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Written submission from Frank Greening

Mémoire de Frank Greening

In the Matter of the

À l'égard de l'

Ontario Power Generation Inc.

Ontario Power Generation Inc.

Request by Ontario Power Generation Inc. to request to remove the hold point associated with Licence Condition 16.3 of the Pickering Nuclear Generating Station Power Reactor Operating Licence

Demande par Ontario Power Generation Inc. visant à supprimer le point d'arrêt associé à la condition 16.3 du permis d'exploitation de la centrale nucléaire de Pickering

Commission Public Hearing

Audience publique de la Commission

May 7, 2014

Le 7 mai 2014

Canada

March 5, 2014

<u>Submission from Frank Greening</u>:

For all of the 18 CANDU reactors that currently supply about 50 % of the electrical power used in Ontario, pressure tubes are the most important life-limiting components and, as such, have to be closely monitored for a number of critical parameters relating to their in-reactor performance. A measure of the level of effort required in this regard is the fact that more than $8 million *per year* has been spent by the CANDU Owners Group (COG) <u>*over the past 30 years*</u> on pressure tube research and development.

Examples of past research projects carried out by COG Fuel Channel Working Groups are:

Research Project	Annual Expenditure (k$)
Effect of Microstructural and Microchemical Changes on Corrosion	900
Zirconium Surface Studies Relevant to Pressure Tube Behaviour	180
IR Spectroscopy for Oxide Thickness Determination on Zr Alloys	116
Bruce A Thick Oxide Patches	75
C-13 Oxide dating	60
Removal of AGS Deposits	170
In-Reactor Studies of Annulus Gas Chemistry on Zirconium Oxides	1000
Measurement of Hydrogen and Deuterium in Zirconium Alloys	273
Pressure Tube Oxide Microstructural and Microchemical Characterization	320
Chemistry, Corrosion and Hydrogen Pickup in Irradiated Annulus Gas	286
Annulus Gas System Chemical Characterization	75
Laser Profiling of Hydrogen and Deuterium in Zirconium Alloys	80

These 12 projects alone involved <u>*a total annual expenditure of more than $3.5 million*</u>. But this is not the full scope of research efforts pertaining to pressure tubes since other COG Working Groups also have projects on pressure tube characterization including:

- Pellet Sampling and Metallography of Surveillance Pressure Tubes
- Hot Vacuum Extraction Mass Spectrometry of Zirconium Alloys
- Inert Gas Fusion of Zirconium Alloys
- Standards and Quality Assurance for Zirconium Alloy Analysis
- Depth Profiling of Irradiated Pressure Tube Samples by SIMS
- Nuclear Reaction Analysis
- Chlorine in Zirconium by Neutron Activation Analysis

Given the level of effort expended on CANDU pressure tube research over the past 30 years one would expect that a full understanding of the effects of long-term exposure of pressure tube material to high temperatures in aqueous environments under neutron irradiation would have been developed by organizations such as AECL, OPG, and Kinectrics. And such an understanding would include an assessment of the end-of-life behavior of pressure tubes. Nevertheless, it is abundantly clear that OPG remains quite unsure about issues such as accelerated corrosion and deuterium pickup by Zr-2.5%Nb, delayed hydride cracking, (especially near inlet and outlet rolled joints), the effects of thermal cycling on hydride precipitates and garter spring failure by embrittlement.

What is the basis for this assertion?

Consider first the following COG Presentation entitled: ***The Role of COG in CANDU Fuel Channel Life Management*** Presented by E. J. Bennett COG Program Manager on May 27th, 2010.

Key parts of this presentation were:

Two Joint Projects that have been initiated to provide the data necessary to demonstrate fitness-for-service in pressure tubes up to the expected life of 210,000 EFPH *and to explore the possibility of operation beyond that*:

●COG Joint Project 4363 –Fuel Channel Life Management Project

●COG Joint Project 4299 –Pressure Tube End-of Life Hydrogen Equivalent Fracture Toughness Testing and Assessment

N.B. Other pressure tube work is also on-going under COG R&D Programs as well as other on-going *operational* programs e.g. the surveillance program

In the area of *deuterium ingress and fracture toughness* the R & D program objectives are:

1. Improve long range deuterium ingress predictive capability

2. Determine the effect that the presence of volume fractions of hydride at operating conditions have on material properties of irradiated pressure tube materials

The Major Issues to be addressed are listed as:

●Some reactors have comparatively high body-of-tube deuterium uptake rates

●Rolled joint deuterium ingress data is limited

●CSA limit on H_{eq} may be exceeded in RJ region in late life

●Fracture toughness (slit burst test) data is generally representative of low H_{eq} found in surveillance tubes (~20 ppm and normally tested at 250°C)

●H_{eq} and hydride orientation have an adverse impact on properties

In the area of spacer integrity and P/T contact the R & D program objectives are:

Assure that annulus spacers:

1. Retain adequate material properties to fulfill design function

2. Remain in position

The Major Issues to be addressed are stated to be:

1. Spacer Integrity: Assumptions have been made concerning the capability of annulus spacer to fulfill design function for the full service life of the station
2. Neutron irradiation adversely affects mechanical properties of metals through atomic interaction
3. Testing of ex-service spacers has indicated a reduction in ductility (particularly I-X750)

External OPEX involving nickel based alloys (Inconel) in LWR applications exhibit sensitivity to stress corrosion cracking in wetted environments

4. PT to CT Contact: Several reactors have loose fitting Zr-Nb-Cu spacers that have exhibited a few instances of post-SLAR spacer movement. In addition there is recent OPEX concerning spacer wear

Tasks related to spacer structural integrity are stated to be:

Perform literature survey of degradation mechanisms that affect nickel based alloys and I-X750 material

Review spacer design requirements and failure modes (in particular I-X750)

Understand the operating environment of spacers

Evaluate improvements to procedures/tooling to protect spacer condition during SFCR

Perform surveillance examination of spacers:

Generate an inventory of ex-service spacers at CRL

Develop surveillance examination requirements (examination, testing and acceptance criteria)

Test/examine a selection of spacers

Consider update to next edition of CSA N285.4 to include spacer surveillance requirements

The above lists of research topics and issues provided by COG Program Manager E. J. Bennett clearly show *the extensive knowledge gaps in the area of Fuel Channel Life Management*.

My question would therefore be if OPG is so sure about the safe and reliable operation of pressure tubes beyond 210,000 hot hours, what is the basis of the need for the COG research projects noted above?

Consider next the *Ontario Energy Board* hearing held on 17th Aug, 2010 (EB-2010-0008) and in particular the *PWU Interrogatory #014*. Here we find these statements:

Until recently, Pickering B was not expected to exceed EOL limits during the pressure tube nominal operating life of 210k EFPH. This expectation was related to the lower operating temperatures in Pickering B. However, the hydrogen and deuterium profiles through the inlet and outlet rolled joint regions of surveillance tube P6 M14 have challenged this belief (report issued December 2008). It appears that P6 M14 has much higher deuterium uptake in the compressive regions of the pressure tube and H_{eq} *exceeds the solubility limit at both inlet and outlet rolled joint burnish marks*.

Work to address the long-term integrity of pressure tubes has been ongoing for many years through the COG Fuel Channel R&D program. The FCLM Project was started in 2009 to supplement and accelerate the work of the COG R&D program, allowing OPG to more aggressively address the uncertainties in the plan for Pickering B Continued Operations. The principal issues that led to the creation of the FCLM Project have only come to light fairly recently, relatively late in the life cycle of the units. For example, the issue of anomalous hydrogen pick-up in Pickering B Generating Station's rolled joints was highlighted by the results of the inspection of a surveillance tube removed in 2007. The concern over garter spring degradation at Pickering B Generating Station developed following the replacement of the fuel channel A13 in 2008, and the potential embrittlement of garter springs was noted during the removal of a Darlington surveillance tube in 2005.

These measurements of hydrogen/deuterium pickup by Pickering B pressure tubes show that OPG's predictive model <u>is *not* reliable</u>. Therefore, how does OPG reconcile the above statements to the OEB with its request to the CNSC to extend continued operation of Pickering Units 5-8 to 247,000 EFPH?

And please note that as recently as September 2013 OPG acknowledge that the issues of D-uptake near pressure tube rolled joints and irradiated spacer degradation <u>*have not been resolved*</u>. See, OPG's *Business*

Case Summary filed with the *Ontario Energy Board* on 27th September 2013 (EB-2013-0321) in a Section entitled: *Fuel Channel Life Management Project 10-62444 & Spacer Retrieval Tool Project 28-66567*. Here we find a Risk Analysis Report Summary that mentions the need to:

> *"Pre-establish performance criteria and monitor results .... to ensure expected performance is attained"*

Why would OPG submit a request to operate Pickering B beyond the industry norm of 210,000 EFPH *__before__* it has established, through experimental measurement and testing, a safety case to operate to 240,000 EFPH? And I would also ask OPG to provide an estimate of the confidence level it expects for fuel channel service to 240,000 EFPH

And consider finally the following discussion concerning the need to refurbish the CANDU reactor at Point Lepreau:

New Brunswick Energy and Utilities Board Commission: IN THE MATTER of the application for the Point Lepreau Nuclear Generating Station Deferral Account Matter Held at the Delta, Saint John, N.B. on January 10th 2013

MR. PASQUET: My name is Paul Pasquet.

Q.7 **-** And your position with the organization?

MR. PASQUET: I am the Vice-President and Chief Nuclear Officer of the Lepreau Nuclear Plant.

MR. PASQUET: So when we talk about industry, we talk about industry experience. So clearly there are the operators of the particular plants and then there is AECL, being the original equipment manufacturer/vendor associated with it. And it would be AECL, who would be specifying the design life for the actual facility and the actual operators would be doing it based on their operating experience.

Q.112 **-** All right **----** so you are relying on AECL mostly, for an estimate of their life expectancy?

MR. PASQUET: AECL is the original equipment manufacturer or the designer of the reactor and they have done the analysis and the assessment to specify that in fact the design life is 210,000 effective full power hours.

Q.113 **-** So the short answer is yes?

MR. PASQUET: That's correct.... in fact there is quite a large body of evidence that has gone in to support why we believe, or why AECL believes that 210,000 effective full power hours is an appropriate and reasonable number to have for the life of the critical (pressure tube) component. So, do I believe the

210,000 effective full power hours is a reasonable and achievable design life for the pressure tubes? The answer is yes, I do.

Evidently the CANDU industry has historically accepted 210,000 EFPH to be a reasonable and safe limit on pressure tube life. Why then would OPG now reject this view in favour of a significantly higher value?

<u>End of Submission</u>

Submitted by:

Dr. F. R. Greening

Hamilton ON.

March 12, 2014

Additional Material for Submission to the CNSC Concerning Pickering B's License Extension Beyond 210,000 EFPH of Operation:

● In OPG's 2010 *Business Case for Pickering B Continued Operation* Paul Pasquet, Pickering B's Senior Vice President, made the following remarkable statement:

"*The decision on whether a nuclear unit is at the end of its life is primarily an economic one*."

What this statement ignores is the simple fact that decisions about the operation of any CANDU reactor are actually based on *safety issues* stipulated in the station's operating license issued by the CNSC. If a system or component fails to meet fitness-for-service guidelines it must be repaired or replaced. Certainly, the decision to repair or replace something in a reactor is the responsibility of the operator. However, *the decision on whether a nuclear unit is at the end of its life or not is embodied in its operating license*.

Vice President Paul Pasquet's 2010 statement also ignores the fact that when it comes to end-of-life decisions, just *one* unexpected problem in an ageing component can lead to the failure of an entire system. And the notion of what is "safe" in a reactor becomes more and more problematic and poorly defined as the reactor ages. So I would say to Mr. Pasquet, predictions of rates of degradation of key components such as pressure tubes define a reactor's end of life, *not economics*.

In any case, Mr. Pasquet's 2010 declaration is at odds with OPG's position presented at a 2008 Ontario Energy Board Hearing, when OPG clearly indicated that a refurbishment of Pickering B *was required*:

“*Because of major equipment degradation issues that have developed over the operating life*

of the units which, if not addressed, could be life-limiting to the units.”

And as proof of its commitment to dealing with “equipment degradation”, OPG’s Board of Directors had, (in 2007), included a budget of $300 million for preliminary work on Pickering B refurbishment. Furthermore, in January 2009 the CNSC approved OPG’s plan to refurbish Pickering B, and in November of the same year OPG published its Pickering B refurbishment schedule which indicated that the first Pickering B Unit refurbishment would start in October 2016 and end in September 2019.

Question: Why in February 2010, after many years of arguing the need for the refurbishment of Pickering B, and less than 6 months after publishing the Pickering B refurbishment schedule, did OPG announce that it no longer believed it was necessary to refurbish Pickering B?

● At OPG’s Public Hearing on its *Application to Renew the Power Reactor Operating Licence for the Pickering Nuclear Generating Station*, held February 20 and May 29 to 31 2013, Mark Elliot, Chief Nuclear Engineer at PNGS, noted that the design limit of hydrogen in zirconium codified in the CSA Standard is 100 parts per million, and stated that Pickering NGS was currently at 53 parts per million of hydrogen. Mr. Elliot further stated that this is expected to be around 80 parts per million by the end of 2020, which he notes is still well below the design limit.

These statements by OPG’s Chief Nuclear Engineer do not accurately depict the truth about deuterium pick up by Zr-2.5%Nb pressure tubes, since they imply that ***all*** pressure tubes at Pickering currently contain ***exactly*** 53 ppm of deuterium, and this will increase to “around 80 ppm” by 2020. In reality the extreme variability of deuterium uptake in the body of operating pressure tubes has been recognized for a long time, (See for example AECL Report No. 00-31100-200-005 issued in July 1998), and, in the case of Pickering NGS, the observed tube-to-tube variation in D uptake ***is up to a factor of 3***. Such a large statistical variation in D pickup cannot be ignored in assessing pressure tube fitness-for-service. Thus, while it may be true that the ***average*** concentration of deuterium in

Pickering pressure tubes is currently 53 ppm, the 95% upper confidence level would be about 80 ppm. This means that out of the 1520 pressure tubes in service in the four Pickering B Units, over 75 have already picked up more than 80 ppm deuterium. And what is not mentioned by Mr. Elliot is the fact that pressure tube rolled-joints experience much higher D ingress rates than the body of the tube. Thus, at the 1995 *CANDU Maintenance Conference*, P. J. Richardson from AECL's Reactor Engineering Services presented a paper which stated that:

The rolled joints experience a much higher ingress rate than the body of the tube because of the additional ingress routes provided by the end fitting. This results in rolled joints reaching a hydrogen equivalent (Heq) concentration (= initial hydrogen + deuterium/2) above the Terminal Solid Solubility (Dissolution) (TSSD) at operating temperature during the design lifetime. As a result, rolled joint flaws capable of having flaw tip stresses exceeding the threshold for Delayed Hydride Cracking (DHC) initiation could accumulate hydrides, during cool-down cycles, and eventually initiate DHC.

This expectation is confirmed in AECL's 2003 Licensing Submission: *The Technology of CANDU Fuel Channels*, Report No: ACR USA 108US-31100-LS-001. Figure 9-13 of this AECL report shows that at the outlet end of removed pressure tubes, deuterium concentrations in excess of 120 ppm are already present at outlet rolled joints after only 149,000 EFPH of reactor operation.

<u>Data Request:</u> I am asking that OPG provide data for the concentration of deuterium picked up at the rolled joints of Pickering pressure tubes after 175,000 EFPH of reactor operation, and provide predictions of rolled joint deuterium concentrations to 210,000 EFPH and beyond.

● In OPG's April 2005 submission to the CNSC entitled: *Pickering GS B Outlet Feeders Required Wall Thickness Values*, (Document No. NK30-CALC-33126-00021 R000), and *Component Disposition Form* NK30-EVAL-33160-00003-R00, Pickering B outlet feeder operational lifetimes were estimated. The data submitted by OPG to the CNSC shows there are many feeders for which the required thickness limit would be reached before

2020. Examples of these pipes are:

Unit 5: B11E, B13E, C08E, C15W, D06W, F09E

Unit 6: C06E, K02W, N02E, O21E, Q03E

Unit 7: O02W, U06W

Unit 8: C17W, E04E, F20W, H02W, J02E, K02W, K21E, N02E, P21W, Q03E

Information Request: **I am asking OPG to provide data on the current status of these feeder pipes.**

● In July 2013 David Gunn, from CANDU Energy Inc.'s Fuel Channel Engineering Department, gave a presentation at the *International Congress on Advances in Nuclear Power Plants* (ICAPP 2013) entitled: ***Extensions to Fuel Channel Life***. This presentation opened with the following statement:

> *CANDU® fuel channels age during operation as a result of a number of different mechanisms, the most life limiting of which are due to degradation of the Pressure Tube (PT). The EC6 Reactor design requirement for fuel channel life has been extended from 210,000 to 245,000 EFPH which represents a 17% increase in planned performance. To achieve this extended operational requirement, a number of design improvements have been implemented or are being explored for the Enhanced CANDU 6® (EC6®) reactor.*
>
> *Irradiation induced deformation causes the pressure tube ID to increase which limits the ability of the primary heat transport system to remove heat from the fuel forcing the full power operation of the reactor to be reduced. To combat this, two modifications have been implemented; the wall thickness of the PT has been increased by 13% to lower creep response and secondly, a modified microstructure shown to be more resistant to in-service diametral strain has been targeted through tighter manufacturing controls.*

<u>Question:</u> To what extent are diametral creep and tube elongation the limiting factors on pressure tube life at Pickering as they were for Wolsong Unit-1 in 2009.

● Also at the ICAPP 2013 meeting, CANDU Energy Inc. staff member David Gunn stated in reference to on-going problems with first generation CANDU reactors:

> *Absorption of excessive hydrogen isotopes into pressure tube material can lead to failure through delayed hydride cracking. When the hydrogen concentration in the pressure tube exceeds the Terminal Solid Solubility (TSS) limit, brittle hydrides can form and reduce fracture toughness. Hydrogen ingress at the rolled joints is of particular concern due to three possible causes; Direct pickup as a result of general corrosion by the coolant, corrosion along the crevice between the PT and End Fitting (EF) and hydrogen migrating through the EF hub and into the PT. To reduce the galvanic effects between the PT and EF and to provide a barrier for migration of hydrogen from the EF into the PT, an ultra-thin Chromium barrier is being explored for incorporation into the EC6 design.*

<u>Question:</u> If the reactors at Pickering are considered to be safe to operate to 245,000 EFPH, why would the next generation of CANDU reactors require chromium plated rolled joints to operate beyond 210,000 EFPH?

● Designers of the next generation of CANDU reactors have stated that one of their goals is to prevent calandria tube failures in order to limit in-core damage in the event of a catastrophic rupture of a pressure tube. Thus new reactor designs have seamless calandria tubes that are significantly thicker than calandria tubes used in first generation CANDU reactors. This modification is necessary to increase calandria tube survivability in the event of spontaneous pressure tube rupture due to unstable crack propagation, and also to provide a stiffer support for the pressure tube to resist sag. It is noteworthy in this regard that in the 2013 Pickering Unit 5 outage, unexpected reductions in pressure tube to calandria tube gaps were measured during routine inspections.

<u>Questions:</u> If pressure tube-to-calandria tube contact is well understood for Pickering's fuel channels, how is an "unexpected" pressure tube-to-calandria tube gap measurement possible? How reliable are OPG's deformation codes? And why do next-generation CANDU reactors require thicker calandria tubes (with better deformation properties) than those used today?

● In order to determine the fracture properties of Zr-2.5%Nb pressure tubes irradiated to the end of design life, test samples are prepared and irradiated to high neutron fluences in high-flux test reactors such as OSIRIS in Saclay, France. However, even ten years of exposure in this reactor to its maximum available fast neutron flux of 2×10^{18} $n/m^2/s$ (E > 1MeV) is only about half of the 3.0×10^{26} n/m^2 (E > 1MeV) exposure received by a Pickering pressure tube after 30 years.

<u>Question:</u> Does OPG have any data on the in-reactor exposure of Zr-2.5%Nb pressure tube material to fast neutron fluences greater than 3.0×10^{26} n/m^2? If the answer is no, could OPG please provide examples of measurements of diametral creep, wall thickness, channel sag, PT/CT gap and tube elongation of pressure tube material with the highest known fast neutron exposures, and also provide the allowed limiting values of these quantities.

● It has recently been discovered that the Inconel X-750 garter spring spacers removed from currently operating CANDUs have become severely brittle and can no longer be expected to function as required over long-term reactor exposure. The CANDU industry has therefore initiated research to address the aging of Inconel X-750 spacers in order to determine service conditions for risk assessments and to demonstrate on-going fitness-for-service for currently emplaced spacers. It has been proposed that recoil energy deposition from the Ni-59(n,α)Fe-56 reaction in the garter spring inconel substrate leads to embrittlement and failure of this component.

Question: In view of this embrittlement mechanism, how can garter spring embrittlement be prevented and what evidence does OPG have to indicate that the garter springs in Pickering B Units will be fit for service to 247,000 EFPH?

● Pressure tube flaws such as scratches and dents in as-installed tubes, or debris and fuel bearing pad fretting marks, are important fuel channel aging issues. In order to prevent tube failures developing from these flaws, CSA N285.4 & N285.8 Standards were issued in 2005 as the codes for flaw fitness-for-service assessments of CANDU pressure tubes. When tensile stresses are applied to a crack tip the crack can grow when the hydrogen (or deuterium) concentration exceeds the so-called terminal solid solubility limit (TSS) for the precipitation of hydrides. For flaws with peak notch tip stresses greater than the threshold stress for delayed hydride cracking the flawed pressure tube is restricted to a limited number of thermal (heat-up and cool-down) cycles to prevent the onset of DHC.

Question: How many flaws have been detected in Pickering B pressure tubes where crack initiation criteria are not satisfied? What are the heat-up and cool-down cycle restrictions imposed on these flawed tubes? What are the requirements for re-inspection of these tubes?

Dr. Frank Greening

March 21, 2014

Subject: Additional Material for my Submission Re: Pickering B Hold Point

Dear Ms. Levert,

Please accept the text below as additional material *to be added to my previous (2) submissions* to the Commission as a written intervention at the May 7, 2014, public hearing.

Sincerely,

Frank Greening

Additional Submission:

● The CNSC requires OPG to estimate the maximum probable emission rates (MPERs) of radioactive species in airborne and waterborne effluents from Pickering NGS. For each effluent pathway, a maximum abnormal release scenario (MARS) is identified. A MARS is defined as a station event that occurs with a frequency of approximately once in ten years, and results in the highest predicted emission rate compared to other postulated scenarios for that pathway. This means that one such MPER event should occur if PNGS A & B are operated for another 5 years.

Questions: 1. What is the predicted waterborne effluent MPER for tritium in Pickering reactor building service water? 2. What is the predicted airborne effluent MPER for tritium in a Pickering reactor building exhaust stack. 3. What are the MARS for these postulated tritium releases?

● OPG Standard P-ST-03480-10000: *Radionuclide Effluent Monitoring System Requirements* states that uncertainty estimates should be made for each radionuclide group in each performance or control monitored effluent stream, and such estimates should be verified every 10 years or whenever the system is changed. This Standard stipulates that the error bounds on effluent monitoring data should not exceed ± 50 % for a radionuclide group emitted at ≥ 0.5 % of the monthly DRL and ± 100 % for a radionuclide group emitted at < 0.5 % of the monthly DRL.

Question: When were uncertainty estimates for Pickering's radiological emissions last carried out?

Request: Please provide copies of the most recent reports on:

(i) ***Measurement Uncertainty Calculations for Gaseous Radioactive Emissions***
(ii) ***Measurement Uncertainty Calculations for Radioactive Emissions from Active Liquid Waste***

● PNGS active liquid waste (ALW) sampled in Oct/Nov 2002 (as part of a COG project) detected U-235, U-238, Pu-239, Am-241 and Cm-244. (See: "*Characterization of Radionuclide Species in CANDU Effluents – Final Report on Analysis of Samples Received in 2002*" COG Report No: COG-03-3046, December 2003). The data show that gram quantities of uranium were being released from Pickering to Lake Ontario at that time.

Request: Please provide examples of more recent analyses of Pickering ALW samples for uranium and transuranic isotopes.

●Alpha particulate derived air concentrations (DACs) used for dose assessments at OPG nuclear sites are based on Am-241 because EPRI Guidelines *assume* that this radionuclide is a good representative of an unknown mixture containing TRU contamination. This assumption works reasonably well for American reactors. However, it is uncertain how well it works for CANDU reactors.

Over the next 5 years, the potential for the spread of alpha contamination will be heightened at Pickering B because of the increased number of reactor core inspections required to demonstrate pressure tube fitness-for-service compliance. These inspections involve the use of techniques such as SLAR and UDM. Source term characterization surveys carried out during these inspection campaigns have measured alpha-emitting radionuclide distributions that show very high levels of Cm-244 in many samples.

Request: Please provide data to demonstrate that alpha radionuclide distributions in inspection tool contamination smears collected at Pickering are consistent with the EPRI Guidelines noted above.

● Niobium-94 (Nb-94) is an activation product that is produced by neutron irradiation of Zr-2.5%Nb. Pressure tube ID surface spalling of the ZrO_2 corrosion film and fuel bearing pad wear lead to the mobilization and transport of Nb-94 throughout the primary heat transport system (PHTS) of an operating CANDU reactor. The mobilized Nb-94 is initially present as suspended particulate (crud) but eventually deposits on end fittings, feeder pipes and steam generator tubing. In addition, a portion of the mobilized Nb-94 is collected by the PHTS purification filters and ion-exchange columns. However, the *sum* of these contributions to the out-of-core inventory of Nb-94 is a measure of pressure tube corrosion and wear.

Short-lived zirconium-95 and niobium-95 may also be used to monitor pressure tube corrosion and wear but these radionuclides are also derived from fuel as cladding debris or fission product releases; thus Zr-95 and Nb-95 are not unambiguously attributable to pressure tube degradation. Nevertheless, studies have shown that for Zr/Nb-95 activity routinely detected in CANDU circuits, about 50 % is from pressure tube material – See CANDU Owners Group reports COG-96-22 and COG-01-044.

OPG report No. NK054-REP-07730-00027 Rev 000, issued in August 2009, provides Nb-94 data for Pickering B steam generators as follows:

Total Nb-94 inventory per SG = 7.3 E+8 Bq

Comparable data for Bruce steam generators has been reported by the CNSC in CMD: 10-H19, issued in September 2010 as follows:

Total Nb-94 inventory per SG = 1.5 E+7 Bq

These data show that corrosion and wear of pressure tubes is about 50 times higher in Pickering than in Bruce Units. This conclusion is supported by the observation of much higher reactor face radiation fields and dose rate contributions from Zr/Nb-95 for Pickering Units compared to Bruce Units.

Questions: Does OPG have an explanation for the higher pressure tube erosion/corrosion rates observed at Pickering compared to Bruce? How do these

higher pressure tube erosion/corrosion rates impact on the fitness-for-service of Pickering's aging pressure tubes over the next five years?

● As shown in Table 4.4 of the 2009 CNSC Report INFO-0792 entitled: *Investigation of the Environmental Fate of Tritium in the Atmosphere*, the levels of tritium in precipitation collected in the vicinity of Pickering have always been considerably higher than tritium concentrations in equivalent samples collected near Bruce and Darlington. Thus in 1988, tritium in precipitation at Pickering was reported to be 2900 Bq/L, or over 40 % of the Ontario Drinking Water Quality Standard, while it was only 440 Bq/L at Bruce and just 31 Bq/L at Darlington. Nevertheless, the CNSC INFO-0792 report shows that the tritium in precipitation at Pickering had dropped to about 300 Bq/L by 2002.

On the other hand, a 2008 IAEA Study entitled: *Modeling the Environmental Transfer of Tritium and Carbon-14 to Biota and Man* reported tritium concentrations in precipitation collected in 2002 at a location about 1 km NE of Pickering as follows:

Measured Monthly HTO Concentration in Pickering Precipitation in Bq/L								
Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sept
3670	1350	347	474	525	579	205	442	452

These data are clearly inconsistent with the CNSC data noted above for tritium in Pickering precipitation in 2002. IAEA rata are also inconsistent with OPG's "site study area" data for tritium in precipitation which (based on 2005 measurements) reportedly peaked at 411 Bq/L. On the other hand, OPG's Pickering B Environmental Assessment Report NK30-REP-07701-00002 issued in 2007, states that for the site study area tritium wet deposition is "*about 37,000 Bq/L*".

Most of these inconsistencies stem from the known variability of airborne tritium concentrations with distance from the source location, especially under conditions of wet deposition. Thus, even concurrent tritium-in-precipitation measurements at Pickering vary by more than a factor of two between nearby sites such as the locations identified as Montgomery Park Road and Alex Robertson Park, (which are both less than a few kilometers from Pickering).

Questions: Would OPG confirm that tritium wet deposition at Pickering is currently about 37,000 Bq/L? Is this level of tritium contamination likely to continue as long as Pickering Units operate?

● Highly elevated levels of tritium in groundwater have been identified at both PNGS A and B since 1997. Subsequent studies carried out between 2001 and 2006 have revealed the extent of this groundwater contamination:

PNGS A Unit1 moderator purification room pit had tritium concentrations up to 1.04×10^{10} Bq/L

PNGS A & B foundation drain sumps had tritium concentrations up to 1.3×10^{5} Bq/L

PNGS A reactor auxiliary bay environmental sumps had tritium concentrations up to 1.9×10^{8} Bq/L

PNGS B reactor auxiliary bay environmental sumps had tritium concentrations up to 8.0×10^{6} Bq/L

PNGS B irradiated fuel bay ground-tubes had tritium concentrations up to 4.0×10^{6} Bq/L

Several of these concentrations show the presence of contamination levels at the Pickering site above the CNSC limit of 3×10^{6} Bq/L for tritium in non-potable water. This is noteworthy because the sumps listed above discharge via a settling basin directly into Lake Ontario. Nevertheless, OPG's Pickering B Environmental Assessment Report NK30-REP-07701-00002 issued in 2007, states:

Slightly elevated concentrations of tritium in groundwater may occasionally occur under the foundations of the plant as a result of minor seepage of tritiated water.

Questions: What are these "*slightly elevated concentrations of tritium*" under the Pickering site? (Please provide tritium concentration data in foundation drainage sumps over the past five years and projected values to 2020)

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To:

Ms.Louise Levert

Secretariat Canadian Nuclear Safety Commission (CNSC)

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Subject: Additional Material for my Submission Re: Pickering B Hold Point

Dear Ms. Levert,

Please accept the text below as additional material ***to be added to my previous submissions*** to the Commission as a written intervention at the May 7, 2014, public hearing.

Sincerely,

F. R. Greening

Additional Submission:

● On March 24th, 2014 the CNSC issued CMD: 14-H2 describing its decision and reasons for its approval of OPG extending its operation of Pickering Units 1, 4, 5-8 beyond 210,000 EFPH. In CMD:14-H2 the CNSC provide a table summarizing the results of OPG's PSAs for Pickering A & B. Some of these PSA results are shown below:

Pickering PSA Results (Reported by the CNSC March 2014)

Plant Damage State	Pickering A (Core Damage Frequency)	Pickering B (Core Damage Frequency)
Internal at Power Event	1.6E-5	4.0E-6
High-Wind at Power Event	2.7E-5	8.0E-6

In order to further consider these *core-damage-frequency* (CDF) values it is instructive to look at PSA results for other CANDU reactors as reported by OPG, AECL and COG, (See for example the report COG-07-9012):

Estimated CDF Values for Internal (at-power) Events for CANDU Reactors

Station	Internal (at-power) Event Core Damage Frequency	Date of Estimate	EFPH at Time of Estimate
Pickering A	1.6E-5	2013	201,000
Pickering A	5.5E-5	2007	170,000
Pickering B	4.0E-6	2013	200,000
Pickering B	8.5E-6	2007	166,000
Bruce A	5.7E-5	2007	151,000
Bruce B	2.2E-5	2007	156,000
Darlington	3.4E-5	2007	105,000
Point Lepreau	1.7E-5	2008	161,500
Generic CANDU 6	2.0E-5	2007	-

These Core Damage Frequencies (CDFs) represent the likelihood of a serious accident occurring in a nuclear plant that is caused by an internal event such as a feeder pipe rupture, and is a measure of the *risk* associated with continued operation of the plant. What is remarkable about the CDFs reported above is that the values indicate that there has been a *decrease* in the risk of operating Pickering A & B Units over the past 6 years in spite of their approach to end-of-life status in this period. The CDFs presented above also show that Pickering A Units, after more than 200,000 EFPH, are *less likely* to experience core damage than Units at Bruce and Darlington with only 100,000 to 160,000 EFPH of operation.

These observations suggest that a nuclear plant becomes *safer* as it ages, which is certainly counter-intuitive and calls into question the validity of OPG's safety analyses of Pickering. The methodology used by OPG in making these PSAs is described in documents such as NK30-REP-03611-00021-R000 entitled: *Pickering B Risk Assessment Summary Report*, issued in 2013. This report is over 100 pages long, but nowhere in the document is the issue of *plant aging* mentioned. This omission is of concern, especially when there are detailed discussions of the impact of plant aging on safety assessments in OPG reports such as *Pickering B Safety Report*: NK30-SR-01320-00001 Rev 001 issued in 2007.

And the CNSC has been fully cognizant of the need to incorporate aging effects in CANDU plant safety assessments for many years. Thus, in August 2009 in E-Doc # 3413831 the CNSC presented a discussion of *Risk-Informed Regulatory Positions for CANDU Safety Issues* where we read:

One such issue is the adverse impact of plant ageing on safety and safety related systems to prevent or mitigate accidents. In particular, the concern is whether all the plant ageing mechanisms are identified and their impact are determined, addressed in an integrated manner and adequately accounted for in the shutdown system trip parameter set-point adjustments. The issue is that there is a concern that existing ageing management programs do not include a complete assessment of all the implications of plant ageing on the safe operating envelop. Therefore the primary risk *area related to this issue is “Negative Impact on Safety”. Licensees need to make sure that ageing effects are taken into account when establishing appropriate operating limits and conditions.*

And in December 2011 the CNSC re-asserted the importance of incorporating plant aging into safety assessments when it wrote:

Probabilistic Safety Assessment (PSA) is one of the most effective tools for the risk-informed decision-making. Thus, it is important to have credible and defensible PSAs representing adequately the actual risk profile of the plants. The current standard PSAs do not address important aging issues. For instance, reliability models of components are based on the "component constant failure rate" assumption, which is not valid for some components in the long term. Consequently, in order to be more realistic, aging-related models and the effects of test and maintenances in controlling the aging of safety components have to be applied.

The most important plant aging effects that impact on the PSAs for Pickering A & B include the following:

1. **Slow Loss of Regulation (LOR) and Neutron Overpower Protection (NOP)**: An analysis of this effect requires consideration of all possible neutron flux states.

Questions: (i) Has OPG carried out such a NOP accident analysis to 240,000 EFPH of operation?

(ii) Are the estimated trip set-points conservative over all flux shape groups?

2. **Loss of Flow**: Here the issue is to demonstrate that fuel sheath dry-out is precluded prior to the onset of flow oscillations and that trip coverage under aged conditions is the same as in the reference un-aged analysis.

Question: Has the impact of PHTS aging effects on electrical system failures been determined for a single heat transport pump trip event up to 240,000 EFPH of operation?

3. **Small Break Loss of Coolant Accidents**:

Question: Have analyses of small break LOCAs with initial break discharge rates up to 1000 kg/s been performed for aged conditions corresponding to 240,000 EFPH?

4. **Large Break Loss of Coolant Accidents**: Irradiation induced pressure tube diametral creep increases the coolant volume contained in a channel. This additional coolant volume results in a significant increase in the reactivity worth of the coolant void under LBLOCA conditions, and the initial power pulse predictions for a LBLOCA scenario would therefore be increased.

Question: Have all the thermal-hydraulic effects of PT diametral creep on LBLOCA been analysed up to 240,000 EFPH of operation?

5. **Steam Supply and Feed-water System Failures**: For Pickering B, steam line breaks and feed-water line breaks are the largest contributors to the Core Damage Frequency (CDF) and the Large Release Frequency (LRF). The HTS aging mechanism that has the largest potential impact on this event is fouling in the steam generators leading to a reduction in the heat transfer to the secondary side, which results in higher steady state RIH temperatures.

Question: The heat sink effectiveness is affected by the steam generator recirculation ratio and separator efficiency, which deteriorate with age. Has the quantity of water available for an interim heat sink under accident conditions been evaluated up to 240,000 EFPH of operation?

6. **Moderator System Failures**: The most important of these are loss of inventory and loss of heat sink. In a loss of moderator inventory event, a top-to-bottom flux tilt develops, raising concerns over reactor trip coverage and safe Unit operation. Similarly, in a moderator cooling failure, pressure tube diametral creep can affect the margin to dry-out following the collapse of the moderator level after the rupture discs burst.

Question: Has the impact of moderator system failures been assessed up to 240,000 EFPH of operation?

● The following is a list of some of the more important systems, structures and components (SSCs) that should be included in a full-scope aging management study of Pickering NGS:

- Fuel channels
- Calandria
- Control Rod mechanisms
- Reactor/Calandria supports
- Primary Heat Transport Systems
- Emergency Core Cooling Systems
- Steam Generators
- PHTS circulating pumps

- High safety-significant pumps, valves and connecting piping

- Emergency diesel generators

- Containment structures

- Containment penetrations (mechanical and electrical)

- Containment isolation valves

- Containment liners

- Steam and Feed water piping, pumps and valves

- Safety related heat exchangers

- Piping supports

- Spent fuel bays

- Containment ventilation systems

- Fixed radiation monitors

The most important degradation mechanisms for these SSCs are as follows:

- Hydriding

- Fretting and wear

- Fast fracture

- Low cycle fatigue

- Irradiation embrittlement

- Creep

- Stress corrosion cracking

- Erosion-corrosion including FAC leading to wall thinning

- Chemical corrosion with or without galvanic effects

- Localized corrosion (e.g. pitting, crevice, etc)

- Microbiologically induced corrosion

Other station aging issues are:

- Continued operation of circuit breakers with dirty and/or oxidized contacts
- Loss of protective coatings and degradation of cabling
- Deterioration of fuses, resistors, switches, coils, etc
- Computer and I&C equipment failures (See IAEA-TECDOC-1402, Page 9)
- Plastic and elastomer degradation leading to seal failures
- Concrete degradation

Questions:

1. Has OPG identified which of these degradation mechanisms/aging issues are serious in terms of their impact on the function of the SSCs listed above?
2. Could OPG please describe what PSA studies it has performed, or plans to perform, that include an assessment of ageing effects?

● In 2007 OPG issued the report: *Refurbishment and Continued Operation of Pickering B NGS: Environmental Assessment*, NK30-REP-07701-000014 (December 2007). In this document we read:

Pressure tube refurbishment of CANDU nuclear generating stations is an element of the plant design assumed to be required at some point in the life of the ageing plant, generally after 25 to 30 years of operation. Refurbishment would allow PNGS B to continue to operate beyond its current life expectancy.... During refurbishment outages, after the reactors have been de-fuelled and dewatered, the fuel channel assemblies and feeder pipes will be replaced. In addition, the steam generators in each of the units will be replaced....... For EA purposes, operation of all four refurbished reactors is anticipated to continue to the new end-of-life which has been conservatively assessed to be 2060 for the last of the four units. Also for EA purposes a probabilistic risk assessment (PRA) is deemed to be most appropriate for bounding accident scenarios.

The Pickering B PRA results provided in the 2007 report NK30-REP-07701-000014 show that the highest predicted core-damage-frequency (CDF) *for the refurbished station* is 1.3×10^{-6} per year. This should be compared to the CDF of 4.0×10^{-6} per year quoted by the CNSC in March 2014 for the continued operation of Pickering B.

Question: Could OPG please provide an explanation for the differences in the CDFs noted above that were calculated for Pickering B in 2007 and 2014?

● In the CNSC Document: *Record of Proceedings, Including Reasons for Decision In the Matter of Applicant Ontario Power Generation Inc*. Subject Application to Renew the Power Reactor Operating Licence for the Pickering Nuclear Generating Station Public Hearing February 20 and May 29 to 31, 2013, we have the following item:

3.11.1 Effluent and Emissions Control, Item 216:

OPG noted that there were no Derived Release Limit or Action Limit exceedances for Tritium, Beta/Gamma or Carbon-14 emissions to water on an annual basis during the current license period.

This is an entirely meaningless and misleading statement by OPG because the DRLs for Pickering are such that the station *can never exceed its waterborne tritium release limit, even if it was to discharge its entire inventory of tritium into Lake Ontario*. This is in fact true for all the DRLs currently adopted for tritium releases to the environment in liquid effluent streams from Bruce, Pickering and Darlington. These DRLs vary somewhat between stations but a value of 2×10^{18} Bq/y is a reasonable average for the purposes of illustrative calculations. Now 2×10^{18} Becquerels is equal to 54 million Curies. So let's consider the implications of releasing this amount of tritium – 54 million Curies – into the Great Lakes, bearing in mind that our nuclear safety experts consider this amount of tritium - *released each and every year* - to have no detrimental effects on the local environment or those living close to the Bruce, Pickering or Darlington nuclear power stations.

The main source of tritium in a CANDU reactor is the moderator which typically contains about 300,000 kilograms of heavy water, or D_2O. "Virgin D_2O" contains no tritium, but tritium slowly builds up in a moderator during reactor operation so that a mature reactor typically contains about 20 Ci of tritium per kilogram of moderator D_2O. Thus we see that one CANDU reactor moderator contains 20 (Ci/kg) × 300,000 (kg) of tritium, or 6 million Curies of tritium. Since there are four CANDU Units operating at Bruce A, Bruce B, Pickering B and Darlington respectively, we have a total inventory of 24 million Curies of tritium at each of these nuclear power stations. However, we also see that 24 million Curies is less than ½ of the 54 million Curie annual release limit or DRL for waterborne tritium from these stations.

This simple calculation reveals the remarkable fact that the Bruce, Pickering and Darlington nuclear power stations can ***never*** exceed their waterborne tritium release limits, *even if they discharge their entire inventories of tritium into Lake Huron (for Bruce) or Lake Ontario (for Pickering and Darlington)*. Similar calculations show that C-14 DRLs for Bruce, Pickering and Darlington can never be exceeded.

Question: What is the purpose of imposing a radionuclide emission limit on a nuclear power station that can never be exceeded no matter how poorly the station is run and why would OPG boast that there were "no Derived Release Limit or Action Limit exceedances" for Tritium or Carbon-14?

● In the CNSC *Record of Proceedings, Including Reasons for Decision In the Matter of Applicant Ontario Power Generation Inc.* we have in Section *3.11.1: Effluent and Emissions Control*, Item 218:

Some intervenors ... expressed concerns regarding releases of Iodine-131 and the possible links to thyroid cancer in the region. The Commission asked for more information concerning these releases. CNSC staff responded that the levels of radioactive iodine released by the Pickering NGS were not detectable and that there was no link to thyroid cancer at these levels.

This claim is reiterated by “CNSC staff” in Section 5.2 of the RADICON report where we read:

Radioactive iodine, which is the primary cause of radiation-related thyroid cancer, was below detection limits of the in-stack sampling monitors at all three NPPs for the entire study period.

It is also reiterated in Section 2.3 of the "Technical Report" for the RADICON Study where we read:

Radioactive iodine releases from Pickering NPP have consistently remained below the limits of detection.

Unfortunately these claims by the CNSC are simply incorrect when one looks at actual radioactive iodine **emissions** data reported for Pickering, Darlington and Bruce for the study period of the RADICON investigation, namely 1990 – 2008. And what is most remarkable about these data is that they are reported in two documents from the CNSC, the publishers of the RADICON report. I am referring to:

(i) CNSC Report: INFO-0210/REV.10: *Radioactive Release Data from Canadian Nuclear Generating Stations 1990 to 1999* [R5]

(ii) CNSC Report: INFO-0210/REV.13: *Radioactive Release Data from Canadian Nuclear Generating Stations 1999 to 2008* [R6]

These reports include data for the maximum airborne radioiodine releases from each station as presented in Table 1.

Table 1: Maximum Airborne Radioiodine Released by Pickering, Darlington and Bruce

(For the Period 1990 – 2008)

Station	Maximum Release GBq/year	Year of Maximum Release
Pickering A	0.32	1990
Pickering B	0.10	2001
Darlington	0.15	2002
Bruce A	0.06	1990
Bruce B	0.12	2007

Contrary to the claims of the RADICON Report, the data in Table 1 clearly show Pickering, Darlington and Bruce had significant radioiodine emissions in the period 1990 – 2008. Furthermore, an analysis of actual station data shows that the detection limit of a stack sampler for radioiodine, calculated as 2 times the standard deviation in the background, is typically about 0.2 MBq per weekly sample, showing that a station's radioiodine background is about 0.01 GBq/yr. Thus it is undeniable that Pickering, Darlington and Bruce had measurable radioiodine emissions between 1990 and 2008. The fact that the RADICON Report claims radioactive iodine, "*was below the detection limits <u>of the in-stack sampling monitors</u> at all three NPPs for the entire study period*", raises questions about the veracity of the RADICON Report and the reliability of information coming from the CNSC.

I-131 is a high-yield fission product that accumulates in CANDU fuel bundles throughout their approximately 1-year residence in a reactor core. In the event of a fuel defect, high-pressure D_2O penetrates the Zircaloy cladding and directly contacts the irradiated UO_2 resulting in the release of I-131 and other fission products to the primary heat transport system (PHTS). Gamma-spectrometry of grab samples of PHTS D_2O readily detects the presence of elevated levels of I-131. Irradiated fuel, including defective fuel, is discharged from core by the fuelling machine and transferred by a conveyor to the primary irradiated fuel storage bay. This operation, which takes about 10 minutes to complete, is carried out with the discharged fuel bundle maintained under cooling water except for a period of about 4 minutes when the bundle is exposed to air in the fuel transfer duct, (See Pickering Safety Reports for more details). At this time I-131 may be released from a fuel defect and enter the primary irradiated fuel bay ventilation exhaust system. An estimate of the amount of I-131 that could be released from a defective fuel bundle during its transfer from core to the primary storage bay may be derived from the safety analysis of a related event – an off-reactor fuelling machine accident at the irradiated fuel port described in Pickering Safety Reports. These reports estimate the amount of I-131 that could be released from damaged fuel to the irradiated fuel bay exhaust system under a variety of conditions and are typically 1 - 2 % of the bundle inventory. The highest releases assume a 10 minute air exposure of the damaged fuel with a momentary temperature excursion to 1000°C. For the more typical freshly discharged fuel exposure period to air of 4 minutes, an I-131 release of 0.5% of the bundle inventory of 13,700 Ci, (5.07×10^{14} Bq), or 68.5 Ci, (2.535×10^{12} Bq), is usually assumed. For an expected removal efficiency of 99.9 % for the HECA filters in the stacks, this would correspond to an acute release of 0.0685 Ci or 2.53 GBq of I-131.

Questions: 1. Does the CNSC stand by its claim that **"***radioactive iodine was below the detection limits of the in-stack sampling monitors at Pickering for the entire study period 1990 – 2008"*?

2. Does the CNSC understand the difference between "atmospheric sampling" and "stack sampling"?

3. Would the CNSC please acknowledge the fact that I-131 environmental releases from Pickering Units are dominated by *spike emissions* from defective fuel?

4. Who peer reviewed the CNSC's RADICON report?

● In the 2013 CNSC Pickering Licensing Hearings we also have Item 228:

The Commission asked for more information concerning OPG's monitoring practices and enquired about the possibility of releasing raw monitoring data to the public. An OPG representative explained that OPG conducts continuous monitoring on a daily basis with low thresholds for investigation and action levels. The OPG representative noted that OPG conducts its environmental monitoring in accordance with CSA Standards. The OPG representative stated that while OPG does not regularly publish its monitoring data, it provides quarterly reports to the CNSC and publishes an annual report.

Unfortunately, averaged emissions data provide poor dose estimates as demonstrated by the example of tritium. The CNSC sponsored RADICON study was largely based on tritium release data for Pickering, Darlington and Bruce for the period 1990 to 2008. The study used station emissions data averaged over one-year intervals to determine doses to the public using water and atmospheric dispersion modeling. This is similar to the approach generally used to determine DRLs for NPPs operating here in Canada – see

D. Hart: *Derived Release Limits Guidance* CANDU Owners Group Report COG-06-3090-R2-I, November 2008. However, this approach is subject to significant error when the emissions are highly variable over time, especially if the emissions include large short-term spikes caused by leaks or spills from systems containing high activity liquids such as tritiated moderator D_2O. In order to understand how data averaging leads to erroneous dose estimates we need to look at real station emissions data such as those presented in Figure 1 (not included) which show actual airborne tritium releases from an operating CANDU station. Figure 1 (Available on rquest) shows a large tritium emission spike of 1730 Ci in the 8^{th} week of data collection. This spike is almost 9 times the weekly average of 200 Ci/week and was caused by a major spill of 20 Ci/kg of moderator D_2O. What is noteworthy about this tritium release is that Environment Canada records show there was a period of heavy rain, (totaling 14.2 mm), at the time of the tritium release. In addition, rainwater samples collected on-site after this period of rainfall showed tritium levels up to 30,000 Bq/liter.

The occurrence of large spikes in a station's airborne tritium releases leads to an increased dose to residents living near the facility. However, the increased dose will be underestimated if the spike release is simply added to other weekly data and averaged as part of the yearly releases. Furthermore, the chance coupling of an emission spike with a period of heavy rainfall serves to further amplify the dose to nearby residents, particularly if they are farmers using well water. For off-site doses from short-term releases it is normally assumed that the wind is blowing steadily in the direction of a receptor under stable atmospheric conditions. Such conditions assume no crosswinds or plume meander and are also assumed to prevail for periods up to 2 hours following an accidental release. The receptor is thus exposed to the maximum possible concentration of the released radionuclide. The associated atmospheric dilution factor is usually referred to as the "2-hr (X/Q)" where X is the atmospheric concentration of the species of interest, in units of Bq/m^3, and Q is its release rate from the source in units of Bq/s. By comparison, for the assessment of annual "routine" releases, a sector-averaged Gaussian plume model is usually adopted rather than one in which the plume is directed at the receptor location throughout the exposure period. For this reason, routine release calculations typically use site-specific meteorological data to construct joint-frequency tables which categorize hourly observations of wind speed and direction into 7 stability classes, 16 compass directions, and 6 wind-speed ranges. Such tables are then used to calculate annual-average atmospheric dilution factors, also referred to as long-term (X/Q)s.

To demonstrate the importance of the proper choice of dilution factors in making reliable dose estimates one needs to consider both short-term and long-term (X/Q)s and determine the doses from the tritium emissions such as those presented in Figure 1. The dose calculations need to consider inhalation, skin absorption, potable water and vegetable consumption for an individual residing close – e.g. 1 kilometer - to the plant. In addition, because the large spike release of tritium seen in Figure 1 was accompanied by a period of heavy precipitation, the effects of wet deposition of tritium must be included in the calculations.

The wet deposition flux F_w ($Bq.m^{-2}.s^{-1}$) is assumed to be proportional to the tritium release rate multiplied by a washout coefficient Λ which is the fraction of HTO removed per second. Allowing for plume dispersion we then have,

$$F_w = Q \Lambda \Phi / (x \bar{u} \theta)$$

Where,

Q is the tritium release rate in $Bq.s^{-1}$
Λ is the washout coefficient in s^{-1}
Φ is the joint frequency of occurrence of wind direction and rainfall in the receptor sector
x is the distance from source to receptor in m
ū is the average wind speed in $m.s^{-1}$
θ is the angular width of the sector in radians

It follows that the concentration of tritium in rainwater, C_w(Bq/liter), is given by:

$$C_w(\text{Bq/liter}) = F_w T / (1000 P)$$

Where,

T is the duration of the study period in s
P is the total amount of rain in the study period in m
1000 is the factor to convert $Bq.m^{-3}$ to Bq/liter

This washout coefficient formalism was used to calculate the concentration of tritium in rainwater for the annual average (200 Ci/week) and short-term (1730 Ci) releases shown in Figure 1. The average wind speed was set at 3 $m.s^{-1}$ and the washout coefficient was conservatively assumed to be $1 \times 10^{-5}\ s^{-1}$. The annual average rainfall at the site was 1045 mm and the short-term (24 hr) rainfall accompanying the tritium emission spike was 14.2 mm. The resulting tritium concentrations in rainwater were 217 Bq/liter and 38,000 Bq/liter for the annual average and short-term releases, respectively. The calculated tritium dose estimates for these exposures shows how the averaging of emissions data effectively masks the contribution of emission spikes. This effect comes about because tritium concentrations downwind from a source decrease with time due to increased plume meander. The long-term calculation averages out the dose impact to a specific group by sharing it with a hypothetical “average dose recipient". However, *real* emission peaks are capable of delivering significant doses to individuals located at the plume's point of impingement, and the dose impact is only further increased by concurrent periods of heavy precipitation.

<u>The Example of Radioiodine</u>:

In spite of the RADICON Report's claims that CANDU NPPs do ***<u>not</u>*** release measureable quantities of radioiodines – I-131 is routinely detected in airborne emissions from Pickering, Darlington and Bruce. This I-131 is not only emitted as a steady "background", but more typically as a short duration spike. All of these emissions are subject to rain washout if the release coincides with a local rain event. Thus although Pickering, Darlington and Bruce typically have relatively constant I-131 emissions ~ 1×10^{-5} Ci/week, *<u>single short-lived spikes up to 10 mCi have also been observed</u>*.

As shown above for the example of tritium, studies such as those described in the RADICON Report use station emissions data *<u>averaged over one-year time periods</u>*. This averaging results in tritium doses that significantly underestimate the dose to members of the public living close to the point of impingement of an emission plume. It is readily shown that this type of data averaging also effects the estimation of doses from I-131 and significantly underestimates the dose-to-public when the emissions are dominated by spike releases. Similar conclusions have been reported by other researchers. However, it is important to note that the dose calculated above for an acute release of I-131 is based solely on dry deposition and does *<u>not</u>* include the effects of wet deposition. Wet deposition of iodine is difficult to model, but empirical evidence from the Chernobyl and Fukushima accidents shows that it can substantially increase iodine uptake by plants, animals and man.

Information on the deposition of iodine from the Chernobyl accident is documented in: "*The Chernobyl I-131 Release: Model Validation and Assessment of the Countermeasure Effectiveness*", Report of the Chernobyl I-131 Release Working Group of EMRAS Theme 1, August 2007. Data for the contamination of rural locations near Prague in the Czech Republic are especially useful in assessing the contribution of wet deposition. The first indications of the presence of a contaminated plume over Czech territory were detected during the night of April 29th 1986. In the morning of April 30th measurements were started by the Czech monitoring network which subsequently detected three passages of contaminated air through the territory. The first one occurred during the night between the 29th and 30th of April; the second one occurred on May 3rd and 4th, and the third one began on May 7th and ended on May 10th, 1986. Data reported in Annex IV of the EMRAS Report show that daily I-131 fallout activities in three locations near Prague rose sharply after periods of rainfall on April 30th and May 9th, 1986. Thus, while surface concentrations of I-131 in the Czech Republic were typically less than 1000 Bq/m^2 throughout this period, I-131 “hot-spots” in excess of 8000 Bq/m^2 were observed immediately after these rain events.

The radiation dose from the Fukushima accident was impacted by iodine washout in a similar manner. Thus, starting on March 15th, 2011, radioactive releases from Unit 2 of the Fukushima Dai-ichi plant were dispersed over Japan during a period when meteorological conditions were changing rapidly. The releases occurring in the morning of March 15th are believed to have moved in a southerly direction, along the coast, whereas those occurring during the nights of the 15th and 16th of March moved to the northwest, crossing an intense precipitation front moving in the opposite direction – see for example: “*Fukushima, one year later: Initial analysis of the accident and its consequences*” IRSN Report IRSN/DG2012-003, March 2012.

Villages within 50 km of Fukushima Dai-ichi, but located southwest of the plant, experienced very little rainfall and deposition of I-131 was typically < 10 kBq/m^2. By comparison, the villages of Iitate, Namie, Kawamate and Katsurao, all located within 50 km of the plant and all more or less directly in the path of the Fukushima plume, experienced heavy rainfall on March 16th, 2011, and were subject to I-131 deposition > 1000 kBq/m^2. This illustrates how detailed weather patterns, and particularly periods of heavy precipitation, influence wet deposition of radionuclides and must be factored into the estimation of doses due to the consumption of contaminated milk.

In summary, the RADICON Report on radiation and the incidence of cancer around Ontario NPPs from 1990 to 2008 contains a number of false assertions which invalidate most of its conclusions. One of the most remarkable claims in the RADICON Report that is demonstrably untrue is that CANDU NPPs have no detectable radioiodine emissions. Nevertheless, a simple check of CNSC reports on radioactive releases from Canadian NPPs shows these plants typically release 2×10^8 Bq of I-131 per year.

However, the most serious problem with the RADICON study is its use of ***averaged*** meteorological data coupled with ***averaged*** annual emissions to estimate doses to members of the public living near Canadian NPPs. Now it is true that this type of data averaging is commonly employed in the calculation of derived release limits (DRLs) and is arguably a valid approach to dose estimation for relatively constant (continuous) emissions. However, CANDU NPP’s emissions, such as those from Pickering, are far from constant; on the contrary, they are dominated by short-term spike releases and are therefore subject to far less dispersion than long-term “routine” emissions. In addition, doses resulting from the wet deposition of radionuclides – especially doses from spike releases that coincide with periods of heavy precipitation - are inevitably grossly underestimated by long-term averaging.

The use of long-term averaging has additional problems besides the loss of detail of specific events; the very concept of “an average value” losses its meaning if the data being averaged do not exhibit a Gaussian distribution. A detailed analysis of CANDU emissions over extended periods of time (up to 10 years) shows that the data invariably exhibit power or log-normal, rather than Gaussian, distribution laws. Gaussian distributions drop off quickly because under this statistic large release events are extremely rare; by comparison, power law or log-normal distributions drop off more slowly. Thus large release events - the events in the tail of the distribution - are more likely to happen in a power law or log-normal distribution than in a Gaussian.

Question: Given that detailed (weekly) emissions data, and the associated meteorological conditions, are available for all NPPs operating in Ontario, why did the RADICON study use averaged, rather than disaggregated, data to assess doses to exposed individuals?

Information Request: Could OPG please provide the weekly airborne emissions data for tritium, radioiodine and C-14 from the contaminated and non-contaminated stacks of Pickering A & B for the past five years?

● **Uranium:** CANDU reactors are fuelled with about 120 tonnes of natural uranium. This fuel consists of 6240 “bundles”, each containing about 20 kg of uranium dioxide, UO_2, in the form of cylindrical pellets packed inside thin-walled zirconium alloy fuel “pencils”. During reactor operation some uranium is invariably present in the primary heat transport system (PHTS) from two sources: (i) “tramp” uranium, and (ii) defective fuel. Most of the uranium that enters a reactor coolant is retained on ion-exchange (IX) columns in the heat transport D_2O purification circuit. However, resin slurrying operations inadvertently permit the direct transfer of uranium-contaminated water to ALW tanks because a significant portion of the uranium in a PHTS consists of species that are present in a colloidal particulate form that passes through IX columns.

The principle isotope of natural uranium is U-238 with an abundance of 99.27%. DRLs for waterborne emissions of U-238 from large CANDU reactors are typically ~ 8×10^{14} Becquerels or 21,600 Curies per year. The condenser cooling water flows for these reactors are in the range 155 to 168 m^3/s. It follows that the DRL for U-238, expressed as an average liquid effluent concentration at the station outfall, is about 160 Bq/litre.

It is very instructive to consider uranium DRLs and effluent concentrations in mass units instead of the more familiar radiochemical units of Becquerels or Curies. Thus, DRL values for U-238 reported in Bq may be expressed in units of micrograms using the conversion factor 1 Bq of U-238 = 80 μg of Uranium. Hence on a mass basis, the DRL for waterborne U-238 is:

$$8 \times 10^{14} \text{ Bq} \times 80 \text{ µg/Bq} = 640 \times 10^{14} \text{ µg} = 640 \times 10^{2} \text{ Mg} = 64{,}000 \text{ tonnes of U-238}$$

This unit conversion of the DRL for waterborne U-238 reveals the absurdity of the currently accepted value because it is more than 100 times a station's entire inventory of uranium. But there are further ramifications to the magnitude of these CNSC approved uranium releases. For the vast majority of radioactive species identified in liquid effluents from CANDU stations the detrimental effects of the alpha, beta or gamma emissions, (responsible for the radiation dose), far outweigh any additional chemical toxicity that may be associated with these species. However, this generalization does ***not*** apply to U-238 because of its low specific activity compared to its well documented nephrotoxicity, (See, for example, *The Fernald Dosimetry Reconstruction Project Final Report*, U.S. CDC, September 1998). This chemical toxicity is reflected in the low concentration of uranium recommended in the Canadian Water Quality Guideline for the protection of aquatic life which was set at 15 µg of U per litre by the Canadian Council of Ministers of the Environment in 2011.

The nephrotoxicity of uranium is due to its propensity to accumulate in the kidneys of mammals exposed to low concentrations of uranium – See data reported by R. W. Leggett in "*The Behaviour and Chemical Toxicity of Uranium in the Kidney: A Reassessment*", Health Physics 57, 365 – 383, (1989). In this respect uranium is similar to other heavy metals such as arsenic, selenium, cadmium, lead and mercury in their tendency to induce significant negative health effects in humans after exposure to remarkably low concentrations of these species.

Question: Does the CNSC stand by its DRL for uranium in waterborne effluents from Pickering of 8×10^{14} Becquerels per year or 64,000 tonnes of U-238?

● OPG's Pickering B Environmental Assessment Report NK30-REP-07701-00002 issued in 2007, states in Table 4.6.2:

> *C-14 emission levels to air have increased since the summer of 2005 due to a leak in the Annulus Gas System (AGS) of Unit 7. Since a permanent repair cannot be completed until the refurbishment outage and timing of a partial repair is uncertain due to the potential impact on other systems, the elevated emission level of 1.57×10^{13} Bq/yr will be used.*

Annulus gas systems in OPG's CANDU reactors use CO_2 gas at slightly above atmospheric pressure. While the Pickering Unit 7 "CO_2 leak" was first detected in 2005, (COG OPEX 38103, "*Indications of Annulus Gas CO_2 Leakage into the Moderator System*" 04-Sep-2005), a series of engineering reviews (including a third party assessment) determined that Unit 7 could safely operate until 2010. However, it was later found (Pickering SCR P-2008-09222, issued 08-Apr-2008) that the *calandria tube* in channel A13 had a through-wall crack which allowed CO_2 and D_2O exchange between the AGS and the moderator, ***and*** there was in fact no D_2O leaking from a rolled joint *as previously assumed*. The CNSC have noted that the OPG management decision to allow Unit 7 to continue to operate for over two years based on this incorrect assumption put the plant at increased risk.

Questions:
1. What were the AGS and moderator cover gas purge frequencies for Unit 7 between 2005 and 2008?
2. Please explain the alleged connection between higher than normal C-14 emissions in the period 2005 -2008 and an AGS leak, when an AGS produces very little $^{14}CO_2$?
3. In what way was the ALARA principle being followed by allowing a leak to go unrepaired for 2 years and thereby substantially increase Pickering B's C-14 emissions over this time period?
4. In what way did the Unit 7's AGS leak compromise the early detection of D_2O ingress from a pressure tube failure? (As required for the leak-before-break methodology specified in Section 5.3 of CSA Standard N285.2)

● The history of technology over the past 100 years is replete with accidents involving catastrophic failures of some of the most exalted examples of technological excellence. These accidents were often accompanied by great loss of life, and in many cases the cause of the accidents were traced back to simple design flaws.

Examples of Accidents Caused by Unrecognized Design Flaws

Accident	Date	Cause	Fatalities
BOAC Comet Crash	January 10th 1954	Fatigue Cracked Rivets	35
Space Shuttle Challenger	January 28th 1986	O-ring Failure	7
Air France Concorde Crash	July 25th 2000	Burst Tire Debris	113
Space Shuttle Columbia	February 1st 2003	Insulating Foam Debris Impact	7

All of these accidents have the common characteristic that *they were preventable had the design flaws been recognized before the accidents took place*. And it is important to note that the designers of the engineering marvels noted above believed that their creations were safe and essentially "accident-proof". Not surprisingly, the history of nuclear power worldwide shows the same pattern of initial success with a new technology, followed by unforeseen disasters and protracted periods of damage control – Chernobyl and Fukushima being two well-known examples of this sequence of events.

So, it is reasonable to ask: are CANDU reactors so much better than other reactor designs that they will prove to be immune to unforeseen failures? Certainly there is no *a priori* reason to believe CANDU reactors are inherently foolproof. But let's consider for a moment how OPG tries to convince us that Pickering's nuclear facilities are safe. We are told that safety is assured through the process of probabilistic safety assessment. And this is based on "*a description of specific, important malfunctions and accident events that have a reasonable probability of occurring during the life of PNGS B*". Furthermore, the so-called "*design basis accidents*" are to be "*stylized scenarios intended to test the conceptual limits of the design*."

The *Oxford English Dictionary* defines the word *scenario* as follows:

Scenario: *noun* 1. A table of the appearances of characters in a dramatic work.

2. Complete plot of a film or play with details of scenes.

3. Imagined sequence of future events.

Evidently OPG is using definition 3 – "imagined events" - when it refers to accident scenarios. Thus, OPG's much vaunted PSA process starts with the compilation of a list of all the accident scenarios that can be *imagined* to occur over the lifetime of the station. For a CANDU station such as Pickering B the list of imagined accident scenarios classified by their initiating event includes:

1. Loss of Reactivity Control from Nominal Flux Shape
2. Loss of Reactivity Control from Distorted Flux Shapes
3. Unbounded Power Excursions
4. Bounded Power Excursions
5. Heat Transport System Pressurization
6. Heat Transport System Depressurization
7. Inadequate Trip Coverage (for some power levels and coolant loss rates)
8. Out-of-Core LOCA
9. End-Fitting Failure
10. Pressure Tube/Calandria Tube Failure
11. Flow Blockage in a Fuel Channel
12. Fuelling Machine Induced LOCA
13. Stagnation Feeder Break
14. Pressure Tube Failure With and Without Intact Calandria Tube
15. Heat Transport Pump Gland Seal Failure
16. Failure in the Heat Transport Feed and Bleed Circuit
17. Feed Line Break
18. Bleed Line Break
19. Instrument Tubing Failure

20. Steam Generator Tube Failure
21. Large Break LOCA
22. Loss of Class IV Power
23. ECIS Impairments - Failure of Boiler Crash Cool-down
24. ECIS Impairments – Failure of ECI Injection
25. Turbine Trip
26. Loss of Steam Generator Pressure Control
27. Loss of Steam Generator Level Control
28. Level Control Failure
29. Steam System Pipe Break
30. Feed-water Failure
31. Loss of Heat Transport Pump Power
32. Heat Transport Pump Seizure
33. Shutdown Cooling System Failure
34. Maintenance Cooling System Failure
35. Moderator Inventory Failure
36. Deuterium Gas Deflagration
37. Tube Failures in the Moderator Heat Exchanger
38. Cable Insulation Failure
39. Fire or Explosion at Power
40. Seismic Event at Power

41. Flooding Event at Power

42. High-Wind Event at Power

43. Malevolent Act

44. Aircraft/Missile Impact

45. Operator Error

It is important to note that most of these scenarios have never occurred in the less than 1000 reactor-years of operation of large CANDUs worldwide, so there is very little statistically meaningful “real world” data on the frequencies of these accident initiating events. Nevertheless, many initiating event frequencies estimated by OPG are reported to two, or sometimes three, significant figures even though they are less than 1×10^{-3} per year.

Questions:

1. How can OPG be sure it hasn’t missed, (i.e. failed to imagine), a potentially significant

accident scenario?

2. Because OPG’s PSAs are prone to under-counting accident scenarios – since risk is only

estimated for enumerated reactor states – doesn’t failure to account for unknown and

serially cascading beyond design-base accident scenarios leave an un-measurable model

error in the core damage frequency estimates?

3. Given that reliability distributions follow a “bathtub curve”, how does OPG estimate

initiating event frequencies for component failures that have never occurred, and what are

the uncertainties in such estimates?

4. Is OPG able to provide an indication of the level of confidence in the reported PSA results?
5. How is “operator error” quantified in the 2014 Pickering B PSA?

● At the March 27th 2014 Public Hearing on Pickering’s Continued Operation beyond 210,000 EFPH, Bruce Power VP, Mr. Frank Saunders, provided his thoughts on PSAs and the topic of large release frequencies and stated that a station’s Safety Report looks at “*each and every accident sequence, the isotopes that would be released and the impact on the critical group.*” He then goes on to state that for all of these postulated accidents, the PSA ‘*just takes the worse-case scenario and says what it would look like*”. Mr. Saunders concludes: “*<u>Prevent the worst case, and you will prevent the rest</u>*”.

Question: Could Mr. Saunders please explain what this comment means, and show how PSAs "prevent" accidents? Also could he show how the identification and assessment of a steam line break could prevent a stagnation feeder break or a pressure tube failure?

● At the March 27th 2014 Public Hearing on Pickering’s Continued Operation beyond 210,000 EFPH, Bruce Power VP, Mr. Frank Saunders notes that in PSA modeling: “*You need to look at that modeling and make sure it’s accurate; otherwise, whatever output you get won’t be very meaningful*.”

Question: Could Mr. Saunders please explain how to make a model “accurate” when it’s based on a postulated, rather than an actual event?

● OPG Initiative Action Plan RP-05 Issued in 2009 stated:

In recent years, increased work activities at the reactor face associated with feeder and fuel channel work in all units have contributed to a steadily increasing dose trend and challenged the station's ability to meet industry standards.

Question: Is this not a compelling reason to close Pickering NGS immediately?

● On April 20, 2012 the Ontario Energy Board issued its report: *Incentive Regulation Options for Ontario Power Generation's Prescribed Generation Assets*. In the Executive Summary of this report we read:

With respect to nuclear operations, the benchmarking analyses that have been performed

indicate that OPG's nuclear units have performed poorly, and dramatically so with respect to the Pickering units.

Indeed, the performance of OPG's nuclear facilities against worldwide nuclear industry benchmarks shows that the Pickering A and B plants have among the worst, *and on some measures the worst*, operating records among the plants in the WANO and EUCG data bases. Furthermore, on certain of the 19 indices in the Scott Madden report, including some key indicators, the Pickering units are not only the worst performers in North America, they achieve this distinction by a wide margin. For example, Pickering A1, B7 and A4 all have forced loss rates (FLR) over 30%; no other North American PWR or PHWR unit has an FLR over 20%. The Scott Madden report quantifies the causes for the gaps between the OPG plant performance and the best quartile among the CANDU benchmark group. It attributes problems to one of three categories:

● Equipment Reliability: Failure of component or equipment that directly forced or extended an outage (includes material condition problems)

● Design Basis: Equipment operated as per design but an inadequate design margin directly forced or extended an outage

● Human Performance (HP): Event caused by HP issues that directly forced or extended an outage

It is important to note that performance in two of these three categories (Equipment Reliability and Human Performance) can be influenced by operator decisions and actions. For Darlington, the gap on FLR is 0.25%. Scott Madden attributes most (83%)

of that gap to equipment reliability, 11% to material condition and 6% to human performance. The gap for Pickering A to the best quartile in the CANDU group was 37.2%. Scott Madden attributes 42% to equipment reliability, 51% to design basis, and 7% to human performance. The Pickering B gap was 17.5%, with, 75% attributed to equipment reliability, 5% to design basis, and 20% due to human performance.

OPG acknowledges that the performance of the Pickering plants has been poor. In its submission to the OEB for EB-2010-2008, OPG asserted that there are essential characteristics of CANDU nuclear plants that make them more expensive. They list complexity, generation technology (the fact that OPG owns the first large-scale CANDU reactors), *safety and regulation*, training, high standards for materials, and the radioactive work environment as the drivers of high costs. However, many of these drivers apply to all nuclear generation plants, so they do not explain the large gaps between OPG performance and the top quartile in the Scott Madden benchmarking study. *OPG recognizes that the poor material condition of these plants contributes to their poorer performance and higher operating costs*.

Question: Are these not compelling reasons to close Pickering NGS immediately?

Summary and Conclusions

I watched the March 27th 2014 Public Hearing on Pickering's Continued Operation beyond 210,000 EFPH on the CNSC website with great expectations, hoping that I would be witnessing a well-informed and open discussion of the many issues surrounding OPG's plans to operate Pickering's reactors for at least another 6 years. Unfortunately, I was greatly disappointed by what I saw and heard. First of all, I discovered that OPG were only telling half the story – the positive half – while completely ignoring the many contentious or negative issues that come with operating an old, and nearly worn out, nuclear plant. The neglected issues include the following:

- Very high deuterium pickup in the rolled joints of pressure tubes
- Extremely variable deuterium pickup in the body of pressure tubes
- The embrittlement of garter springs
- The uncontrolled movement of garter springs
- The cracking of a calandria tube
- The deliberate release of $^{14}CO_2$ by moderator cover gas purging

● Gadolinium oxalate precipitation during GSS

● The unexpected reduction in pressure tube to calandria tube gaps discovered during a Unit 5 outage

● The less than 100% coverage of feeder pipe and pressure tube inspections

● The increasing need to replace excessively thinned feeder pipes

● The increasing levels of alpha contamination caused by fuel channel inspections

● Increasing maintenance activities at the reactor face associated with feeder and fuel channel work in all

Units leading to increasing doses to personnel

● PSAs that fail to consider the detrimental effects of system, structural and component aging

● PSAs that fail to quantify uncertainties in calculated CDFs

● The massive amounts of tritium leaking into the foundation drains at Pickering A & B

That OPG failed to so much as mention any of these issues is outrageous and inexcusable, but the fact that the CNSC Staff also failed to address any of these concerns at the March 27th Public Hearing is totally disingenuous on their part, and suggests the CNSC is quite happy to keep the public and its Commissioners in the dark with regard to Pickering's true state of dilapidation.

So it was hardly surprising to see Commissioners Velshi, Harvey, Tolgyesi and McEwan struggle to ask even the most simple and generic questions of OPG and the CNSC staff on the topic of the fitness-for-service of Pickering's fuel channels; thus these Commissioners totally missed the important issues noted above. And President Binder himself joined in the charade when he stated towards the end of the proceedings:

"*This is pressure tube 101, so I'm allowed to ask dumb questions*."

Could any statement be more undignified and insulting to Canadians or show more contempt for the seriousness of the issues under debate? The total lack of due diligence displayed at the March 27th Public Hearing by our Nuclear Regulator is truly disgraceful. But perhaps this simply shows who the CNSC really serves. Certainly, if the CNSC desired to serve the best interests of Canadians it would *have to* reject OPG's license extension request on the grounds of too great a risk to the public and too much detriment to the environment. However, having seen the CNSC in action, I predict that the CNSC will have no problem renewing OPG's license for the Pickering Nuclear Generating Station, including permission to operate beyond 210,000 EFPH. When this happens, I hope the CNSC will, as is its usual practice, provide its "reasons for decision". And if the CNSC was truly honest about its reasons, it would admit: *The CNSC works primarily on behalf of the Canadian nuclear industry and strives to protect its business interests and gloss over its problems and mistakes*.

Yes, power tends to corrupt; but nuclear power corrupts absolutely

Dr. F. R. Greening

To:

Ms.Louise Levert

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Subject: Additional Material for my Submission Re: Pickering B Hold Point

Dear Ms. Levert,

Please accept the text below as additional material ***to be added to my previous submissions*** to the Commission as a written intervention at the May 7, 2014, public hearing.

Sincerely,

F. R. Greening

Additional Submission:

Dear Commissioners,

From reading the transcripts of the Public Hearings on OPG's application to the CNSC to extend its operation of Pickering Units 1, 4, 5-8 beyond 210,000 EFPH, it is clear that many complex issues need to be considered in order to assess the potential risks involved. However, this complexity is substantially reduced if the likelihood and consequences of individual accidents that result in damage to a reactor core are quantified – a process known as a Probabilistic Risk Assessment or PRA.

Appendix A of the CNSC Report RD-301 - "*Safety Analysis for Nuclear Power Plants*" – provides a list of component failures that could result in significant radionuclide releases to the environment, the most important being:

- A stagnation feeder pipe break
- A pressure tube failure
- An end-fitting failure

In the OPG Report NK30-REP-03611-00021-R000 entitled: "*Pickering B Risk assessment Summary Report*" accidents involving these and other component failures are analyzed and grouped into several release categories (RCs) depending on the magnitude of their anticipated radionuclide releases. For example:

RC1: Greater than 3% core inventory of I-131/Cs-137

RC2: Greater than 10^{14} Bq of Cs-137 but less than RC1 occurring mainly within 24 hours

RC3: Greater than 10^{14} Bq of Cs-137 but less than RC1 occurring mainly after 24 hours

RC4: Greater than 10^{15} Bq of I-131 but less than RC2 occurring mainly within 24 hours

RC5: Greater than 10^{15} Bq of I-131 but less than RC3 occurring mainly after 24 hours

In the PRA process these RCs are combined with component failures such as those noted above to arrive at a set of "*core damage frequencies*" and "*large release frequencies*". The OPG Report NK30-REP-03611-00021-R000 provides data on some of these frequencies for Pickering B, but with very little explanation as to how the reported values were calculated. Thus, for example, the predicted frequency for an RC1 event is stated to be in the range 2 to 4 ($\times 10^{-6}$) per reactor year, but the causative accident scenario is not identified but simply described as an "*internal at-power event*". Nevertheless, it is certain that the most damaging internal at-power events stem from the failure of a feeder pipe, a pressure tube or an end fitting; therefore an analysis of the consequences of these failures is essential to any meaningful PRA of a CANDU reactor.

Feeder Pipe Breaks:

After 30 years of operation of Pickering B Units, feeder pipe thinning from flow accelerated corrosion has reached the point where over 20 feeders are now approaching end-of-life safety limits. (See for example, OPG Report "*Pickering GS B Outlet Feeders Required Wall Thickness Values*, Document No. NK30-CALC-33126-00021 R000, and *Component Disposition Form* NK30-EVAL-33160-00003-R00). Studies in the open literature show that a complete rupture of an inlet feeder will reverse the flow in the affected channel, while a partial feeder pipe break will cause only a small flow reduction. However, in both of these cases there will be sufficient flow to provide adequate cooling to the fuel. Nevertheless, for break sizes *in-between these limits*, so-called flow stagnation will occur causing fuel bundles, and the pressure tube connected to the failed feeder pipe, to rapidly overheat. A 1989 study - "*Consequences of a Single Feeder Break Accident in a CANDU Power Reactor*", J. King Saud Univ., Vol. 1 - noted that feeder break sizes in the range $11 cm^2$ to $13 cm^2$ lead to flow stagnation and in the event of impaired containment the environmental releases were estimated to be 130 TBq of I-131 and 300 TBq of Xe-133 or approximately 2% of the average single channel inventory. The resulting site boundary dose would be about 150 mSv (whole body) and 1800 mSv (thyroid). **Comment:** It would be helpful to know if OPG agrees with this assessment of the consequences of a feeder pipe break at Pickering.

Pressure Tube Failures:

A design basis accident potentially resulting in major fuel damage and the release of fission products is a pressure tube rupture. In 1975 cracks were observed just inboard of the rolled joints in some Zr-2.5%Nb pressure tubes in Pickering NGS and it was subsequently recognized that deuterium picked up at a rolled joint will diffuse inboard into the body of the pressure tube. Once the terminal solute solubility (TSS) is exceeded, hydride precipitation occurs in high stress regions followed by delayed hydride cracking. For Pickering Units known to have sufficiently large/sharp flaws, thermal cycle restrictions must be imposed. As early as 1994, it was predicted that by

2014 more than 50 % of Pickering A pressure tubes would have a total hydrogen concentration exceeding TSS in the high stress region (80 mm from the tube end), and thus be susceptible to DHC. The amount of fuel damage caused by a fast fracture of a pressure tube is similar to the fuel damage induced by a feeder pipe break, but the associated radiological releases to the environment depend on the degree of containment impairment assumed in the risk assessment analysis.

End Fitting Failures:

An end fitting failure accident is defined as a failure of a fuel channel that occurs somewhere between the feeder pipe and the rolled joint (i.e., the connection between the pressure tube (PT) and the end fitting body). The demonstrated worst case scenario for an end fitting failure is the high speed ejection and impact of a maximum power bundle that breaks up and exposes a large amount of UO_2 surface area to a steam environment. Typically up to 5 % of the available I-131/Cs-137 may be released within a few minutes of an end fitting failure due to rapid overheating of the affected fuel. However, as with the failures noted above, the associated radiological releases to the environment depend on the degree of containment impairment assumed in the risk assessment analysis.

The Source Term Issue:

The first quantity that needs to be determined in these accident scenarios – whether we are considering a feeder pipe, pressure tube or end fitting failure – is the so-called *source term*. This may be defined as the amount of each radioactive species released by a postulated accident. Apart from a few activation products such as tritium and C-14, the most important species to consider are the longer-lived volatile fission products Xe-133, I-131, Cs-137 and Ru-106. Once a realistic source term for an accident scenario is determined it is then a relatively simple matter to calculate the associated dose-to-public. For example, the dose from radionuclide inhalation is given by:

Inhalation Dose = *Source Term* (Bq) × *Dose Conversion Factor* × *Breathing Rate* (m^3/s) × *(X/Q)* (s/m^3)

Here ***Q*** (in Bq/s) is the release rate of a specific radionuclide, and ***X*** is the radionuclide concentration (in Bq/m^3). The **(X/Q)** term, also referred to as the atmospheric dilution factor, drops off rapidly with distance from the emission source. At a station's regulatory "site boundary", generally taken to be about 1 km from the emission source, **(X/Q)** is about 10^{-5} s/m^3.

The dose conversion factor, breathing rate and atmospheric dilution factor are essentially constant parameters for an exposed individual located at a known distance from a NPP, thus the *source term* is the only true variable in a dose-to-public calculation. This is why knowledge of the source term for each specific accident scenario is essential to a safety assessment of an NPP. And this is also why it is reasonable for independent scientists, engineers and concerned members of the public to be given access to accident source term data if so requested.

Most regrettably, however, in December 2011 the Ontario Information and Privacy Commissioner ruled in favor of arguments presented by OPG and the CNSC and rejected a FOIA request from an intervener with regard to access to CANDU accident source term data. The Commissioner, Dr. A. Cavoukian, summarized her reasons for this decision as follows:

"I note that, as referenced in Order PO-2960-I, the confidentiality of a probabilistic risk assessment, as a totality, was upheld by the CNSC in April 2008. On this point, I commented as follows in deciding to uphold OPG's claim that the records are exempt under section 16 of the Act:

"*I find it to be both relevant and persuasive that the CNSC, a body charged with protecting the public interest in the licensing of nuclear power facilities, has previously refused the appellant's request for access to a record containing the source term data at issue in this appeal for the Pickering B facility. As already noted, this occurred during a CNSC licence renewal hearing, during which the appellant was an intervener. The CNSC denied access on the grounds that disclosure "may be prejudicial to the security interests of Canadians*."

Thus we see that the December 2011 decision by the Ontario Privacy Commissioner to deny an intervener access to source term data was based on a prior April 2008 decision by the CNSC's to deny the same intervener access to the same source term data.

Nevertheless, in May 2013 CNSC President Binder stated in a letter to me (File No: ccm: 22013-000235):

> "*The CNSC is committed to transparency. However, an examination by the Ontario Privacy Commissioner concluded that source term information, should not be put in the public domain*"

With this comment President Binder deceptively lays the blame for the CNSC's "lack of transparency" on the Privacy Commissioner, when it is clear that the Privacy Commissioner took her lead from an earlier CNSC ruling. Unfortunately, for the public at large, all that these rulings achieve is to stifle debate on the safety of OPG's reactors by denying independent parties the opportunity to review information that is vital to an estimation of the dose-to-public resulting from accidents or malevolent acts at OPG's plants. Henceforth, nuclear safety assessments in Canada will be based exclusively on the unassailable authority of those on the payroll of the nuclear industry without any meaningful input from concerned scientists and environmentalists seeking answers to basic questions such as the precision and accuracy of dose-to-public calculations.

The only argument offered by OPG and the CNSC for withholding data on CANDU accident source terms is that such information would enable terrorist to attack these reactors, and the buildings in which they are housed, with greater rapidity and with larger impact than without this information. This rationale for withholding CANDU accident source term data from public scrutiny is untenable for the following reasons:

● OPG and the CNSC claim that existing and proposed reactors in Ontario meet all the safety goals specified in RD-337 which includes an assessment of the radiological consequences of malevolent acts. Indeed, OPG explicitly state in Report NK054-REP-07730-00024 that any credible malevolent act "*would **not** cause a significant release of radioactivity to the public*" – an assessment endorsed by the CNSC in its Report No. PMD: 11-P1, issued January 2011. Furthermore, it is universally acknowledged that the most serious credible "terrorist" threat to a nuclear power plant would be an attack involving insider collusion. This is because well-trained insiders would know how to disable emergency cooling systems and backup power sources such as diesel generators. Nevertheless, while OPG and the CNSC assert that Pickering and Darlington are secure from <u>all</u> credible malevolent acts they also claim to the contrary that in the hands of the public "s*ource term data could be used to both plan attacks to maximize impact* (and cause) *catastrophic harm*" – a statement that is completely at odds with the conclusions presented in OPG's Report NK054-REP-07730-00024.

● The CNSC claim that "*source term data would <u>not</u> assist interveners in addressing whether Pickering B could operate safely with due regard for the protection of the environment*". However, this claim comes from the same organization that, through its Regulatory Guide RD/GD-369, requires mandatory safety studies of nuclear power plants that should include "*a source term analysis and assessment of off-site consequences of credible accidents*". It would therefore appear that the CNSC considers accident source term analysis to be a requirement of a would-be operator, while at the same time believing that such analyses have no bearing on an assessment of the environmental impact of a CANDU NPP – surely a fine example of "**Doublespeak**" [See Ref 1], and a viewpoint that would be contested by people living close to these plants.

● A great deal of accident source term information, for a wide variety of reactor designs (including CANDUs), is currently available in the public domain. Published reports that provide source term data often include lists of critical plant component and the doses to the public that would result from their failure. However, this is precisely the information that OPG and the CNSC fear would result in "*catastrophic harm if it fell into the hands of trained terrorists*". Nevertheless, I would humbly suggest that it is highly improbable that "trained terrorists" are being thwarted in achieving their sinister ambitions simply because they do not have access to accident source term data!

However, what makes the CNSC's position all the more untenable is that training courses and workshops are available on a regular basis that provide instruction on how to carry out source term calculations for accidents at nuclear power plants – <u>*including courses specific to CANDU reactors*</u>. See, for example:

1. "*Workshop on Severe Accident Analysis for Nuclear Power Plants*" IAEA Knowledge and Experience Sharing Workshop held in Dubrovnik, October 2010

2. “*Source Term Assessment Training*” Course currently offered by International Safety Research Inc.

3. “*Workshop on Regulatory Safety Review of the Source Term Assessment*” IAEA Workshop held in Jakarta, March 2012

● By their very nature, source terms for postulated accidents are always speculative in nature and subject to a high degree of uncertainty. For this reason, rating the consequences of one postulated accident scenario as being significantly worse than another, is generally not statistically valid and certainly of no use to would-be terrorists because, as we have seen, radioactive releases caused by terrorist attacks on nuclear reactors have been rated by OPG, (and the US NRC), as being no worse than releases from aleatory events whose probabilities are well-documented in the open literature.

In conclusion it is evident that OPG and the CNSC have no legitimate reasons to deny concerned scientists and environmentalists access to CANDU accident source term data. Furthermore, the position taken by OPG with regard to accident source term data - a position fully supported by the CNSC - only serves to stifle debate on nuclear safety and expedite the Canadian nuclear industry’s desperate ambition to run Pickering NGS "into the ground". Regrettably, OPG and the CNSC appear happy to pursue their self-serving goals not by logical and reasoned arguments - because they have none - but through fear mongering tactics involving false alarms to the Ontario Information and Privacy Commissioner. This is reprehensible behavior by so-called “public servants” that is nothing short of capricious abuse of power.

References:

1. "***Doublespeak**" is language which pretends to communicate but doesn't. It is language which makes the bad seem good, the negative seem positive, ... It is language which avoids, shifts or denies responsibility; language which is at variance with its real or purported meaning. It is language which conceals the truth*".

F. R. Greening