technically competent, especially in the field of severe accidents. As a result, a strong culture of privately or silently agreed obfuscation has emerged. Public safety has become secondary to corporate need for uninterrupted power production & regulator's need to exist in significant denial of lessons of Fukushima.

Almost none of the operating 400 odd nuclear power reactors incorporated severe accidents within their design basis. So, all multi-unit CANDU reactors, just like their single unit counterparts and most all operating LWRs share some of the same vulnerabilities to onset of severe core damage accidents. They also share their inability to adequately avoid severe core damage early, incorporate enough passive systems to delay its onset, provide adequate means of early arresting their progression, provide ample opportunities to successfully apply external resources to accident management, include enough design margins to reduce releases into the containment and have strong and tight enough containments to keep the accident source terms from releases by leaks, over-pressurization or explosive outcomes. While LWRs also are of a vintage design and vulnerable to severe core damage, not all have taken the path of denial. Manly utilities, like with NRC's State-of-the-Art Reactor Consequence Analysis Project, are doing a much job of better critical self-examination and risk reduction. Overseas CANDU utilities cite Canadian CNSC actions to justify their inaction and apparent lack of technical expertise.

Detailed technical analyses including sophisticated computer simulations reveal that many of the severe accident related vulnerabilities of multi-unit CANDU PHWR design at Darlington (4 units) Bruce A / Bruce B (4+4 units) and Pickering (8 units) reactors are common with single unit CANDU reactors in Canada, Korea, Argentina, India, Pakistan, Romania and China. A number of inadequacies in severe accident mitigation capabilities are also shared with LWR designs of the same vintage. As discussed in a number of earlier papers on the same issues [12], an evaluation of a station blackout (SBO) accident at the multi-unit Bruce, Darlington and Pickering CANDU stations reveals significant challenges to accident management options. There, however, are easily identifiable indicators and sources, instigators of potentially unacceptable off site radiological consequences as well as engineering fixes to reduce risks. It is unfortunate that only another severe core damage accident will likely force the required change. Right now, the Canadian utilities have the regulator CNSC in a firm capture and are in no mood for a serious dialogue on the topic, irrespective of risk or consequences. It is hoped that professional forums such as ICONE and public awareness will propel the regulators and/or utilities into action.

Design analyses and numerical simulations reveal that opportunities for design improvements and alternate mitigation measures are abundantly clear for certain challenges and not so much for others. But in all cases regulators and utilities reject them in their preference for wild and untrue claims of easy operator actions to bring the reactor under control and benign severe accident consequences even without any operator actions. The regulator has put out glorifying videos without doing any analyses and accepted utility submissions without any meaningful critical technical reviews. These evangelical pronouncements of eternal and near absolute safety in the presently operating, albeit of obsolete design reactors, portray severe core damage accidents in a distorted positive light in defiance of engineered realities (by claiming physically impossible long times to bring in emergency equipment - [13] that claims 5 hours for boilers as heat sinks instead of likely 1 hour when an engineering analyses is undertaken [14]and in defiance of expected professional integrity in ensuring public safety (by claiming extremely low releases of ~ 100 TBq of Cs-137 instead of likely 30,000 TBq from leaky, weak containments without ever doing <u>any</u> numerical analyses or modelling - [3]).

The CANDU PHWRs concept started in 1950s with a 22 MWe Nuclear Power Demonstration (NPD) going critical in 1962 and a first full scale power plant at Douglas Point (220 MWe) in 1966. The 600 to 800 MWe units first entered commercial operation as multi unit power plants in 1971.The basic design of twelve or so, 10 cm diameter 50 cm long fuel bundles in about three to five hundred horizontal Zircaloy pressure tubes within a thermally isolated low pressure, low temperature D_2O moderator has not changed much over these 60 years. Improvements in rolled joints, end fittings and pressure tube materials have increased their reliability but the degradation of Zircaloy pressure tubes has required previously unforeseen 'mid-life' replacements and extensive 'refurbishments' which involve removal and replacement of very radioactive core structural materials and have typically cost more than the original plant did. All units are at the end of their design life, under 'refurbishment' to replace degraded core components (pressure and Calandria tubes, feeders and boilers) or already rebuilt back to the original specs of 1960s and 1970s.

Further development of the CANDU technology has since been almost abandoned in Canada with the design organization AECL, into which literally billions of dollars were invested by the Government of Canada to develop the CANDU reactor concept, was sold with most all its assets minus the liabilities to a private company SNC-Lavalin for the price of a well used corporate jet and all future plans have now shifted to commercializing the so called Small Modular Reactors with renewed promises of riches and safety. The national regulator CNSC is playing the bandleader once again. Attention has shifted away from the high risk obsolete multi unit reactors at Bruce, Darlington and Pickering with their long term licenses in the utility pockets with risk reduction opportunities of no immediate interest to anyone. Unless of course, if we engineers recognize the disservice this does to our future and force them to act in public interest alone.

What has changed over these 60 years is our understanding that these reactors, like all others of that vintage, were more complex than other power reactors and were certainly not designed with core damage accidents within the design basis. Some have been thankfully taken off service at the end of their life (Gentilly-2) or earlier (Gentilly-1, Pickering A units 2,3) while one at Wolsong 1 was removed from service after it was refurbished with new reactor internals at great expense but did not satisfy safe operating envelope expectations. In Canada, Gentilly-2 single unit CANDU was wisely retired after its' design life. It was not the regulator that initiated its closure; it was the utility that did not particularly need the associated risk.

3. CONTRIBUTIONS BY THE NATIONAL REGULATOR

The challenge to public safety is further exasperated by a diminishing safety culture at the regulatory body CNSC that glorifies the obsolete designs, disregards known safety issues and discourages real public discourse and input from outside the regular payroll of the industry[1]. It also spends inordinate times in self adulation and is looking more like a public relations arm of the utilities it is supposed to regulate.

The Canadian regulator CNSC has taken the lead in producing misleading information about CANDU severe accident progression [13] and its consequences [3]. Reactor vulnerabilities have been ignored in defiance of basic science by siding with corporate interests that have had the regulators in firm capture for over a decade. This behaviour is in stark contrast to the practices south of the border where rule based regulations are more the norm; rules are scientific fact based and comprehensive analyses and supporting research are routinely commissioned. A comparison of CNSC generated claims in [3], [13]is instructive with reports such as 'The State-of-the-Art Reactor Consequence Analyses' (SOARCA) project [15] that were undertaken by NRC to systematically summarize accident progression pathways and mitigation strategies with actual numerical analyses using state of the art computational aides without resorting to hyperbole on one hand and artificially generated fog as in [3], [13]on the other.

The continuing insistence by the regulator to be the bugler for an obsolete technology that it explicitly says need not be comprehensively, systematically rejuvenated before its further exploitation and claims in its reports that severe accident consequences are nothing more than benign - are all appalling facts. When the issue of high oxidation potential of feeders, the carbon steel pipes downstream of hot fuel, their first reaction was that feeders cannot get warm and hence any issues of carbon steel oxidation were humbug. The regulator has even told the local emergency management organizations that the worst off-site releases after a severe accident are expected to be as minimal as total releases of 100 TBq of Cs-137 (and other species in proportion) which is from about 0.15% of the fuel and that health effects of a severe accident would be benign. This pronouncement was not based on any analysis but was camouflaged under words deceptively implying that specific analyses for the worst accident without operator intervention were undertaken. This has emboldened the utilities to do practically nothing meaningful to reduce residual risk and push for even longer operating licenses well beyond the original design life of plants whose materials degrade faster than in any other reactor with age (Zircaloy pressure tube thinning, elongating, thinning and increasing in diameter with creep, hydriding and being replaced prematurely at exorbitant costs) and normal exploitation (e.g. thinning of carbon steel feeder pipes that connect the fuel channels to pumps and boilers).

Given the unexpected nature of any accident and severe potential for extreme damage to the environment if the accident results in severe core damage as in a sustained loss of heat sinks after a station blackout as in Fukushima, one would imagine that the regulators would be insisting and utilities would be installing proper measures to reduce the likelihood of occurrence of multiple failures that can lead to severe core damage; and incorporating measures to identify, control, manage and arrest the progression of the accidents early; and ensuring measures to contain the consequences to within the reactor units and most of all, accepting the limitations of the technology and their understanding of it to invest in fundamental research. None of that has happened to a degree consistent with needs. As a result the reactors today are not much better able to mitigate severe accidents than they were before Fukushima and before the shiny pumper fire trucks were bought to provide low pressure heat sinks, filtered containment venting systems installed and a few symbolic but dangerous hydrogen recombiners scattered around the plants. These measures are poorly thought out and even more poorly executed with the large number of other vulnerabilities largely unaddressed. Of course such behaviour has consequences.

The official report of The Fukushima Nuclear Accident Independent Investigation Commission concluded in part that:

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

and actions, rather than issues relating to the competency of any specific individual."

It has become very clear that the situation in a number of countries is exactly the same as summarized above for Japan in 2011. While this paper concentrates on the issues arising out of multi unit CANDU PHWR operation in Canada, the path taken by the regulators in other countries with CANDU reactors is not much different. After 3 decades of severe accident progression and consequence assessment evaluations and trying to get the industry to recognize that the reactors do not meet the evolving public expectations of risk, it has become apparent to me and many others that a combination of design weaknesses, corporate intransigence, and regulatory weakness has come together in a form that is detrimental not only to public safety but also to the future of nuclear power. The regulators have recently bestowed on the multi unit reactor utilities unprecedented 10 year license extensions, in some cases in defiance of overwhelming evidence that these reactors pose large risk under SBO accident conditions not dissimilar to Fukushima.

I will make my point by first discussing the design specifics that have cried out for new and innovative mitigating measures as our understanding of severe accidents have matured and then pointing out the specific decisions made by specific people in the Canadian nuclear industry to put the issues under the rug. As a nuclear safety engineer with over 30 years of work in nuclear safety and as one who has developed a dozen computer codes to model accident progression in CANDU reactors, including the CANDU specific parts of the now obsolete MAAP-CANDU code that the industry still uses to analyze severe accidents, I consider it my ethical duty to present the arguments in favour of stepping our game up to meet the unmet challenges to successfully mitigating severe accidents in multi unit CANDU reactors or shutting them all down in interest of public safety and security. In interest of clarity I will use the multi unit reactors at Darlington and Bruce in Canada as examples, although the malaise of poor severe accident mitigation permeates to all CANDU / PHWR units in all countries.

4. CANDU DESIGN VULNERABITY IS NOT A NEWLY DISCOVERED PROBLEM

A number of red flags have been raised over the years and a systematic design evaluation has uncovered a long list of vulnerabilities that make severe core damage accident consequences from multi unit reactors alarmingly unacceptable. The response of the Canadian nuclear industry has varied from silence to outright lies and bullying. The Canadian national regulator has taken the lead in spewing technically impossible positions on severe accident consequences [3] and the collusion between the industry and the regulators has deteriorated progress in resolution to such an extent that the latest position from a utility Bruce Power VP during relicensing hearings is that they will soon see no conditions under which these reactors will release any long lived isotopes following a severe core damage accident [7] and hence a reduction in planning zones is to be in order . This for a design that has a containment unable to be tested above 0.45 atmospheres and a leak rate at design pressure of 2% per hour (500 times more than at a light water PWR such as at Surry), not to mention the other design features that make these multi unit reactors un-licensable in any other jurisdiction in the world. The same VP smugly claimed that Bruce Power adopted new standards faster than others and in special contrast to overseas utilities that would do so only every 30 - 40 years on relicensing. The regulator CNSC similarly makes claims of being the 'world leader' in safety regulation. Their mutual admiration is evident in transcripts of public meetings where the two practically finish each other's sentences. CNSC has also quietly sidestepped its own already watered down regulations to allow the utilities to pressure test their containment every 12 years [16] instead of the already unusual for a nuclear reactor containment leakage test frequency at full design pressure of every 6 years per the CNSC regulatory guide R7 [17].

5. CANDU REACTOR DESIGN VULNERABILITIES TO UNSATISFACTORY OUTCOMES AFTER STATION BLACKOUT

CANDU PHWRs suffer from a number of vulnerabilities to unacceptable outcomes after severe accidents and a number of design features that accelerate failures or exasperate the accident consequences and hence risk to public. Some of these are specific to the D_2O cooled and moderated horizontal fuel channel concept just as the RBMK is with its vertical boiling light water cooled, hot graphite moderated fuel channels. While utilities and the national regulators have long sung the CANDU design praises, some fundamental CANDU vulnerabilities cannot be rectified for existing reactors. For example, absence of a pressure vessel around the core will always directly expel activity into the containment once the channels experience structural damage. Thus the leaky containment becomes the only barrier to release of activity.

The strange choice of pressurizer location below the boilers and reactor headers (in 12 reactor units at Darlington and Bruce stations) will cause draining into it of primary coolant from boiler tubes in a SBO to an extent that boilers will become useless as heat sinks and no amount of emergency measures to add water to boilers will restore cooling to the reactor core unless the primary cooling system was replenished as well, something that cannot be done after an SBO in the present design and presently configured SAMGs.
The low pressure retention capacity (<<0.9 bar) of the rectangular slab industrial buildings that surround the reactor cores and their design leak rate at 2%/hour which is 480 times greater than the 0.1%/day in modern PWRs will always make the containments ineffective repositories of fission and activation product activity put unfiltered into them from the disassembling fuel channels and also make them traps for combustible Deuterium that will come out of the same path into inverted cup like inter-connected rooms called reactor vaults that surround the reactors. Gas explosions in any one reactor vault will cause a huge containment bypass.

What is fundamentally disturbing is that certain long well known design features that may cause an unwarranted pressure boundary failure (*because the primary heat transport system (PHTS) overpressure steam relief capacity is too low*) or accelerate onset of core disassembly (*such as an un-necessary, forced expulsion by flashing of a critical amount of moderator upon onset of boiling because rupture disks actuate instead of a controlled relief through relief valves*) or a lack means of direct depressurization of PHTS or cause the containment to leak profusely at relatively low pressures have not been accepted or rectified. Certain challenges to the containment integrity, such as from high amounts of hydrogen and deuterium produced by oxidation of outside and inside surfaces of feeders after a core damage accident of LOCA+LOECC have been ignored for even design basis accidents without the regulator ever highlighting the omission or recognizing that Carbon steel is more oxidation reactive than Zircaloy at all temperatures.

Potential for steam and air oxidation to produce copious amounts of combustible Deuterium and Hydrogen from the large amount of Zircaloy and carbon steel associated with the fuel channels during a severe core damage accident is easy to see. There is about 50,000 kg of Zircaloy with an oxidation surface area of over 12000 m^2 and over 120 tons of low carbon steel piping over 10 km long with a surface area greater than 1800 m^2 in a Darlington CANDU PHWR associated with fuel channels where a loss of cooling can elevate fuel temperatures such that rate of oxidation of feeder carbon steel will always be greater than that for Zircaloy(Figure 222) and the amount of Deuterium produced by oxidation of steel will exceed that from Zircaloy very early (Figure 3).

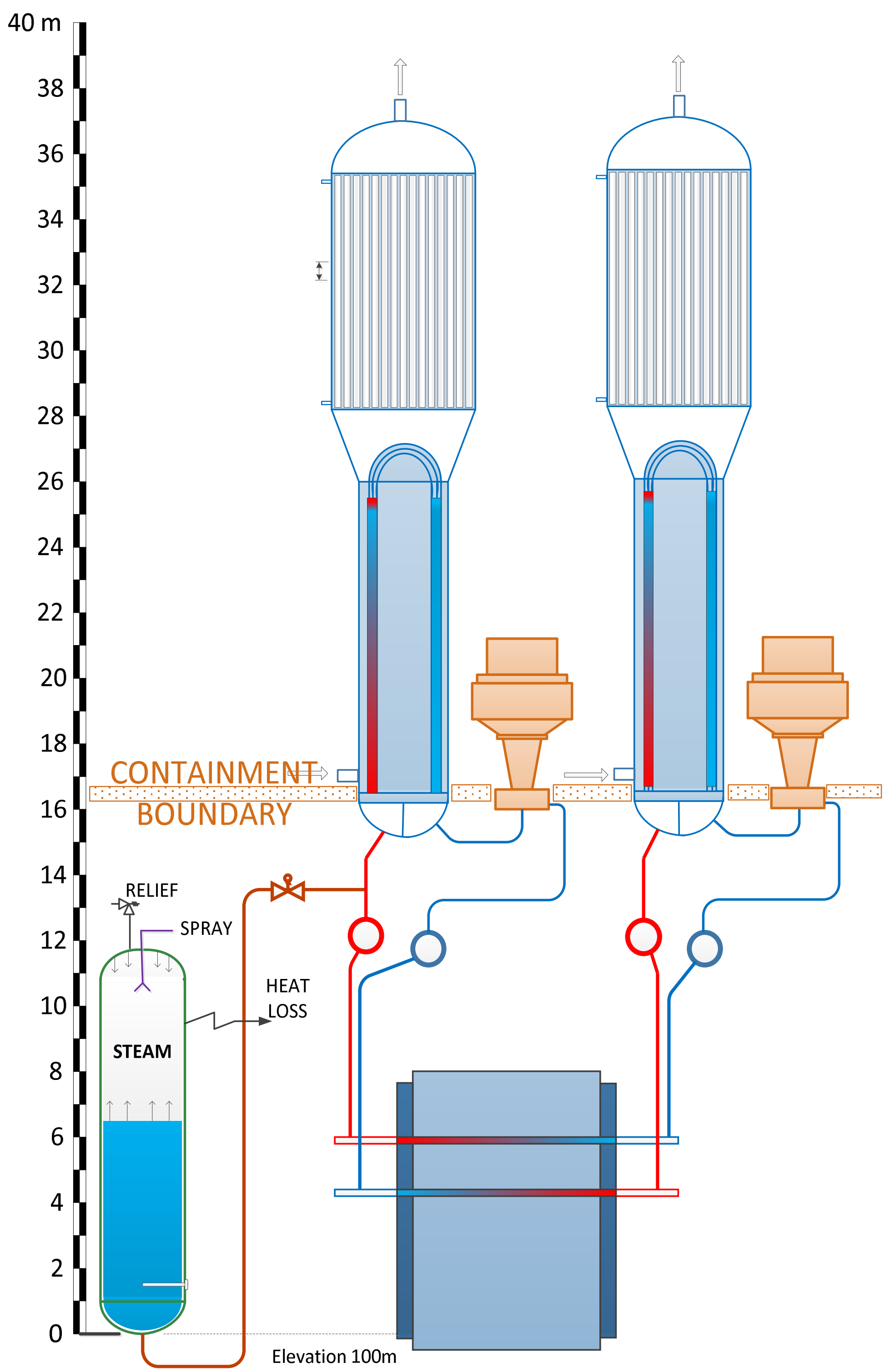

Figure 1 : Lower than core placement of pressurizer that will drain boiler tube inventory at Darlington/Bruce reactors in SBO

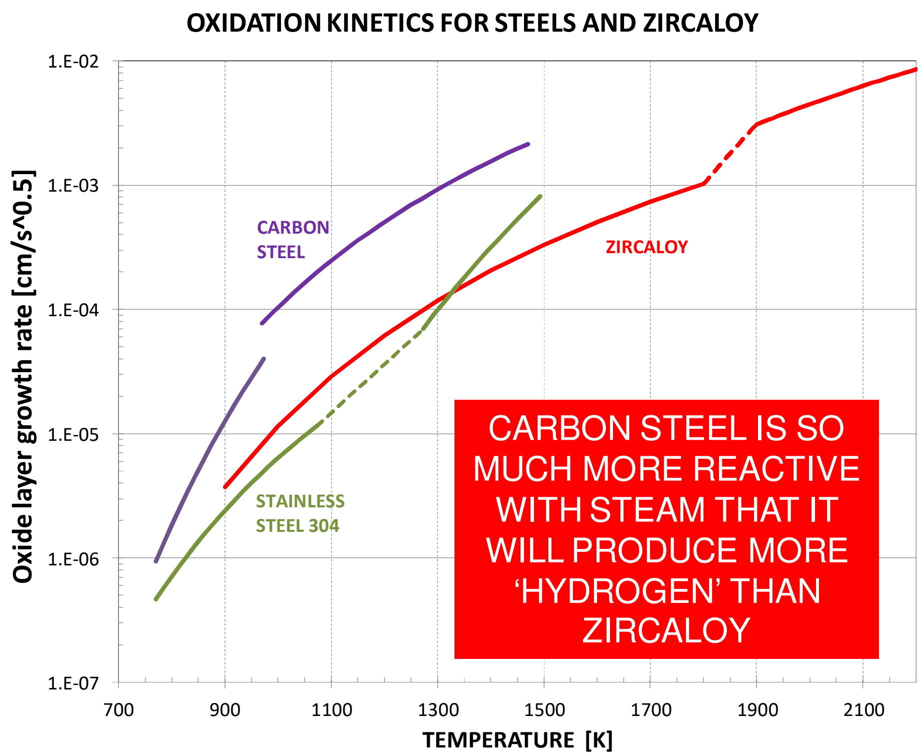

Figure 2: Oxidation kinetics for Zircaloy, carbon steel and stainless steel

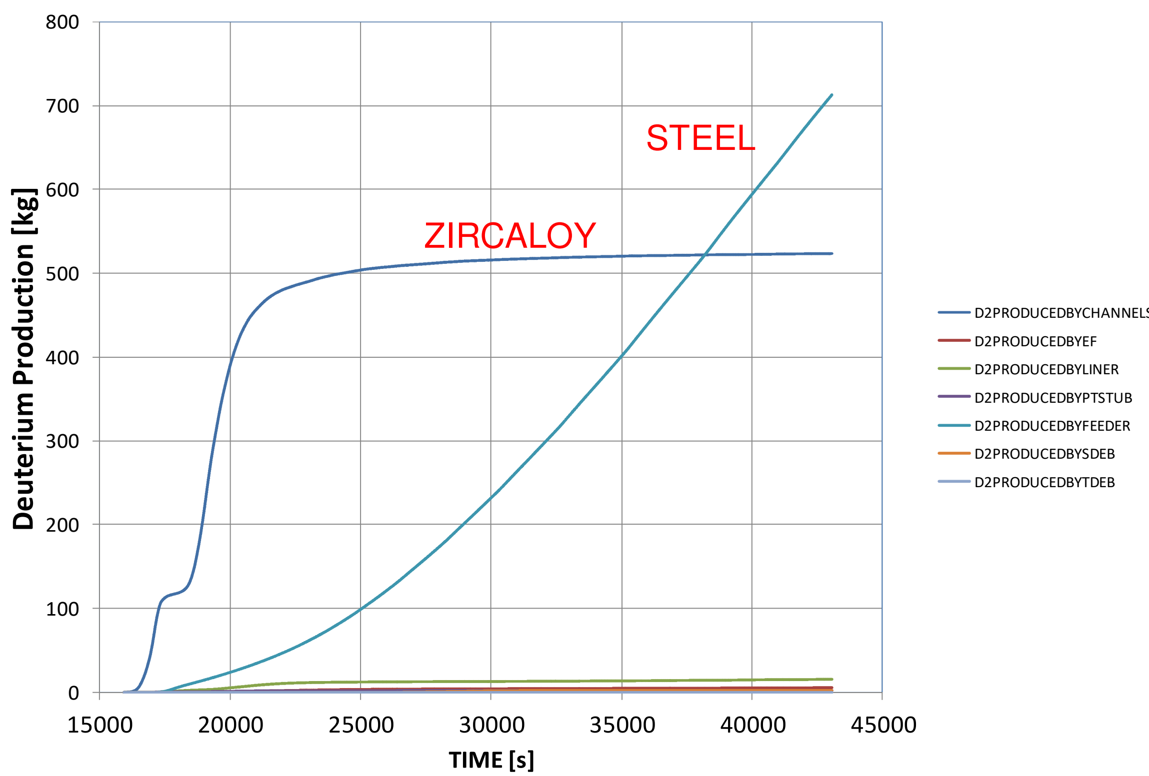

Figure 3 : Sample results for the first 12 hours of combustible gas production in a 600 MWe single unit CANDU

A station blackout (SBO) scenario which with its sustained loss of engineered heat sinks represents a large number of accidents with other initiating failures and is a representative scenario undertaken for all reactors worldwide to assess effectiveness of engineered passive systems that may come into play and of opportunities for emergency mitigating measures.

Engineering analyses reveal that reactor risk profiles are in the alarm territory for a number of very obvious reasons. Early passive heat removal by steam generators after a station blackout is not only short lived (~1.5 hours as opposed to claimed 5 hours) but can also be compromised even earlier by primary coolant from boiler tubes getting drained into a large cooling pressurizer located well below the boilers and the reactor core. (Figure 1, for Darlington). Thus any delay in restoring secondary heat sinks and primary inventory drained by gravity into pressurizer may make the boilers irrelevant and ineffective.

Over-pressure protection systems on main core cooling system is indirect (goes through another vessel and requires two sets of valves in series to successfully actuate) and functionally inadequate to satisfy the heat load. On a loss of boilers as heat sink, a steam relief through relief valves is the only heat sinks for decay heat but the PHTS steam relief capacity is only ~20% of decay heat equivalent, let alone for other anticipated severe accident loads [18]. This can likely cause an early over pressure failure and hence a containment bypass by steam generator tube ruptures or another uncontrolled pressure boundary rupture, something that the ASME code or common engineering sense would require that not ever happen. This very fundamental error in design has been known to the utilities for 20 years without resolution or nay an understanding of its consequences. When an industry starts accepting a pressure boundary failure as an acceptable outcome rather than re-engineer the safety valves, it is time for that industry to shut operations or as an ex NRC chairman Gregory Jaczko put it 'is Going Away'[19]. As a nuclear engineer with great confidence in my peers, I find such a direction and such an outcome for my industry also likely but otherwise unacceptable.

Inability to manage a loss of heat sinks accidents is exacerbated by handicaps like no external emergency means of high pressure water addition to the heat transport system. Any addition of emergency coolant requires that boilers be manually depressurized successfully first for the PHTS to be hopefully, indirectly depressurized. A manual depressurization of boilers is actually an operator assisted process of forcing the relief valves to stay open in a process that forcibly removes a third or so of the boiler liquid inventory by flashing and dumps it into the atmosphere without a foolproof guarantee that any subsequent action to replenish the same inventory would be successful. A high pressure makeup feedwater injection with a passive steam driven turbine would have easily solved both problems without breaking a sweat. This has been the logical backup solution at a number of PWRs but the CNSC brass totally trashed the idea a number of times citing some unrelated steam turbine failures at Fukushima. A steam driven auxiliary feedback system is as passive as they get. In fact one has been at the single unit CANDU at Pt. Lepreau forever. The issue is really not the merit of this or that solution to the various vulnerabilities in multi unit CANDU stations; the issue is the attitude and a collusive decision to do absolutely nothing more than what little they have done, even if the decisions such as low pressure pumper fire trucks to add water to the boilers is now recognized in private conversations to be not the wisest one.

The current SAMGs erroneously credit gravity feed of water into the boilers after their depressurization through the feedwater train from de-aerator. This will really not work. Flashing of the ~160ºC water inventory and unavoidable high pressure back leakage of boiler inventory through the check valves would vapour bind the feedwater flow path. In addition, there will not be enough driving force to open and then keep the feedwater check valves open.

With more and more channels losing their heat sinks and dumping their decay and chemical heat into the moderator, onset of moderator boiling causes the rupture disks on the Calandria vessel to open up, creating a direct path for release of steam, fission products, hot and combustible gases into the inverted-cup reactor vault over the common duct in the containment.

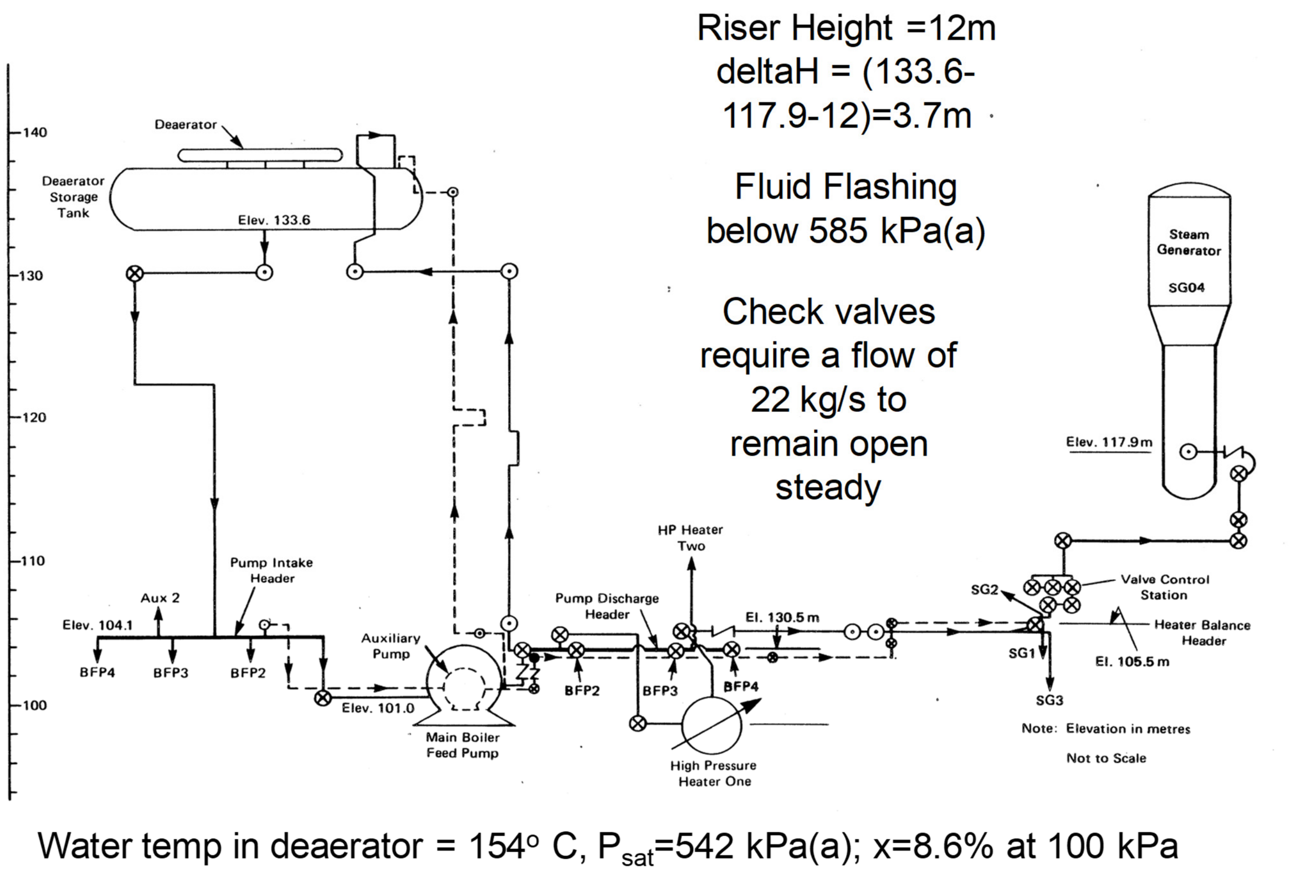

Figure 4 : Looking into gravity feed into the boilers from de-aerator

This happens as a lack of a decay heat level controlled steam relief on the Calandria vessel (which contains the moderator) accelerates severe core damage by ejecting a significant amount of moderator when it becomes the dominant heat sink for fuel channels, boils and causes the rupture disks on its large piping to burst to eject water by flashing and carryover. By ignoring the extra 30-60 min such a design omission subtracts from time to onset of severe core damage, the industry reinforces its intransigence and inability to think through that public safety supersedes all other considerations.

An important claim by the industry on debris creation and potential retention of 'melt' in the Calandria vessel is examined below:

Disassembly of a reactor fuel channel is its partial breakup into single or multiple bundle length pieces and in some cases even separation from the rolled joints at end shields. Breakup into pieces can occur when both the primary coolant from inside of the fuel channel and the moderator coolant from outside the fuel channel are depleted and the pressure tube perforates with it being unable to sustain the weight of fuel within or above it. Fuel channels begin to heatup individually once they are devoid of coolant and the moderator becomes the sole heat sink. A widespread core damage accident in a CANDU would only occur gradually because of the large variability in the inventory of water associated with each channel, variability in channel powers and variability in time at which the moderator outside each channel may drain or be boiled off. In all cases a fuel heatup to temperatures high enough to cause the pressure and Calandria tubes to deform and perforate are required and disassembly of different channel segments would take a finite time and with a finite stagger between channels.

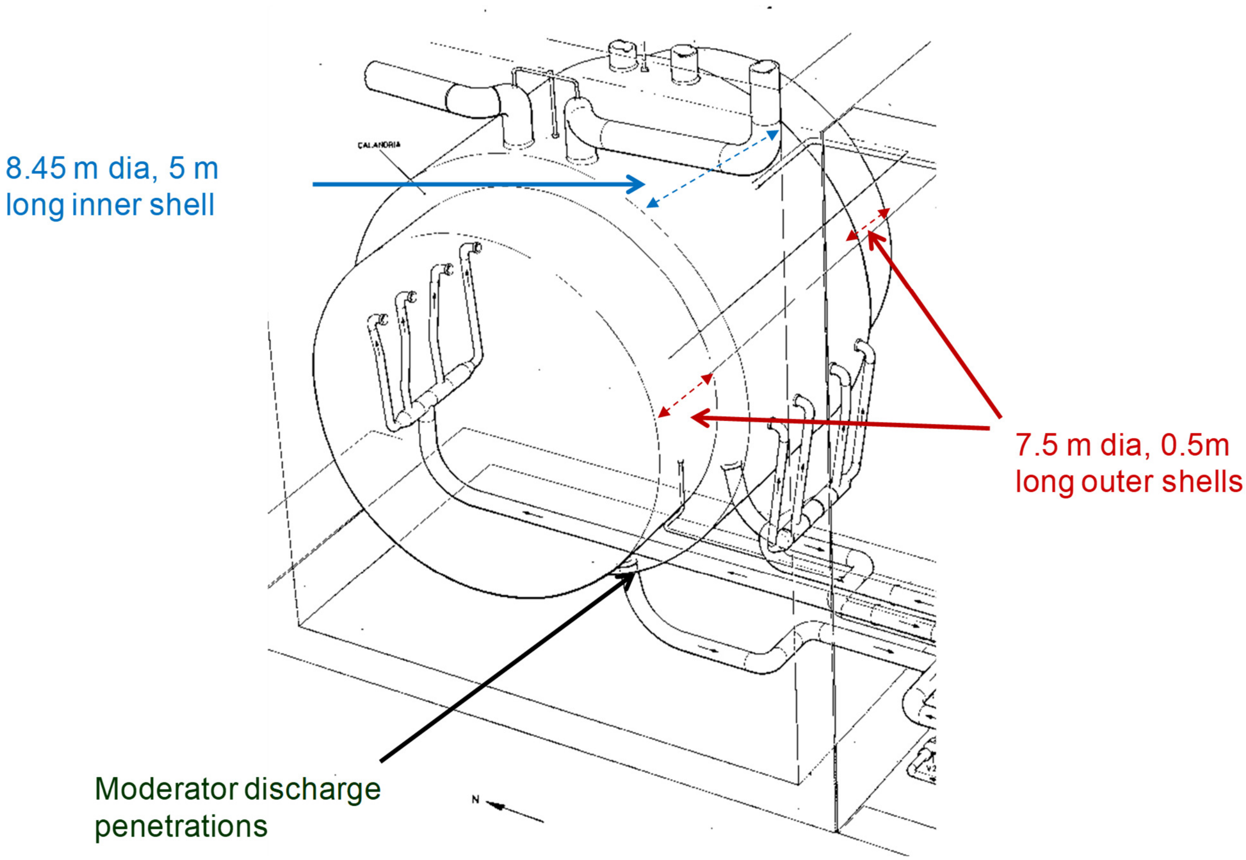

Figure 5: Calandria vessel, a stepped stainless steel vessel with welded annular plate

Accident termination by retention of molten core debris in a vessel has been adopted from PWRs without consideration of the design specifics of the stepped low pressure and thin CANDU moderator vessel. The debris formation in a CANDU reactor is in solid chunks of fuel channel and its eventual retention upon Zircaloy melting in the Calandria vessel cannot be guaranteed as the relatively thin walled stepped and welded vessel (wall thickness varying between 19 and 28 mm) may fail at welds by thermal loads long before any gross melting thus violently introducing water from the shield tank onto hot debris.

Any claims of an LWR like in-vessel retention of molten uranium debris are not credible or consistent with the gradual core disassembly of CANDU cores in case of a station blackout scenario with a sustained absence of heat sinks. The Calandria vessel has a wall thickness that varies between 19mm at annular plates to 28 mm in main shell. The weld failure upon differential expansion of the two shells, with outer shell constrained, is easy to demonstrate (Figure 6 and Figure 7).

The effect of Calandria vessel weld failure can vary from additional hydrogen production, accelerated fission product releases as one mode of outcome for small weld cracks and slow leaks, to catastrophic vessel failures by energetic interactions of incoming water with the hot and molten solid-liquid debris at the bottom of the Calandria vessel as the other mode.

As a result of absence of a retaining vessel, direct un-attenuated releases into the containment, weak containment structures and significant likelihood of energetic interaction of hot debris with water and Deuterium burns /explosions causing challenges to containment integrity, large releases of radioactivity from failed containment structures are inevitable.

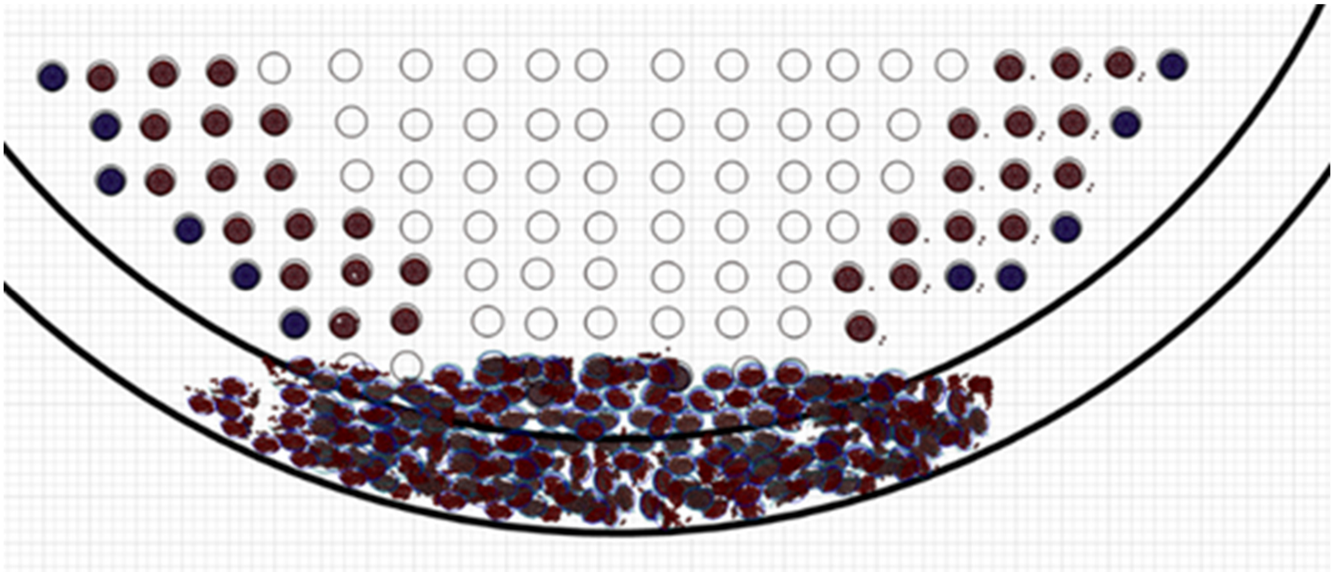

Figure 6 : Likely state of debris upon Calandria weld failure

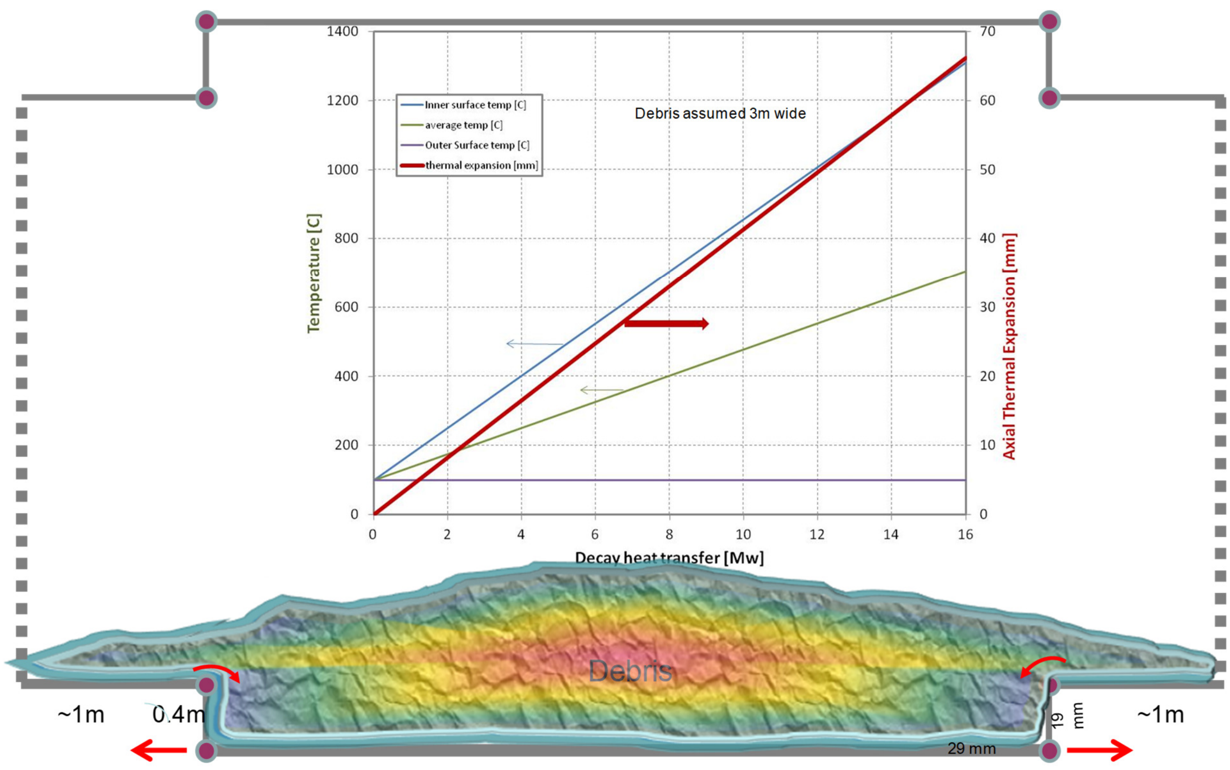

Figure 7: Calandria shell elongation as an indicator of stresses that will cause weld failure

6. WEAK AND LEAKY CONTAINMENT STRUCTURES

In all cases in absence of a retaining LWR like pressure vessel, the disassembling channels would continuously and over many hours release fission products without attenuation through the rupture disk pipes and directly into the box like containments (Figure 9) that are at 48% per day design leak rate at design pressure very leaky and at less than 1 bar design pressure, structurally weakest of all operating reactor containments (typical PWR building design pressure is 5 times higher and leakage at design pressure is 480 times lower).

Another containment bypass potential is in high temperature disassembly of in-core devices along with hot channels. Recall that the in-core device controllers and drives are outside the containment on the reactivity deck. So certain release of fission products onto the reactivity deck cannot be avoided once these devices heatup and melt.

The reactor buildings around each individual reactor core are inverted cup like traps for combustible gases (Figure 8). A large number of safety significant components like the steam generators, pumps and the reactivity control devices are all outside the containment envelope and vulnerable to failures by external impact or otherwise of the weak structures on top of the reactivity decks. These are some of the vulnerabilities that can be fixed.

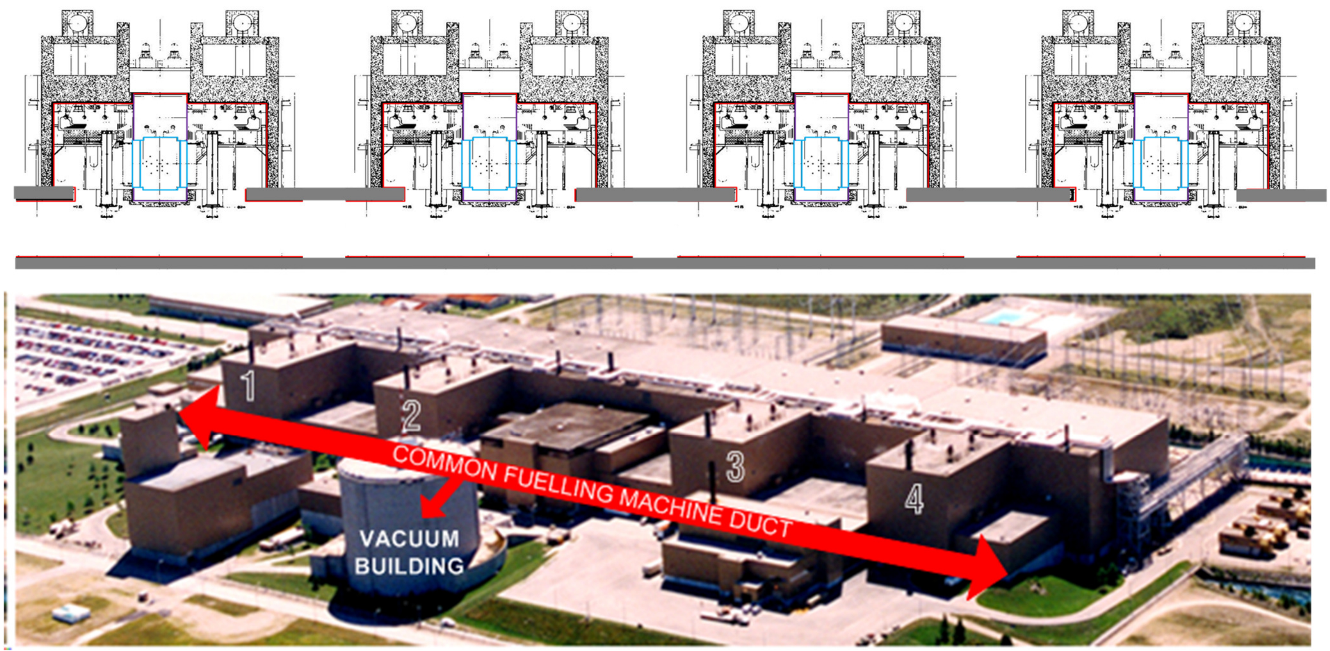

Figure 8: Multi unit layout at Bruce station

7. REACTOR VAULT A TRAP FOR HYDROGEN

The containment layout is such that even a 1% oxidation of any of these materials will cause stagnated and explosive pockets of combustible Deuterium and Hydrogen in reactor vaults shaped like interconnected inverted cups. The reactor vault is the direct recipient of products of reaction with hot fuel as the moderator relief pipes vent into the reactor vault.

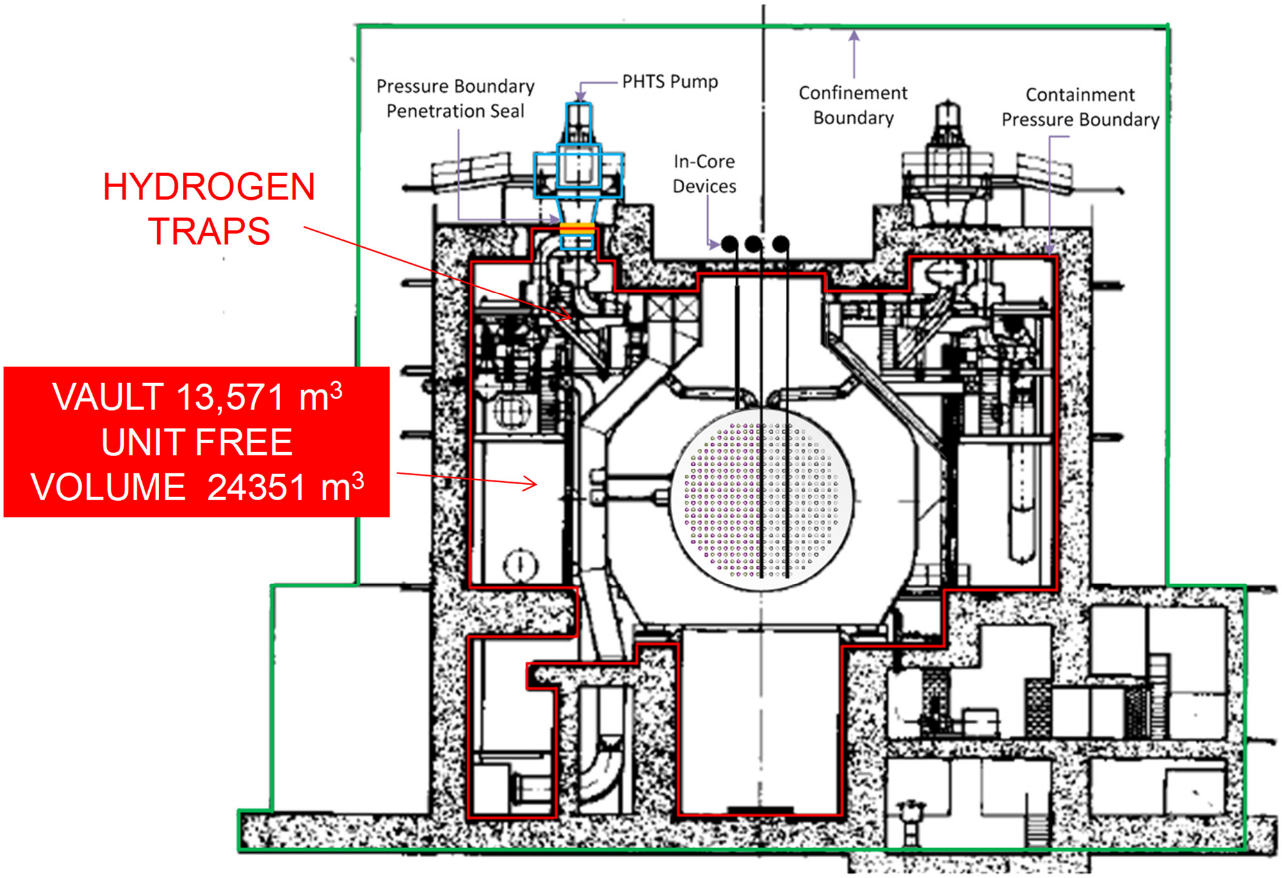

Figure 9: Single unit reactor assembly in a crowded vault with common fuelling machine duct below

The production of combustible Deuterium gas from over ten km of carbon steel piping and over 50 tons of Zircaloy in each Darlington / Bruce unit can be extremely high with steel oxidation more problematic over the longer term; making the installed numbers and types of PARS not only inadequate but as early ignition sources also dangerous.

These conclusions are based on thirty years of working on severe accident related issues at CANDU reactors, conducting extensive design reviews and developing integrated computer codes (MAAP-CANDU [20] and ROSHNI [21]) and supporting numerous analytical methods for PHWR accident progression and consequence assessments.

It was hoped that open discussions by professional engineers would foster change in name of public safety. That has not happened for a number of reasons, allegiances and self-interests. It is now feared that nothing will change unless an accident occurs and an ensuing national inquiry unveils a naked collusion between the regulator and the utility as in Japan prior to Fukushima. A lax and uninformed regulatory regime blindly supporting an intransigent industry resisting basic design enhancements has further exasperated, like it did in Japan, the severe accident related risk from continued operation of these reactors.

It is unfortunate that assumptions in evaluations of accident progression are made by the industry and the regulatory bodies acting in unison that make the accident consequences look benign. One such assumption of a 'core collapse' due to disassembly of higher elevation channels is essentially a numerical trick that gives a false impression of the whole core suddenly falling into cold water in the moderator and ceasing to emit radiation for 8 hours or so. As a result the accident consequences can actually be engineered to look benign.

Actual improvements after Fukushima are perfunctory and the analytical methods in support of severe accident management procedures are outdated and incomplete. A widely used computer code MAAP-CANDU (developed 25 years ago by this author on the MAAP LWR code) is incapable of providing the source terms required to evaluate containment response, design mitigation equipment or off-site releases. I developed that code over 25 years ago integrating CANDU specific models with the LWR code MAAP at a time when we used Pentium 286 machines and have made public [12] a large number of limitations that make it ill suited to meet today's post Fukushima requirements.

8. A PHWR WILL PRODUCE DEUTERIUM, NOT HYDROGEN :D_2-H_2 DIFFERENCE

Given that the reactor is cooled and moderated with D_2O, one would expect the mitigation measures and detection measures designed for D_2, but almost all research and development and implementation has been for H_2. On top of public denial of the almost two fold difference in transport properties between D_2 and H_2, differences in recombination rates have been loudly professed by the regulator and the utilities to be negligible, in defiance of hosts of research papers that have shown that except for chemical reaction of formation, the two gases are really not identical in any meaningful way that would allow the utilities to treat them as one and the same. In addition to differences in transport properties differences in recombination on metallic catalysts has also shown to be different for the two gases [22, 23]. In addition, there are scenarios in which H_2 would also be produced by external surface air oxidation of carbon steel feeders. This also has to be considered in design of systems for mitigation and detection.

8.1 MANAGEMENT CLAIMS OF NO FUTURE RELEASES

Meanwhile the largest of multi unit reactors continue to operate with 5 to 10 year license extensions in the middle of the most densely populated parts of Canada with almost no new systems in place to retard the progression of a severe core damage accident with the management claiming publically[7], to the horror of those who understand these reactors that the improvements made so far will make the chances of long lived radioactive species escaping from these reactors after a severe accident an impossibility.

8.2 A 'HOLISTIC' APPROACH TO SAMG

Utilities have recently touted a new and bizarre 'holistic' approach to severe accident management. For example, Bruce Power say that by claiming in-vessel retention of core melt and a filtered containment venting it needs to not install adequate hydrogen mitigation systems or over pressure protection systems or rectify any one of the dozens of design deficiencies[10]. In denying the risk reduction capability of such simple measures such as adequate safety relief valves for over pressure protection of the primary and the moderator cooling loops, it is acting against public interest, forgetting that according to good engineering practices and IAEA guidelines probabilistic analyses should not be considered as a substitute to a design approach based on deterministic requirements but as a part of the process to identify potential safety enhancements and to judge their effectiveness.

9. STATION BLACKOUT AT A MULTU UNIT CANDU

Let us go back to a Station Blackout scenario with an unmitigated loss of all AC power in a multi-unit CANDU plant at Darlington or Bruce station. This scenario implies that no AC power is available for a specified recovery period, usually taken at 12-24 hours for consequence analyses.

As the reactor trips, turbines trip and feedwater flow ceases, nuclear steam discharges to the atmosphere through Main Steam Safety Valves (MSSVs). Necessary condition for the atmospheric discharge of steam to remain a heat sink is that fluid inventory in maintained both within the boiler tubes and outside the boiler tubes. Early passive heat removal by thermosyphoning flows from core to the steam generators is maintained as long as the primary fluid inventory can be carried over the U tubes. It is unfortunately jeopardized early at Darlington and Bruce multi unit stations by the low elevation positioning of the large pressurizer vessel. It's free steam volume a couple of minutes after a reactor trip is about equal to the volume of the coolant in the boiler tubes (65 m^3) upon a loss of power to the pressurizer heaters[24]. So the pressurizer can slowly swallow the volume of the heat transport coolant in the boiler tubes. As a result, the boilers stop being a heat sink even before they run out of water on the secondary side. No further addition of water to the boilers by AFW pumps or any other means will restore a heat sink for the core decay heat.

Even if the lost water inventory in the boilers tubes can be replenished by a major change in emergency management procedures, with no passive steam driven auxiliary feedwater pumps or a method to easily replenish the steam generators with a high pressure emergency water injection the boilers stop being an effective heat sinks after less than 2 hours. Back leakage through the feedwater line check valves will cause vapour binding in the feed pumps and only alternate paths for water addition to the boilers will be effective.

With no effective heat sinks, the primary cooling system re-pressurizes and with an inadequate steam relief capacity of the safety relief valves on the degasser condenser vessel in path of the relief, an uncontrolled over-pressurization leads to a pressure boundary rupture. There neither are any provisions for passive or manual depressurization of the reactor loops after a loss of steam generator heat sinks nor a capability for a high pressure coolant injection into the pressurized heat transport loops and an uncontrolled rupture becomes an unnecessary inevitability with a potential for an early containment bypass as the most atypical of any reactor overpressure protection system fails to provide adequate relieve steam through dual valves in series qualified only for liquid relief. In absence of a retaining pressure vessel like in LWRs, an ensuing gradual onset of fuel channel heatup and disassembly upon loss of moderator coolant puts energy, radioactivity and combustible gases directly into the relatively weak reactor buildings. These structures are quite different from a traditional PWR cylindrical dome building and are rectangular structures built to old industrial standards. There are significantly high sources of combustible Deuterium gas (‘heavy hydrogen’) from large amounts of carbon steel in feeders and Zircaloy in fuel and fuel channels. Given the layout of the reactor units mimicking four inverted volumes interconnected at the bottom by a common duct, separation and accumulation of combustible gases in these unventable, inverted cup like geometries makes for impracticable combustible gas control. The small number of Passive Autocatalytic Recombiners planned and/or installed are neither quantified / qualified for severe accidents nor for the actual gas (Deuterium) they must recombine and can become early ignition sources. There is an enhanced potential for energetic interactions of fuel debris with bodies of water enveloping the hot fuel channels. Pressure relief in relevant reactor systems (PHTS, Calandria, Shield Tank, and Containment) is inadequate for anticipated severe accident loads. With the reactor units directly attached to the containment pressure boundary and a significant number of reactor systems outside the containment, a containment bypass, as for example from reactivity device failure following fuel and debris heatup, is a likely outcome after a severe core damage. The Calandria Vessel, long heralded as a core catcher, is a thin ~1” thick stainless steel welded low pressure vessel that has been assessed to fail catastrophically at welds and not able to contain hot molten debris. This failure can not only lead to enhanced combustible gas production but also severe energetic explosions leading to failure of structures at the containment pressure boundary. The Shield Tank also cannot contain pressure upon boiling and can fail.

Given that unmitigated expulsion of hot gases and fission products targets the small reactor buildings, there is potential for poor equipment survivability. The in-reactor instrumentation for monitoring and control is neither adequate nor qualified for conditions after a severe accident. Severe accident simulation methods are outdated, crude and in dire need of upgrades. There are no dedicated simulators for severe accidents and the perfunctory desktop exercises with high-level Severe Accident Management 'Guidelines' are inadequate. No significant design changes have been implemented since Fukushima that may prevent a severe core damage scenario and some well known design problems like inadequate over pressure protection have been ignored. Yet, there are opportunities for engineered upgrades that can substantially eliminate a large number of vulnerabilities. However, the regulatory regime in Canada is lax and regulatory staff does not have the technical capability or guidance to independently verify assessments and analyses presented by the utilities not motivated to invest in design upgrades for low probability events they want to ignore. As a result, a continued exploitation of an outdated design with refurbishments that extend the life by another couple of decades is not only a risk to public but also to the utilities.

10. CONTAINMENT STRUCTURE VULNERABILITIES

The multi-unit CANDUs at Darlington and Bruce house four reactor units in an interconnected slightly sub-atmospheric containment attached to a normally isolated vacuum building maintained at about 7 kPa(a) and of about 75% of the containment volume. Each reactor sports a containment structure that is common and contiguous to 4 relatively large reactor power units. Each reactor is capable of putting un-attenuated fission products from the ~2700 MW(th) fuel fission sources as well as combustible Deuterium from over 50,000 kg of Zircaloy and 2000 m^2 of the 120,000 kg of carbon steel. As a result, any accident that results in activity releases into the containment, whether within the design basis or not, is likely to contaminate and disable from service all four reactor units.

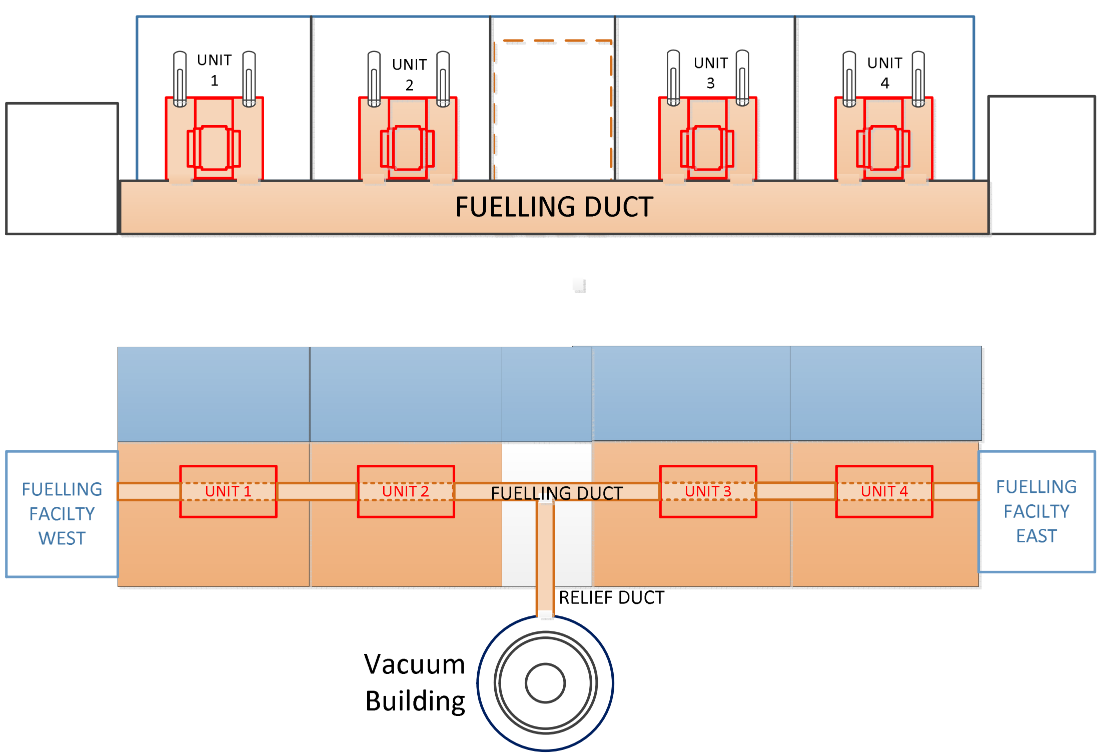

Figure 10 : Bruce / Darlington station layout for 4 units with common containment

With an over pressure retention capability of less than a bar (at design and significantly deteriorated after 20+ years such that all containments are no longer tested at design pressure or at the required frequency of 6 years) and a containment structure made up of rectangular concrete slabs and about 500 times leakier than the 1%/day leakage at design pressure for PWRs, a number of critical equipment are outside the containment and some critical equipment like pressurizers are placed below the mid elevation of the reactor core inside the containment. A common Fuelling Machine Duct underlying the 4 reactors connects the containment volume via a Pressure Relief Duct to a Vacuum building whose volume is deemed adequate for most design basis accidents in a single unit but its effectiveness to mitigate a severe accident in all 4 units is very obviously lacking as the effective volume per reactor unit is less than half of that for a typical PWR and the structures are weaker and with greater likelihood of trapping combustible gases . The containment is built to the National Building Code as are the access requirements, fire protection, smoke detection, etc. It is not built to modern nuclear containment standards.

A number of reactor systems including the reactivity control mechanisms, primary pumps and steam generators are located outside the containment boundary above the reactor cores. The reactor core related structures themselves are within a tank attached at the containment pressure boundary. Critical structures essential for maintaining core cooling being outside the containment are likely vulnerable to certain externally induced challenges. The stations have not considered reactor building reinforcements to avoid building failure or added additional reinforcements with special emphasis on confinement space on top of reactivity decks to mitigate external impact hazards. While a PWR containment may be expected to withstand an aircraft impact, there is such no protection in a multi unit CANDU.

There are no new improvements to pressure suppression system in reactor building as the vacuum building is an inadequate volume supplement to avoid building failure after a multiunit core damage accident or even due to pressurization caused by hydrogen burns. Measures to reinforce the confinement pressure boundary (space occupied by safety and process systems outside the containment) are missing.

The basement of the reactor buildings (fuelling machine duct and the pressure relief duct) is located below the level of the water in the lake. To the credit of the utilities, new portable Emergency Diesel Generators have been to be located at elevations higher than the original backup Diesel Generators that are at lower grade elevations, about 3m higher than the water, not dissimilar to what sank Fukushima. They are yet to be relocated to higher elevations. If that was done a failure similar to that at Fukushima could be avoided.

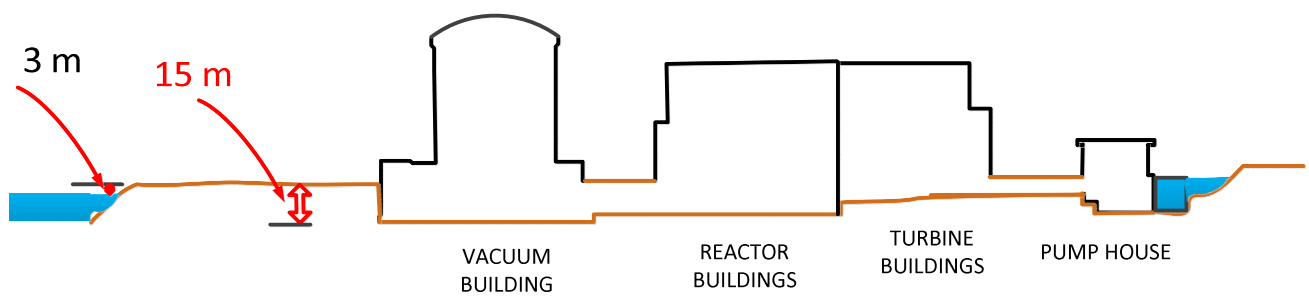

Figure 11 : Building basement layout below Lake water level. The emergency power supply generators are at grade level with cable tunnel 6m under.

The containment structures are rectangular slabs different significantly from typical cylindrical PWR containments and have a relatively weak design pressure (0.6 to 0.9 bar) with relatively high design leakage at design pressure (up to 2% volume per hour or up to 48% per day comparing very unfavourably to a typical PWR with 0.1% leakage per day (Figure 4) at a design pressure that is typically 5 times higher).

The containments are tested for pressure retention most infrequently of any power reactor in the world. Darlington now tests containment for pressure every 12 years while the regulations under which it was originally licensed required a 6 year test interval. The last pressure test was described as a difficult and arduous process that took 6 months of planning.

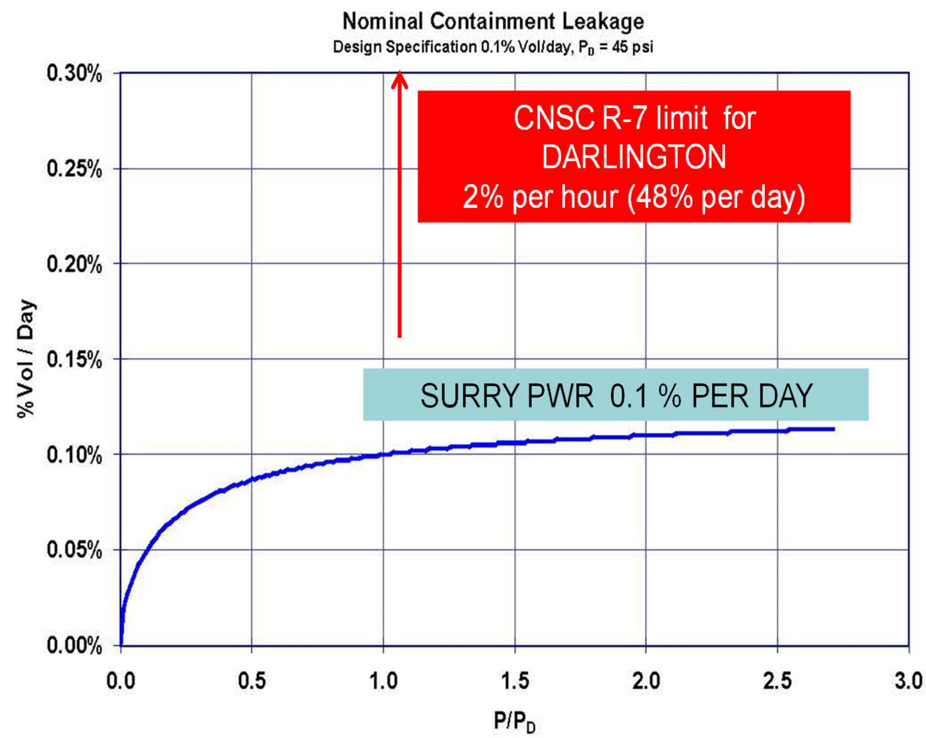

Figure 12 : Comparison of a PWR containment design pressure leakage with that for a Multi unit CANDU.

The individual reactor buildings can be envisioned to be inverted cups on top of a common duct such that retention of flammable gases and fission products after the vacuum building becomes ineffective is a concern. The reactor building volumes are about 14000 m^3 each with a combined volume of the 4 unit reactor buildings and the common fuelling machine and pressure relief ducts of about 120,000 m^3. The normally isolated vacuum building is an additional 95,000 m^3 and it is maintained originally at an isolated pressure of 7 kPa(a) with the main containment volume slightly sub atmospheric. For a multi unit severe accident, the containment volume per unit power is among the smallest of any other similar power reactor in the world.

11. SUMMARY OF MULTI UNIT SEVERE ACCIDENT PROGRESSION & MITIGATION CHALLENGES

11.1 Containment

- Low containment design pressure (<0.9 bar) and high design leakage at design pressure(48% per day)
- Reactivity devices, steam generators, pumps and other equipment critical for long term heat removal are outside the containment and located under an industrial building .
- Containment bypass from over-pressure and thermal creep induced steam generator tube ruptures and from reactivity device failure a likely outcome after a severe core damage.
- Reactor vaults shaped and arranged to be highly likely traps for combustible gases.

11.2 Poor Overpressure Protection Design

- Safety relief valves not directly on the main cooling circuit (ASME section III , NB-7141 (b) requires a direct and unobstructed relief path) and require another pair of downstream valves to open. All valves designed for liquid relief.
- Only two safety relief valves (called 50% capacity valves but the 'capacity' is misrepresented) - contravenes single failure criteria
- Undersized over pressure protection with steam relief capacity of the 2 safety relief valves by a factor of up to 10 - contravenes common sense - relief capacity must exceed anticipated loads, which will always exceed decay heat.
- Inadequate primary cooling circuit relief inherently forces reactor damage by uncontrolled over-pressurization even before an ECC is given a chance to avoid severe core damage. An uncontrolled relief through a rupture in pressure boundary is an unacceptable outcome.
- Accelerated depletion of moderator inventory due to expulsion through pressurized Calandria rupture disks upon channel voiding and fuel heatup to cause moderator boiling.
- Shield Tank cannot contain anticipated pressurization upon boiling and can fail. Restoration of cooling after water depletion problematic as pump flow inlet at the top of vessel that can be voided.

11.3 Poor Pressure and inventory control

- No provisions for direct manual depressurization of the Primary Heat transport system.
- Pressurizer located well below the core can drain water from primary coolant system upon cooling upon loss of power and inhibit thermosyphoning flows.
- No systems for high pressure ECC or any emergency measures for high pressure primary makeup intervention / injection.

11.4 Lack of a pressure vessel causes direct containment contamination

Onset of severe core damage puts activity directly into the containment. There is no isolation of damaged core and its activity in a closed vessel like in a PWR pressure vessel.

11.5 Poor Deuterium Hydrogen mitigation systems

Significantly higher sources of hydrogen from large amounts of carbon steel and Zircaloy.

Currently planned hydrogen mitigation systems (igniters + a small number of PARS) inadequate and potentially dangerous. Poor combustible gas mitigation measures. Small number of Autocatalytic Recombiners inadequate for severe accident scenarios and will cause explosions.

11.6 Moderator vessel an unlikely core catcher - poor

Energetic interactions of disassembling core debris with underlying boiling moderator water in the low pressure Calandria vessel can cause vessel structural failures.

Calandria vessel failure by weld failures is a likely outcome even before debris melt. There are a number of pipe penetrations at the bottom of the vessel that can fail by thermal interactions with hot debris.

Should the Calandria vessel fail, interaction of hot debris with Shield Tank water also similarly challenging to integrity of structures holding the reactor vessels connected to the reactivity deck at the containment pressure boundary pressure relief in ALL relevant reactor systems in inadequate (PHTS, Calandria, Shield Tank, Containment) to remove decay heat

Calandria vessel likely cannot contain melting reactor core debris and can fail catastrophically at welds causing energetic interactions with potential for gross structure failures.

11.7 Spent Fuel Storage.

The spent fuel medium term storage in spent fuel pools is poorly designed and highly susceptible to Zircaloy fires.

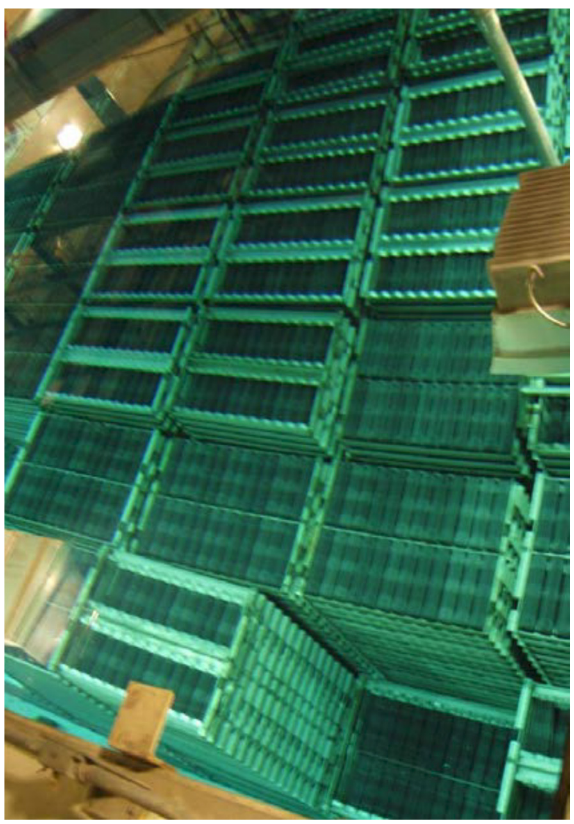

Figure 13: Spent fuel bundles stacked like fish in a fish basket, 16 or more trays high

11.8 Backup Diesel Generators

These are located at the lowest grade elevation in the plant and are no more than 3m above the water line at Darlington and ~4m at Bruce. The tunnel carrying the cables is below the water line by about 4m and can get deluged with water. Pickering station has seen its basement level flooded in the past from water swell in the lake. Location of backup diesel generators has been pointed out as the single most critical error at Fukushima; something that has escaped the CNSC despite repeated warnings. In fact, CNSC staff provided misleading information as to the actual location of the diesel generators in Bruce reactor relicensing public hearings in 2018by claiming that they were located 40-50 feet higher than the lake water.

In Bruce the diesel generators are at 591' elevation while the grade varies from 614.5' at the north side to 590.5 ft at the south side. So the diesel generators are at lowest grade in the station grounds and certainly below the average grade. The tunnel carrying the cables from the water treatment plant (where the generators are) is at 569' with the water line higher by 10' at 579'. So the tunnel is below water line. The tunnel can be flooded by a deluge or a flood or a seismic activity. So if the grounds which are at 591' in that area ever get flooded by a wave, a tsunami, ice dam or whatever, the diesel generators will get flooded at Bruce, just like at Fukushima. The cables from the diesels are in the trench at a much lower elevation, below water line. Design of structures housing backup diesel generator at Darlington is a copy of the one at Bruce.

And…

- Inadequate instrumentation and control for severe accidents
- Poor equipment survivability due to poor containment layout
- No dedicated operator training / simulators for severe accidents.
- Severe accident simulation methods are outdated, crude and inadequate.
- No significant design changes implemented. Known problems ignored for decades.
- Current SAMGs are unrealistic and inadequate. Many potentially favourable emergency hook-ups not implemented.
- Environmental assessments for off-site releases after severe accidents performed with a source term that represents barely 0.15% of the total core inventory

The lessons learned from Fukushima disaster have been poorly accepted despite the hoopla surrounding development of Fukushima Action items by the National Regulator. Risk to public can only be reduced by much needed design upgrades, starting with an open discussion of the severe accident related vulnerabilities, and an acknowledgment that the reactors not designed with consideration of any severe accidents within the design basis; cannot be expected to provide mitigation measures necessary to meet the newly emerging understanding of progression and consequences of a severe accident and current public expectations.

12. SUMMARY MAJOR AVENUES OF DESIGN UPGRADES

Following is a partial list of design improvements that require serious and immediate consideration to meet some of the vulnerabilities of the multi-unit CANDU design.

1. Passive makeup by steam driven auxiliary feedwater pumps for high pressure water addition to boilers
2. PHTS overpressure protection enhancements for avoidance of uncontrolled ruptures (replace PHTS relief valves)
3. Emergency power hook-ups to pressurizer heaters for early re-establishment of pressure control, or Relocate pressurizer to higher elevation
4. High pressure makeup of PHTS inventory loss (high pressure emergency water injection pumps)
5. Pressure relief valves on Pressurizer for manual PHTS depressurization
6. Calandria vessel overpressure protection enhancements for avoidance of deliberate voiding (relief valves with decay heat capacity in addition to rupture disks)
7. Calandria vessel structural design enhancements for better likelihood of retention of core debris
8. Shield tank overpressure protection enhancements for avoidance of structural failure
9. Shield tank heat removal capacity enhancements for retention of debris in Calandria vessel
10. Containment penetration reinforcement for avoidance of overpressure failures
11. Containment pressure suppression improvements: local sprays and external support to coolers
12. Instrumentation enhancements for detection of important accident parameters
13. Filtered containment cooling for avoidance of imminent structural failures
14. Emergency hook-ups for water and power to safety critical systems at appropriate pressures
15. Improved Class 1 batteries., better definition of anticipated loads over prolonged periods of loss of AC power.
16. Combustible gas detection, measurement and recombination systems calibrated for Deuterium
17. External water makeup to a stranded fuelling machine after a LOCA
18. External water makeup and heat removal from the spent fuel bay
19. Off-site measurements of activity magnitude and energy for identification of radioactive species in releases and correlating them to source terms;
20. Upgraded consequence assessment codes dedicated for PHWRs (current codes are not entirely fit for intended use)

13. UNEXPLORED AVENUES OF RESEARCHOR THINGS WE DO NOT KNOW / UNDERSTAND WELL ENOUGH

There are a number of phenomena associated with accident progression that require separate effect quantification with research and have not been addressed properly. These include:

1. Effect of uncontrolled pressurization of the heat transport system before core degradation. With over-pressure relief valves unable to remove decay heat an uncontrolled re-pressurization of the PHTS is inevitable. Typical design failure pressures in a CANDU reactor for level C conditions in Table 1indicate that the ever so vulnerable boiler tubes have the lowest pressure retention capacity and are thus are prime candidates for failure. However, the degradation of feeders by thinning (~0.1 mm/yr) and of pressure tubes by hydriding, creep - thinning, elongating etc. makes the issues more complex.

Table 1: PHTS COMPONENT PRESSURE RETENTION CAPACITY

CANDU 6 REACTOR COMPONENT	Level B	Level C	Level C with Seismic
Inlet Header	14.81	18.96	18.96
Outlet header	12.49	17.58	17.58
Pressure Tube Outlet	11.81	22.22	15.10
Rolled joint outlet	12.46	21.65	10.14
thickness	12.11	16.52	13.71
SG tubing	13.43	14.34	12.72
Pressurizer	12.13	16.00	16.00
Degasser Condenser	11.77	15.51	15.51
header interconnect	17.46	30.99	11.68

2. Reflux condensation holdup of water in boiler tubes on feedwater recovery. This becomes important in case of boiler recovery after PHTS is voided and can lead to early channel heatup of voided channels and their failures at high pressures when natural circulation flows cannot be re-established.

3. Core and Channel thermal hydraulics under loss of forced circulation - during PHTS blowdown, voiding by boiloff and depressurization ; intra channel fluid interactions
4. Mechanisms of high temperature fuel bundle deformations and quantification of bundle geometry parameters
5. Fuel bundle oxidation with air, oxygen at various stages of its disassembly.
6. Mechanisms of high pressure rupture failure of CANDU channels by hot fuel and melt interactions with pressure tubes
7. Channel failures by their deformations; melt through to channel disassembly at low pressures
8. Gross core disassembly, debris retention, displacements, interactions and collapse of individual columns of channels
9. Effect of recovery actions to reflood fuel channels

10. Steam explosion potential during debris and melt relocation to underlying water in Calandria vessel
11. Solid debris behaviour in Calandria with accumulation over many hours and without water ingress
12. Solid debris interactions with air drawn from Calandria overpressure relief ducts
13. Thermo-mechanical behaviour of stepped welded Calandria vessel under load of hot debris
14. Response of boiler tubes following core heatup (consequential boiler tube failure) at high pressures and at low pressures with boilers dried out
15. Component and system failure modes for interfacing systems, in-core device failures that may create containment bypass.
16. Interaction of debris with an intact loop in case of coolant loss and core damage restricted to one loop.
17. Oxidation of end fittings, feeders, Calandria by steam and air
18. Fission product release mechanisms under different fluid conditions from fuel pins in bundles, debris, corium
19. Effect of recovery actions in Calandria, shield tank, fueling machine duct in presence of debris
20. Effect of Calandria vessel weld failures including interaction of water ingress on solid and molten debris
21. Containment response to sharp pressurization loads (energy, mass addition ; hydrogen combustion)
22. Hydrogen / Deuterium distribution in reactor vaults and rest of containment
23. Hydrogen / Deuterium burns, detonation, deflagration in reactor vaults and failure modes of structures
24. Effectiveness and adverse effects of recombiners, igniters (auto-ignition and explosions)
25. Containment response to sharp pressurization loads (energy, mass addition ; hydrogen combustion)
26. Potential and effects of consequential floods, fires in containment
27.

14. COMPUTER CODES USED FOR SEVERE ACCIDENT PROGRESSION & CONSEQUENCE ASSESSMENTS

The currently used computer code MAAP-CANDU suffers from the following errors and deficiencies :

1. No consideration of heavy water, deuterium gas (light water and H_2properties used)
2. No momentum equation for PHTS
3. Channel degradation during channel boiloff before dry steam/D_2 heatup not modelled - Initial fuel temperatures at onset of heatup are arbitrary
4. Channel hydraulics based on assumed header to header Δp and no overall core thermal hydraulics. No intra channel flows. No consideration of fluid discharge paths.
5. A limited number of channels modelled.
6. No explicit fuel sheath modeling.
7. No modelling of out of flux pressure tube lengths.
8. No modelling of water retention in end fittings after boiloff or blowdown
9. No thermal modelling of feeders and end fittings
10. No consideration of differences in burnup and power profiles between various channels
11. No modelling of in-core devices and their effect on individual fuel bundle displacements.
12. No modelling of piping into Calandria vessel.
13. Crude modelling of core disassembly & a physically impossible model of 'core collapse'
14. Primitive modelling of suspended solid debris
15. Solid debris interactions with air not modelled
16. Deuterium / Hydrogen generation by steel oxidation and Uranium-steam oxidation ignored.
17. Fission product releases from debris crudely modelled.
18. Fission products do not decay.
19. As ‘engineered’ codes with specific accident progression pathways – many scenario paths not considered.
20. Difficult I/O; primitive post processing

It is incomprehensible that above deficiencies have not been rectified in the 25 years since the control of code development left Canadian hands. No SAMGs, design assist or other accident management or training measures are possible without properly modelling the reactor, its phenomenology and all potential accident progression pathways. Without modelling the behaviour of each fuel channel individually, for example, the erroneous conclusions drawn from models such as for a total, global core collapse can give misleading and dangerously inaccurate results.

15. CONCLUDING REMARKS

The purpose of the paper is to foster an open discussion of design vulnerabilities so that the industry feels encouraged to develop engineering solutions that can help reduce risk. Our commitment to reactor safety should not just be defensive but also honest. The recent claims by the industry of the infallibility of the design and relative benign consequences of a severe accident prompting the regulator to suggest that local authorities may not even need any evacuation for 24 hours is a very disturbing trend. The purpose of this paper is also to reiterate that as professional engineers, we are bound by our professional ethics to keep the public safety first in our list of concerns and that sweeping under the rug of design deficiencies and knows vulnerabilities to unfavourable outcomes or knowingly stretching the truth is contrary to our professional obligations. Damage done by radiation to the countryside at Chernobyl and Fukushima far exceeds the economic and other benefits from operation of all 400 odd nuclear reactors that have been operated so far. In both cases it was not just the technology that was wanting; it was also the organization and the abysmal safety culture that was the root cause. Those accidents required a number of professionals to be complicit in putting the corporate priorities first over public safety concerns.

Fukushima reviews by the industry have increased the awareness of the potential of a CANDU severe core damage but not understanding of its implications. Some newly implemented mitigating measures like mobile emergency hook-ups will partly reduce the likelihood of progression to a severe core damage but are not passive or well thought through. Some long planned measures like PARS and Containment Venting will help reduce off-site consequences but implementation is dangerously incomplete and backup analyses are questionable. There is significant resistance to understanding, acceptance and targeting of inherent reactor design deficiencies by the utility management. Restructuring of CANDU nuclear industry has affected progress and granting of long term licenses by the CNSC has further retarded progress towards risk reduction from ageing and obsolete reactors. Significant opportunities exist in further reducing risk to owners and the public but regulatory actions have had a negative effect. There has been little progress in enhancing analytical capabilities to help identify and quantify vulnerabilities; justify and introduce new risk reduction measures. Unless corrective action is taken, a disaster is looming. At a minimum it will consume the utility with no upper limit in sight of the damage it can cause to the nation.

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Comments on OPG's Environmental Impact Statement for New Nuclear Build at Darlington NGS

by F. R. Greening – for le Mouvement vert Mauricie's Intervention on the Darlington New Build Environmental Assessment Hearings in 2011

The complete MVM Intervention is found at www.ccnr.org/MVM_final.pdf

Radioactive Emissions from Darlington New Build

1.0 Introduction

OPG's September 2009 Environmental Impact Statement (EIS) for the construction of up to 4800 MW of new nuclear generating capacity on the Darlington NGS site shows that the projected radioactive emissions from such a significant addition to the pre-existing nuclear facilities at Darlington are potentially very large.

This simple fact underscores the need for these emissions to be properly assessed in relation to the applicable release limits for radioactive species in gaseous and liquid discharges. Indeed, only a detailed assessment of all such emissions can ensure the new nuclear build is in compliance with these limits.

The question of the projected radioactive emissions from the proposed new nuclear build at Darlington is discussed in two Technical Support Documents (TSDs) issued as Volumes 15 and 16 of OPG's EIS:

Volume 15: "Radiation and Radioactivity Environment Existing Environmental Conditions"

Volume 16: "Radiation and Radioactivity Environment Assessment of Environmental Effects"

From these documents we see that OPG's approach to assessing the radiological impact of new nuclear reactors at the Darlington NGS site is to first consider the concentrations of natural and man-made radioactive species already present in the air, soil and groundwater around the site, and then to predict the expected increases in the concentrations of radioactive species emitted as a consequence of adding up to 4800 MW of nuclear generating capacity at the Darlington NGS site.

OPG's methodology for predicting the environmental emissions from the proposed reactors involves the evaluation of parameters that influence the release and dispersal of radioactive species from normal operation of the new reactors. For simplicity, and as a reasonable approximation, the reactors are considered to be one or more point sources of emission of a particular radioactive species that is subsequently traced in the near-field and far-field environments using atmospheric dispersion and water dilution factors in suitable plume-tracing models.

What is most significant about this approach is that while the air dispersion and water dilution factors of many radioactive species are well-known from studies by organizations such as the U.S. EPA and the NCRP, the source terms for the numerous radioactive species emitted by a newly designed, but yet to be operated, reactor tend to be quite uncertain.

Thus we need to ask the simple question: is OPG's EIS based on sound scientific principles whereby radioactive emissions are accurately predicted or is it merely a self-serving prophecy based on wishful thinking by OPG?

2.0 Issues Arising from Alternative Reactor Designs

At the present time OPG is considering four reactor designs:
• The ACR-1000, a heavy water reactor offered by AECL
• The AP1000, a PWR offered by Westinghouse
• The US EPR, a PWR offered by Areva
• A "modified" CANDU-6, based on AECL's existing CANDU-6

There are a number of issues arising from the fact that the type of reactor to be built at Darlington is yet to be selected by OPG:

(i) Modern nuclear reactors are very complex facilities that utilize a wide range of water and gaseous process streams and also generate large quantities of solid wastes. A detailed accounting of how radioactive wastes will be produced, managed and disposed of is required for each reactor design for a meaningful assessment of the environmental impact of these reactors to be made.

(ii) Interim storage of some effluent streams and solid wastes may (or may not) be used to delay the environmental release of relatively short-lived radioactive species; the potential for varying degrees of holdup of effluents for each reactor design serves to add uncertainty to environmental impact assessments.

(iii) The ACR-1000 and CANDU-6 utilize heavy water as a moderator – technologies that produce, and inevitably release, far more tritium than any comparable light water reactor design. This is of special concern to the Darlington EIS review because of the on-going debate as to an appropriate standard for tritium in Ontario's drinking water supply (as reflected in the ACES and ODWAC recommendations).
In view of these issues it is necessary to closely examine not only the conclusions reached by the requesting party – OPG in the present case – but also the claims made in the submissions to OPG by the reactor vendors. In this regard it is perhaps a happy coincidence that the three companies that have submitted proposals to OP – namely, AECL, Westinghouse and Areva – have all recently made similar submissions to the UK's Environment Agency (UK EA) for the purpose of assessing the expected performance of new nuclear power stations to be built in England and/or Wales.

Thus it is possible to compare the vendors' predictions for the environmental impact of the ACR-1000, the AP1000 and the EPR reactors with the responses of two requesting parties: namely, OPG and the UK EA. Fortunately the three different reactor designs currently under scrutiny by OPG and the UK EA employ similar radioactive gaseous and liquid waste management systems. Nevertheless, to be in compliance with regulatory emission limits, it must first be proven that the proposed monitoring techniques for each reactor design are adequate to quantify the radioactive content of a particular discharge at the required level of detection. In addition the vendors must demonstrate that the various wastes arising from their respective reactors meet appropriate criteria for disposal in waste repositories.

It is significant that the UK EA's initial comments on the submissions it received in 2008 from the vendors of the ACR-1000, the AP1000 and the EPR, has been to state over and over again that: "insufficient information has been supplied for us to draw any conclusions". In the case of AECL's submission, the UK EA have requested that detailed information on the source/location, height, diameter and volume flow of gaseous and liquid discharges should be provided and add that "designs rather than concepts should be described".

It is rather telling, and somewhat disturbing, that no complaints about insufficient information on the three reactor designs under assessment have been forthcoming from OPG. Furthermore, as recently as November 2009, the UK's Health and Safety Executive said it could not recommend plans for new reactors because of wide-ranging concerns about their safety.

3.0 Radioactive Emissions: General Comments

As listed in Table 3.1 below, a large number of radionuclides are produced by the operation of water-cooled reactors. Most of these radioactive isotopes are created either through neutron activation or uranium fission (yielding "activation products" and "fission products"). In addition, a number of "transuranic isotopes" are created when non-fissile uranium atoms absorb one or more neutrons, subsequently transmuting into various isotopes of neptunium, plutonium, americium, curium, and so forth.

After an induction period, varying from a few days to several years, most of the radionuclides in question attain relatively constant (equilibrium) concentrations within the various systems in which they are produced – such as the reactor fuel bundles, coolant pipes, moderator tanks, heat exchanger tubes or cover gas plenums.

Inevitably some radioactive isotopes leak or otherwise escape from the systems in which they are produced and enter one or more liquid or gaseous waste effluent steams. It is these streams that must be assayed by continuous monitoring, or by the analysis of frequent "grab" samples, to determine the radionuclide content of the systems involved.

This type of data is essential for the control of radioactive emissions because it allows a reactor operator to follow the movement of radioactivity throughout the nuclear station under his or her control. Furthermore, only with this level of detailed radiation monitoring may all radioactive releases from a nuclear facility be reliably reported to the appropriate regulatory agencies as a "source term" for each radionuclide. Radiation dose calculations require radionuclide source terms – usually expressed as a time averaged flux – to determine the rate of release of a radioactive species and derive an associated radiation dose.

However, as we have seen, source terms for a "first-of-a-kind" reactor are problematical because they cannot be measured beforehand. Even a longstanding nuclear power station generally has insufficient data to accurately quantify all of its radioactive emissions and radiation doses.

Consider the problem of estimating the radiation dose at a location 1 km from an operating nuclear reactor. The expected dose could be calculated from a measurement of the mean annual concentration of radionuclides at the location of interest – but such data are usually not available.

The only practical way to make up for this lack of knowledge of the detailed dispersion of escaping radioactive species is to use source terms measured at the outlet of a contaminating stack or liquid effluent pipe and then determine the dose at a remote location using plume tracing models. However, this approach still requires reliable analytical data for the rate of emission of all the radionuclides, including those shown in Table 3.1. This entails the measurement of the concentration of at least forty radionuclides in every effluent stream.

The analysis of a wide range of radionuclides, such as those listed in Table 3.1, is not a trivial task.

Gamma spectrometry is probably the most useful technique to quantify the gamma emitters (γ-emitters) in a sample using a single detector, but is of no use in quantifying the so-called "pure" beta-emitters (β-emitters) such as

- H-3 (tritium, which is radioactive hydrogen, usually given off in the form of radioactive water molecules),
- C-14 (carbon-14, usually given off as radioactive carbon dioxide),
- Cl-36 (chlorine-36),
- Ni-63 (nickel-63),
- Sr-90 (strontium-90) and
- I-129 (iodine-129).

These pure beta-emitters require specialized, isotope-specific, analytical techniques.

The same holds true for uranium and most of the transuranic isotopes in Table 3.1 such as Pu-239 (plutonium-239), where α-spectrometry must be used on specially prepared samples.

Reactor operators, faced with the daunting task of measuring the concentrations of up to 40 radionuclides in all the gaseous and liquid effluent streams in a nuclear power station, generally resort to collecting analytical data for a much-reduced list of “high priority radionuclides”, leaving the remaining radioactive species to be checked occasionally or not at all (see Section 3.4 for more details on this).

However, as we shall see, many of the most important radionuclides, such as tritium and carbon-14, are also the most difficult to determine with good precision and accuracy – an issue that is not addressed in OPG’s EIS for Darlington new build.

TABLE 3: Important Long-Lived Radionuclides in Reactor Waste Streams
EC = Electron capture; UF = Uranium fission; α = alpha; β = beta, γ = gamma

Radio-nuclide	Half-life	Mode of Production	Mode of Decay	Principal Gamma Energies (keV)
H-3	12.3 y	^{2}H(n,γ)	β	No γ-rays
C-14	5730 y	^{14}N(n,p) ^{17}O(n,α)	β	No γ-rays
Cl-36	3.0 x 10^{5} y	^{35}Cl(n,γ)	β	No γ-rays
Ar-41	1.8 h	^{40}Ar(n,γ)	β, γ	1293
Cr-51	28 d	^{50}Cr(n,γ)	EC	320
Mn-54	313 d	^{54}Fe(n,p)	EC	835
Fe-55	2.7 y	^{54}Fe(n,γ)	EC	No γ-rays
Fe-59	45 d	^{58}Fe(n,γ)	β, γ	1099, 1292
Co-60	5.27 y	^{59}Co(n,γ)	β, γ	1173, 1332
Ni-63	100 y	^{62}Ni(n,γ)	β	No γ-rays
Zn-65	244 d	^{64}Zn(n,γ)	EC	1115
Kr-85	10.7 y	UF	β, γ	517
Sr-90	29 y	UF	β	No γ-rays
Zr-95	66 d	UF, ^{94}Zr(n,γ)	β, γ	724, 757
Nb-94	2.0 x 10^{4} y	^{93}Nb(n,γ)	β, γ	703, 871
Nb-95	35 d	UF, ^{95}Zr(β)	β, γ	766
Tc-99	2.1 x 10^{5} y	UF	β	No γ-rays
Ru-103	40 d	UF	β, γ	497
Ru-106	369 d	UF	β, γ	512, 622
Ag-110	252 d	UF	β, γ	658, 884
Sb-124	60 d	UF,^{123}Sb(n,γ)	β, γ	603
Sb-125	2.73 y	UF, ^{125}Sn(β)	β, γ	176, 428
I-129	1.6 x 10^{7} y	UF	β	No γ-rays
I-131	8.0 d	UF	β, γ	364
Xe-133	5.3 d	UF	β, γ	81
Cs-134	2.1 y	UF	β, γ	605, 796
Cs-137	30 y	UF	β, γ	662
Ce-141	33 d	UF	β, γ	145
Ce-144	284 d	UF	β, γ	133
Eu-152	13 y	UF	EC	122, 1408
Eu-154	8.6 y	UF	β, γ	725, 1272
U-235	7.0 x 10^{8} y	Natural	α	No useful γ-rays
U-238	4.5 x 10^{9} y	Natural	α	No useful γ-rays
Pu-238	88 y	^{238}U(n, β), etc	α	No useful γ-rays
Pu-239	2.4 x 10^{4} y	^{238}U(n, β), etc	α	No useful γ-rays
Pu-240	6540 y	^{238}U(n, β), etc	α	No useful γ-rays

Pu-241	15 y	$^{238}U(n, \beta)$, etc	β	No useful γ-rays
Am-241	433 y	$^{238}U(n, \beta)$, etc	α	59
Cm-242	163 d	$^{238}U(n, \beta)$, etc	α	No useful γ-rays
Cm-244	18 y	$^{238}U(n, \beta)$, etc	α	No useful γ-rays

3.1 Tritium

In light water reactors such as AP-1000 and EPR, tritium (hydrogen-3) is produced by ternary fission within the fuel assemblies or by neutron activation of lithium (added for pH control), or boron (added for chemical "shim"), in the cooling water.

By comparison, for an advanced CANDU reactor such as the ACR-1000, or a modified CANDU-6, a far greater amount of tritium is produced by the neutron activation of non-radioactive heavy hydrogen atoms (hydrogen-2) contained in the heavy water molecules that are used as the moderator.

The relative magnitudes of the various tritium production routes in the three reactor designs under consideration by OPG shows that an ACR-1000 reactor or a modified CANDU-6 produces about 100 times more tritium than either the AP-1000 or the EPR reactors. Nevertheless, experience with the operation of OPG's fleet of heavy water reactors suggests that tritium emissions from large CANDUs can be controlled to some degree by the implementation of strategies to limit heavy water spills and leaks and the optimization of vapor recovery drier performance.

This probably explains why AECL's estimated HTO release to water, reported in Table D.2-1 of OPG's EIS, is only about ten times (rather than 100 times) higher than the equivalent tritium release data estimated by the vendors of the AP-1000 and the EPR reactors – but is this number realistic? First note that none of the estimated tritium discharges provided by the three vendors is accompanied by documentation showing any rationale behind the reported values, nor the extent of any possible variability in the discharges. Neither is information provided on how specific events such as start-up, shutdown, maintenance, system leaks, fuel failures, etc, might impact on the reported tritium discharges.

Available tritium release data for OPG units show that high tritium emissions are associated with maintenance activities on certain systems.

Thus variable tritium emissions should be expected if an ACR-1000 or CANDU-6 is selected as the Darlington new nuclear build.
This conclusion is further supported by tritium monitoring data for CANDU units at Bruce, Pickering and Darlington over the past 20 years, which show that tritium emissions can vary by more than a factor of two for a given unit from one year to the next.

Tritium emission data for AECL's CANDU reactors at Point Lepreau and Gentilly-2 also show a very similar degree of year-to-year variability.
But let's take a closer look at the projected HTO ("tritiated water") emissions for four projected ACR-1000 reactors as reported in Tables D.1-1 and D.2-1of OPG's EIS. The projected airborne tritium release for the ACR-1000 is stated to be 0.48 Peta-Bq, while the projected waterborne release of an ACR-1000 is about three times higher at 1.4 Peta-Bq.

This is somewhat surprising because CANDU reactors traditionally release more tritium in the gas phase than in the aqueous phase.

What is more, Bruce A's four-unit airborne tritium emissions in 2008 were reported by Bruce Power to be 1.15 Peta-Bq – more than double the projected airborne emissions for the new CANDUs offered by AECL.
One is compelled to ask how AECL plans to maintain tritium emissions at or below the maximum projected levels of 0.48 Peta-Bq (airborne) and 1.4 Peta-Bq (waterborne). Our experience with the long-term operation of more than twenty large CANDUs here in Canada shows that current CANDUs are in some cases already above these emission levels.

Years of effort in trying to reduce tritium emissions from existing CANDU reactors have largely been unsuccessful. As a case in point, Darlington's waterborne tritium emissions more than doubled from the levels seen in the late 1990s to the levels reported in the period 2002 - 2007.

It is also noteworthy that OPG recently announced that it failed to meet its overall 2008 tritium emission targets.

Finally, as a cautionary note, there are reasons to believe that airborne tritium emissions are actually higher than currently measured by station monitors because, as AECL has reported, tritiated species tend to plate out on the walls of the sampling lines, thereby producing artificially low readings.

What is also not mentioned in OPG's EIS with regard to projected tritium emissions for an ACR-1000 is the fact that the tritium concentration in the moderator builds up over several years of unit operation as the function: C (tritium) = 2.5 [1 − exp (− 0.0563 t)] Tera-Bq/kg.

To make matters even worse, waterborne tritium emissions also increase over time because larger leaks tend to form in aging reactor systems such as the steam generators.

Now there is a way to alleviate some of the expected increase in tritium emissions from a heavy water reactor, namely, detritiation. However we are not informed by AECL or OPG if there are plans to detritiate heavy water from new ARCs, should this reactor design be selected.

Certainly, OPG has since 1990 used cryogenic distillation to detritiate heavy water from its CANDU reactors using the Darlington Tritium Removal Facility (TRF). This facility has the capacity to detritiate up to 3000 tonnes of D_2O (heavy water) per year. It has significantly reduced the average tritium content of OPG's inventory of 10,000 tones of D_2O.

Indeed, it has been estimated that without this facility OPG would be emitting an additional 7.4 Peta-Bq of tritium per year to the environment, which is more than three times its actual tritium emission rate. It must be noted, however, that such calculations typically ignore the fact that OPG's TRF is itself a significant source of tritium emissions.

Nevertheless, if the ACR-1000 or CANDU-6 is selected for the Darlington new nuclear build, substantially higher tritium emissions from the Darlington site are to be expected, either from the buildup and escape of moderator tritium in the new reactors, or from substantially increased use of the existing TRF.

Whatever the case, the projected use of detritiation for moderator heavy water in new ACRs needs to be addressed by OPG in its EIS for Darlington new nuclear build.

3.2 Carbon-14

Radioactive carbon-14 (C-14) is produced in both light water and heavy water reactors by neutron activation of N-14 (non-radioactive nitrogen-14) and/or O-17 (non-radioactive oxygen-17). However, among the three reactor designs under consideration by OPG, the highest projected C-14 emissions of 1.1 Tera-Bq correspond to the projected airborne C-14 emissions from the ACR-1000 heavy water reactors.

Unfortunately however, as we saw for the projected tritium emissions, none of the estimated C-14 discharges provided by the three vendors is accompanied by documentation showing the rationale behind the reported values, and the extent of any possible variability in the discharges. Neither is information provided on how events such as start-up, shutdown, maintenance, system leaks, fuel failures, etc, might impact on the reported C-14 discharges.

What is more, as we will show below, C-14 in CANDU reactor waste (such as ion-exchange resin) is a major environmental concern because of the very long, 5730-year, half-life of C-14.

OPG's original fleet of CANDU reactors commissioned in the early 1970s at Pickering NGS, used nitrogen gas (N_2) to fill their annulus gas systems. Most regrettably, prior to 1979, no one at AECL or OHN recognized the possibility that nitrogen could produce vast quantities of C-14 particulate under neutron irradiation.

Indeed, I have seen documents from AECL Chalk River written in 1981 stating that solid C-14 was not present in the annulus gas systems of Pickering reactors, even though I had reported the presence of solid C-14 in deposit removed from Pickering Unit 4 in 1980. (See: "Analysis of Pickering NGS "A" Unit 4 N_2 Annulus Gas Filter Deposit", OHRD Report No. C81-04-K, January 1981).

Unfortunately for AECL's alleged "experts" on this topic, we now know that thousand of Curies of C-14 particulate were produced in all four Pickering Units prior to the large-scale fuel channel replacement operations in the mid-1980s.

Today OPG no longer uses N_2 in its annulus gas systems, but residual N_2

from air enters moderator systems where it is readily converted to C-14 through the N-14 (n,p) C-14 thermal neutron reaction.

The fact that O-17 (oxygen-17) is enriched in heavy water relative to natural, light water, only adds to the C-14 production problems with CANDUs through the O-17 (n,alpha) C-14 reaction.

This certainly makes one wonder why OPG has no gaseous C-14 emission data for Darlington from 1993 to 1998.

While some C-14 is emitted during reactor operation, however, most of the moderator C-14 is collected on ion-exchange (IX) resin columns used for moderator water quality control.

Storage and/or long-term disposal of carbon-14-contaminated resins is already a major problem for OPG because of the potentially high collective radiation dose (63 person-Sieverts per gigawatt of electric power) from the long-lived C-14.

In light of these facts I would ask OPG to provide answers, with supporting experimental data and/or calculations, to the following questions concerning the production and fate of C-14 from four new ACR-1000 reactors at Darlington:

- What is the projected end-of-life C-14 inventory on spent IX resin from these reactors?
- Where and how will the spent resin be stored and at what repository costs?
- What is the expected condition/integrity of these IX resins to 2050 and beyond?
- What are the expected effects of self-irradiation on the retention of C-14 by the resin?
- What is the probability that microbial action could mobilize the C-14?

3.3 Noble Gases

The radioactive noble gas emissions from nuclear reactors are mostly shortlived fission product isotopes of krypton and xenon. However, Ar-41 (argon-41) from the activation of the small amount of non-radioactive argon in air (0.94 %), is invariably present in the gaseous emissions from operating reactors.
The day-to-day amounts and isotopic composition of noble gas emissions from operating reactors are variable and complex because the numerous radioactive species of interest are short-lived, ($t_{1/2}$ ~ 15 min to 12 days), with continually changing activities.

To add to this complexity, some noble gases escape containment directly and enter the environment via the "non-contaminated" stack, while other species find their way into gaseous effluent streams that use activated carbon beds to delay noble gas release.

The monitoring of noble gas emissions from CANDU reactors has been accomplished in many different ways over the years. Problems such as insufficient detector resolution and sensitivity remained unresolved until well into the 1980s. AECL also encountered similar problems at Point Lepreau and Gentilly-2, and has acknowledged that noble gas emissions reported for these reactors "were flawed", at least until 1994.

Even today, however, CANDU reactor operators do not provide a detailed, isotope-specific, breakdown of their noble gas emissions but simply report gross noble gas emission data in "energy-compensated" units of gamma – Bq.MeV.

This approach is based on the assumption that the radiation dose received by a population exposed to a radioactive noble gas mixture is proportional to the average gamma-ray energy per disintegration. But this is true only if the isotopic composition of the gaseous effluent is relatively constant.

And, as we have already seen, that is simply not the case for CANDU reactors.

Large variations in noble gas composition are caused by variable holdup times as well as by routine operational activities such as startup, refueling and shutdown.

Nevertheless, OPG and AECL continue to use such energy-compensated units when reporting noble gas emissions – even though there are internationally accepted standards, such as ISO60761, for gaseous effluent monitoring from nuclear reactors, and most jurisdictions do indeed report noble gas emissions for individual radioisotopes of argon, krypton and xenon in units of Bq.

Another important requirement of noble gas monitoring at a nuclear station is that the measuring instrument should be able to provide on-scale readings under accident conditions so that the station operator is able to provide meaningful release information for off-site emergency planning and actions. OPG does not address this issue in its Darlington EIS.

What we do find in OPG's EIS (TSD No. 27) is an analysis of "a stylized accident radioactive release scenario" in which a scaled source term, assumed to be a small portion of the reactor core inventory, is released from damaged fuel; this postulated release is subsequently used to determine a dose to the public.

However, this approach also assumes that reactor containment is not breached for 24 hours, artificially allowing the short-lived noble gases to decay. I would ask OPG to justify the assumption of a 24-hour delay. Regrettably, OPG's accident "scenario" has little to do with anticipated reactor accidents that have actually been postulated and studied by nuclear agencies around the world; on the contrary, OPG's approach appears to be an exercise in radioactive bean-counting to satisfy emission/dose limits.

OPG's imagined accident "scenario" is not realistic because it considers a radioactive release from only one fuel element or assembly even though the Canadian nuclear industry and its regulators know that power pulse transients and temperature excursions could damage much more than that. Indeed, a recent CNSC risk assessment for CANDU reactors mentions the likelihood of more than one fuel element being damaged:

"Most accidents involve deteriorated cooling conditions, resulting in elevated fuel temperatures which in some events may reach very high values…. In (feeder) stagnation break or flow blockage, several bundles in a single channel are predicted to experience melting".

I would therefore ask OPG to explain how it arrived at "the post-accident gaseous release source term" data in Table 4.4 of its report N-REP-01200-10000 entitled: "Use of Plant Parameters Envelope to Encompass the Reactor Designs Being Considered for the Darlington Site" In particular I would ask OPG to explain (in relation to Table 4.4 in N-REP-01200-10000):

- How it determined, and how it would validate, the noble gas and radio-iodine emissions in Table 4.4?
- How it modeled the gas, vapor and aerosol release, transport and retention in containment for the postulated accident scenario?
- Why a more realistic accident scenario, involving the melting of several fuel bundles, was not considered?

3.4 "Missing" Radioisotopes

There are a number of radioisotopes, known to be produced in nuclear reactors, which are quite difficult to analyze and are therefore not monitored or reported by reactor operators. Nevertheless, these isotopes are of concern for long-term disposal of reactor wastes.

I would therefore ask OPG to provide production, emission and dose estimates for the following unmonitored long-lived isotopes that may be released or found in the waste generated by Darlington new build reactors:

Al-26 (7.3×10^5 y)	aluminum-26	730,000 years
Cl-36 (3×10^5 y)	chlorine-36	300,000 years
Fe-60 (10^5 y)	iron-60	100,000 years
Cs-135 (2.3×10^6 y)	cesium-135	2,300,000 years
I-129 (1.59×10^7 y)	iodine-129	15,900,000 years
Zr-93 (9.5×10^5 y)	zirconium-93	950,000 years
Nb-92 (3.2×10^7 y)	niobium-92	32,000,000 years
Ar-42 (33 y)	argon-42	33 years

3.5 Accumulation of Radioisotopes in the Near-Field Environment

An important issue that is not addressed in OPG's EIS for Darlington New Build Reactors is the potential for the accumulation of long-lived radioisotopes in the near-field environment around Darlington.

Radioisotopes of particular interest are H-3 (tritium), C-14 and Cs-137 and the near-field environment of concern would be any location within about 10 km of the Darlington NGS site. Within this region radioisotope emissions from the Darlington site accumulate in exposed vegetation, soil and groundwater as the result of natural dry and wet deposition processes.

While it is difficult to accurately measure the rate of accumulation of H-3, C-14 and Cs-137 in the near-field environment around a nuclear facility, such rates may be inferred from several years of data from suitable environmental samples and comparisons to the concentrations of these species in "background" samples. Background concentrations of H-3, C-14 and Cs-137 reveal the occurrence of these radioisotopes from natural and anthropogenic sources such as cosmic rays and nuclear weapons testing.

Nuclear weapons testing, especially in the 1950s and early 1960s, injected considerable amounts of H-3, C-14 and Cs-137 into the earth's atmosphere, much of which found its way into soil and surface waters around the world. Nevertheless, since the Test-Ban Treaty of 1963, the concentrations of these species have been slowly declining so that current environmental levels are quite low and predictable. Representative maximum background concentrations in environmental samples collected around Darlington are:

H-3 in water from the Great Lakes and/or inland lakes and rivers: 4.5 Bq/L
C-14 in soil: 226 Bq/kg-C
Cs-137 in soil: 7.0 Bq/kg

OPG's EIS TSD Volume 15: "Radiation and Radioactivity Environment Existing Environmental Conditions" provides data for these species at various sites around Darlington. The maximum reported values are:

H-3 in water within the study area: 29.2 Bq/L or 6.5 times background
C-14 in soil: 301 Bq/kg-C or 1.3 times background
Cs-137 in soil: 11.5 Bq/kg or 1.6 times background
These data clearly show that radioactive contamination from the existing Darlington site, which has been in operation for only about 15 years, is already spreading into the local environment.

OPG likes to claim that the radioactive emissions from its nuclear facilities are within regulatory limits and therefore pose no threat to the local environment. However, such claims ignore the accumulation of long-lived radioactive species in the environments around OPG's nuclear facilities due to years of exposure to controlled emissions, uncontrolled leaks and accidental spills. Radioactive species such as Cs-137 have a tendency to bioaccumulate in select species of flora and fauna such as berries, fungi and fish.

What is more, there is evidence that radioactive emissions tend to increase as a nuclear facility ages because more and more radioactive material such as irradiated fuel is stored on-site and radioactive circuits such as annulus gas systems tend to develop leaks and/or require more frequent purging.

Thus it is to be expected that an ever expanding and deleterious radioactive "footprint" will grow around Darlington NGS over the predicted 50-plus years of operation of new nuclear reactors at this site.