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**Written submission from  
Anna Tilman and  
Eugene Bourgeois**

**Mémoire d'Anna Tilman et  
Eugene Bourgeois**

**Regulatory Oversight Report for  
Canadian Nuclear Power  
Generating Sites: 2018**

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**Rapport de surveillance  
réglementaire des sites de centrales  
nucléaires au Canada : 2018**

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Commission Meeting

Réunion de la Commission

**November 6, 2019**

**Le 6 novembre 2019**

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October 10, 2019

To:

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From:

Anna Tilman  
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**Re: Comments on CNSC's Regulatory Oversight Report (ROR) for Canadian Nuclear Power Generating Sites: 2018** (CMD 19- M30, September 6, 2019)

## **Introduction**

The following comments relate to specific elements of the Report, specifically with respect to the method of evaluation and inspections of nuclear facilities applied by the CNSC, the setting of Derived Release Limits (DRLs) for specific radionuclides released from these facilities, public accountability with respect to reporting of accidents, and public access to radioactive releases via the National Pollutant Release Inventory (NPRI). In addition, comments include pressure tube issues (models and testing), related to Nuclear Power Plants.

The *Regulatory Oversight Report for Canadian Nuclear Power Generating Sites: 2018* provides Canadian Nuclear Safety Commission (CNSC) staff's assessment of the overall safety performance of Canadian nuclear power plants (NPPs) and their adjacent waste management facilities (WMFs) for the year 2018. The rating categories used by CNSC staff in these assessments of defined safety control areas (SCA) are: FS - fully satisfactory, SA – satisfactory, BE - below expectations, and UA - unacceptable (ROR p. 7, 8, p. 245, p. 250).

Not one facility has scored less than SA. In other words, the Report categorically portrays to the public that there no problem has arisen in the various categories that are “assessed”, or that any incident that may have arisen would be of concern. Nor is it clear what the rating categories imply or incorporate. Furthermore, no adequate explanation is given as to precisely how the individual safety areas are “assessed” or what is incorporated in the inspections carried out.

But what exactly do these ratings mean? If the majority of operations are ‘satisfactory’, that could also indicate that some things were not satisfactory but, on the whole, things OK. And is it even possible that not one issue has arisen in all these facilities is of concern – to workers, the public, or to operational issues?

## Inspections – Compliance Verification Program (CVP)

According to the ROR:

“The safety assessments presented in this report were based on the results of activities planned through the CNSC compliance verification program (CVP). In 2018, these activities included inspections led by inspectors and supported by subject matter experts. CNSC inspections include:

- Type II inspections, which evaluate the outputs and outcomes of licensee programs and typically involve documentation review and on-site activities
- field inspections, which are limited in scope (e.g., focusing on a specific area of the facility) and involve on-site activities to collect data on the outputs and outcomes of licensee programs.”<sup>1</sup>

However, a repeated concern that has been raised in my previous submissions is the continuing use of Type II inspections, especially with respect to Nuclear Power Plants (NPPs), and the lack of use of Type I inspections.<sup>2</sup>

The purpose of Type I inspections is to determine whether licensees’ programs comply with all applicable regulatory requirements and to verify that the programs have been carried out. Type I inspections are normally broad, program-based inspections similar to audits or evaluations.

- They are in-depth examinations of licensees’ processes and operations.
- They normally require a multidisciplinary inspection team due to their broad scope.
- The Canadian Nuclear Safety Commission (CNSC) did not conduct Type I inspections on nuclear power plants during the 2013–14 to 2014–15 fiscal years.

In comparison, Type II inspections are intended to verify the delivery (results) of licensees’ programs through routine item-by-item checklist inspections. Type II inspections are usually inspections of specified equipment, facility material systems, or of records, products, or outputs that result from the process the licensee must follow.

- They are an on-site snapshot of the licensees’ operations.
- They can be conducted by only one inspector.

In 2016, as noted in previous submissions, the Commissioner of the Environment and Sustainable Development under the Office of the Auditor General (OAG) issued a report that commented on and made recommendations on inspections of nuclear power plants.<sup>3</sup> In particular, the OAG Report reviewed two types of inspections, Type I and Type II.

The OAG Report noted that:

“Overall, we found that the CNSC could not show that inspectors always followed CNSC procedures when carrying out and documenting inspections of nuclear power plants. This

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<sup>1</sup> ROR, p. 19 Section 1.4.4

<sup>2</sup> Citing from a submission (CMD18-H4.54, May, 2018, renewal of Bruce Power’s Operating Licence):

<sup>3</sup> [http://www.oag-bvg.gc.ca/internet/English/parl\\_cesd\\_201610\\_01\\_e\\_41671.html](http://www.oag-bvg.gc.ca/internet/English/parl_cesd_201610_01_e_41671.html)

has led to inconsistencies, gaps in documentation, and missed opportunities for identifying improvements in conducting inspections.”<sup>4</sup>

The Report recommended that CNSC develop detailed criteria to help identify when to conduct Type I inspections. CNSC’s response to that recommendation was that several major relicensing or refurbishment activities for nuclear power plants entailed comprehensive compliance reviews (including desktop reviews, site inspections, and reviews of unplanned events) during the audit period, and would provide the required information needed to ensure regulatory compliance, and as a result, Type I inspections were not required during that period.

In response to the findings of OAG’s 2016 report and its recommendations that the CNSC develop detailed criteria to help it identify when to conduct Type I inspections respect to nuclear power plants, while the CNSC updated criteria to trigger a Type I inspection, no such inspections have been carried out to date. For example, no Type I inspection has been carried out at Bruce Power since 2005. This is particularly disconcerting, especially in light of the alpha incident at Unit 1 in 2009, which would have and should have heightened the need to conduct detailed inspections of the nature of Type I inspections and on a routine basis. This is all the more important given the nature of the refurbishment work being carried out at Bruce Power, Darlington, for example, and the continuing operation of the Pickering Plant well beyond its predicted lifespan.

The notable lack of thorough inspections is also contrary to the safety ethic or “safety culture” which CNSC and Nuclear Power Plant operators to uphold.

## **Derived Release Limits (DRLs)**

“Derived Release Limits” (DRLs) are the legal upper regulatory bounds set by the CNSC for releases of radioactive substances to the environment. A DRL represents the quantity of a radionuclide that, if released from the specified facility in a year, would result in a dose to the most exposed member of the public of 1 mSv/yr, i.e., the International Commission on Radiological Protection (ICRP) public dose limit. Exceedances of the DRL trigger reporting to the CNSC, followed by a formal investigation and regulatory oversight.

DRLs are calculated for specific radionuclides expected to be found in the airborne and liquid operational effluents as defined in CSA Standard N288.1.

Licensees also establish Action Levels (ALs), which are specific quantities of a radionuclide (released as an airborne emission or waterborne effluent) that, if reached, could indicate a loss of control of part of a licensee’s environmental protection program and the need for specific actions to be taken and reported to the CNSC.

According to the ROR, “releases of radionuclides to the environment in 2018 were well below the DRLs for each facility; hence no radiological releases to the environment from the facilities exceeded the regulatory limits. Further, no environmental action levels were exceeded in 2018 at the NPPs and WMFs.”<sup>5</sup>

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<sup>4</sup> [http://www.oag-bvg.gc.ca/internet/English/parl\\_cesd\\_201610\\_01\\_e\\_41671.html](http://www.oag-bvg.gc.ca/internet/English/parl_cesd_201610_01_e_41671.html)

<sup>5</sup> P. 66 ROR Report CMD 19-M30; Appendix H

A review of both DRLs and ALs for a number of nuclear facilities clearly illustrates a total disconnect between the value of the DRLs and the reported emissions, which seriously questions the veracity and effectiveness of DRLs as regulatory tools. For example:

- DRL models are prepared by the licensee and reviewed by the regulator. Models (e.g., the Environmental Transfer Model) are used in preference to monitoring actual emissions as a basis for determining the limits.
- The determination of DRLs also involves many factors including identifying and characterizing representative persons, exposure pathways, meteorology, and dose conversion factors which provide the estimated radiation dose imparted to a cell, tissue or organism by the radioactive decay of one atom of that radionuclide.
- Licensees may choose model parameters that underestimate doses without the benefit of public or independent expert peer review.
- Dose estimates for air emissions are based on assumptions about the behaviour of stack plumes, which are notoriously difficult to model. The local meteorology also compounds the difficulty in determining such estimates.
- Estimates of public doses arising from waterborne discharges of radionuclides are based on the dilution capacity of receiving waters using the average rather than the minimum water flow. The latter would be more appropriate because of variations in water flow caused by climate change and other factors.
- The cumulative effects of exposure to multiple radionuclides over time are not taken into account.

Nuclear licensees and the CNSC typically report emissions as percentages of DRLs, in addition to reporting the actual emissions. For example, the following table illustrates the Annual Airborne Emissions of Tritium Oxide from Bruce A and B, the Central Maintenance and Laundry Facility (CMFL), and the Western Waste Management Facility (WWMF) for the years 2014-2016 and their respective DRLs in Becquerels per year (Bq/year).<sup>6</sup>

Facility	Airborne Emissions (Bq/yr) of Tritium 2014-2016			DRL (Bq/yr)
	2014	2015	2016	
Bruce A	7.51E+14	7.05E+14	5.66E+14	1.98E+17
Bruce B	4.13E+14	3.74E+14	5.70E+14	3.16E+17
CMFL	6.55E+09	1.06E+10	6.99 E+9	2.45E+17
WWMF	7.17E+12	4.14E+12	2.06E+13	2.96E+17

The DRL for the CMFL is a hundred million times greater than the reported emissions. Furthermore, it is almost identical to the DRLs for Bruce A or B and the WWMF. There is no explanation as to why a laundry facility would have such a large DRL.

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<sup>6</sup> Refer to Environmental Monitoring Reports, 2014 to 2016. Tthe Douglas Point Waste Management Facility (DPWMF), also on the Bruce site reports releases of Tritium of 2.74E+11, 1.33E+10, and 1.59E+11 for those years.

Because of the sheer magnitude of difference between DRLs and emissions, these percentages lead to treating the actual emissions as insignificant, which is seriously misleading. Rather than showing that the emissions are insignificant relative to the DRLs, these differences could very well demonstrate that the DRLs are not realistic and thus not appropriate.

### **Action Levels (ALs)**

The CNSC requires licensees to determine ALs to serve as an early warning to indicate when releases may be deviating from the norm.<sup>7</sup> ALs are typically about 10% of DRLs. Exceedances of ALs trigger reporting to the CNSC. Unlike DRLs, which are established on an annual basis, the ALs is established to compare with monitoring data.

While ALs may indicate spikes or irregularities, as in the case of DRLs, they are inordinately much greater than the monitored emissions. For example, the AL for emissions of tritium to air for **one week** is approximately the same as the **annual** monitored emissions of tritium to air. The AL for emissions of tritium to water is 100 times greater in one month than its emissions in a year.

Besides being non-regulatory, the methodology for establishing or calculating ALs, which is in the hands of the licensee, is not consistently applied.<sup>8</sup>

In that both DRLs and ALs are orders of magnitude greater than the reported monitored releases, they do not serve as effective and meaningful measures to protect the public and the environment. It is essential that the CNSC review and address this matter.

### **Aging Management Issues – Pressure Tubes<sup>9</sup>**

The aging of fuel channels (calandria and pressure tubes) is considered to be the single greatest cause of declining performance in CANDU reactors. The fuel channel components most affected by degradation are pressure tubes. Issues related to pressure tube problems include leakages at the pressure tube rolled joints, neutron-induced creep, embrittlement and blister formation due to excessive hydrogen pickup, fretting and corrosion. Factors such as the weight of the fuel bundles, the high temperatures, pressures and radiation fields (neutron radiation) in the reactor core, the absorption of hydrogen, the embrittlement of their metal walls (zirconium alloy), corrosion and deterioration contribute to these problems.<sup>10</sup> For example:

#### **i) Fracture Toughness - Hydride Formation**

A reduction in fracture toughness caused by an increase in hydrogen concentration (also referred to as deuterium uptake) is the dominant contributor to the failure pressure tubes.<sup>11</sup> As the operation time of the reactor increases, so does the concentration of hydrogen. The accumulation of hydrogen results in the formation of blisters and cracks, a

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<sup>7</sup> <http://nuclearsafety.gc.ca/eng/resources/educational-resources/feature-articles/radiation-dose-limits-release-limits-and-action-levels.cfm>

<sup>8</sup> CNSC DIS-12-02: Process for Establishing Release Limits and Action Levels at Nuclear Facilities February 2012

<sup>9</sup> ROR p. 50

<sup>10</sup> [http://www-pub.iaea.org/MTCD/publications/PDF/te\\_1037\\_prn.pdf](http://www-pub.iaea.org/MTCD/publications/PDF/te_1037_prn.pdf)

<sup>11</sup> CMD 15-H2 Aging Management p. 49-53: CNSC Commission Public Hearing Part 2 – Licence renewal re Bruce Power CNSC staff February 27, 2015

process referred to as hydrogen embrittlement. This can result in a loss of toughness, and cause a stable, time-dependent crack growth mechanism called Delayed Hydride Cracking (DHC). This is most pronounced during the reactor's transition states between shut down to full power and vice versa. During DHC, hydrides migrate to stress regions and promote crack growth. When a critical condition is reached, probably related to size, a fracture develops, the crack extends, and the process continues on the newly exposed metal.

## **ii) Deuterium Ingress and Corrosion**

During "hot" operation conditions, pressure tubes react with the heavy water coolant, resulting in an increase of the concentration of hydrogen (that is, deuterium in terms of the equivalent hydrogen concentration,  $H_{eq}$ ) over time. The pressure tubes absorb deuterium in two main locations, the inside surface of the main body of the pressure tube and the end fittings where the ingress of hydrogen is much more rapid than in the body of the tube.

Pressure tube material has a limited solubility of hydrogen, referred to as terminal solid solubility (TSS), which increases with increasing temperature. If sufficient quantities of deuterium are absorbed the TSS is exceeded, leading to the formation of zirconium hydrides. These hydrides weaken the cladding of the pressure tubes by decreasing its hardness, ductility and density, making them susceptible to DHC, which could cause pressure tube failures.

These stressors change the dimensions and material properties of pressure tubes. Over time, the tubes increase in diameter (known as diametrical creep) and length, causing their walls to thin out and sag and potentially come into contact with the outer calandria tube, which increases the likelihood that they will rupture. As a result, their useful life and the maximum power a reactor can provide are limited.

These issues are particularly critical in light of the work going on regarding refurbishment of reactors, and extending the end-of-life of the fuel channel components beyond the current licence limits, referred to as the "hold point". With increasing age, fuel channels become more vulnerable to these problems.

With respect to aging management, the ROR states that: (p.51-52)

Pressure tube aging management activities include inspections to verify the condition of the tubes, surveillance activities to monitor material property changes, and the development of assessment methodologies and fitness-for-service guidelines.

The licensees demonstrate the ability to safely operate pressure tubes through assessments of the current and expected conditions of the pressure tubes that are based on an understanding of relevant degradation mechanisms. Research activities as well as inspection and maintenance programs provide data to periodically validate the input parameters for these assessments. To assess mechanisms or parameters that are dependent on neutron flux (e.g., diametric creep of pressure tubes), EFPH is the best indicator.

During the PROL renewals for the PNGS and Bruce A and B in 2018, the Commission approved new EFPH limits for pressure tubes in those units, which were identified as



compliance verification criteria in the LCHs for the PNGS and Bruce A and B (see sections 3.2.6 and 3.3.6, respectively).

However, for in-service changes in pressure tube properties (e.g., fracture toughness), equivalent hydrogen (Heq) concentration is more important than EFPH. Fracture toughness is an important parameter that is modelled and used for assessments of leak-before-break and fracture protection of pressure tubes. For temperatures below 250°C, Heq content in the pressure tube is a critical input to the fracture toughness model. **The analytical fracture toughness model that CNSC currently accepts for use in this temperature range is only valid up to a Heq concentration of 120 ppm. [Emphasis added].**

CNSC staff considers the current regulatory process to monitor additional validation of the existing fracture toughness model up to Heq of 120 ppm to be adequate to ensure to support CSA-mandated assessments. For units approaching the validity limit of the existing toughness model (120 ppm Heq in any pressure tube), licensees are required to develop a revised toughness model (capable of predicting toughness beyond 120 ppm Heq) and submit the technical basis for CNSC staff’s approval well before any pressure tube reaches 120 ppm.

Appendix G (p.271-2) of the ROR provides the current and predicted status of key parameters and models for pressure tubes in Canadian power reactors (key dates – time to reach 120 ppm Heq, the maximum EFPH and Heq concentrations and the validity of fracture test models). (In the case of Bruce Power, while test models are valid, the estimates made by CNSC staff for Units 5, 7 and 8 were based on the most recent reports of hydrogen isotope concentration in pressure tube rolled-joints.)

No facility is expected to surpass 120 ppm Heq, which is notable in itself. At this stage, there is no evidence for this to be the case, especially given the likelihood that predicted schedules are could be fluid, and not necessarily the reality.

The following table illustrates the estimates of the predicted EFPH for Units 3-8 at the time of their corresponding Major Component Replacement (MCR) date and the estimated maximum Heq at that time.<sup>12</sup>

Unit	Estimated Year to reach 120 H <sub>eq</sub> ppm	MCR Outage Date	Predicted EFPH at MCR	Estimated H <sub>eq</sub> ppm at MCR
3	n/a	2023	245,000	102
4	n/a	2025	255,000	104
5	2020	2026	294,000	151
6	Dec. 2019	2020	245,000	121
7	2022	2028	300,000	147
8	2027	2030	300,000	139

Notably, Units 5, 7 and 8 have Heq estimates greater than 120 ppm.

<sup>12</sup> CMD 18-H4 Supplementary Presentation by CNSC March 14, 2018, p. 47 (Note, EFPHs estimated by CNSC are approximately 2,000-3,000 EFPH higher than predicted by Bruce Power (CMD 18-H4.1A Supplementary Presentation by Bruce Power March 14, 2018, p. 27)

As a closing comment on this issue, one has to question the validity of the models, what degree and thoroughness of testing and verification will be carried out, and at what frequency. Will the CNSC find the need to not approve further increases in Heq in that this would impose a risk that cannot necessarily be measured but may be probable? That is why there is a reason for imposing hard limits, and for not taking risks that push the envelope, with no hard evidence to do so.

## **Public Accountability – Access to Information**

### **A. Reporting Events - CNSC Requirements**

Since 2003, Nuclear Power Plant operators have been required to submit “Event Reports”, known as S-99 Reports, to the CNSC on a yearly basis under the S-99 regulatory standard.<sup>13</sup> While the regulatory document, REGDOC-3.1.1, *Reporting Requirements for Nuclear Power Plants* has replaced the S-99 regulatory standard as of June 2015, the requirements under this regulation are similar to those of the S-99 regulations.<sup>14</sup>

Accordingly, for every “reportable” event a nuclear facility must file a full report that provides details regarding the event, including the effects on the environment, the health and safety of persons, and the maintenance of security that has resulted or may result from the situation, as well as the actions that the facility has taken or proposes to take with respect to the reportable event.<sup>15</sup> These reports are posted by the licensees on their respective websites.

The regulation also states that “Licensees should use the situation or event reporting according to this regulatory document as an input to their public disclosure protocol.”<sup>16</sup>

However, there are limitations as to what is considered a reportable event and the information that is made publicly available. For example:

- The websites of licensees include only a list of events that have occurred at a specific station for a specific year and a report number.<sup>17</sup> No further information is provided as to how one could access reports on any of these events. Thus, the public has no indication as to the cause of the accident/incident; its relative severity; or whether there were releases of radioactive and other hazardous substances that resulted in exposure by workers and/or the public.
- The lists of events exclude those considered to involve confidential or security-based information. While a request can be made for such information under Access to Information, some of that information could still be redacted. This limits public access to reports of events beyond those submitted under REG 3.1.1.

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<sup>13</sup> Section 6.3 of the S-99 regulatory standard, *Reporting Requirements for Nuclear Power Plants* CNSC March 2003 [http://nuclearsafety.gc.ca/pubs\\_catalogue/uploads/S99en.pdf](http://nuclearsafety.gc.ca/pubs_catalogue/uploads/S99en.pdf) (Criteria - p. 24, 25)

<sup>14</sup> [nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory.../regdoc3-1-1/index.cfm](http://nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory.../regdoc3-1-1/index.cfm)

<sup>15</sup> CNSC CMD 17-H.3 p. 31: Sections 29 and 30 of the *General Nuclear Safety and Control Regulations* outline specific scenarios under which a licensee must file a report to the CNSC.

<sup>16</sup> <http://nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/published/html/regdoc3-1-1/index.cfm#appA>

<sup>17</sup> <http://www.brucepower.com/2017-reportable-events/>: <http://www.brucepower.com/site-updates/>

- Only those events that meet specific criteria are required to be reported to the CNSC. Even when an event is reported to the CNSC, its staff decides whether it has enough “significance” to warrant further review by the CNSC Commission. There are no clear guidelines as to what would determine the significance of an event or whether it should be reported.
- Even if an event is initially reported under Event Initial Reports (EIR), there may be no follow-up action that the CNSC deems warranted.
- There is no indication as to whether the findings of the investigation of the causes of an event are verified by an independent third party. Nor is there a requirement for these reports to include recommendations as to specific measures that should be taken to remedy/prevent these events, especially those that are recurring.

Despite weaknesses in the system of reporting these events, it does inform the public of incidents, irregularities etc., at these facilities, some of which result in the release highly toxic substances to the environment, both radiological and non-radiological.

However, one must question whether the change in regulation as to reporting events had a negative impact on the public’s ability to access event reports. For example, for Bruce Power’s Licence Hearing in 2015, S-99 reports and follow-up reports from the CNSC were readily obtainable by request from the CNSC, which no longer seems possible, which is unfortunate, if so.<sup>18</sup>

## **B. Update on the National Pollutant Release Inventory (NPRI)**

As stated in the ROR, “the CNSC is making radionuclide release data more readily accessible to the public as part of its commitment to open government and its mandate to disseminate this information to the public. In addition to including the data in the ROR, the CNSC and the National Pollutant Release Inventory (NPRI) are working together to establish active links between the CNSC and NPRI web sites. Stakeholder sub-group consisting of environmental non-governmental organizations and industry are beta testing the links between the NPRI site and existing CNSC data products (RORs, etc). The CNSC has also commenced the creation of downloadable, digital databases of radionuclide releases, further supplementing the range of CNSC environmental data products linked to the NPRI website. The downloadable databases are expected to become part of the active beta testing activities in the latter part of 2019.”<sup>19</sup>

As an environmental non-governmental organization (NGO) member of this stakeholder subgroup, I can confirm that to date, there has been no meeting of the committee as a whole. Therefore, NGOs have not yet participated in discussions on the testing of links between the NPRI and existing data products.

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<sup>18</sup> Refer to Submission from Eugene Bourgeois and Anna Tilman re re-licensing Bruce Power March 2015 CMD 15-H.210 p. 83-102, p.118 -120

<sup>19</sup> Appendix H: DERIVED RELEASE LIMITS AND RADIOLOGICAL RELEASES TO THE ENVIRONMENT (p.273)

On a further note, prior to the establishment of this committee, comments as to including emissions data directly on the NPRI site as the preferred path were made in a joint letter by several NGOs to ECCC.

### **Emergency Planning**

The concerns of Bruce residents about emergency planning remain unresolved following the relicensing hearing in May 2018, although discussions have been underway with CNSC staff with a goal of resolving them, including seeking further input from Health Canada concerning the type of modelling used to describe the meteorology of the Inverhuron region, and in particular, whether two-dimensional Gaussian models are used rather than three-dimensional dynamic modelling. These discussions are ongoing.