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Email from Frank Greening

Courriel de Frank Greening

In the Matter of

À l'égard du

Regulatory Oversight Report for Canadian Nuclear Power Generating Sites: 2017

Rapport de surveillance réglementaire des sites de centrales nucléaires au Canada : 2017

Commission Meeting

Réunion de la Commission

November 8, 2018

Le 8 novembre 2018

Canada

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From:	Frank Greening [redacted]
Sent:	Tuesday, October 30, 2018 9:52 AM
To:	Velshi, Rumina (CNSC/CCSN)
Cc:	jeffrey.lyash@OPG.com; Rinker, Michael (CNSC/CCSN); d.reiner@opg.com; brian.duncan@opg.com; Amarjeet.Sohi@parl.gc.ca; Interventions (CNSC/CCSN); Levert, Louise (CNSC/CCSN); Media relations / Relations avec les médias (CNSC/CCSN)
Subject:	The Darlington Alpha Event
Attachments:	NWMOLET.docx; LUNGSOL.docx

Dear President Velshi,

I am writing to you at this time in the hope of resolving a number of on-going issues pertaining to the February 2018 alpha contamination incident at Darlington NGS during refurbishment operations on Unit 2. This incident has already been addressed by OPG and the CNSC – e.g. at a CNSC Public Meeting held in August of this year – and is slated for further discussion on Nov 7-8 2018. Nevertheless, as someone who was directly involved in a similar alpha contamination event at Bruce NGS in December 2009, I would like to offer my thoughts on the February 2018 Darlington event. OPG has provided some details of how alpha contamination of two workers occurred in the Darlington RWPB - see for example: OPG's April 23, 2018, letter to the CNSC containing the *Detailed Event Report for D-2018-04257*. However, I believe it is important to review all of the details of the alpha contamination event to fully understand exactly what happened at Darlington on February 5th 2018:

A. Refurbishment Operations that Led to the Contamination Event

Two workers were assigned to "lidding" operations in the Retube Waste Processing Building (RWPB) which involved placing lids on the so-called Darlington Storage Overpacks (DSOs). These large box-like containers are designed to hold refurbishment waste metal components. At the time of the contamination event, two components were being processed:

(i) End Fitting stubs from the inboard or "hot" end of the reactor. These tube sections were disposed of without additional volume reduction and were therefore essentially free of cutting debris.

(ii) Pressure tubes (PT) and calandria tubes (CT) which are first processed in the Darlington Volume Reduction System (VRS) located in the RWPB. The VRS is a mechanical press that crushes and cuts PTs and CTs into two-inch coupons which then fall through a chute into a DSO located below the press. This process creates a considerable amount of debris in the form of cutting "fines" and therefore has the potential of creating loose airborne radioactive contamination.

B. Discovery of Worker Contamination in the RWPB

When leaving the rubber area associated with the RWPB lidding operations, the two workers were found to be contaminated. Nevertheless, decontamination procedures enabled the workers to clear the Whole-Body Monitor (WBM) and the workers exited the zoned area. However, the workers subsequently alarmed another WBM and were sent for a whole-body gamma scan which indicated both workers were internally contaminated.

C. Investigation of the Cause of the Contamination Event

Radiological control of the lidding operations that led to the contamination of two refurbishment workers was under the supervision of an OPG Radiation Protection Coordinator (RPC). This RPC has acknowledged that a pre-job survey found loose contamination in a groove on the lid of the pressure tube waste container – contamination that was subsequently removed. As a result, and prior to the workers undertaking the lidding operations, the work area in question was incorrectly declared to be free of loose contamination.

Discussion:

OPG's Darlington refurbishment records show that in the period November to late December 2017, only end fittings were being processed in the RWPB and smears collected in the RWPB during this period indicated that an Alpha Level I designation was appropriate for work at this location. However, the February 2018 exposure event showed that for the processing of pressure tubes in the RWPB, an Alpha Level III designation was more appropriate. Nevertheless, as discussed below, the Alpha Level I designation was not changed when lidding of pressure tube waste containers commenced.

These alpha contamination designations assume that an internal dose from an alpha emitter present in loose contamination will either *not exceed 10 %* of the beta-gamma inhalation dose for Alpha Level I, or *is likely to exceed 90 %* of the beta-gamma inhalation dose for Alpha Level III. If we consider Pu-239 as representative of a typical alpha emitter, and Sr-90 as representative of a typical beta emitter, an Alpha Level III implies that workplace contamination will exhibit a beta/alpha activity ratio less than about 30. Significantly, OPG's Station Condition Record (SCR) for the Darlington alpha event, (Document N-2018-03429), notes that a smear sample collected in the RWPB after the contamination incident measured a beta/alpha ratio of 4.

The Darlington SCR N-2018-03429 also provides a rationale for the Radiation Protection Coordinator's decision *not* to change the Alpha Level I designation for the RWPB prior to the processing of pressure tube waste. Thus, the following statement is found on page 3 of the SCR:

> Up to December 2017, *only End Fitting (EF) inboard ends were placed in the Retube Waste Containers. The severed EF stubs were never volume reduced so the Volume Reduction Press (VRP) had not been used. The VRP is designed to be used on the pressure tubes (PTs) and calandria tubes (CTs). It is a mechanical press that breaks the PTs and CTs into two-inch square coupons… This process has the potential to create a large amount of loose contamination. The waste container that was being lidded by the two workers that received the internal uptakes was the first pressure tube waste container to be lidded in the RWPB. The RPC involved in this event failed to realize that as the work changed from EF removal to PT removal, the magnitude and nature of the hazards associated with the waste containers also changed.*

And later, on page 4 of the Darlington alpha event SCR, we read the following:

> *The process for lidding the waste containers for pressure tubes (PTs) is the same as was used on the end fittings (EFs). Since the EFs did not show high levels of contamination, … it was believed that the contamination levels would remain low throughout the work. It was not anticipated that high levels of contamination existed elsewhere in the area and that the motion of the lids would result in higher levels of contamination being introduced to the lidding area.*

It is useful to consider these statements from OPG's Darlington alpha event SCR in some detail because other statements made by OPG management pertaining to these issues contradict what was said in the alpha event SCR. Thus, as noted above, the SCR states that:

(i) *The waste container that was being lidded by the two workers that received the internal uptakes was the first pressure tube waste container to be lidded in the RWPB*

(ii) *It was not anticipated that high levels of contamination existed elsewhere in the area and that the motion of the lids would result in higher levels of contamination being introduced to the lidding area*

However, these two statements are difficult to reconcile with the following:

(i) At the CNSC August 2018 Public Hearing: From Mr. Ian Edwards, Darlington Health Physicist:

On February 5th we were processing a second (pressure tube) waste container, which is where this incident occurred, and all the information we had to date supported an Alpha Level 1 classification

(ii) From an OPG *Risk Report* issued March 17th 2017:

Spread of Contamination during RFR Waste Processing of crushed PT

Execution Phase Risk: There is a risk that there will be the spread of loose contamination in the RFR WTS and RWPB, with negative impacts on the schedule and worker dose due to the need to cleanup. CAUSE: This risk is postulated to occur by loose contamination being spread from the PT chute that links the VRS press to the RWC debris cover. There is an 'O-ring' like barrier that seals the connection between the chute and RWC debris cover during PT crushing, however, when the chute is retracted, it is speculated that some of the fine particles in the chute would become loose and spread beyond this ring. By design, it is expected that the 'dirty' pins on the WTS lidding station will become contaminated over time as they lift the debris cover; there are separate 'clean' pins for lidding the RWC lids. However, it is speculated that the partly contaminated debris cover will be lifted over clean RWC lids and transfer some of this loose contamination to the clean lids, therefore requiring cleanup. IMPACT: Based on OPEX with a similar WTS VRS press used at Lepreau Refurbishment - when PTs were crushed, the loose contamination led to high dose rates in the vault that required significant cleanup.

I intend to return to these contradictions later in this email, but first we need to consider what level and type of contamination was to be expected on end fittings and pressure tubes removed from Darlington's Unit 2 during the refurbishment of this reactor.

CANDU End Fitting and Pressure Tube Comparisons

In all CANDU reactors, a stainless-steel end fitting is attached at the two reactor faces, (east and west), to each end of a zirconium alloy pressure tube which in combination facilitate the transport of hot pressurized D_2O through the reactor core. Table 1, below, provides a comparison of some of the key properties of these reactor components.

Table 1: Some Properties of CANDU End Fittings and Pressure Tubes

Reactor Component	Material	Mass (kg)	Average Neutron Flux ($n/cm^2/s$)
Pressure Tubes	Zr-2.5%Nb	61	1×10^{14}
End Fittings	403 Stainless Steel	163	1×10^{10}

From the point of view of radio-activation, the pressure tubes are situated inside the reactor core in a very high neutron flux and therefore become highly radioactive over time. By comparison, end fittings are located outside of the reactor core where the average neutron flux is about 1000 times lower than in core. This means that neutron activation of end fittings is negligible compared to the neutron activation of pressure tubes. Nevertheless, end fittings may exhibit residual activity due to the deposition of corrosion and fission product (crud) that is suspended in the circulating D_2O heat transport water.

Using available data for Darlington Unit 2 heat transport water crud deposits, and generic data for irradiated pressure tubes, (assuming 1 mg of fines is produced by cutting a pressure tube into 2-inch square coupons), I have estimated the principal loose activities derived from these reactor waste components as shown in Table 2.

Table 2: Estimated Loose Activities: Pressure Tube Cutting Fines and End Fitting Smears

Reactor Component	Radionuclide Activities (Pressure Tube data in Bq/mg fines) (End Fitting data in (Bq/cm^2) × 5)							
	Co-60	Cs-137	Nb-94	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
Pressure Tubes	5500	1200	7000	2	3	4	50	800
End Fittings	1900	250	100	0.4	0.9	0.9	2.0	30

It should be stressed that the activities in Table 2 are approximate *estimates* that represent the surface activities associated with these components to within a factor of about two. Nevertheless, the data in Table 2 are sufficiently accurate to draw a number of important conclusions about Darlington Unit 2 refurbishment wastes that have been processed in the RWPB:

(i) The beta/alpha ratio for Darlington end fitting waste was approximately 68. This means the RWPB work area for processing this waste was correctly designated as an Alpha Level I area.

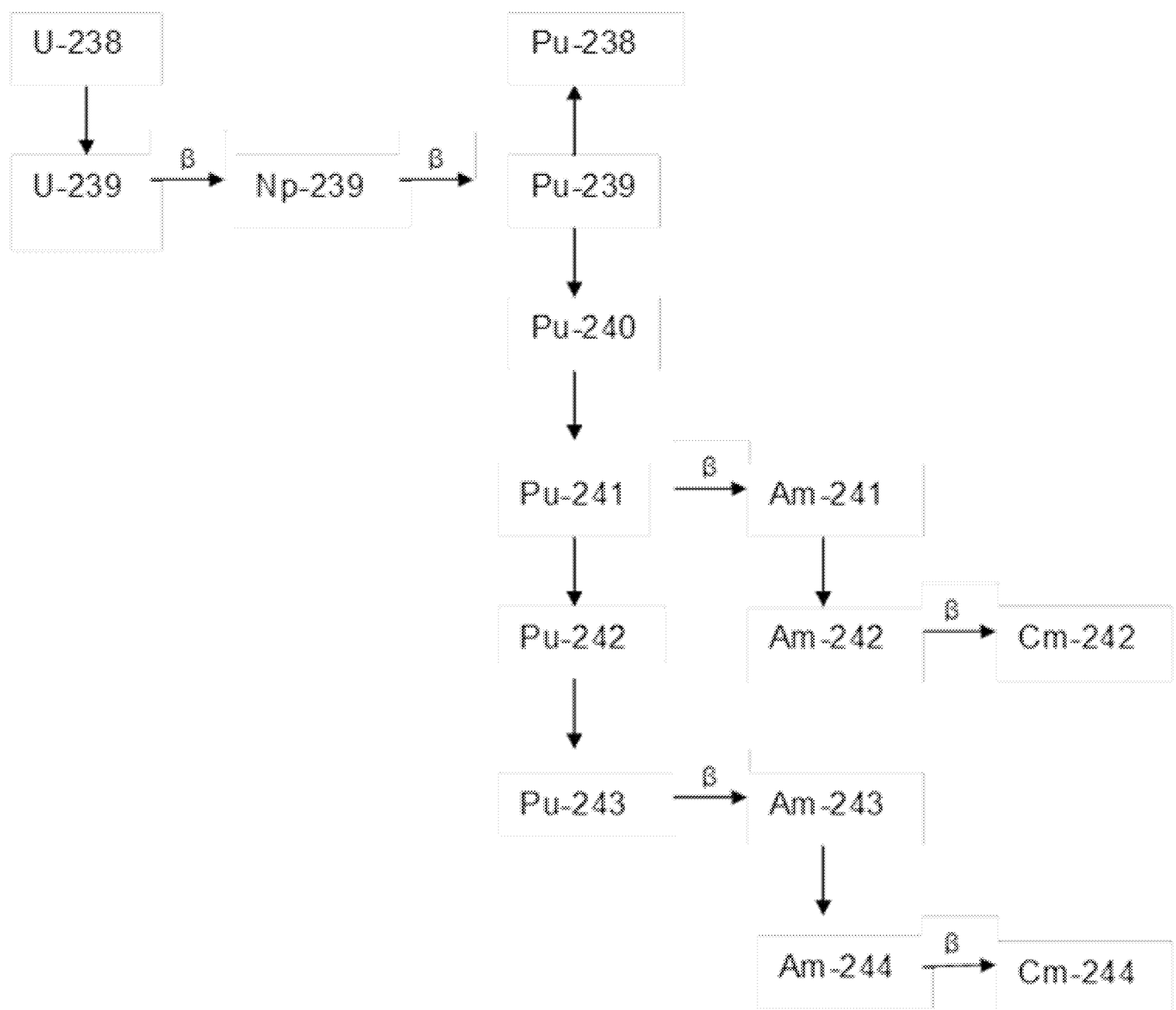
(ii) The beta/alpha ratio for Darlington pressure tube waste was approximately 16. This means the RWPB work area for processing this waste should have been designated as an Alpha Level III area, rather than an Alpha Level I area.

(iii) The alpha source term for Darlington refurbishment waste is dominated by Cm-244.

Why Cm-244?

To understand the predominance of Cm-244 in the Darlington refurbishment waste alpha source term we must first consider its production/depletion pathways in irradiated UO_2 fuel, as summarized in Figure 1 below.

Figure 1: Major Pathways for the Production and Depletion of Transuranic Radionuclides in Irradiated Uranium



The ORIGEN computer code is widely used to calculate the in-growth of Pu, Am and Cm in CANDU fuel for exposure times of interest. Typical results of such calculations are summarized below in Table 3.

Table 3: ORIGEN Predictions for CANDU Fuel

Isotopic Species	Specific Activity for Specified Exposure Time (GBq/kg initial U)

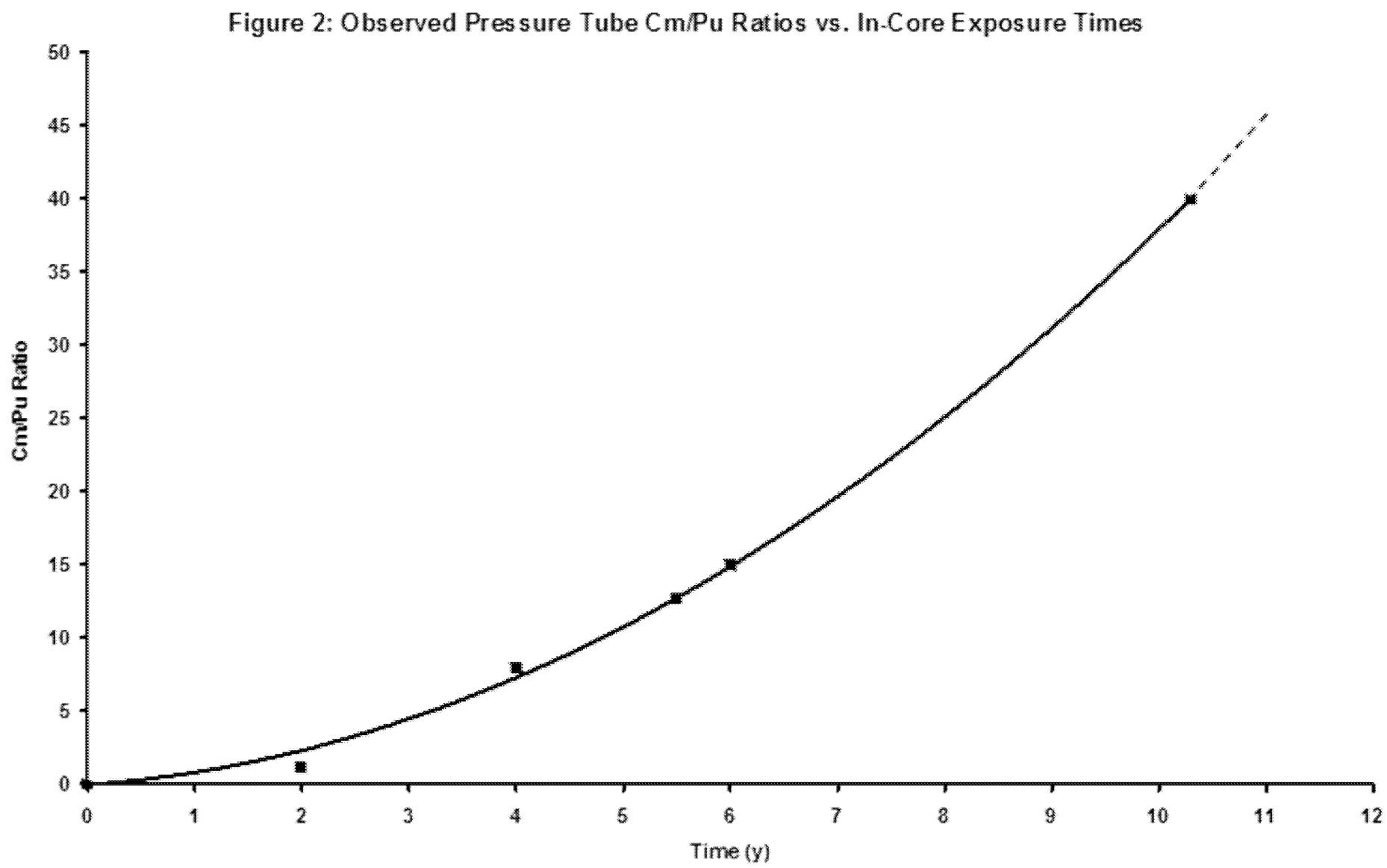
	40 days	150 days	200 days	240 days	350 days
Pu-238	0.081	1.44	2.63	4.28	9.02
Pu-239	2.74	5.39	5.80	6.02	6.11
Pu-240	0.89	5.90	8.18	10.33	14.18
Am-241	0.063	0.88	1.39	1.94	3.08
Cm-244	-	0.13	0.44	1.21	5.64
Pu239+Pu240	3.63	11.29	13.98	16.35	20.29
Pu238+Am241	0.14	2.32	4.02	6.22	12.10
Ratio Cm/Pu	-	0.01	0.03	0.07	0.28

Table 3 shows that there is a steady build up of Cm-244 relative to (Pu-239 + Pu-240) as the in-core exposure of the UO_2 fuel increases. ORIGEN calculations are not very accurate for longer irradiations than shown – about 1 effective full power year (EFPY) – but other evidence confirms that Cm-244 becomes the principal α-emitting species in CANDU fuel for in-core exposures in excess of about 3 EFPY.

Alpha-contamination up to 20,000 Bq/cm^2 has been detected on the inside surfaces of sections of pressure tube removed from mature CANDU Units. These measurements were made on samples that were cut from known locations on pressure tubes with in-core exposures between 1.5 and 15 EFPY. A common characteristic of these samples is the high level of Cm-244 in comparison to the associated (Pu-239 + Pu-240). Thus, Cm-244/(Pu-239 + Pu-240) ratios measured on removed pressure tubes are observed to be in excess of 30 for material with more than about 10 EFPY in-core reactor exposures.

It is also significant that the Cm/Pu ratios of a typical set of pressure tube samples exhibit a good correlation with the sample's in-core exposures as shown in Figure 2 below. This behaviour indicates that the uranium responsible for the alpha-contamination on these samples has been in place for a major portion, if not all, of the sample's in-core exposure. The activity of Pu-239 saturates within 3 EFPY, while the activity of Cm-244 continues to build up under longer exposures – a process that ultimately leads to very high Cm/Pu ratios.

If an initial surface uranium concentration of 10 µg/cm^2 is assumed to be present on an as-installed pressure tube, and it is exposed to an average in-core thermal fluence of 2.5 x 10^{21} neutrons per cm^2 per year, the activities of Pu-239, Pu-240 and Cm-244 extrapolated from the ORIGEN calculations reported in Table 3, are in reasonable agreement with the observed levels of these species on removed pressure tubes.



In the case of pressure tubes from Darlington Unit 2 we are dealing with in-core exposures ~ 30 years so that Cm-244/(Pu-239 + Pu-240) ratios much greater than 100 are to be expected. In 1991 I participated in a study of the production of actinides in CANDU pressure tubes, (See AECL Report RC-697), which measured Cm-244 concentrations > 3000 Bq/µg U, and Cm/Pu ratios > 100 in sections of pressure tube irradiated in Pickering NGS for periods up to 10 years. I should add that a recent paper presented at the 3rd CNS Canadian Nuclear Waste Management Conference reports similar data for Cm-244 in pressure tubes and shows that the Cm-244 specific activity in the samples under investigation saturates at about 10,000 Bq/µg U for exposures of about 18 EFPY.

Cm-244 Contamination During the 2009 Point Lepreau Refurbishment

The February 2018 alpha contamination incident during the operation of the pressure tube VRS at Darlington is not the first example of the spread of airborne alpha contamination from the handling of pressure tube waste during a CANDU reactor refurbishment. Thus, an almost identical event occurred at New Brunswick Power's Point Lepreau NGS in February 2009. Details of this event are described in a November 2010 CNSC Staff Submission CMD 10-M72 as follows:

> (During the refurbishment of the Point Lepreau reactor, because general contamination was present in the vault), *workers wore respirators on the fuel channel platform during the removal and crushing of pressure tubes. Then, on February 25th 2009, during the removal and crushing of the third pressure tube, the continuous air sampler in the west vault went into alarm due to a malfunction of the active ventilation system. As a result, respiratory protection became mandatory for all vault workers, (not just those on the platform). Whole-body counts for personnel who were working in the vault were conducted. The subsequent investigation confirmed alpha contamination on smear samples from the continuous air monitors. The alpha contamination identified was primarily due to Cm-244. New Brunswick Power immediately implemented an enhanced sampling and analysis program involving additional swipes and smears and conducted additional whole-body counts.*

The results of the radiochemical analysis of a VRS smear are shown in Table 4, below. It is also noteworthy that approximately 25 nanograms of U-238 was detected in this Point Lepreau smear.

Table 4: Activities of the Radionuclides Detected in a Point Lepreau Smear

Radionuclide	Co-60	Cs-137	Nb-94	Pu-238	Pu-239/240	Am-241	Cm-244
Activity/Smear (Bq)	22,000	540	6700	4.5	2.7	7.2	870

It is encouraging that the data in Table 4 show reasonable agreement with the data previously reported in Table 2 for the radionuclide activities estimated to be in loose material collected by smearing Darlington pressure tube waste. This agreement is not unexpected because Point Lepreau and Darlington had been operating for about 26 years at the time of their respective refurbishments. Thus, the buildup of the activities of Nb-94 and the transuranics Pu, Am and Cm on pressure tube surfaces should be about the same. – as observed.

It is very commendable that the Point Lepreau refurbishment workers were able to deal with the Cm-244 contamination released from the VRS *as soon as it was detected*; this is because a continuous air sampler was operating nearby at the time which alarmed as soon as the airborne alpha contamination reached a pre-set action level. By comparison, and 9-years after the Point Lepreau incident, OPG totally failed to learn from this OPEX because this time around it appears that, when it comes to OPG's RP staff and the refurbishment workers under their direction, (and protection!), the blind were leading the blind with disastrous consequences!

Does OPG have a Cm-244 "Blind Spot"?

As we have seen, the Darlington alpha exposure event led to the uptake of airborne Cm-244 by two refurbishment workers. But OPG's failure to anticipate this hazard makes one wonder: *Does OPG have a Cm-244 "Blind Spot"?* This may appear to be a very strange question, but the history of OPG's recognition of the presence of Cm-244 in its CANDU reactor systems – and in the radioactive waste generated by a refurbishment – shows why such a question needs to be asked. Thus, over the past two decades OPG has spent a great deal of time and money developing a low and intermediate-level radioactive waste inventory for its Deep Geological Repository, (DGR) – the proposed final resting place of OPG's past, present and future radioactive waste. This inventory includes the all-important reactor refurbishment wastes, and, in terms of the total radioactive inventory predicted to have accumulated by 2062, activities associated with removed pressure tubes, are the largest single contributor, (by reactor component type).

OPG published its first report on the DGR Inventory in August 2008 – See OPG Report 00216-REP-03902-00003-R01 – and followed up with a revised version in December 2010 – See OPG Report 00216-REP-03902-00003-R003. The August 2008 report was notable in that it listed *no* transuranic species whatsoever in the estimated 145 m^3 of pressure tube waste predicted to have been produced by 2018, even though transuranic activities, (including minor species such as Pu-242 and Am-243), were quoted for OPG's steam generator waste – See Table 3.2 in the August 2008 Inventory Report.

Nevertheless, transuranic inventories in pressure tube waste became even more anomalous in OPG's December 2010 DGR Repository Inventory Report, because quantitative data for pressure tube waste were now presented for the

transuranic alpha-emitters Am-241, Am-242m, Cm-243, Pu-238, Pu-239, Pu-240 and Pu-242 – *but Cm-244 was still nowhere to be seen* – see Table 3.2 in the December 2010 Inventory Report. As noted above, Cm-244 is predicted to be the principal alpha-emitting species in CANDU fuel for in-core exposures in excess of 3 EFPY, and is expected to exceed Pu-239 and Pu-240 activities by orders of magnitude after 10 EFPY.

In the 23 years I worked for Ontario Hydro/OPG I spent about half of my career analyzing pressure tubes removed from CANDU reactors, and much of this work was carried out in conjunction with researchers at AECL Whiteshell Laboratory. Thus, by 2013, I had a very extensive set of data for the radionuclide content of pressure tubes, based on carefully conducted *measurements*, which I was able to compare with the data reported by OPG in its Inventory Reports. What I discovered was that there were *many* discrepancies between my data and OPG's data, (in addition to OPG's total omission of data for Cm-244), and I subsequently discovered that OPG's pressure tube radionuclide activities were based on calculations using a so-called "used fuel ratio" methodology, *not on measurements*. Furthermore, and most regrettably, OPG's calculations were clearly prone to large errors which invariably led to the *underestimation* of the reported activities.

As a result of these findings, in January 2014, I wrote to Mr. Ken Nash, President of the Nuclear Waste Management Organization, (with c.c.'s sent to Dr. S. Swanson, Panel Chair of the Deep Geological Repository Panel and Dr. M. Binder, President of the CNSC), to report my concerns about the reliability of OPG's radionuclide inventory of pressure tube waste. (Please see the attachment NWMOLET to this email for a copy of this letter). The letter contained, among much additional supporting material, the following Table and comments:

A Comparison of OPG/NWMO Data with SIMS and Alpha Spectrometric Measurements on Bruce Pressure Tubes

Radionuclide	Estimated Values	Measured Values	
	OPG/NWMO Activity (Bq/kg ZrNb)	Averaged Data from Table 5 (Bq/kg ZrNb)	SIMS Data for B3U11 (Bq/kg ZrNb)
U-235	0.363	-	0.701
U-238	29.4	214	29.4
Pu-238	7910	205,000	-
Pu-239	14,200	62,000	104,000
Pu-240	19,400	85,000	133,000
Am-241	23,400	318,000	146,000
Cm-244	Not Reported	50,000,000	14,900,000

A number of important conclusions may be drawn from the data presented in the above Table:

- The *measured* activities of alpha-emitting transuranic radionuclides in irradiated pressure tubes are considerably higher than the OPG/NWMO "used fuel ratio" estimates

- Cm-244, which is *not* included in the OPG/NWMO's estimates, is by far the highest alpha activity measured in irradiated pressure tube samples

- There is a high degree of variability, (by up to a factor of 3), between measurements from different pressure tube samples

In May 2014, in response to my January 2014 letter, OPG issued a correction to its December 2010 DGR Repository Inventory and acknowledged for the first time that Cm-244 was indeed the largest contributor to the alpha source term for pressure tube waste – See Attachment to OPG Letter, Ms. Laurie Swami to Dr. Stella Swanson, Deep Geological Repository Project: *OPG Responses to Information Requests in IR Package #13 from Joint Review Panel*, CD# 00216-CORR-00531-00235, issued May 9th, 2014.

In addition, as a result of the need for increases to many other radionuclide inventories in pressure tube waste, the dose rates to workers handling pressure tube waste containers in a RWPB were estimated to have increased by a factor of about four: namely, from 5.7×10^{-3} mSv/h to 2.2×10^{-2} mSv/h, necessitating the use of additional shielding and/or a reduction in the allowable occupancy for work in the pressure tube waste packaging and loading work areas.

So, as it now stands, OPG estimates that the inventory of alpha-emitting radionuclides in pressure tube refurbishment waste is as shown in Table 5:

Table 5: OPG's Current Estimate of Alpha-Emitting Radionuclides in Pressure Tube Waste

Alpha-Emitting Radionuclide	Pressure Tube Inventory (Bq)
U-235	3.1×10^{5}
U-238	7.8×10^{7}
Pu-238	3.1×10^{10}
Pu-239	1.0×10^{10}
Pu-240	1.7×10^{10}
Am-241	6.9×10^{10}
Cm-244	1.9×10^{12}

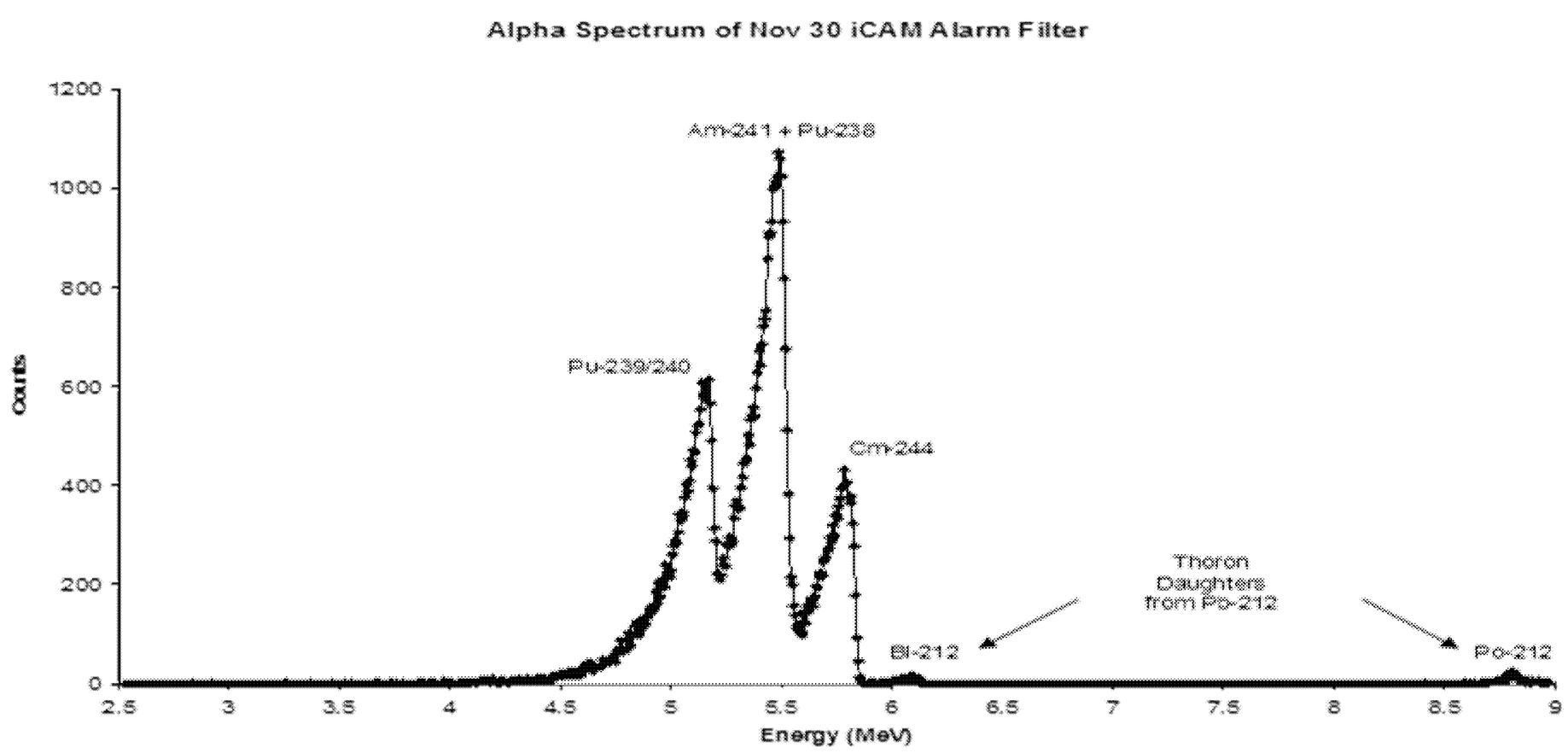
These radionuclide activities predict a Cm-244 / (Pu-239 + Pu-240) ratio of 70 which is typical of high burnup CANDU fuel. Unfortunately, however, OPG's *Responses to Information Requests* document does not elaborate on how the revised inventory was determined – thus we are left wondering: what pressure tubes were analyzed, what analytical procedures were used, what are the uncertainties in the data, etc. Of interest in this regard is OPG's statement, *made without proof*, about the nature of the alpha activity in pressure tube refurbishment waste:

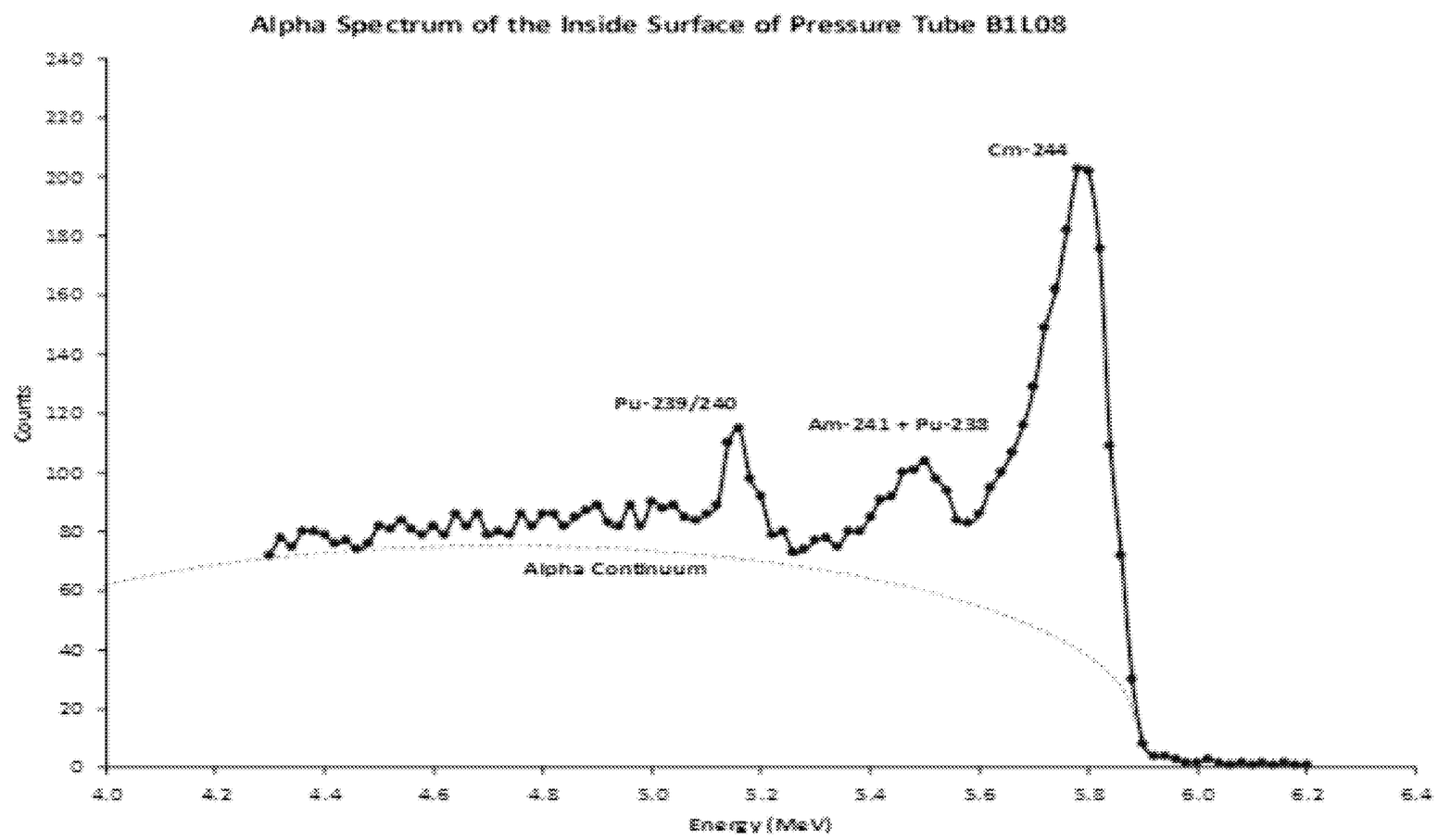
The revised radionuclide inventories are primarily due to surface deposit from coolant on the pressure tubes. Therefore, they are assumed to be located in the surface layer of the pressure tubes, and assumed to be quickly released on contact with water.

This statement is highly suspect because we know that pressure tubes operate in a hot aqueous environment. If transuranic species are "*quickly released on contact with water*" one must question how such deposits could be present on the surface of removed pressure tubes. And, as discussed later in this email, the issue of the solubility of a transuranic radionuclide is of great importance to the dosimetry of these species in the event of an internal uptake.

As reported in a paper I co-authored for the *Waste Management 94' Conference* in Tucson (AZ), direct alpha spectrometry of the inside surface, (coolant side), of pressure tube material from Pickering NGS shows Cm-244 with activities ~ 7×10^3 Bq/cm^2. Significantly, however, the activity observed when a duplicate sample was dissolved, and the Cm-244 extracted and counted, was 5 times *higher* than the activity observed on the inside surface of the sample. This shows that the Cm-244 activity associated with irradiated pressure tubes is present not only in surface deposits, but in the base metal (Zr-2.5%Nb) matrix as well – a conclusion that is contrary to OPG's position on this question.

As further evidence of the unusual distribution of Cm-244 in irradiated pressure tubes it is useful to compare the alpha spectrum of airborne particulate collected on a filter in the vault of a CANDU Unit during refurbishment, with the alpha spectrum of the inside surface of a pressure tube, as shown below. What I have labelled as the "alpha continuum" is due to the presence of sub-surface Cm-244 in the pressure tube sample. The spectral continuum is symptomatic of the increasing energy loss of alpha particles as they escape from ever greater depths within the sample matrix. For a 5 MeV alpha particle traversing a metal oxide such as UO_2 or ZrO_2, the energy loss dE/dx is ~ 0.3 MeV/µm, so alpha particle emissions with energies less than 2 MeV originate at depths greater than 10 µm within the sample.





By way of a concluding comment on the present status of OPG's assessment of the alpha source term for pressure tube waste, it is noteworthy that in April 2014 OPG made a commitment to the CNSC to implement a so-called Inventory Verification Plan (IVP) for its Low and Intermediate Level radioactive waste. Details of this IVP may be found in OPG's *Consolidated Responses List*, IR Package No. IR-EIS-13-514, page 474, where we even have a "Timeframe" that commits OPG to "*the characterization of at least two samples of all retube waste types*" in the period 2015 – 2017. This initiative was also announced at a CNSC staff presentation held on September 9th, 2014, which included the following overhead:

CNSC Staff Assessment of the Inventory Verification Plan (IVP)

- OPG commits to collect:
  - 3 samples per waste type for all waste types to screen levels of radionuclides, hazardous substances and organic materials
  - a minimum of 10 measurements per key radionuclide per waste type to quantify a conservative activity value
  - at least 2 samples for each nuclear generating station and covering operating timeframes
- IVP will quantify variability in the packages and will provide representative and conservative levels of radionuclides

This Inventory Verification Plan (IVP) is clearly quite ambitious, and the CNSC and OPG are to be commended for initiating such a rigorous verification protocol. However, I have been unable to find *any* indication that this plan is in fact

being carried out as described above, and I would ask the CNSC to provide an update on the status of the Refurbishment Waste IVP and in particular, report on how it has been applied to the validation of OPG's pressure tube waste inventory.

Dosimetry of the Darlington Alpha Event

That ingestion and/or inhalation of Cm-244 are harmful to human health is well documented. Thus, for example, studies of human data on the biokinetics and radiotoxicity of Cm-244 have shown that lung and bone tumor induction and chromosome aberrations are the major hazards from inhaled and systemically deposited Cm-244 – See, for example, Appl. Radiation & Isot. Vol 66(5), 632, 2008. However, first, let's consider some previous investigations of the dosimetry of inhaled alpha emitters, (including curium isotopes), for material derived from CANDU reactors. In 2004, W.M. Smith at AECL carried out ORIGEN-S code calculations to estimate the doses from the ingestion or inhalation of natural UO_2 irradiated up to 30-years in a CANDU thermal neutron flux of 1.21×10^{14} n/cm^2/s – see COG Technical Note TN-03-3012. Because these irradiation conditions are similar to the exposure conditions of a Darlington pressure tube, I have adapted Smith's computational approach to calculate the radionuclide-specific doses resulting from the inhalation of 1 µg of uranium contamination assumed to be associated with pressure tube cutting fines in the refurbishment waste processed in the Darlington RWPB. The results of these calculations are presented in Table 6 below.

Table 6: Calculated Alpha Doses from the Inhalation of 1 microgram of Uranium Irradiated for 30-years in a Neutron Flux of 1.2×10^{14} n/cm^2/s

Transuranic Radionuclide	Alpha Dose (mSv)	Activity Uptake (Bq)
Cm-244	61.1	4700
Cm-242	0.356	284
Pu-238	0.893	56
Pu-240	0.181	11
Pu-239	0.069	4
Cm-243	0.014	1

Table 6 clearly demonstrates the major contribution of Cm-244 to the inhalation dose resulting from an internal uptake of heavily irradiated uranium. Interestingly, two other curium isotopes – namely, Cm-242 and Cm-243 – contribute to the overall alpha dose. Cm-242 is difficult to quantify precisely because, with a half life of only 163 days, it is quite short-lived and decays rapidly after a reactor is shut down to Pu-238. On the other hand, Cm-243 is of interest because OPG's dosimetry methodology for the Darlington uptake assigned *all* of the alpha inhalation dose to this species. However, this assignation appears to have no scientific basis, and I feel quite certain that Cm-244, *not Cm-243*, was by far the largest contributor to the Darlington alpha event source term.

OPG has reported that the inhalation dose to each of the exposed refurbishment workers was about 0.3 mSv – See *CNSC Memorandum RIB/BIR, Action 14051 for CMD 18-M41*. If we accept this value, the data in Table 6 indicate that about 0.005 µg, or 5 ng, of irradiated uranium was inhaled by the exposed workers.

It may be calculated that for the hypothetical case of a single cubic-shaped particle, 5 ng of UO_2 will have a diameter of about 8 µm. This represents the *largest* size of 5 ng of UO_2; or equivalently, we could have 5 ng of UO_2 present as *four* 2 µm particles, etc. This particle size range – i.e. 2 to 10 µm – is significant because it corresponds to the size range of so-called respirable particles, which are particles that, if inhaled, are retained by the lungs.

The average particle size of an airborne dust is also important because it determines the settling velocity, which in turn limits the degree of dispersion of the dust. For particles in the size range 1 - 100 µm, the terminal settling velocity v_{TS} of a particle of diameter d is given (in c.g.s units) by the equation:

$v_{TS}(cm/s) = \rho\, d^2\, g\, /(18\, \eta)$
Where,

ρ is the particle density in g/cm^3 = 10.96 g/cm^3 for UO_2 particles
g is the acceleration due to gravity = 980 cm/s^2
η is the coefficient of viscosity of air in poise = 1.81×10^{-4} poise

Thus, we see that for a 10-micron UO_2 particle, the settling velocity is:

$v_{TS}(cm/s) = \{10.96 \times (10 \times 10^{-4})^2 \times 980\} / \{18(1.81 \times 10^{-4})\}$

$v_{TS} = 3.3$ cm/s

This shows that a 10 µm UO_2 particle released from a height of 1 meter takes 30 seconds to settle. However, if we consider a 1 µm UO_2 particle released from the same 1-meter height, the settling time becomes 50 minutes. Clearly, if the air into which 1 µm UO_2 particles are released is subject to even the slightest horizontal laminar flow – e.g. from ventilation blowers – particles are capable of travelling considerable distances before settling out. This demonstrates the ease with which fine UO_2 particulate from refurbishment waste debris – i.e. material derived from reactor components such as feeder pipes and pressure tubes that invariably contain transuranic contamination – may become widely dispersed once it is released into a reactor vault.

The extent to which airborne alpha contamination has been dispersed into and beyond the reactor vault at Darlington's Unit 2 may be seen in the recent detection of significant levels of alpha particulate in Darlington's environmental emissions. First, note that since 2015, OPG has published weekly airborne radionuclide release data for Pickering A, Pickering B and Darlington. And prior to 2017, these alpha particulate emissions were consistently *below* the detection limit of the emission monitors at all three stations. However, in April 2017, alpha particulate was detected for the first time in Darlington's airborne emissions and has continued to be detected to this day, (Oct 2018). Significantly, mid-2017 and shortly thereafter coincides with the time period when highly contaminated radioactive components, such as feeder pipes and pressure tubes, were being removed from the Darlington Unit 2 reactor as part of its refurbishment. And let's not forget, it was the removal of feeder pipes that created a severe airborne alpha hazard during the 2009 refurbishment of Bruce Unit 1.

Since mid-2017, airborne alpha emissions from Darlington have typically been ~ 20,000 Bq/week and, should this emission rate be maintained well into 2018, the total alpha activity released to the environment for the complete Unit 2 refurbishment project will be ~ 1 MBq. This is comparable to the annual airborne alpha emissions from nuclear fuel reprocessing plants but is unheard of for an operating nuclear power station anywhere in the world! In addition, because of the long-term irradiation and micron particle size of pressure tube cutting fines, Darlington's airborne alpha-

contaminated dust has a higher specific activity than the notorious "hot particles" found in the 1986 Chernobyl accident fallout. And it is this highly radiotoxic dust that is settling over the Durham Region and beyond, and will continue to do so, *as long as Darlington NGS Units are being refurbished*! And please note that *four* schools – Waverley, Holy Family Catholic, Dr. Ross Tilley and Saint Stephen's Elementary – are located less than 5 km downwind from Darlington NGS.

Unfortunately, it has not been reported to the public which stack(s) at Darlington, (and their exhaust flow rates), contribute to these emissions, but the Unit 2 non-contaminated stack is likely to be a significant source. Another alpha particulate emission source would be the RWPB exhaust stack(s) because, as we have seen, an airborne alpha particulate hazard developed very quickly in this building in February 2018 – the very first time pressure tube waste was processed in the Darlington VRS.

It is possible to estimate how much alpha contamination could be dispersed from an operating VRS using surface activity data for this type of equipment. Thus, as reported in Table 4 of this email, a smear of the Point Lepreau VRS collected in 2009 during refurbishment operations, picked up 885 Bq of lose alpha contamination. If we assume the smeared area was ~ 5 cm^2, the average surface contamination of the Point Lepreau VRS was 1.77×10^6 Bq/m^2.

There is a considerable body of experimental data in the open literature – See for example ORAUT-OTIB-0070 (2008) – that has been used to make estimates of internal doses from surface activity measurements using a so-called resuspension factor, K, defined as follows:

K (m^{-1}) = Air Concentration in Bq/m^3 / Surface Concentration in Bq/m^2

For indoor workplace scenarios a typical value of K is 1×10^{-5} m^{-1}. Then, for an operating VRS, and using the above definition of K, we calculate:

Alpha air concentration above a VRS = $(1 \times 10^{-5}\ m^{-1}) \times (1.77 \times 10^6\ Bq.m^{-2}) = 17.7\ Bq/m^3$

From this airborne concentration of alpha particulate, we may estimate the committed effective inhalation dose, D_{In}, (in mSv), for an individual standing next to an operating VRS and working there for a period of time, t hours, (without respiratory protection), as follows:

$$D_{In}\ (mSv) = DCF_{In} \times B \times t \times C$$

Where,

DCF_{In} is the dose conversion factor for the inhaled alpha-emitter in mSv/Bq

B is the worker's breathing rate in m^3 per hour

C is the airborne concentration of the alpha-emitter in Bq/m^3

If we assume the principal alpha-emitter was Cm-244, the DCF_{In} is ~ 0.013 mSv/Bq. Then for an individual who spent 1 hour working beside the VRS in the Darlington RWPB with a breathing rate of 1.4 m^3/hour, his or her committed effective inhalation dose would be:

D_{In} (mSv) = 0.013 mSv/Bq × 1.4 m^3/hour × 1 (hour) × 17.7 Bq/m^3 = 0.32 mSv

This is very close to OPG's estimated dose for each refurbishment worker, (in the RWPB in Feb 2018), of about 0.3 mSv.

Had OPG taken the time and trouble to carry out such a simple, "back of the envelop" calculation *prior to undertaking pressure tube waste container lidding operations*, the inhalation dose to two refurbishment workers in February 2018 *could have been avoided*. [redacted] Nevertheless, once an alpha uptake *was* recognized by OPG, a detailed appraisal of the dosimetry, based on bioassay data, is called for in order to evaluate the alpha source term and assign dose coefficients to the exposure. Unfortunately, however, OPG has provided no information on how the dosimetry for the alpha event was carried out, which raises four important questions:

(i) For the alpha contamination event, what analytical data for urine and fecal samples are available? And, in particular, are there data for ^{239}Pu, ^{240}Pu, ^{238}Pu, ^{241}Am, ^{242}Cm and ^{244}Cm, in both fecal and urine samples? Also, are there analytical data for beta emitters such as ^{14}C, ^{60}Co, ^{94}Nb, and ^{241}Pu and were any of these radionuclides considered in the dose assessment?

(ii) Since the mode of exposure is suspected to be via inhalation, the urine and fecal results must be interpreted through application of a lung model. For ^{244}Cm an intake value can be calculated. But this probably represents only a fraction of the total intake from all radionuclides, including non-alpha emitters such as ^{241}Pu. Therefore, source term information is needed. The best source term data would be from an air sample collected *at the time of the event*, because for inhalation exposure, the intake is proportional to the air concentration, breathing rate and time of exposure. But I have to ask: are any air samples pertaining to the actual exposure event available?

(iii) If we assume the alpha contamination event was caused by inhalation, a key assumption has to be made regarding the solubility of the inhaled material in the lung. The transuranics of interest are formed in uranium dioxide fuel, and there is evidence that shows the fuel used in CANDU reactors behaves like a Type S (or Class Y in ICRP 1979) material in the lung. Therefore, for preliminary dose assessments, it is a reasonable initial assumption that the transuranics will behave in a similar manner. However, in order to test this assumption a second bioassay method is needed, but has one been used?

(iv) The availability of results from independent bioassay methods for the two exposed individuals would provide the opportunity to adjust the Type S lung model parameters. A key assumption in such calculations is that the associated americium and curium radionuclides have the same lung solubility as plutonium.The key parameters involved in such dose calculations are particle size (AMAD value) and three ICRP Publication 66 lung absorption model parameters: (i) The **initial dissolution rate** S_p (i.e., rate at which particles absorbed to blood without being transformed), (ii) The **transformation rate** S_{pt} (i.e., rate at which particles in initial state are converted into the transformed state) and (iii) The **final dissolution rate** S_t (i.e., the rate at which particles in the transformed state are absorbed into blood). Thus, we need to ask: What particle size and dissolution rates were assumed in OPG's dosimetry calculations?

It is worth noting that the four questions listed above were very much at issue at the time of the Bruce alpha contamination event it late 2009 – See for example Bruce Power's *Alpha Dosimetry Report* issued in September 2011. In addition, the uncertainty in the lung solubilities of plutonium, americium and curium was a topic I raised at my intervention for the Bruce Relicensing Hearing in April 2015 – See Intervention 15-H2.18A.

As a follow-up to this intervention, I wrote to the CNSC in May 2015, and specifically addressed the unique and unusual solubility behavior of Cm-244: (See the attachment LUNGSOL to this email for a copy of this correspondence). Thus, in relation to some solubility studies I had carried out on Bruce Unit 1 feeder pipes, I noted the following in my May 2015 email:

It is also important to note that, in these dissolution tests, data for the actinide Cm-244 were also collected. It was found that the distribution of Cm-244 in the Bruce feeder oxide is quite different to that of other actinides. Thus, curium tends to be enriched near the surface of the feeder oxide and is probably incorporated independently of uranium, plutonium or americium. Indeed, curium appears to be co-adsorbed with Co-60 and shows very similar depth-profiles to this ubiquitous activated corrosion product. Given that the distributions of U, Pu, and Am in Bruce feeder oxides are congruent and quite consistent with the composition of high burn-up UO_2 fuel debris, the anomalous behavior of Cm is significant and will be discussed later in this email.

And I concluded my May 2015 email with the following set of questions for the CNSC:

1. Could the CNSC please explain how the dose assessment methodology used by Bruce Power in its Alpha Dosimetry Report, NK21-REP-03416-00001, was validated?

2. Could the CNSC please provide estimates of the precision and accuracy of the dose assignments for the Bruce alpha event?

3. Does the CNSC now agree that lung solubility studies are required for a proper dose assessment of the Bruce alpha event?

4. Will the CNSC state for the record that the results of any future lung solubility studies pertaining to the Bruce alpha event will be made available for review by independent scientists or other interested parties?

These questions were never answered by the CNSC. Therefore, I would now like to ask the CNSC the same questions, but with "Bruce alpha event" replaced by "Darlington alpha event".

And, as a final comment on this topic, I note that lung solubility studies of irradiated UO_2 samples were undertaken in 2015 by the Radiobiology Research Group at Canadian Nuclear Laboratories, (CNL). – See *Characterization of Alpha Radiation Hazards*, Report No. 153-121110-REPT-080, issued April 2016. This CNL report ends with the following statement about the results of the study:

Currently within Canada, the most commonly used method to identify and quantify the Committed Effective Dose, (CED), following intakes of irradiated fuel is either a semi-annual or annual urine bioassay to quantify Pu-239 intake, which is followed by dose attribution for the other associated radionuclides. It is assumed that the various actinides, fission products and other fuel constituents will 'leach' at comparable rates due to the bulk dissolution of the irradiated UO2 fuel upon deposition in the lung. Again, accuracy of this approach is dependent on the assumed isotopic composition inhaled by the worker and the assumption that each radionuclide within the material matrix dissolves at the same rate. The results indicate that this is perhaps an invalid approach, as some actinides and fission products are leaching from the bulk irradiated material in the lungs at considerably faster rates. This allows for further excretion or transportation to targeted tissues/organs where residing will deliver a localized dose other than at the site of initial deposition.

This conclusion by CNL researchers especially applies to their findings on the solubility of Cm-244. This is well illustrated by the Figure 33, below, which is taken from their *Characterization of Alpha Radiation Hazards* Report:

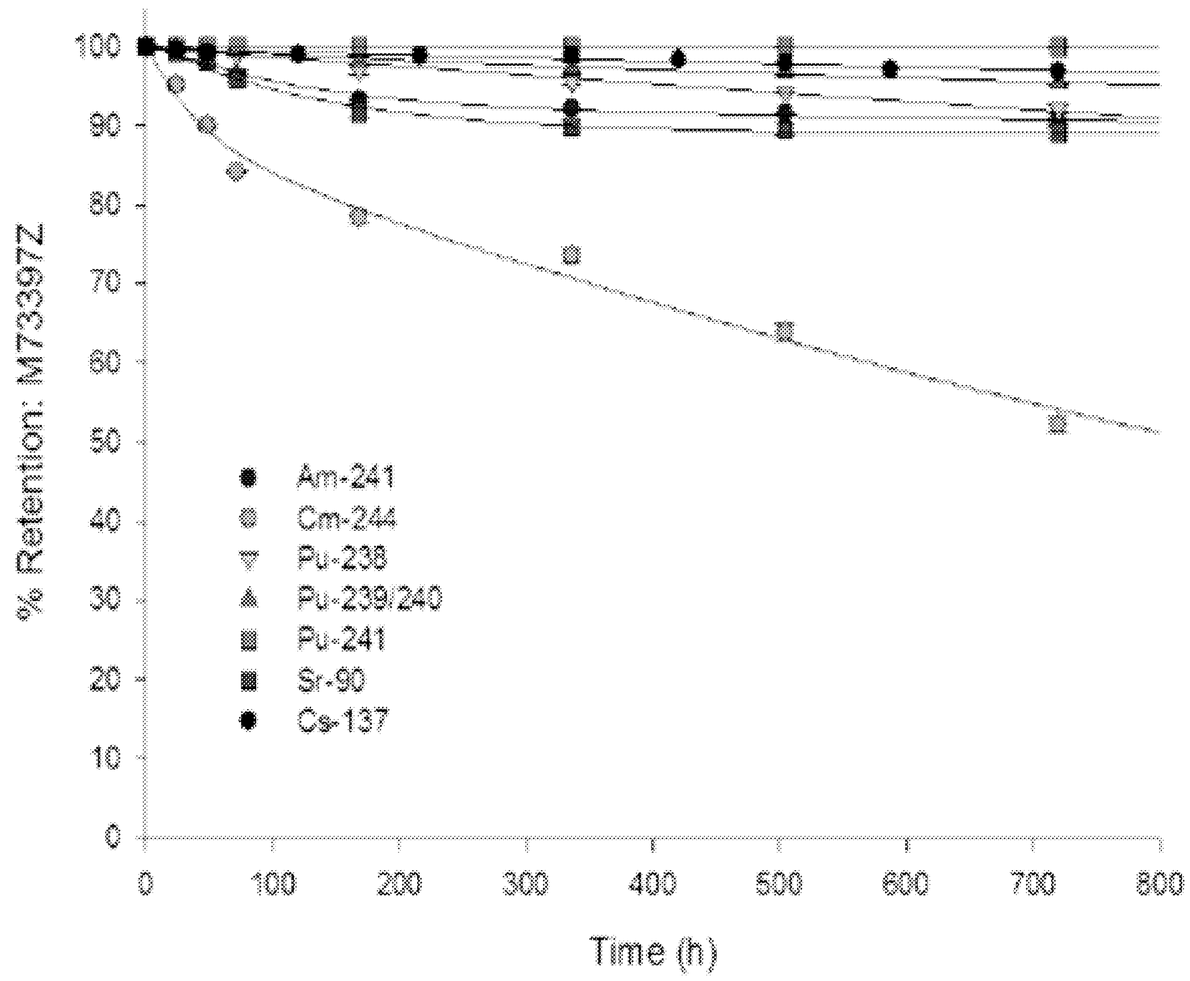


Figure 33 Individual leach rate measurements for radionuclides associated with irradiated UO_2 fuel ID M73397Z.

CNL's Figure 33 clearly shows the unique behavior of Cm-244 in comparison to other actinides such as plutonium and americium. Indeed, it confirms that alpha dosimetry based on the assumption that curium's lung solubility is similar to that of plutonium is likely to lead to highly erroneous results.

These findings by CNL researchers once again emphasize the need for *material-specific lung solubility data* in order to make reliable alpha inhalation dose assessments when Cm-244 is a major contributor to the alpha source term. And, as we have also shown, this is especially true for the inhalation of pressure tube cutting fines where the transuranic isotopes are present in two radically different matrices – the pressure tube surface oxide and the pressure tube base metal. At the present time, there appears to be no experimental data available on the solubility of Cm-244 within a heavily irradiated Zr-2.5%Nb matrix.

Additional Remarks:

1. First I would like to say a few words about the CNSC's role in the Darlington alpha contamination event. This is an important issue because the CNSC said it was satisfied with OPG's radiation protection plan (RPP) for the refurbishment of Darlington when it gave its approval for the project to proceed. This declaration of confidence in OPG's RPP was stated explicitly at least two times: First, at a CNSC Public Hearing held on November 3rd, 2015, and secondly in early December 2015 when the CNSC announced its reasons for the renewal of the Operating Licence for Darlington NGS.

The first declaration of CNSC's position on OPG's RPP for Darlington was made by Ms. Suzanne Karkour, a CNSC site inspector at Darlington, as follows:

> *I want to confirm that we conduct type 2 inspections on radiation protection specifically on radiological hazard control. An inspection was just conducted in September on radiological hazard control. This inspection is conducted every three years, and …. we have found that Ontario Power Generation is in compliance with regulatory requirements.*

The second declaration of confidence in OPG's RPP was stated in the CNSC's January 2016 Darlington's Licence approval document "*Reasons for Decision*":

> *The Commission is of the opinion that, given the mitigation measures and safety programs that are in place or will be in place to control radiation hazards, OPG has provided, and will continue to provide, adequate protection to the health and safety of persons and the environment. The Commission is satisfied with the measures taken by OPG to minimize the exposure of workers as well as that of members of the public, and notes that the doses received by workers and estimated doses to members of the public are well below regulatory limits.*

There are two crucial points of note in these CNSC pronouncements:

(i) In the first CNSC statement noted above, Ms. Karkour expresses satisfaction with OPG's ability to provide adequate radiation protection for its nuclear workers at Darlington. However, this satisfaction is based on *one* inspection carried out in September 2015 – *a full year before* any refurbishment activities had taken place. Furthermore, the CNSC's September 2015 assessment of OPG's RPP was based on the performance of *four operating reactors* for which all systems are closed and sealed so as to be completely leak tight. Under such circumstances, the only significant radiation hazard would have been the gamma radiation fields near the reactor face or PHTS pipework; at the same time, airborne contamination from alpha particulate would *not* have been an issue.

(ii) In the second CNSC statement on OPG's radiation protection program for Darlington we read the following: *The Commission …. notes that the doses received by workers are well below regulatory limits*. Again, the "doses to workers" under consideration here would be doses from *operating reactors* which, as we know, do not entail any contributions from airborne alpha particulate hazards.

However, as I have demonstrated in this email, cutting open a CANDU reactor is guaranteed to release radioactive contamination to the reactor vault in the form of fission, transuranic and activated corrosion products such as Cs-137, Sr-90, Pu-239, Am-241, Cm-244, Co-60, Nb-94, etc. And the method used to dispose of waste pressure tubes – namely the crushing and segmentation into 2-inch coupons in a mechanical volume reduction system – also creates a fine dust from the friable surface oxide on the inside surface of each pressure tube, as OPEX from the 2009 Point Lepreau and Bruce refurbishments has shown. What is particularly concerning about this waste disposal technique is that the pressure tube cutting fines contain respirable particles of the potent alpha emitter Cm-244 with specific activities in excess of 3000 Bq/µg U. This is not the type of airborne particulate to be taken lightly!

Nevertheless, and in spite of the well-documented hazards associated with the removal and disposal of irradiated pressure tubes, end fittings and feeder pipes, it appears that the CNSC considers a reactor undergoing refurbishment to be radiologically equivalent to an operating reactor, *which is quite clearly an unreasonable and dangerous assumption*. Nevertheless, one of the key documents issued by the CNSC that allowed OPG to proceed with its refurbishment plans – **The Darlington Licence Conditions Handbook, LCH-DNGS-R000 Effective Date: January 1, 2016** – is written as if a full reactor refurbishment presents no additional radiation protection issues than those arising from the routine maintenance of said reactor!

Remarkably, however, the 2016 Darlington LCH has very little to say about *any* refurbishment activities, even though the refurbishment of four CANDU reactors is a billion-dollar project involving thousands of workers that will take place over much of the 10-year Darlington licensing period granted to OPG by the always compliant CNSC. In fact, in the 145 pages of the Darlington LCH, the word "*refurbishment*" appears only "half-a-dozen" times! To be fair, the Darlington LCH does concede that:

> *Refurbishment outages differ from planned maintenance outages in that the duration is longer, work activities are more complex, and the configuration of the unit is significantly altered to allow work to proceed.*

But does this statement offer any advice on how routine radiation protection procedures should be revised for a refurbishment?

And on the topic of worker training for the refurbishment of Darlington we read:

> *With regard to refurbishment activities planned for Darlington NGS, it is recommended that OPG start preparing, in advance, for the additional future training-related requirements, including continuing training for personnel requiring certification or maintenance thereof.*

Here again, we get platitudes about "training requirements", but not a word about what these requirements might be!

But on the vitally important topic of Radiation Protection Programs for the refurbishments of Darlington Units 1 – 4, we find there are <u>*no special provisions for refurbishment activities noted anywhere in the Darlington LCH*</u>! [redacted]

2. Here I would like to address the topic of *consistency* and whether or not it's being displayed by OPG with regard to its various pronouncements about the radionuclide content of the particulate matter in a pressure tube waste container. We know that an RWC-PT contains up to 30 pressure tubes, (sectioned into 2-inch square coupons), constituting a total mass of 1830 kg of irradiated Zr-2.5%Nb. In the *Western Waste Management Facility Safety Assessment Report*, W-REP-01320-00008-R000, issued in November 2016, we find in Section 3.2.5 – entitled *Accident Scenario: Dropped Retube Waste Container* – a detailed discussion of the quantity of each radionuclide that is likely to be released from a retube waste container in the event of a drop accident. Here, the key parameter is the amount of metal dust estimated to be produced by cutting pressure tubes. The report uses a value of 0.01% of the waste pressure tube mass as an estimate of the fraction of material present in each RWC-PT in the form of settled dust. This implies there are 183 grams of radioactive zirconium-alloy dust per waste container.

OPG's accident scenario next considers that a dropped retube waste container's release fraction is ~ 10^{-3}, implying that 183 mg of pressure tube dust becomes airborne as a result of the postulated accident. Then, from pressure tube inventory data, releases of the highest activity radionuclides such as Nb-94 and Co-60 are determined. The resulting values prove to be in reasonable agreement with the activities of these species that I previously estimated to be in Darlington pressure tube cutting fines, as reported in Table 2 of this email. (So far, so good!).

However, OPG's *Western Waste Management Facility Safety Assessment Report* then uses these estimated radionuclide releases to calculate a hypothetical inhalation dose for a worker assumed to spend ½ hour close to the dropped retube waste container at the time of the accident, with his/her fractional uptake of the release assumed to be 10^{-3}. But now things are not so good because this is where consistency problems with OPG's results for the hypothetical pressure tube waste container accident and the <u>*real*</u> February 2018 container lidding accident become evident. Thus, OPG calculates that 95% of the container drop inhalation dose, or 0.64 mSv, is contributed by C-14, with no contribution at all from Cm-244, while OPG's analysis of the February 2018 container lidding event indicates that "*all*" of the inhalation dose of 0.30

mSv was due to Cm-244. Therefore I have to ask OPG to explain this inconsistency and wonder if *either* of these dose estimates has any validity!

3. And continuing the topic of *consistency,* I would ask that the obvious contradictions in OPG's account of the Darlington alpha event be resolved. In particular OPG needs to confirm that (i) The waste container that was being lidded by the two workers that received the internal uptakes was without question ***the first*** *pressure tube waste container to be lidded in the RWPB*. And (ii) OPG needs to acknowledge that it ***was*** anticipated that high levels of contamination would be present in the RWPB after operation of the VRS because motion of the lids would result in higher levels of contamination being introduced to the lidding area.

4. For completeness, I ask OPG and the CNSC to consider the potential contribution of tritium, H-3, to Darlington's environmental emissions and worker dose during the Unit 2 refurbishment. This may appear to be a non-issue because conventional wisdom would argue that the main sources of tritium, namely the moderator and heat transport D_2O, are drained from the reactor in the earliest stages of a CANDU refurbishment, thereby removing this hazard from further consideration.

Nevertheless, there are several reasons, with compelling supporting evidence, to view tritium as a neglected, but significant radiological hazard throughout many refurbishment activities. As for evidence supporting this contention, first and foremost, airborne emissions data for reactors undergoing refurbishments *show an increase in airborne tritium emissions* as soon as the reactor is opened up for the removal of major components such as feeder pipes, end fittings, pressure tubes and calandria tubes.

Thus, for example, emissions data for Bruce Unit 1, which had been shut down for many years prior to 2009, showed a sudden unexpected spike of about 10 TBq in tritium emissions from September 2009 to February 2010 – precisely the period when pressure and calandria tubes were being removed from this Unit. Similarly, for Darlington in the period April 2017 to March 2018, we see airborne tritium emissions that are about 16 TBq in excess of the station's normal emissions, which may reasonably be attributed to tritium releases from Unit 2 refurbishment activities during that time.

So we need to consider: What is the source of this "*refurbishment tritium*"? The answer appears to be pressure and calandria tubes which are known to contain copious quantities of tritium. Thus, in the 1990s I made direct measurements on 17 samples of irradiated pressure tube and found an average of 2.27×10^6 Bq of H-3 per gram of Zr/Nb. For Darlington, with 480 pressure tubes, each weighing 61 kg, we then predict a pressure tube inventory of 66.5 TBq of H-3 per Unit. Unfortunately, there appears to be no published data available for H-3 in calandria tubes; nevertheless, we conservatively estimate it to be 50% of the H-3 activity in pressure tubes, or 33.2 TBq per Unit. Thus, we expect a total tritium inventory for Darlington Unit 2 pressure and calandria tubes of 100 TBq.

As previously noted, the airborne tritium emissions for Darlington in the 1-year period from April 2017 to March 2018 show several spikes which correlate well with refurbishment activities on Unit 2. Thus, for example, tritium emission spikes that appear from April to December 2017 may be attributed to the removal of Unit 2 pressure tubes and calandria tubes which led to a substantial release of HTO to the Unit 2 vault and a tritium emission of about 10 TBq from the Unit 2 contaminated stack. However, we can safely assume that refurbishment workers in the Unit 2 vault at that time were wearing plastic suits and therefore received minimal or zero internal doses.

However, if we consider airborne tritium releases from Darlington in the first quarter of 2018 we observe a spiked emission of about 6 TBq that correlates well with the February alpha exposure

event; and here it appears that pressure tube waste disposal was the culprit. Thus, the crushing and cutting of pressure tubes (and calandria tubes!) into 2-inch coupons would release significant quantities of tritium (probably as HTO) from the freshly exposed Zr/Nb metal surfaces. Indeed, it was observed in 1985 (See OHRD Report No: 85-248-K) that the *in situ* cutting of Pickering Unit 1 pressure tubes led to tritium concentrations in the Unit's west vault of up to 3×10^6 Bq/m^3. Had OPG considered this well-documented OPEX, the potential for a significant tritium dose to unprotected workers in the vicinity of a pressure tube waste volume reduction system (VRS) should have been anticipated by Darlington's refurbishment Radiation Protection team, but apparently was not.

A simple calculation of the potential tritium dose in question may be made as follows:

First, consider the concentration of airborne tritium in the building housing a VRS, of open volume V, to be represented by the time-varying quantity $C_{H3}(t)$, then the concentration at time t is given by:

$V.dC_{H3}(t)/dt$ = Rate of activity release − Rate of activity removal

Next assume that tritium is released to the building air only during operation of the VRS, and, for simplicity, also assume that the tritium is removed by one mechanism: building air ventilation. Then we may write:

$dC_{H3}(t)/dt = Q/ V - C_{H3}(t).\ n$
where,

Q is the rate of tritium release in Bq per hour

V is the building open volume in m^3

n is the fractional ventilation rate = ventilation rate in m^3/hour / V(m^3)

Now consider that the airborne activity changes in the building air over a time interval of one hour so that short-term fluctuations may be ignored. Under this circumstance it is safe to assume that equilibrium conditions have been attained so that:

$dC_{H3}(t)/dt$ equals zero. The resulting steady-state, (or slowly varying), airborne activity concentration is then given by:

$C_{H3}(eq) = Q / V_R$

where V_R is the building ventilation rate in m^3/hour.

To apply this formula, I have assumed that each waste container holds coupons from the cutting of 30 pressure tubes and a total of 480 Darlington Unit 2 pressure tubes were processed. I have also assumed that the VRS took one hour to process one container. From the airborne tritium emission of 6 TBq, (previously attributed to pressure tube waste processing), one may conclude that 6 TBq of tritium was released to the VRS building air over a total of 16 hours of pressure tube waste processing, so that:

$Q = (6 \times 10^{12})\ /16 = 3.75 \times 10^{11}$ Bq/hr

Finally, taking the building ventilation rate to be 10,000 scf/min or 17,000 m^3/hr, we have:

$C_{H3}(eq) = 3.75 \times 10^{11}$ Bq/hr / 17,000 m^3/hr = 2.2×10^7 Bq/m^3

Now the potential tritium inhalation dose, D_{H3}, in mSv, for a refurbishment worker who spent 50% of his time in the VRS building during pressure tube waste processing is given by:

D_{H3} (mSv) = $C_{H3}(eq)$ (Bq/m^3) × 8 (hr) × Breathing Rate (m^3/hr) × DCf_{H3} (mSv/Bq)

Then for a typical worker breathing rate of 1.8 m^3/hr and using the ICRP Publication 119 DCf_{H3} (Dose Coefficient for tritiated water) of 1.8×10^{-8} mSv/Bq, we find:

D_{H3} (mSv) = 5.7 mSv

This dose from the processing of 16 waste containers is clearly *much larger* than OPG's estimated inhalation dose from Cm-244 of 0.3 mSv. But, in fact, the tritium dose remains marginally higher than the estimated Cm-244 dose, even *if only one container was processed*!

5. And finally, it's encouraging to note that OPG and the CNSC agree that the Darlington alpha exposure event was caused by the following failures in OPG's radiation protection protocols:

• *The radiation hazard characterization performed by Ontario Power Generation (OPG)was insufficient to estimate the radiological risks associated with the alpha hazards in the RWPB*

• *OPG made non-conservative assumptions that resulted in the incorrect classification of alpha hazards*

• *Workers were exposed to alpha hazards in the absence of suitable protective measures due to incorrect alpha hazard classification*

• *Monitoring and oversight were insufficient to provide timely indications as to when alpha hazards were changing and when current protective measures for workers were no longer sufficient for the actual hazards present.*

However, on April 23, 2018, OPG submitted to the CNSC its Detailed Event Report for D-2018-04257 - *Other Events of Regulatory Interest: Internal Uptake from Personnel Contamination Event*. CNSC staff reviewed this Report and found that additional information was required to substantiate the identified causes and to provide assurance that OPG's corrective actions would prevent recurrence of a similar event. In response to the CNSC's comments, OPG provided the results of a common cause analysis to address deficiencies identified during work on the Darlington Unit 2 Refurbishment Project. This analysis identified three (3) common factors that contributed to the RWPB event:

[1] Production focus diverting OPG's RP staff from their primary focus of providing radiation protection safety oversight

[2] RP Coordinator weakness in RP fundamentals due to lack of knowledge and experience

[3] Inadequate focus on basic contamination control and hygiene by workers.

Of these three factors, item [1] stands out as the most serious and concerning of all the causes that have been proposed for the Darlington alpha contamination incident. However, I would like to point out that the notion that a "*production focus diverted OPG's RP staff from their primary focus of providing radiation protection safety oversight*" is entirely untenable for waste disposal operations in the Darlington RWPB, and I question why OPG would make such a claim.

It certainly is true that there have been production pressures on Darlington refurbishment staff since the start of the Unit 2 refurbishment operations in October 2016. And, as I have stated before, I believe that such work pressures contributed to the 2009 Bruce alpha contamination event. But the Bruce incident involved contamination of workers during feeder pipe removal and replacement, *activities that were on the critical path of the Bruce refurbishment*. However, when we compare the Bruce alpha contamination event to the alpha event at Darlington, the main difference is that the Darlington event occurred during pressure tube volume reduction and waste disposal – *activities that were **not** on the refurbishment critical path*. How can I be sure of this? Well, consider the overhead below, which is taken from a recent OPG presentation on the Darlington refurbishment, and especially note the 2nd bullet:

Overhead from a Presentation by Scott Stafford – OPG Station Manager ALARA Refurbishment

Refube Waste Processing Building (RWPB)

- A new dedicated RWPB is planned to perform the waste reactor component processing and packaging operations for the RFR of all four Darlington units. This approach was adopted due to the limited space available in the Darlington reactor vaults. In addition, the use of a dedicated facility for this purpose provides better functionality, increased productivity, improved reliability, and enhanced ALARA and safety compared with performing these same operations in the reactor vaults.
- The separate RWPB also decouples reactor core component processing and packaging from critical path activities in the reactor vault.

OPG has published many details of the Darlington Unit 2 refurbishment operations including the *Critical Path Level 1 Schedule* which, (for the time period of interest), is summarized in Table 7, below.

Table 7: Activities on the Critical Path of the Darlington Unit 2 Refurbishment

Task No.	Description of Activity	Start Date	No. of Days
1	Defuel Reactor	15-Oct-16	113
2	Install Bulkhead	24-Mar-17	38
3	Install Shielding	1-Apr-17	19
4	Remove Fuelling Machine Bridge	10-May-17	44
5	Remove Feeder Cabinet	25-Jun-17	38
6	Feeder Removal	1-Aug-17	28

7	Pressure Tube Sever	10-Sep-17	16
8	Bellows Sever	1-Oct-17	33
9	End Fitting Removal	1-Nov-17	36
10	Pressure Tube Removal	1-Dec-17	37
11	Calandria Tube Release	10-Jan-18	28
12	Calandria Tube Removal	10-Feb-18	32
13	Inspection and Cleaning	20-Mar-18	38
14	Calandria Tube Install	28-Apr-18	101
15	Pressure Tube Install	20-Aug-18	125

From an examination of Table 7, it is abundantly clear that *all* of the critical path activities listed in the table involve the removal or installation of reactor components. Likewise, *none* of the critical path activities listed in Table 7 include activities involving the processing or packaging of reactor waste. Indeed, it is absurd for OPG to claim that the performance of *two* workers, who were simply lidding some waste containers, could be on the refurbishment project's critical path, or could somehow adversely affect the schedule of a billion-dollar project! However, it is easy to understand where this line of thinking is coming from when we consider what Mr. Dietmar Reiner, OPG Senior Vice President Nuclear Projects, said at a November 2015 CNSC Hearing in response to a question about alpha contamination at Darlington:

> *You know, the last place we ever want to be in the refurbishment is in an event where there's a significant safety hazard, because… it can have a significant impact on a project from a scheduling perspective.*

This is a remarkable statement coming as it does from the person who is ultimately in charge of the overall safety of the Darlington Refurbishment Project. The truth is that a CANDU refurbishment is filled with hazardous activities such as the cutting and "J-prepping" of feeder pipes, the removal of pressure tubes and calandria tubes, the operation of the VRS, etc. This is because these activities invariably create an airborne particulate hazard which carries with it a high probability for the spread of alpha-contamination.

So, to say: "*The last place we ever want to be in the refurbishment is in an event where there's a significant safety hazard*", shows a complete lack of understanding of how potentially dangerous many refurbishment operations are. Simply put: When it comes to the refurbishment of a CANDU reactor, airborne radiation hazards come with the job; *its how you deal with these hazards that matters*.

And on a related topic, it is worth commenting on another of the factors that OPG claims contributed to the RWPB alpha event, namely that the RP Coordinator was "*weak in RP fundamentals due to lack of knowledge and experience*". This begs a number of questions:

(i) Was the RP Coordinator qualified to do his (or her) job or not?
(ii) If he *was* qualified, then how could he lack the knowledge and experience to do the job?
(iii) But if he was *not* qualified, *then why was he hired to do the job in the first place*?

Inspection of Darlington's CNSC-approved Radiation Protection Program (RPP) shows that it *requires, as a minimum, adherence to the following regulations*:

• That no person shall be hired as a Nuclear Energy Worker if the person's qualifications, training and experience do not comply with the licence conditions applicable to this person.

• That the station has equipment in place for the analysis of the potential for the creation of hazards.

• That within the facility, provision is made for continuous monitoring of hazard levels where there is a probability of sudden unexpected increases in radiological hazards.

• That for air monitoring, instruments which *accurately* determine the level of airborne contamination are used.

[redacted]

[redacted]

[redacted]

[redacted]

And finally, with regard to the Darlington alpha event, may I summarize my main concerns and questions stemming from this unfortunate internal exposure of two refurbishment workers in February 2018:

(i) Why has the C-14 contribution to the refurbishment workers' inhalation dose from the Feb 2018 lidding event been ignored when OPG considers it to be *the major contributor* to the dose from a hypothetical retube waste container accident?

(ii) Has OPG considered the contribution of *tritium* to the refurbishment workers' inhalation dose from the Feb 2018 event?

(iii) The contradictory claims by OPG as to whether or not the pressure tube waste container being processed in the RWPB at the time of the exposure incident was *the first* such container, needs to be resolved.

(iv) Could OPG justify the claim that it was *not anticipated* that high levels of contamination existed in the RWPB near the VRS and that it did not expect the motion of the lids would result in high levels of contamination being introduced to the lidding area?

(v) Was OPG not aware of the valuable OPEX from Point Lepreau's 2009 discovery of high levels of alpha contamination associated with the operation of a pressure tube waste VRS?

(vi) Does OPG have reliable measurements of *the inventory of Cm-244* in its pressure tube wastes?

(vii) Does OPG have any information on *the surface concentration of loose or fixed Cm-244* on its pressure tube wastes?

(viii) Would OPG and/or the CNSC provide an update on the status of its radionuclide inventory verification plan and in particular, report on how it has been applied to the validation of OPG's pressure tube waste inventory

(ix) Could OPG provide data on the particle size of the alpha-contaminated dust inhaled by two refurbishment workers in February 2018

(x) Could OPG provide information on the calculations it used to estimate the radiation dose to the two exposed refurbishment workers – in particular what particle size and lung solubility were assumed in OPG's dosimetric calculations?

(xi) Is OPG (and the CNSC) aware of the findings of CNL's *Characterization of Alpha Radiation Hazards*, Report No. 153-121110-REPT-080, issued April 2016? Were these findings applied to the dosimetry of the Darlington event?

(xii) Will OPG confirm that, starting in April 2017, alpha particulate was detected for the first time in Darlington's airborne emissions and has continued to be detected to this day, (Oct 2018)? And will OPG acknowledge that the source of these highly radiotoxic emissions is the dispersal of alpha-contaminated dust by refurbishment activities on Darlington Unit 2?

(xiii) OPG has acknowledged that the RP Coordinator in charge of the safety of the two exposed refurbishment workers was "*weak in RP fundamentals due to lack of knowledge and experience*". This begs a number of questions: Was the RP Coordinator qualified to do his (or her) job or not? If he *was* qualified, then how could he lack the knowledge and experience to do the job? But if he was *not* qualified, *then why was he hired to do the job in the first place*?

(xiv) Does OPG stand by its claim that the performance of *two* workers, who were simply lidding some waste containers, could be on the refurbishment project's critical path, or could somehow adversely affect the schedule of a billion-dollar project?

[redacted]

Sincerely

Dr. F. R. Greening

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