ASSESSMENT OF NEUTRON DOSE FROM CAMECO'S PORT HOPE CONVERSION FACILITY

Prepared For:

Cameco Corporation

Prepared By:

SENES Consultants Limited

June 2009

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SUMMARY

The potential dose to workers and members of the public from neutron radiation has been investigated in the past by both Cameco and the Canadian Nuclear Safety Commission (CNSC). In 2000, Cameco carried out a neutron survey at their Port Hope Conversion Facility (PHCF) and concluded that no special restrictions or monitoring of workers were warranted, including when working close to UF_6 cylinders. Neutron doses to all workers at the PHCF were likely to be much less than 1 mSv/y.

The CNSC, in a response to two letters to the president of the CNSC, commented on various environmental matters related to the PHCF. In their comments, the CNSC considered the possible neutron doses at the PHCF. The CNSC noted that the emission of neutrons from a UF_6 cylinder is a well understood phenomenon and that neutron radiation fields produced in this way are measurable but small relative to the gamma fields emitted by the cylinder. The CNSC also concluded that the transport of UF_6 cylinders from and to the PHCF does not represent an undue risk to members of the public. In summary, the CNSC has concluded that the neutron dose rate from UF_6 cylinders is very low and separate monitoring of neutron radiation levels is not warranted.

These evaluations all support the conclusion that, while measurable, neutron dose rates from UF_6 cylinders are low and do not require routine separate measurement. Nonetheless, Cameco has continued to evaluate potential neutron dose rates associated with UF_6 activities. This report describes the various activities carried out by or for Cameco to assess the neutron dose rates arising from the presence of UF_6 at the PHCF, primarily from the storage of UF_6 cylinders at the site.

Recent surveys using Landauer's passive CR-39 monitors around the PHCF and other sites in and around Port Hope also showed quite low neutron dose rates below Landauer's limit of detection. However, using the raw Landauer data, the mean dose rate along the fenceline was calculated to be about 0.008 microsieverts/hour (µSv/h) higher than the mean for other locations removed from the PHCF. The average incremental (above a background of 0.080 µSv/h) gamma radiation dose rate at the fenceline TLDs was 0.041 µSv/h for the same period of CR-39 deployment. This is about 5 times higher than the nominal estimate of an average incremental neutron dose rate of 0.008 µSv/h estimated from the CR-39 measurements.

A portable neutron meter was selected for this study to provide a reasonable counting time for measurements at background and at PHCF fenceline locations. Neutron dose rate measurements were collected at background locations, selected TLD locations and at various distances from a UF_6 cylinder requiring between 0.5 and 6 hours of counting time. The neutron dose rates at three

TLD locations ranged between 0.001 µSv/h above background at TLD-13 and TLD-14 across from the Mill Street receptor, to about 0.018 µSv/h above background near TLD-8.

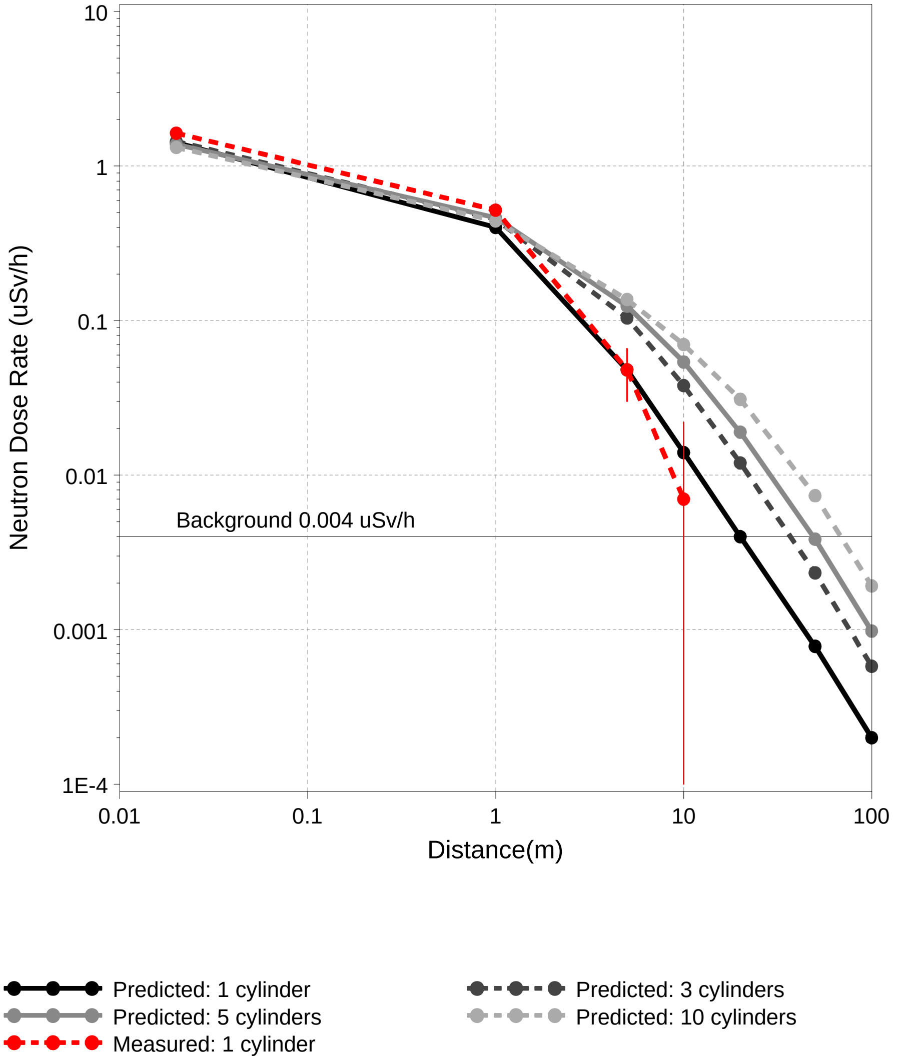
The neutron dose rates around UF_6 cylinders were also modelled using the well-known Monte Carlo N-Particle (MCNP) model and compared with measured neutron dose rates. The measured and predicted dose rates showed a similar pattern, with measured values close to the MCNP modelled values for a single cylinder. The data confirmed that the MCNP modelling and measurements were consistent.

MCNP predictions for one, three, five or ten cylinders along with the measured values for one cylinder are shown in Figure S.1. As illustrated by the figure, the neutron dose rate decreases with distance from the cylinders in a predictable and expected fashion.

Based on observations from work performed to date:

- As a consequence of the low neutron dose rates, it is difficult to obtain precise measurements of neutrons using CR-39 monitors which are currently the best passive, long-term detectors available;
- Although recent Landauer CR-39 data shows neutron dose rates below Landauer's limit of detection, the raw CR-39 data suggest incremental neutron dose rates of the order of 0.008 µSv/h at fenceline locations;
- Preliminary survey data using a survey meter capable of detecting neutrons at background levels and the modelling indicate that the incremental neutron dose rates approach background at distances of 10-20 m from a single UF_6 cylinder and around 70 m for an array of cylinders. Receptors located at distances greater than these listed above would receive an incremental neutron dose no greater than the background level.
- MCNP provides a reliable method of estimating neutron dose rates.

Figure S.1
Dependency of MCNP Predicted and Measured Neutron Dose Rates on Distance from the UF_6 Cylinders



Based on raw data from the recent measurements with Landauer CR-39 monitors, a neutron dose rate of about 0.008 µSv/h represents an estimate of the average incremental (above background) neutron dose rate at the fenceline (TLD stations). On this basis and assuming, as was done for Cameco's DRL (SENES 2008), that a fenceline walker spends one hour each day (i.e. 365 h per year) exposed to a neutron dose rate of about 0.008 µSv/h, he or she would receive an incremental dose from neutrons of about 3 µSv/y.

Based on task-time analyses and estimates of neutron dose-rates, neutron doses to workers who handle or work in proximity to UF_6 cylinders are estimated to be below 1 mSv/y. On this basis, no routine neutron monitoring is required.

Overall, the combined information from long-term monitoring using CR-39 monitors, survey data with a portable neutron meter and modelling show remarkable coherence and suggest that neutron dose rates are reasonably predictable. Moreover, the results of the current modelling compare well to that reported by others. A ratio of 5 between incremental gamma radiation and neutron dose rates could conservatively be used to predict neutron dose rates for receptors at or near the fenceline. The estimated neutron dose to a walker at the fenceline represents 1% of the current Cameco licence limit for public receptors. Receptors located further away from the fenceline are exposed to much lower dose rates, and at distances beyond 70 m from the cylinders, the incremental neutron dose rates are comparable to background.

Furthermore, while measurable neutron dose rates from UF_6 cylinders are low and do not require routine separate measurements, Cameco has continued to evaluate potential neutron dose rates associated with UF_6 activities.

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1.0 INTRODUCTION

Cameco's Port Hope conversion facility (PHCF), located in the Municipality of Port Hope, receives natural uranium trioxide (UO_3) from Cameco's Blind River uranium refinery where it is reduced to uranium dioxide (UO_2) or converted to uranium hexafluoride (UF_6) in separate process streams. In uranium compounds that contain fluorine, such as UF_6, neutron radiation fields are naturally created as a result of a reaction between fluorine and the alpha particle emitted by uranium.

The potential dose to workers and members of the public from neutron radiation has been investigated in the past by both Cameco and the Canadian Nuclear Safety Commission (CNSC). As discussed below, based on a neutron survey at the PHCF, Cameco concluded that there was no need for special restrictions or monitoring of workers as the result of working close to UF_6 cylinders and moreover, that neutron doses to all workers were likely to be much less than 1 mSv/y (Takala 2001).

The CNSC, in a response to two letters to the president of the CNSC, commented on various environmental matters related to the PHCF, including neutron radiation. In May 2005, the CNSC indicated that the emission of neutrons from a UF_6 cylinder is a well understood phenomenon and that neutron radiation fields produced in this way are measurable but small relative to the gamma fields emitted by the cylinder. The CNSC concluded that the transport of UF_6 cylinders from and to the PHCF does not represent an undue risk to members of the public in Port Hope (Pereira 2005a).

Subsequent CNSC memoranda further commented on the potential doses from UF_6 cylinders and concluded that the neutron dose rate from UF_6 cylinders is extremely low and that separate monitoring of neutron radiation levels is not warranted (Hinton 2005, Pereira 2005b). Finally, in a presentation to an OECD workshop, Carisse (2007) reviewed the then-available data on environmental neutron exposures from UF_6.

These previous evaluations all support the conclusion that, while measurable, neutron dose rates from UF_6 cylinders are low and do not require routine separate measurement.

Nonetheless, Cameco has continued to evaluate potential neutron dose rates associated with UF_6 activities. This report describes the various activities carried out by or for Cameco to assess the neutron dose rates arising from the presence of UF_6 at the PHCF, primarily arising from the storage of UF_6 cylinders at the site.

1.1 SOURCE OF NEUTRONS

Spontaneous fission from uranium naturally occurs at a low rate. In addition, neutrons are also produced as the result of cosmic rays interacting with the earth's atmosphere. Thus, neutrons contribute, at a low level, to the natural background radiation dose. However, when elements such as oxygen and fluorine are mixed with alpha-emitting nuclear material, a reaction can occur that releases neutrons. This occurs for example when the alpha particle from the spontaneous radioactive decay of uranium is captured by a fluorine nucleus. The resulting fluorine nucleus is then unstable and may release a neutron to return to a stable configuration. Neutrons are produced at a much lower rate from the spontaneous fission of uranium alone than from fluorinated uranium compounds.

The interaction of alpha particles from uranium with nuclei of fluorine generates neutrons of approximately 2 MeV energy (U.S. DOE 2004). The intensity of the neutron radiation field varies with distance from the source and the uranium enrichment. Based on the alpha-neutron yields for oxides and fluorides of uranium, the neutron yield per gram for uranium isotopes in oxides is 100 times less than in fluorides (i.e., UO_2 vs. UF_6) (U.S. DOE 2004). Therefore, the dose rate expected from the neutron radiation field from UO_2 and UO_3 would be much smaller than from UF_6. In the case of UF_6 cylinders (containing natural to 5% enriched uranium), the typically measured neutron dose rates are approximately 0.1 to 2 µSv/h (U.S. DOE 2004).

Neutrons occur naturally in the environment. According to UNSCEAR 2000 (Annex B, Figure II), the outdoor cosmic ray neutron dose rate at sea level at our latitude (about 43-44 degrees north for the Port Hope – Toronto area) is in the order of 0.01 µSv/h. Potential variations with factors such as season or solar activity can occur. The rate increases with elevation above sea level but the difference between the Port Hope (less than 100 m above sea level) and sea level neutron dose rates is less than 10% (based on atmospheric depth and neutron attenuation factors in UNSCEAR 2000).

Measurements of the neutron dose rate in surveys in Japan (roughly comparable latitude to Port Hope) indicate background neutron effective dose rates of about 0.003 to 0.005 µSv/h (Nagaoka *et al.* 2009, Rasalonjatovo *et al*. 2002).

Measurements of the neutron dose rate at background locations (away from the PHCF) were made with a portable neutron meter in February 2009. These results, as described in Section 3.3, ranged between 0.003 and 0.004 µSv/h. The results (based on total counts of about 25 to 40) have an uncertainty, due to counting statistics, of about 0.001 µSv/h. These levels are therefore somewhat lower than would be suggested by the UNSCEAR number, but comparable to the values measured in Japan.

1.2 OUTLINE OF REPORT

The assessment of neutron dose rates at the PHCF is described in the following chapters:

Chapter 2 – Results of Recent Measurements - describes recent measurements made with passive (CR-39) detectors and with a hand-held neutron survey meter suitable with sufficient precision for measuring at background levels.

Chapter 3 – Modelling of Neutron Dose Rates - describes the results of recent modelling of neutron dose rates arising from loaded UF_6 cylinders using the Monte Carlo N-Particle Model (MCNP).

Chapter 4 – Overview - provides a short discussion of the key observations and conclusions from the recent measurements and modelling.

Chapter 5 – References - lists the references cited in the report.

Appendix A - Photographs of January 2009 Neutron Monitoring.

Appendix B - Supporting Technical Information.

Appendix C - Previous Evaluations of Neutron Dose Rates at Cameco's Port Hope Conversion Facility.

2.0 RESULTS OF RECENT MEASUREMENTS

Cameco has undertaken a number of previous neutron measurements as presented by Carisse (2007). These presentation slides are provided in Appendix C with a summary of results provided in Appendix B.

2.1 NEUTRON MEASUREMENTS WITH CR-39

Cameco measured neutron radiation levels at the PHCF fenceline and other locations over a three month period (early September to early December 2008). The passive instruments were CR-39 monitors from Landauer; however, the laboratory reports indicate that the measured neutron dose over a 2208 hour deployment period was <100 µSv for all locations. This converts to a dose rate of <0.045 µSv/h for these measurements which is larger than the estimated 0.004 µSv/h background neutron dose rate for the area (discussed in Section 3.3).

Landauer provided the transit control reading in addition to the neutron dose at the location monitors. The following section calculates the point estimate of neutron dose rate for those concentrations reported as below detection limit by Landauer. Examining the range of these measurements relative to the site may provide qualitative information on the pattern of neutron radiation.

A detailed description of the CR-39 monitoring data and estimated doses is provided in Appendix B.

In brief, the calculated neutron doses are low and highly uncertain; therefore, information from individual measurements is difficult to assess. However, comparison of averages can be informative since random variations in the measurement process at below detection limit levels are "averaged" out. The UF_6 cylinders are considered to be the main source of neutrons at the PHCF. The mean values were calculated for the fenceline TLD locations and for all other TLDs grouped and the difference in the means is a conservative estimate of the neutron dose rates due to the UF_6 cylinder. The "PHCF" mean was about 0.008 µSv/h higher than the mean for the "Other" TLD locations and this approached statistical significance. The average incremental gamma radiation rate (above a background of 0.08 µSv/h) at the fenceline TLDs was 0.041 µSv/h for the same period of CR-39 deployment. This dose rate is about 5 times higher than the nominal estimate from the CR-39 of an average incremental neutron level of 0.008 µSv/h.

2.2 NEUTRON SURVEY METER

A list of neutron survey meters which were potentially suitable for measuring neutron dose rates at the Cameco's PHCF was compiled by SENES. The criteria for suitability were based on sensitivity of the detector, operational temperature range, gamma rejection, availability of a scaler or integrating option, and portability. The neutron monitors identified as potentially suitable are provided in Appendix B (Table B.2.2-1).

The neutron monitoring instrument used in this study was the high sensitivity He-3 Meridian Neutron Monitor, Model 5085 from Far West Technology Inc. It can be operated as a rate meter or in scaler mode with the instrument reporting either dose rate, or the total number of counts. There is also an export function on the instrument where data can be exported to computers or other recording points. The instrument is relatively light, reasonably portable and quiet during operation.

2.2.1 Locations Measured

Locations were measured beginning with initial exploratory measurements on 30 January 2009 and continuing from 3 February through 9 February. Measurements were collected at various locations in Port Hope and the PHCF to reflect the range of neutron radiation levels including:

i) background locations well away from the facility;
ii) various distances from an isolated UF_6 cylinder to determine the pattern of neutron radiation with distance (potentially for comparison with MCNP monitoring results). These measurements were collected at the height of the centre of the cylinder.
iii) TLD locations of interest (TLD-13 and TLD-14 on the east fence of the centre pier); and,
iv) other miscellaneous locations on the PHCF.

Background measurements were also taken outside Port Hope. Due to the low neutron dose rates, the measurements were collected as scaler counts with extended monitoring times up to about 6 hours. A description of the location, typically with a photograph, the number of counts and the time required to collect those counts were recorded. Photographs are included in Appendix A. The counts and times required are provided in the following section.

2.2.2 Data Reduction

The neutron dose rate was determined using the instrument-specific calibration (i.e. 1.27 μSv/h per count per second). Table 2.2-1 shows these measurements along with an estimate of the uncertainty[1], due to counting statistics in the measured values. The relative uncertainty is higher for lower numbers of counts. Where possible, the background neutron rate was

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[1] The statistical variance due to counting statistics is equal to the number of counts. For example, 100 counts will have a variance of 100 and standard deviation of 10 counts. This indicates that the two sigma uncertainty for a count of 100 would be approximated as +/-20.

subtracted from the measured values to indicate the incremental neutron dose rates at the various locations monitored and their uncertainties.

Table 2.2-1
Neutron Measurements Collected With Portable Neutron Meter

	Counts [a]	**Time (h)**	**Dose Rate (μSv/h)**	**Absolute Uncertainty [b] (2 sigma)**	**Increment (μSv/h)**	**Absolute Uncertainty [b] (2 sigma)**
Background						
Richmond Hill	41	3.26	0.004	0.001		
Toronto	27	3.02	0.003	0.001		
Cobourg (Garage)	25	3.50	0.003	0.001		
Dale Road (Port Hope)	42	4.21	0.004	0.001		
Peter Street (Port Hope)	54	5.41	0.004[c]	0.001		
Station No.8 (near TLD-8)	13	0.22	0.021	0.012	0.018	0.012
TLD-13	52	3.57	0.005	0.001	0.002	0.002
TLD-14	57	4.39	0.005	0.001	0.001	0.002
Building 22, office 22-G-7 (facing East)	60	5.92	0.004	0.001	0.000	0.001
Building 22, office 22-G-7 (facing West)	168	15.37	0.004	0.001	0.000	0.001
Cylinders						
Test Area Background	26	0.33	0.028[d]	0.011		
Side of Cylinder						
contact	1175	0.25	1.660	0.097	1.632	0.097
1 m	774	0.50	0.546	0.039	0.519	0.041
5 m	107	0.50	0.075	0.015	0.048	0.018
10 m	49	0.50	0.035	0.010	0.007	0.015
End of Cylinder						
contact	1073	0.25	1.511	0.092	1.483	0.093
1 m	421	0.50	0.297	0.029	0.270	0.031
5 m	50	0.34	0.052	0.015	0.024	0.018
10 m	15	0.42	0.013	0.007	<0.01	n/a

Notes: Numbers may not add or subtract to table values due to rounding. The number of significant figures shown should not be taken as indicative of the precision of the values.

[a] Count rate of a minimum of 50 counts is recommended for any future measurements to keep the uncertainty (2 sigma) of the measurements below 30%.

[b] Uncertainties calculated based on counting statistics and error propagation for a difference.

[c] Site background away from PHCF and neutron sources.

[d] Background for testing of a single cylinder.

Some of the measurements had relatively low total counts; for example, one measurement result was 15 counts which provided an estimated dose rate of 0.013 μSv/h with an uncertainty of +/ 0.007 μSv/h. This level of count approaches what is usually considered a method detection limit (the uncertainty equals the measured value). Neutron dose rates were measured with the neutron meter at two additional TLD locations (Stations 1 and 31); however, these resulted in very low counts (1 and 3) due to the time available for monitoring. These have not been reported in the table because of their very large uncertainty (nevertheless, the neutron dose rates at these locations would be low).

2.2.3 Interpretation

The measured neutron levels and the increments above the background shows a measureable range of levels. The Peter Street location represents background in Port Hope with a dose rate of 0.004 µSv/h (+/- 0.001 µSv/h) and this required about 5.5 hours of measurement to collect this level. Another background measurement in Port Hope was similar.

Neutron levels at TLD-13 and TLD-14 locations were nominally higher than the Peter Street background measurement; however, the confidence interval for the increment included the value of 0 (no change). Longer measurements could potentially demonstrate a statistical difference. One measurement inside the fenceline near TLD-8 demonstrated an increment of 0.018 µSv/h above the Peter Street background. The photograph of this location in Appendix A demonstrates the presence of UF_6 cylinders which may be emitting neutrons. The building #22 measurement had similar neutron levels to the Peter Street background.

A single cylinder was isolated from other cylinders; however, the relatively nearby cylinders on site contributed to a local background for the measurements of 0.028 µSv/h which is above the Peter Street background. The 0.028 µSv/h background is appropriate for the measurements collected up to 5 m distance from the cylinder. Future studies would benefit from taking the cylinder further away (50 m or more) from other neutron sources to more precisely estimate the contribution at larger distances.

The measurements show a decrease in neutron dose rates with distance from the cylinder ranging from an increment of 1.632 µSv/h (at contact) to below the local area background at a distance of 10 m from the end of the cylinder. The measurements closer to the cylinder are considered more reliable.

2.2.4 Gamma Radiation Exposure and Neutron Measurement

During the neutron measurement program, gamma radiation measurements were also collected using a Bicron microsievert meter at the same locations during the 30 January 2009 measurements.

At Station Number 8, the gamma radiation level was 0.115 μSv/h which demonstrates elevation above the typical background for Port Hope and this would suggest a gamma radiation increment of 0.035 μSv/h at the location (above the 0.080 μSv/h background gamma radiation dose rate for Port Hope). The measured neutron dose rate at this location was 0.021 µSv/h and this corresponds to an incremental neutron dose rate of 0.018 µSv/h +/ 0.012 µSv/h (based on an approximate background neutron dose rate of 0.003 to 0.004 µSv/h from Table 2.2-1). Based on these nominal estimates, the ratio of gamma radiation to neutron dose rate at this location could be a factor of 2; however, the neutron dose rate is quite uncertain and this estimate of the relationship is considered unreliable.

Table 2.2-2 shows a comparison of incremental (above background) gamma radiation and neutron dose rates measured around the test cylinder. The ratios at locations where the higher gamma radiation and neutron dose rates were measured relatively precisely (contact and 1 m) have ratios of gamma radiation dose to neutron dose ranging from 6 to 9. Measurements were not precise enough to be definitive at larger distances. Nonetheless, the data suggest that within about 10 m from a cylinder, the incremental neutron dose rates are close to background levels.

Table 2.2-2
Comparison of Measured Incremental Gamma and Neutron Radiation Dose Rates Near the Test Cylinder

Distance (m)	Gamma Dose Rate (µSv/h)	Neutron Dose Rate (µSv/h)	Ratio of Measured Gamma to Neutron Dose Rate [a]
From Side of Cylinder			
contact	10.365	1.632	6
1	3.865	0.519	7
5	0.865	0.048	18
10	0.115	0.007	16
From End of Cylinder			
contact	10.865	1.483	7
1	2.365	0.270	9
5	0.085	0.024	3
10	<0.01	<0.01	n/a

Note: The number of significant figures shown should not be considered as indicative of the precision of the values.

[a] Ratios at contact and 1 m distances have estimated uncertainties of 20% or less. Ratios at greater distances have larger uncertainties and are considered unreliable.

3.0 MODELLING OF NEUTRON DOSE RATES

3.1 OTHER MODELLING OF UF_6 CYLINDERS

Modelling of neutron and gamma radiation doses at a UF_6 cylinder storage area has been undertaken by the NISYS Corporation (2000). The purpose of the study was to evaluate the gamma dose equivalent to neutron dose equivalent ratios associated with the storage of 14-ton (12,700 kg) UF_6 cylinders in storage yards. MCNP was used to model the gamma and neutron doses at various distances (1 m to 300 m) from an array of cylinders. Natural uranium, enriched uranium and depleted uranium were evaluated.

The specific storage arrangement modelled was an array of two rows of cylinders, stacked two cylinders high and 25 cylinders long. [At the PHCF, cylinders are currently placed on the ground rather than being stacked.] The 14-ton cylinders were assumed to be "full" of UF_6 (i.e., at the maximum weight limit). The space inside the cylinder not occupied by UF_6 was modelled as a void. The cylinders were modelled above a 6-inch (15 cm) concrete pad. Skyshine effects were assumed to be adequately addressed by modelling the air above the cylinders to a height of 240 meters. A decay time of 50 years was conservatively chosen for the source terms to achieve maximum activity levels and maximum dose from the spectra. (A calculation of the activity levels showed that saturated activity is achieved in the period from 5 to 10 years, with a slight increasing activity thereafter.) Thick-walled cylinders (1.59 cm thick, similar to Cameco cylinders) were modelled by NISYS because they yielded the most conservative results, even though the majority of the cylinders assessed by NISYS were thin-walled (i.e., thin-walled cylinders have a higher gamma/neutron ratio).

The NISYS study notes that the formation of solid UF_6 in a cylinder is annular in nature, with a thicker region at the bottom of the cylinder. Two cases were evaluated to bound the effect of UF_6 distribution: (1) a perfect annulus of UF_6 inside the cylinder, and (2) a "slumped" geometry with all of the UF_6 modelled in the bottom of the cylinder. The "slumped" geometry resulting in the greatest gamma-to-neutron ratios was used in their final analysis

The calculations showed that the gamma-to-neutron ratios varied with distance and enrichment. The ratio for any given enrichment initially decreased with distance from the cylinder array. The ratio then appeared to increase after about 200 meters, but the furthest two distances modelled (200 m and 300 m) were reported to carry lower statistical confidence than the other distances. Humidity had no significant effect on the ratios.

The highest neutron dose rate (at 1 m distance) for natural uranium was approximately 1.5 µSv/h. At 50 m and 100 m, the neutron dose rates were approximately 0.0253 µSv/h and 0.007 µSv/h, respectively. The gamma-to-neutron dose ratios were fitted to an exponential relationship that asymptotically approached a constant value. For natural uranium, the relationship was found to be:

$$\text{gamma-to-neutron dose ratio} = 4.5659\ D^{-0.1}$$

where D = distance (m) from cylinder array.

The highest ratio for natural uranium (at 1 m) was 4.50. The bounding (lowest) ratio for natural uranium (at about 200 m) was conservatively determined to be 2.50.

This relationship could underestimate the gamma-to-neutron ratios at the Cameco PHCF due to: the conservatism in the calculations (i.e. the model with lowest gamma-to-neutron ratios was selected); the use of a stacked (two row) array of cylinders which does not occur at the PHCF; and the fact that other non-neutron emitting gamma radiation sources at the Cameco facility would be expected to result in higher gamma-to-neutron ratios.

3.2 MONTE CARLO N-PARTICLE MODELLING RESULTS FOR UF_6 CYLINDER ARRAYS

The assumptions of SENES Monte Carlo N-Particle Model (MCNP) calculations of neutron dose rate at different distances from the side of 1, 3, 5 and 10 UF_6 cylinder arrays is described in Appendix B.

The effective neutron dose rates from the side of the cylinders as a function of distance are shown Table 3.2-1. Figure 3.2-1 shows the neutron dose rates plotted as a function of distance (contact (modelled as 0.02 m), 1, 5, 10, 20, 50, and 100 m) for the 1, 3, 5 and 10 cylinder configurations.

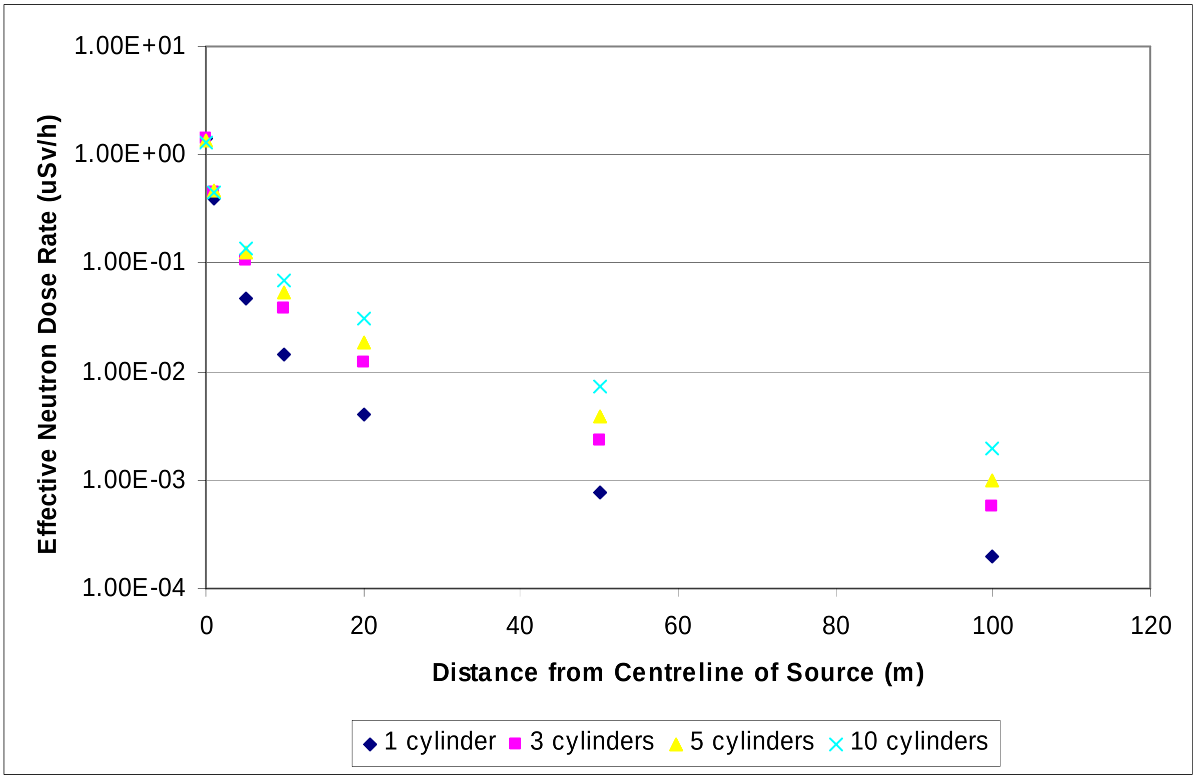
Table 3.2-1
Neutron Dose Rates from Side of the UF_6 Cylinders (1, 3, 5, 10 Cylinders)

Distance from Centreline of Source (m)	**Effective Dose Rate (µSv/h)**			
	Number of Cylinders			
	1 Cylinder	**3 Cylinders**	**5 Cylinders**	**10 Cylinders**
contact	1.42E+00	1.45E+00	1.38E+00	1.32E+00
1	4.01E-01	4.52E-01	4.63E-01	4.42E-01
5	4.77E-02	1.04E-01	1.24E-01	1.37E-01
10	1.42E-02	3.84E-02	5.39E-02	7.03E-02
20	4.05E-03	1.20E-02	1.88E-02	3.07E-02
50	7.81E-04	2.33E-03	3.85E-03	7.36E-03
100	1.98E-04	5.81E-04	9.81E-04	1.92E-03

<u>Note:</u> The number of significant figures shown should not be taken as indicative of the precision of the values.

As expected, the neutron dose rates decreased with distance from the UF_6 cylinders. Measured net neutron dose rates (above background) at the PHCF (see Table 2.2-2) were about 1.6, 0.52, 0.048 and 0.007 µSv/h at contact, 1 m, 5 m and 10 m, respectively from the side of a full UF_6 cylinder. The predicted dose rates for one cylinder shown in Table 3.2-1 compare well with the measured values. The predicted values are also comparable to values predicted by NISYS (2000), especially at the closer distances.

Figure 3.2-1
Neutron Dose Rate as a Function of Distance



Overall, the findings of this MCNP evaluation were:

- The neutron dose rates from the side of a full UF_6 cylinder were calculated with MCNP to range from about 1.4 to 0.014 µSv/h from contact to 10 m distances, respectively. These compared well to measured values and to values predicted by NISYS (2000).
- The neutron dose rate at 100 m from an array of 10 cylinders (i.e., potential off-site location) was less than 0.002 µSv/h, and more than a factor of two smaller than an estimated background dose rate of about 0.004 µSv/h.

4.0 OVERVIEW

4.1 MEASUREMENTS

Previous attempts to measure neutron dose rates around the PHCF using long-term passive detectors showed dose rates generally below limits of detection. Recent surveys using CR-39 monitors from Landauer also show dose rates below Landauer's detection limits. Nonetheless, comparison of averages can be informative. UF_6 cylinders are considered to be the main source of neutrons at the PHCF. The mean values were calculated for the fenceline TLD locations with all other TLDs grouped. The "PHCF" mean was 0.008 µSv/h higher than the mean for the "Other" TLDs and this approached statistical significance. Thus, 0.008 µSv/h represents an estimate of the average incremental (above background) neutron dose rate at the PHCF fenceline. This is consistent with an average incremental (above background) dose rate of 0.007 µSv/h measured at three TLD stations using the portable neutron meter (see Table 2.2-1).

The average incremental (above a background of 0.080 µSv/h) gamma radiation rate at the fenceline TLDs was 0.041 µSv/h for the same period of CR-39 deployment. This is about 5 times higher than the nominal estimate of an average incremental neutron dose rate of about 0.008 µSv/h estimated from the CR-39.

The portable neutron meter was capable of measuring background neutron dose rates in scaler mode with extended monitoring time up to about 6 hours. Measurements were also collected at some TLD locations and surrounding a single cylinder containing UF_6.

4.2 MODELLING

Table 4.2-1 shows a comparison between the MCNP predictions and the measured neutron dose rate from the side of the cylinder. Measurements at 10 m were influenced by the uncertainty of the neutron measurement for short counting times. The dose rates show a similar pattern with measured values close to the MCNP modelled values. The comparison is quite good given the various sources of uncertainty. Small differences may be due to incomplete filling of the cylinder, variations in density and the duration of counting and measurement precision of the portable neutron meter. The data verify that the MCNP modelling for this scenario was reasonable and that based on either measurement or model, incremental neutron dose rates decrease to background levels at a distance of about 10 to 20 m from a single cylinder.

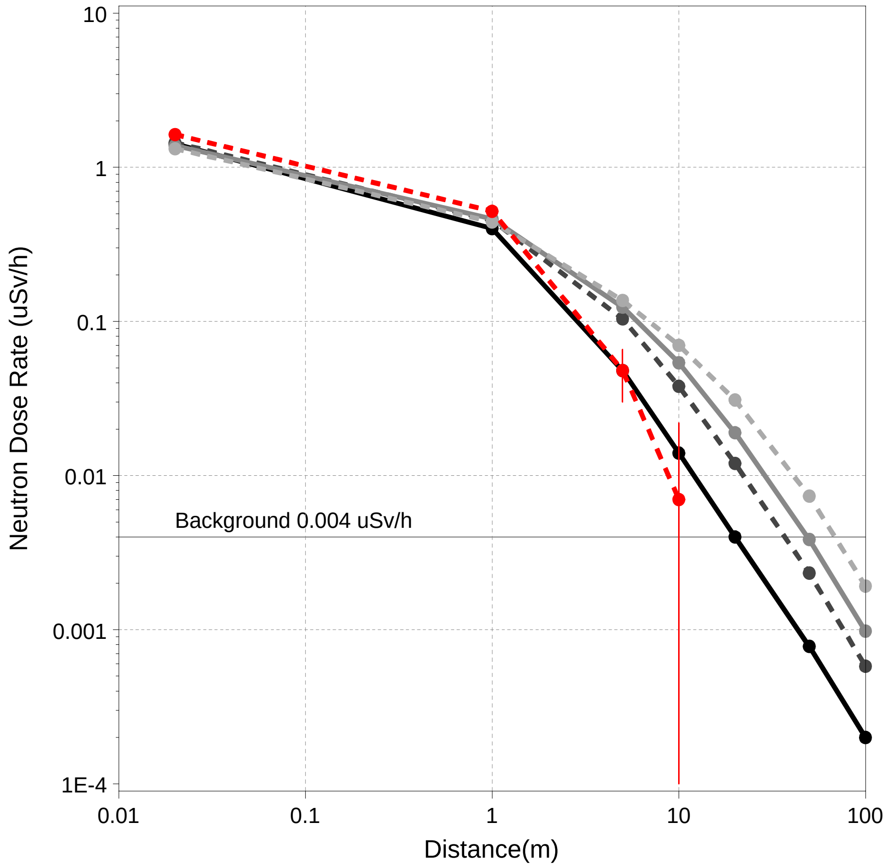
Table 4.2-1
Comparison with MCNP Predictions

Distance (m)	MCNP Modelled Effective Dose (μSv/h)	Measured Neutron Effective Dose (μSv/h)	Measurement Uncertainty (μSv/h)
contact	1.418	1.632	0.097
1	0.401	0.519	0.041
5	0.048	0.048	0.018
10	0.014	0.007	0.015
20	0.004	-	-
50	0.001	-	-
100	0.0002	-	-

Figure 4.2-1 shows the pattern of MCNP predictions, along with the measured values for one cylinder. The receptor is out from the centre of the side of one, three, five or ten cylinders oriented in a line of end to end. The plot indicates that dose near the cylinders (within 1 metre) is relatively independent of the number of cylinders modelled. At larger distances (about 10 to 20 m or more), the dose is proportional to the number of cylinders (i.e. the dose from ten cylinders is about ten times as high as that from a single cylinder at these distances). At these distances, the dose drops off as $1/r^2$ with distance since the source begins to act as a point source relative to dose rate at those distances. Nearer the source, the drop in dose rate with distance is slower and similar to the relationship with distance from a line source. Overall, the data provided in Figure 4.2-1 suggest that, based on modelling, the incremental neutron dose rates approach background levels within a distance of about 70 m from an array of 10 cylinders.

The measured values using the portable neutron meter shown on Figure 4.2-1 are for a single cylinder measured on site on 30 January 2009. The confidence intervals for the measurements are shown. Agreement between the MCNP modelled dose rates and the measurements is present and this indicates a consistency in this study's modelling and the measurements conducted for a single cylinder.

Figure 4.2-1
Dependency of MCNP Predicted and Measured Neutron Dose Rates on Distance from the UF_6 Cylinders



4.3 DOSE IMPLICATIONS

Members of the Public

Based on raw data from the recent measurements with Landauer CR-39 monitors, a neutron dose rate of about 0.008 µSv/h represents an estimate of the average incremental (above background) neutron dose rate at the fenceline (TLD stations). Neutron dose rates for more distant receptors would be lower. On this basis and assuming, as was done for Cameco's DRL (SENES 2008), that a fenceline walker spends one hour each day (i.e. 365 h per year) at an incremental neutron dose rate of 0.008 µSv/h, he or she would receive an incremental dose from neutrons of about 3 µSv/y for the current cylinder configuration. This dose represents 1% of Cameco's current licence limit for public receptors of 0.3 mSv/y (300 µSv/y). Based on an incremental gamma radiation dose of 19 µSv/y to the average fenceline walker (from SENES 2008), this suggests a gamma to neutron dose ratio of about six to one at the fenceline.

Based also on the raw Landauer data, a nominal ratio of about 5 to 1 for gamma dose rate to neutron dose rate is suggested at fenceline locations. Recent measurements with a portable neutron meter suggest a similar ratio of about 5 to 10 between gamma and neutron dose rates at nearby locations. Thus, a nominal gamma to neutron ratio at the fenceline of 5 seems appropriate for future dose assessment assuming yard storage of cylinders. As indicated previously, for the sources at the PHCF, this assumption may be somewhat conservative.

Overall, the findings of this MCNP evaluation were:

- The neutron dose rates from the side of a full UF_6 cylinder were calculated with MCNP to range from about 1.4 to 0.014 µSv/h from contact to 10 m distances, respectively. These compared well to measured values and to values predicted by NISYS (2000).
- The neutron dose rate at 100 m from an array of 10 cylinders (i.e., potential off-site location) was less than 0.002 µSv/h, and more than a factor of two smaller than an estimated background dose rate of about 0.004 µSv/h.

Both the measurements and the modelling performed to date indicate that the incremental neutron dose rates decrease with distance and approach background levels within 10 to 20 m of a single UF_6 cylinder and around 70 m for an array of cylinders. Receptors located at distances greater than these listed above would receive an incremental neutron dose no greater than the background level.

Workers

Workers in close proximity to UF_6 cylinders will also receive a small neutron dose. Using neutron dose rates estimated from Table 4.2-1 and task-time analyses provided by Cameco, annual worker doses can be estimated as shown in Table 4.3-1. Under these conditions, the maximum estimated annual doses are less than 1 mSv/y and hence no routine neutron monitoring is required.

Table 4.3-1
Estimate of Neutron Doses to Workers

Activity	Distance	Dose rate (μSv/h)	Time/cylinder	Cylinders/y	Hours/y	Work share	Hours/person/y	Annual Dose (mSv/y)
UF_6 Plant - cylinder filling, weighing, movement	1-3 m	0.14	2-3 h/cylinder whole process 0.5 h/cylinder moving	12,500 t licence limit = 1,010 cylinders (12 t) 1,250 cylinders (10 t)	~3,300 h	4 shifts	850	0.12
Materials Handling - cylinders movement	>3 m	< 0.1	0.2 h/cylinder move includes out of B#50 (1), cold evac (2), wash (2), inspection (2) and loading (1)	1,100 cylinders moved 6 times (75% up to 8 times) = 8,250 moves/year	~1,650 h	2 shifts	825	< 0.08
Materials Handling - inspection, cold pressure test, loading	<1 m	0.75	0.2 h/cylinder inspection 0.1 h/cylinder cold pressure test 0.1 h/cylinder loading	836 cylinders shipped overseas in 2006	~330 h	2 shifts	165	0.12
Materials Handling - cylinder washing	<1 m	0.75	0.5 h/cylinder	836 cylinders shipped total in 2006	~420 h	2 shifts	210	0.16
Materials Handling - applying thermal protection	<1 m	0.75	0.5 h/cylinder covers	664 cylinders shipped overseas in 2006	~330 h	2 shifts	165	0.12
R&E Technician - swipe collection	<1 m	0.75	0.1 h/cylinder	836 cylinders shipped total in 2006	~84 h	8 techs	40	0.03

a. Worker task-time information provided by Cameco (February 2008).
b. Dose rates estimated from Table 4.2-1 for single cylinder.

Overall Conclusions

Overall, the combined information from long-term monitoring using CR-39 monitors, survey data with a portable neutron meter and modelling show remarkable coherence and suggest that neutron dose rates are reasonably predictable. Moreover, the results of the current modelling compare well to that reported by others. A ratio of 5 between incremental gamma radiation and neutron dose rates could conservatively be used to predict neutron dose rates for receptors at or near the fenceline. The estimated neutron dose for the fenceline walker represents 1% Cameco's current licence limit for public receptors. Receptors located further away from the fenceline are exposed to much lower dose rates and at distances beyond 70 m from the cylinders, incremental neutron dose rates cannot be distinguished from the background. The maximum estimated neutron doses to workers are less than 1 mSv/y and hence no routine neutron monitoring is required.

Furthermore, while measurable neutron dose rates from UF_6 cylinders are low and do not require routine separate measurements, Cameco has continued to evaluate potential neutron dose rates associated with UF_6 activities.

5.0 REFERENCES

Boneh, Y., S. Goshen, Z. Karpas, O. Shahal and A. Wolf 1984. *Neutron yields from uranium-fluorine compounds.* Nuclear Science and Engineering: 86, 106-115.

Carisse, H. 2007. *Environmental and Public Neutron Exposure from UF_6.* Presented at the OECD/NEA/WGFCS Workshop October 16-18, 2007 Wilmington NC Fuel Cycle Safety – Past, Present and Future.

Hinton, A. 2005. *Packaging and Transport Licensing Division Type II Inspection: Neutron Dose Rate Surveys of Uranium Hexafluoride (Type H) Packages at Cameco – Port Hope, ON.* Memorandum to M. O'Brien from A. Hinton, CNSC Inspector, CNSC File 30-1-15/ 30-6-3/ 36-3-1-0, 24 June.

International Commission on Radiological Protection (ICRP) 1996. *Conversion Coefficients for use in Radiological Protection against External Radiation.* ICRP Publication 74 Volume 26 No. 3/4. Pergamon Press, Oxford, U.K.

Los Alamos National Laboratory (LANL) 2005a. *MCNP5 Version 1.40*.

Los Alamos National Laboratory (LANL) 2005b. MCNP- *A General Monte Carlo N-Particle Transport Code, Version 5 Volume I: Overview and Theory.*

Nagaoka K, I. Hiraide, K. Sato and T. Nakamura 2009. *Nationwide Measurements of Cosmic-Ray Dose Rates throughout Japan*. Radiat Prot Dosimetry. 132(4):365-374.

NISYS Corporation 2000. *Determination of Neutron to Gamma Dose Ratios for the UF_6 Cylinder Yards*. Report for Bechtel Corporation, Calculation Number: 1073-018-C001, 25 August.

Pereira, K. 2005a. *Reply to your letter of April 8, 2005.* CNSC e-mail to Mr. Miller and Mrs. More from K. Pereira, Executive Vice-President, Operations, CNSC File 37-26-0-0, 3 May.

Pereira, K. 2005b. *Reply to your letter dated June 1, 2005 concerning Cameco's Fourth Quarterly Report – Port Hope.* CNSC e-mail to Mrs. Faye More from K. Pereira, Executive Vice-President, Operations, 21 June.

Rasolonjatovo DA, H. Suzuki, N. Hirabayashi, T. Nunomiya, T. Nakamura and N. Nakao 2002. *Measurement for the dose-rates of the cosmic-ray components on the ground.* J Radiat Res (Tokyo). 43 Suppl:S27-33, December.

SENES Consultants Limited (SENES) 2008. *Derived Release Limits for Cameco's Port Hope Facility*. Prepared for Cameco Corporation, October.

Takala, J. 2001. *Neutron Measurements at Port Hope.* Interoffice (Cameco) memo to H. Carisse, February 14.

United States Department of Energy (U.S. DOE) 2004. *Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities.* DOE-STD-1136-2004. December.

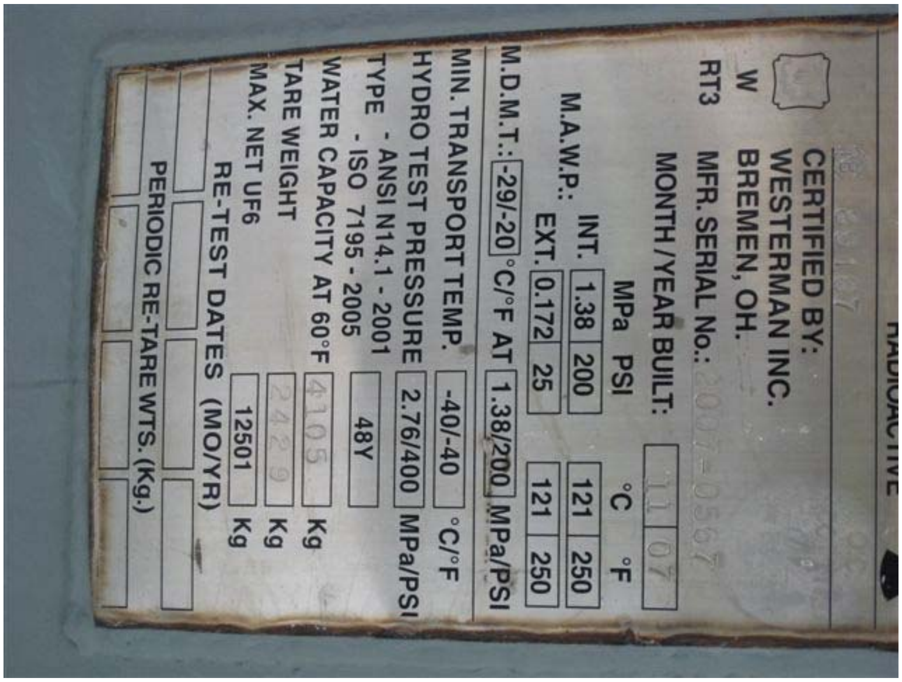
United States Department of Energy (U.S. DOE) 2006. *The UF_6 Manual, Good Handling Practices for Uranium Hexaflouride*, USEC-651, Revision 9, July 2006, USEC Inc.

United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) 2000. *Sources and Effects of Ionizing Radiation- Vol. 1 Sources*. Report to the General Assembly, with Scientific Annexes, United Nations, New York.

APPENDIX A

PHOTOGRAPHS OF JANUARY 2009 NEUTRON MONITORING

APPENDIX A – PHOTOGRAPHS OF JANUARY 2009 NEUTRON MONITORING



RADIOACTIVE

CERTIFIED BY:
WESTERMAN INC.
BREMEN, OH.

W

RT3

MFR. SERIAL No.: 2007-0567

MONTH/YEAR BUILT: 11 07

		MPa	PSI	°C	°F
M.A.W.P.:	INT.	1.38	200	121	250
	EXT.	0.172	25	121	250

M.D.M.T.: -29/-20 °C/°F AT 1.38/200 MPa/PSI

MIN. TRANSPORT TEMP. -40/-40 °C/°F

HYDRO TEST PRESSURE 2.76/400 MPa/PSI

TYPE - ANSI N14.1 - 2001 48Y
- ISO 7195 - 2005

WATER CAPACITY AT 60°F 4105 Kg

TARE WEIGHT 2429 Kg

MAX. NET UF6 12501 Kg

RE-TEST DATES (MO/YR)

PERIODIC RE-TARE WTS. (Kg.)

Photograph No. 1: Identification placard found on opening end of test cylinder used in testing for neutrons. (Cylinder serial No. 2007-0567)

Photograph No. 2: Identification marking painted on side of test cylinder used in testing for neutrons. (Cylinder No. 567)

Photograph No. 3: Meridian Portable Neutron Monitor (Model 5085) in direct contact with side of cylinder No. 567 testing for neutrons.

Photograph No. 4: Set up of the Meridian Portable Neutron Monitor 1m from side of test cylinder No. 567.

Photograph No. 5: Set up of the Meridian Portable Neutron Monitor 5m from side of test cylinder No. 567.

Photograph No. 6: Set up of the Meridian Portable Neutron Monitor 10m from side of test cylinder No. 567.

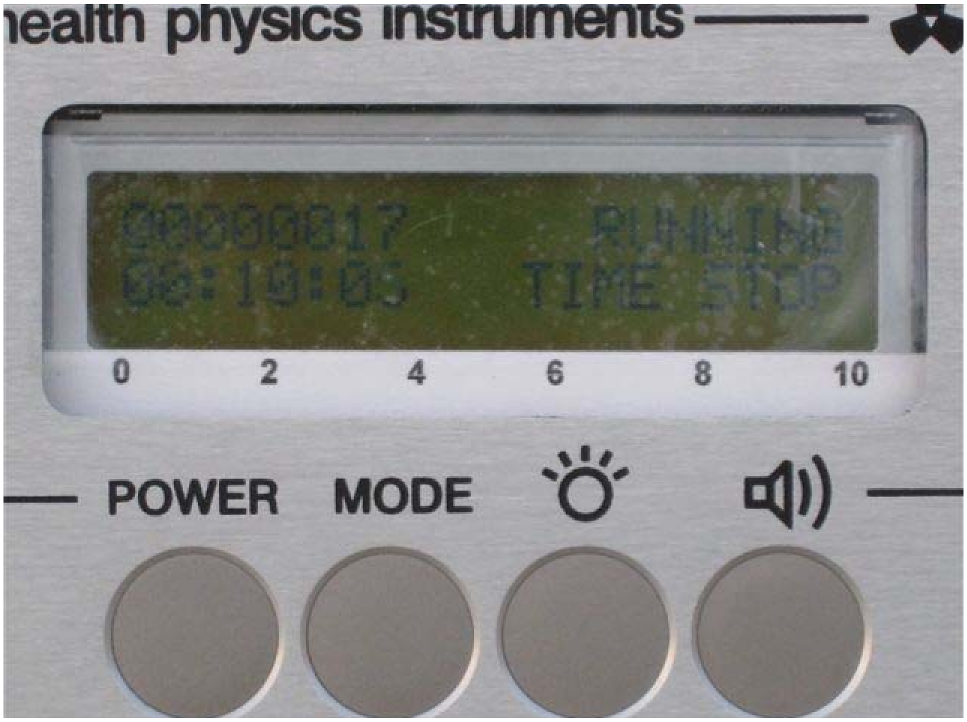
Photograph No. 7: Meridian Portable Neutron Monitor in direct contact with opening end of cylinder No. 567 testing for neutrons.

Photograph No. 8: Set up of the Meridian Portable Neutron Monitor 1m from end of test cylinder No. 567. Note storage of cylinders in background. Closest cylinder, which is ready for shipping, is ~30m.

Photograph No. 9: Set up of the Meridian Portable Neutron Monitor 5m from end of test cylinder No. 567.

Photograph No. 10: Set up of the Meridian Portable Neutron Monitor 10m from end of test cylinder No. 567.

Photograph No. 11: Performing background test 5m from where test cylinder was during test. Testing former side of test cylinder No. 567. Note stored cylinders in upper right.



Photograph No. 12: Looking at screened display of Meridian Portable Neutron Monitor during background test.

Photograph No. 13: Set up at monitoring station No. 1 located north east of plant. Set up 1.8m west of actual location of station.

Photograph No. 14: Looking south west from test location as to what would be tested during test at monitoring station No. 1.

Photograph No. 15: Set up at monitoring station No. 31 located north of plant. Set up 4.6m south east of actual station location.

Photograph No. 16: Looking south east from test location as to what would be tested during test at monitoring station No. 31.

Photograph No. 17: Set up at monitoring station No. 8 located south of plant. Set up 1.4m north of actual station location.

Photograph No. 18: Looking north from test location as to what would be tested during test at monitoring station No. 8. Note cylinder stored near this location.

APPENDIX B

SUPPORTING TECHNICAL INFORMATION

APPENDIX B – SUPPORTING TECHNICAL INFORMATION

B.1 PREVIOUS MEASUREMENTS

Cameco has undertaken a number of previous measurements of neutrons at the PHCF. These have been presented by Carisse (2007). The presentation slides from Carisse (2007) are provided in Appendix C with the results summarized below.

B.1.1 Portable Neutron Survey Meter

Cameco undertook a neutron survey at the PHCF in December 2000 using an ADM 300C Gamma/Neutron survey meter borrowed from Ontario Power Generation (OPG). The meter has an energy-compensated response to neutrons that measures in units of µSv/h. Because of the low dose rates, the meter was run in integration mode, rather than direct reading mode. Both work-occupied locations and in-contact readings with UF_6 cylinders were measured. In addition, some measurements at fenceline TLD monitoring locations were also obtained. (According to OPG and Cameco, some difficulties with electronic noise could have biased some of the measurements high.)

Table B.1.1-1
Summary of Neutron Dose Rate Measurements (December 2000)

Location Type	No.	Neutron Dose Rate (µSv/h)		
		Average	Minimum [a]	Maximum
Engineering	7	1.51	0.05	3.00
Workplace	13	0.14	non-detect	0.56
Fenceline	8	0.03	non-detect	0.05

[a] Non-detect values shown as 0 in Carisse (2007).

Direct contact reading with the cylinders ranged from 1.5 to 3.0 µSv/h. (All measurements included any background neutron dose rates in the area.) The readings decreased quickly with distance. Two metres (2 m) from a row of 15 cylinders, the dose rate was 0.4 µSv/h. In typical workplaces within the fenced area, the readings ranged from non-detectable to about 0.6 µSv/h, with an average of 0.14 µSv/h. The measurements showed that doses to workers from neutrons should be less than 1 mSv/y (Carisse 2007). Fenceline measurements ranged from non-detect to 0.05 µSv/h, with an average of 0.03 µSv/h.

B.1.2 Neutron Dosimeters

Neutron dosimeters from Landauer Inc. (CR-39 monitors) were deployed along the PHCF fenceline in April 2005 for a 30-day period. All dosimeter readings were below the Landauer detection limit of 100 µSv over the deployment period (i.e., dose rates were < 0.14 µSv/h). A second trial deployment of CR-39's was carried out from December 2006 to December 2007. Although most of the detectors were not analyzed due to damage, the remaining detectors showed low levels similar to the earlier deployment.

B.1.3 Bubble Detectors

In addition to studies with the CR-39 monitors, short-term measurements (<1 day per deployment) were also made using bubble detectors manufactured by Bubble Technologies Industries (BTI). These detectors with a reported detection limit of 0.01 µSv were deployed at four fenceline locations in 2001. The results were as follows:

Table B.1.3-1
Bubble Detector Neutron Measurements at the PHCF (2001)

Location	Number of Trials	Highest Results (µSv/h)
South of UF6 plant	5	0.062
West of UF6 plant	5	0.080
North of UF6 plant	5	0.230
Critical receptor at Mill Street	5	0.030

In 2005-2006, extensive measurements with bubble detectors were carried out to look at neutron doses at various locations on and offsite. The average neutron dose rate inside the plant was less than 0.3 µSv/h. The maximum dose rate along the fenceline was reported as 0.28 µSv/h. Based on this limited experience, bubble detectors are quite sensitive to environmental conditions (e.g. very limited temperature range for operation) and of limited use for ambient environmental surveys.

B.2 RECENT MEASUREMENTS

B.2.1 CR-39 Passive Monitors

Neutrons were measured for one quarter of the year using passive monitors (CR-39) located at or near TLD locations and other locations with some outside of Port Hope.

B.2.1.1 Calculation of the Point Estimate of Dose

The transit control was reported to have a dose of 0.06 mSv over the transit and deployment period. This includes dose collected during deployment as well as dose during transit. For each location, the transit control dose was subtracted from the location dose and the dose rate was calculated based on 2208 hours of exposure in the field. The net dose values could be interpreted as above-background levels.

Table B.2.1-1 shows the calculated net neutron dose rate for each location. Some values are negative; however, this could be expected for measurements calculated below the minimum detection limit (MDL).

Table B.2.1-1
Calculation of Neutron Dose Rate using Landauer CR-39 Monitors
(Sept 1 2008 to Dec 3 2008)

Grouping	Location	Dose for 2208 hours (mSv)	Net Dose Rate (µSv/h)
	Transit Control	0.06	
PHCF	Enriched Plant	0.04	-0.01
PHCF	Warehouse	0.10	0.02
PHCF	UF6 East Side	0.05	0.00
PHCF	UO2 East Side	0.05	0.00
PHCF	South Fence	0.02	-0.02
PHCF	Zirc Plant West	0.12	0.03
PHCF	Metals Plant	0.14	0.04
PHCF	UF6 South Side	0.11	0.02
PHCF	Oliver's	0.05	0.00
PHCF	UF6 North Side	0.03	-0.01
PHCF	Crane East Fence North	0.10	0.02
PHCF	Crane East Fence South	0.12	0.03
PHCF	Crane Opposite 119 Mill St	0.02	-0.02
PHCF	Crane Opposite 125 Mill St	0.00	-0.03
PHCF	Crane West Fence North	0.02	-0.02
PHCF	Crane West Fence South	0.08	0.01
PHCF	North-West of building 50	0.16	0.05
PHCF	Building # 26	0.02	-0.02
	Average		0.0038
Vicinity	125 Mill St	0.06	0.00
Port Granby	Granby East	0.02	-0.02
Port Granby	Granby West	0.05	0.00
Welcome	Welcome South	0.11	0.02
Welcome	Welcome North-East Gate	0.04	-0.01
Welcome	Bud's	0	-0.03
Background	Cobourg	0.05	0.00
Background	Oshawa	0.02	-0.02
Background	Mississauga	0.05	0.00
Background	Peterborough	0.14	0.04
Background	Port Hope	0.02	-0.02
	Average		-0.0041

B.2.1.2 Interpretation

The calculated neutron doses are highly uncertain; therefore, consideration of individual measurements is difficult. However, comparison of averages can be informative as random variations in the measurement process at below detection limit levels are "averaged" out. The PHCF is expected to have the highest neutron dose rates due to the presence of the UF_6 cylinders. The mean values were calculated for the fenceline TLD locations with all other TLDs grouped. The "PHCF" mean was about 0.008 µSv/h higher than the mean for the "Other" TLDs and this approached statistical significance. The average incremental background gamma radiation dose rate (above a background of 0.08 µSv/h) at the fenceline TLDs was 0.041 µSv/h for the same period of CR-39 deployment. This is about 5 times higher than the nominal estimate of an average incremental neutron level of 0.008 µSv/h estimated from the CR-39.

B.2.2 Neutron Survey Meter

A list of neutron survey meters which were potentially suitable for measuring neutron dose rates at the Cameco's PHCF was compiled by SENES. The criteria for suitability were based on sensitivity of the detector, operational temperature range, gamma rejection, availability of a scaler or integrating option, and portability.

The neutron monitors identified as potentially suitable are provided in Table B.2.2-1.

Table B.2.2-1
Potentially Suitable Neutron Monitors for Environmental Monitoring

Company	Model	Sensitivity	Optimum Temperature Range	Gamma Rejection	Weight
Ludlum Measurement Inc.	Model 42-41 PRESCILA Probe + Model 2350 Neutron Survey Meter	35 cpm/µSv/h	M4241: -10° to 50°C M 2363: -10° to 50°C	0.5 cpm/µSv/h	~8 lbs (3.6 kg - probe and survey meter)
	Model 12-4	12 cpm/µSv/h	-20°C to 50°C (may be certified for operation from -40°C to 65°C)	Less than 10 cpm through 10 R/h	21 lbs (9.5 kg)
Far West Technology Inc.	Meridian Neutron Monitor, Model 5085	47.2 cpm/µSv/h	-20°C to 50°C	No response up to 500 R/h Cs-137	20.5 lbs (9.3 kg)
Bot Engineering Ltd	Neutron Probe RM-SCLN1 [a]	6 cpm/µSv/h	-15°C to 50°C	-	13.2 lbs (6 kg)
Canberra	ADM300 with NP100H	10 cpm/µSv/h	-30°C to 50°C (ADM300) -10°C to 50°C (NP100H)	-	2.6 lbs (1.17 kg) (ADM300) 22 lb (10 kg) (NP100H)

[a] Availability of integration mode not confirmed.

A key criterion was the sensitivity of the detector. In order to compare all meters, the length of time required to get 50 counts[2] to detect a background dose rate of the order of 0.01 µSv/h (UNSCEAR 2000, p.97, para. 25, Figure II at 43 degrees latitude). The results of this evaluation are summarized in Table B.2.2-2.

Table B.2.2-2
Count Rate and Time for Potentially Suitable Neutron Monitors

Company	**Model**	**Sensitivity**	**Count Rate to Detect 0.01 µSv/h**	**Time to Get 50 Counts (minutes)**
Ludlum Measurement Inc.	Model 42-41 PRESCILA Probe + Model 2350 Neutron Survey Meter	35 cpm/µSv/h	0.35 cpm	143
	Model 12-4	12 cpm/µSv/h	0.12 cpm	417
Far West Technology Inc.	Meridian Neutron Monitor, Model 5085	47.2 cpm/µSv/h	0.472 cpm	105
Bot Engineering Ltd	Neutron Probe RM-SCLN1 [a]	6 cpm/µSv/h	0.06 cpm	833
Canberra	ADM300 with NP100H	10 cpm/µSv/h	0.10 cpm	500

[a] Availability of integration mode not confirmed.

Based in part on the results from Table B.2.2-2, SENES recommended the Meridian Neutron Monitor, Model 5085 from Far West Technology Inc. because it required the shortest count time to get 50 counts at background locations (assuming an approximate background neutron dose rate of 0.01 µSv/h based on generic UNSCEAR (2000) values), thus allowing for multiple measurements at a background location in a normal working day of 8 hours. Subsequently, the background neutron dose rate in Port Hope and area was found to be about 0.004 µSv/h (see Section 3.3.1) and therefore, the actual count time to accurately measure background was about 2.5 times higher than that estimated in Table B.2.2-2.

B.3 MONTE CARLO N-PARTICLE MODELLING AND CALCULATIONS FOR UF_6 CYLINDER ARRAYS

This section provides the assumptions and results of SENES Monte Carlo N-Particle Model (MCNP) calculations of neutron dose rates at different distances from the side of 1, 3, 5 and 10 UF_6 cylinder arrays. The UF_6 cylinders were assumed to be 15.24 cm (6") apart.

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[2] The one sigma relative uncertainty associated with 50 counts is approximately 14% (i.e., square root of the number of counts divided by the counts.)

B.3.1 Input Assumptions

The MCNP5 code (version 1.40, LANL 2005a) requires the user to create an input file which is read by MCNP. This file contains information such as:

1. Geometry specification;
2. Description of materials;
3. Characteristics of the particles (neutron, photon, or electron);
4. Receptor locations;
5. Type of answers or tallies desired; and
6. Any variance reduction techniques used to improve efficiency (i.e. minimizing computing time need to obtain a tally estimate with acceptable error).

Geometry Specification

The dimensions of one Model 48Y UF_6 cylinder are as follows (U.S. DOE 2006):

- The outside dimensions of the cylinders are 48.5" (123.019 cm) which includes a thickness of 5/8" (1.5875 cm) steel - the inner diameter is 47.25" (120.015 cm);
- Length of the cylinder body (not including the skirt) is 145.5" (369.57 cm) (not including the 5/8" (1.5875 cm of steel thickness on each end); and
- The uranium payload of the Model 48Y is reported 12,501 kg.

The cylinders are not true cylinders as the ends are rounded. In addition, the cylinders were assumed to be stored on their sides and never stacked.

Description of Materials

For the MCNP calculations, the UF_6 was assumed to fill the cylinders uniformly. The density of UF_6 is a mass-based density which was calculated by dividing the net weight of the Model 48Y cylinder by the volume of the cylinder (assuming that the cylinder is full) as shown below.

Equation 1:

$$\rho_{source}\left(\frac{g}{cm^3}\right) = \frac{NetWeight\,(kg) \times 1000\,\frac{g}{kg}}{\pi \times \left(\frac{diameter(cm)}{2}\right)^2 \times L_{cylinder}\,(cm)}$$

where

Net Weight	12,501 kg
Diameter	120.015 cm
$L_{cylinder}$	369.57 cm (Model 48Y)

Using Equation 1, the (effective) density of the UF_6 assumed to be uniformly distributed in the Model 48Y cylinder was estimated as 2.99 g/cm^3.

Receptor Locations

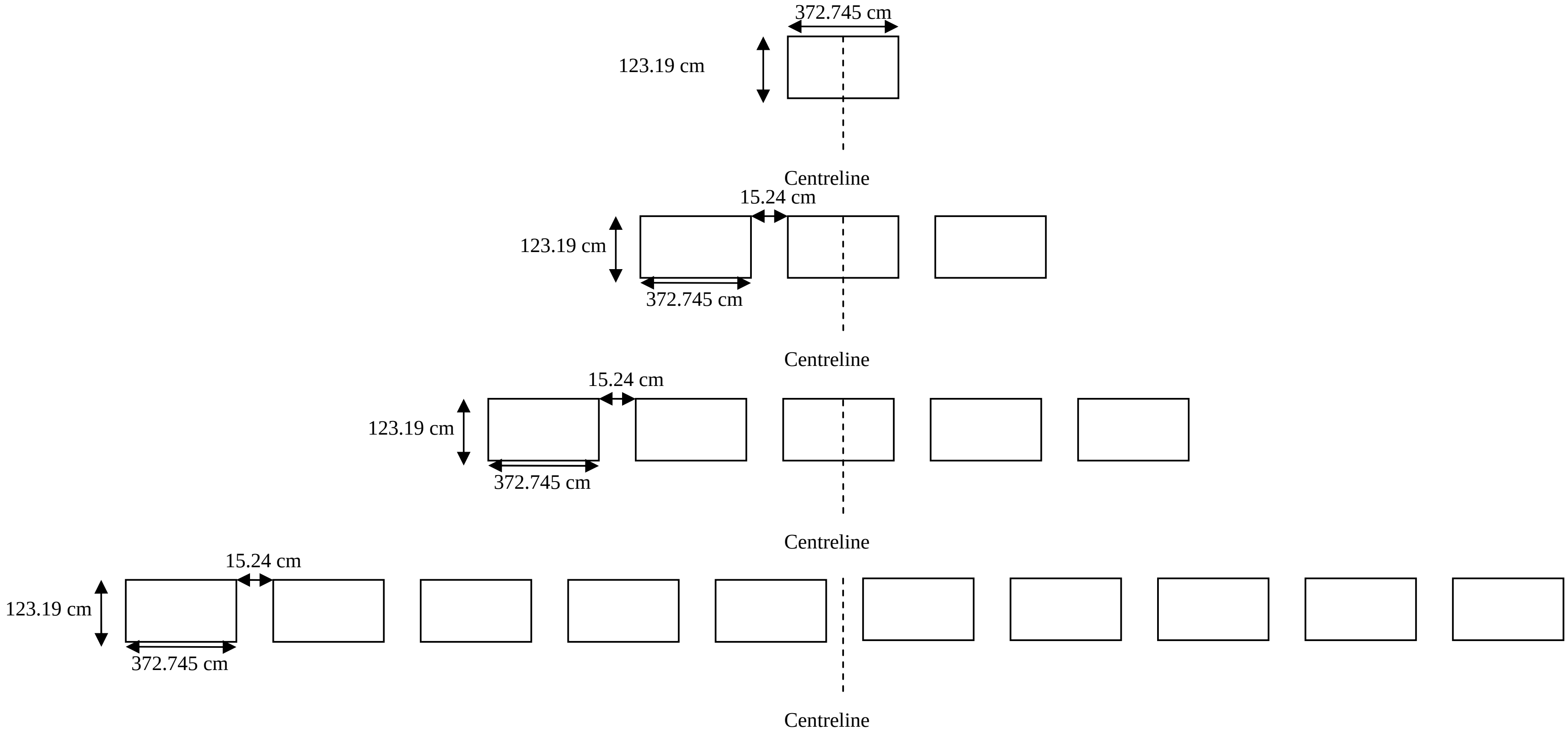
The effective neutron dose rates were calculated for 1, 3, 5 and 10 UF_6 cylinder arrays with the receptor locations at contact (modelled as 0.02 m distance) and at 1, 5, 10, 20, 50 and 100 m from the side of the UF_6 cylinder arrays and along the centreline as shown in Figure B.3-1.

Characteristics of the Particles

The energy of the neutrons used in the MCNP calculation was 2.8 MeV which is a weighted average neutron energy derived from Table 4 (Neutron Energy Spectrum for Natural Assay after 50 years) of NISYS (2000). The whole spectrum of neutrons can be used in future MCNP calculations.

Figure B.3-1
Top View of Modelled UF_6 Cylinder Arrays and Receptor Locations

(Receptors assumed at various distances along centre-line)



B.3.2 Output Calculations

For this evaluation, 250,000 particles were followed. [This was sufficient to produce relative errors of <0.05 which are considered "generally reliable for point detectors" (LANL 2005b)]. The F5a tally result was modified using conversion coefficients to convert flux to effective dose. The energy-dependent effective dose per unit fluence (for rotational geometry, i.e. the receptor was assumed to be exposed on the sides of the body) conversion coefficients were taken from Table A.41 of ICRP Publication 74 (ICRP 1996). The conversion coefficients in units of pSv cm^2 modify the F5a tally result to units of pSv/source neutron as shown by Equation 2:

Equation 2:

$$\text{Modified F5a Tally }\left(\frac{\text{pSv}}{\text{source neutron}}\right) = \text{F5a Tally }\left(\frac{\#\,\text{neutrons}/\text{cm}^2}{\text{source neutron}}\right) \times \text{Conversion Coefficient }\left(\frac{\text{pSv}}{\#\,\text{neutrons}/\text{cm}^2}\right)$$

The modified F5a tally results were converted to effective neutron dose rates by multiplying the modified F5a tally result from MCNP by the neutron flux for UF_6. The calculated neutron yield from UF_6 given per 1 kg content of uranium is 59.07 n/s per kg(U) (Boneh, Y. *et al.* 1984)

The total neutron flux from UF_6 inside one cylinder was calculated using the Equation 3.

Equation 3:

$$\text{Total neutron flux }\left(\frac{\text{n}}{\text{s}}\right) = \text{total neutron flux per g UF}_6 \left(\frac{\text{n}}{\text{s - gU}}\right) \times \frac{\text{gU}}{\text{gUF}_6}\,\text{V}\left(\text{cm}^3\right) \times \text{density}\left(\frac{\text{g}}{\text{cm}^3}\right)$$

where,

V = volume of UF_6 inside the cylinder(s), and
Density = density of UF_6.

Using Equation 3, the total neutron flux from UF_6 in one cylinder was calculated to be 5.0E+05 n/s. Similarly, the total neutron fluxes from an array of 3, 5, and 10 cylinders were 1.5E+06, 2.5E+06, and 5.0E+06 n/s, respectively.

APPENDIX C

PREVIOUS EVALUATIONS OF NEUTRON DOSE RATES AT CAMECO'S PORT HOPE CONVERSION FACILITY

APPENDIX C – PREVIOUS EVALUATIONS OF NEUTRON DOSE RATES AT CAMECO'S PORT HOPE CONVERSION FACILITY

This appendix contains in order the following previous evaluations of neutron dose rates at Cameco's Port Hope Conversion Facility: Carisse (2007), Hinton (2005), Pereira (2005a), Pereira (2005b) and Takala (2001).

Carisse (2007)

Environmental and Public Neutron Exposure from UF_6

By

Hess Carisse

Cameco Corporation

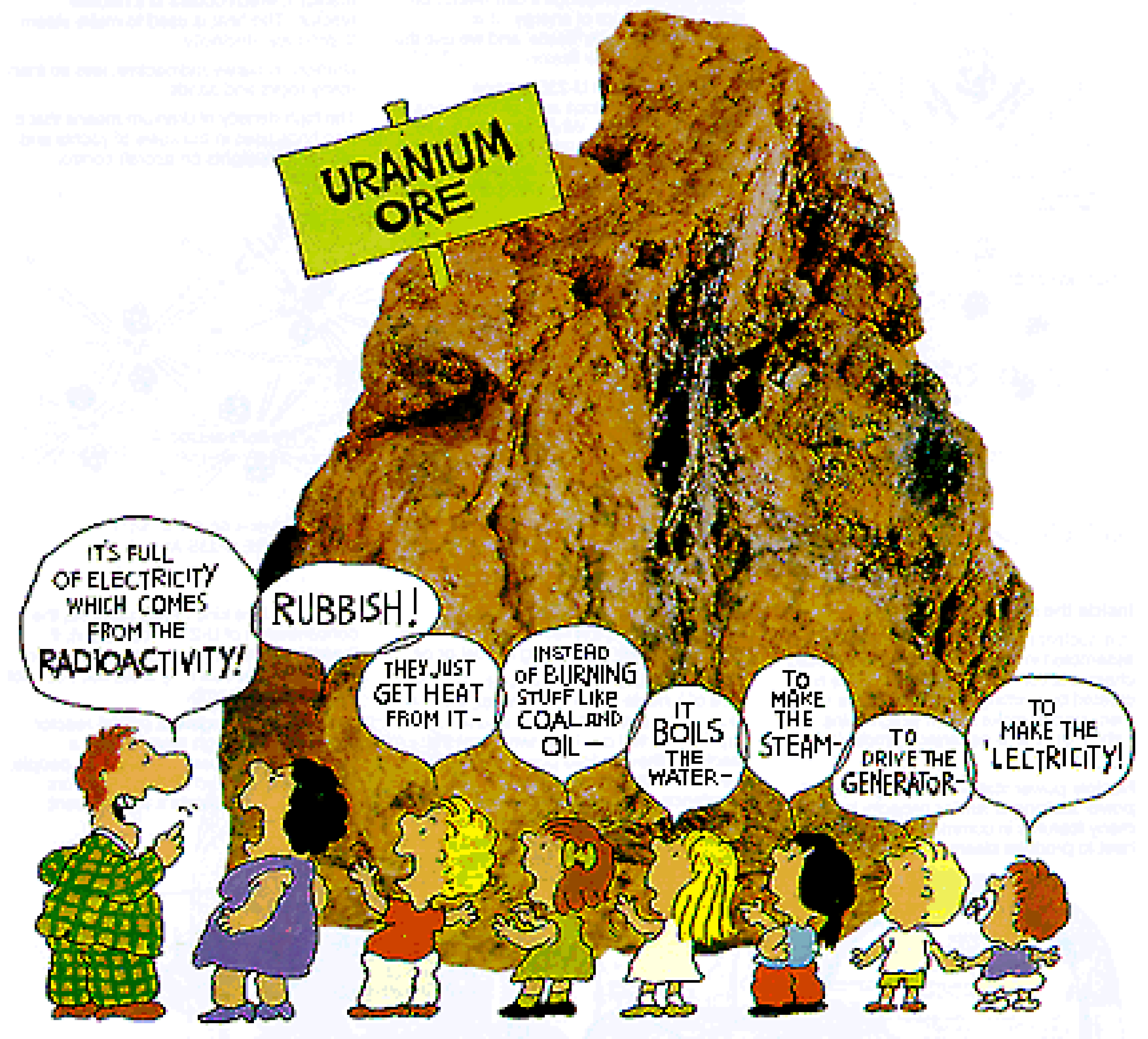
Presented at the OECD/NEA/WGFCS Workshop

October 16 to 18, 2007 Wilmington NC

Fuel Cycle Safety – Past, Present and Future

What is Uranium?

Cameco



Uranium Information Centre Ltd.

Nuclear Fuel Cycle

Cameco

Exploration

Mining and Milling U_3O_8

Refining UO_3

Conversion UF_6

Enrichment UF_6

Conversion UO_2

Fuel Fabrication for light water reactors

Conversion UO_2

Fuel Fabrication for heavy water reactors

Nuclear Reactor

Spent Fuel Storage

Waste Disposal

Reprocessing

Carried on by Cameco and others

Carried on by others

Port Hope Radium Refinery

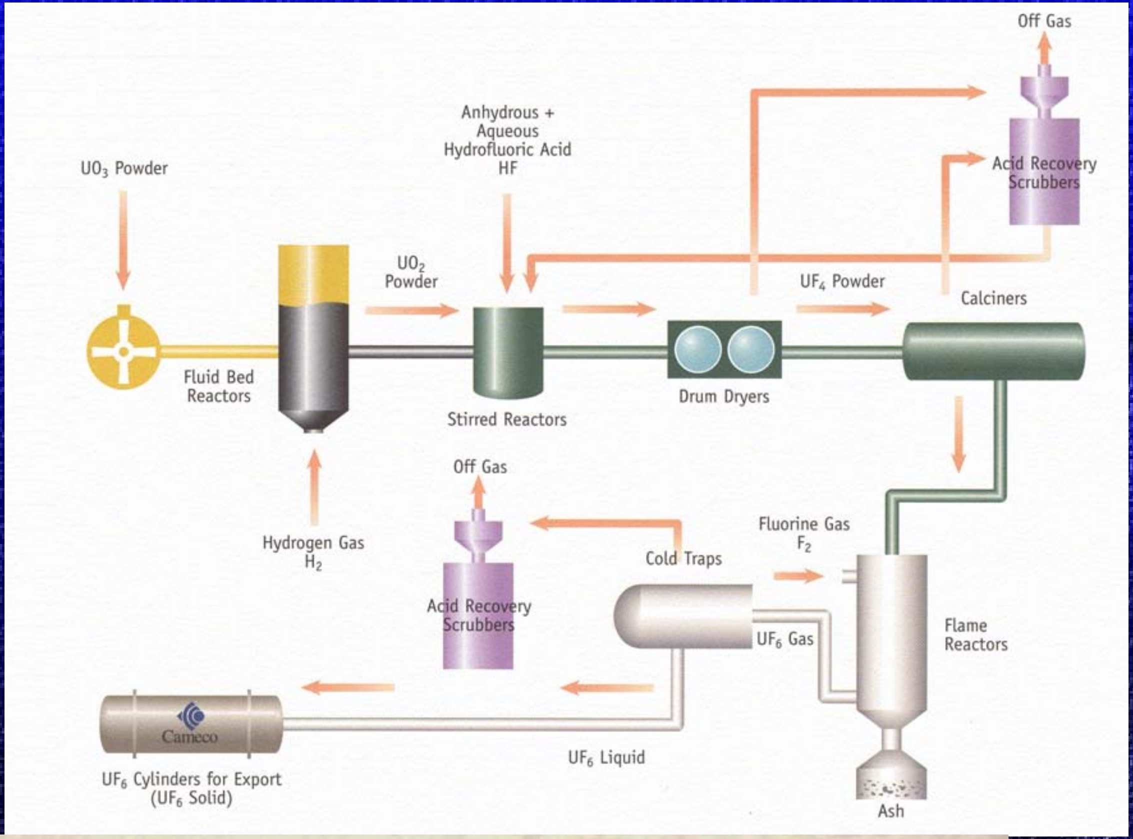
Cameco

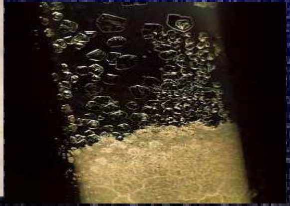
Port Hope conversion facility

Cameco UF_6 conversion facility

UF_6 Conversion Process

Cameco





Solid UF_6

Cameco


3688
3688

◆ ;

Cameco

Cameco

UF_6 Cylinder Transport Canada to USA

Figure 1 Photograph of open trailer road truck loaded with 2 full 48X cylinders for transport to the U.S.A.

Neutron Measurements at PHCF

◆ Neutrons are generated at a low rate from spontaneous fission of uranium and at significantly higher rates from uranium-fluorine compounds through the reaction $^{19}F+\alpha\rightarrow n+^{22}Na$.

◆ Laboratory measurements and theoretical calculations indicate that the dose rate due to neutrons around UF_6 cylinders could be in the range of 0.5 to 2 µSv/h. These values suggest the doses to workers in the UF_6 plant should be less than 1 mSv/y because only a small fraction of a typical work year is spent in close proximity to UF_6 cylinders.

Neutron Measurements at PHCF

- To determine the validity of these studies, as they apply to the situation at Port Hope, measurements were conducted with a neutron survey meter.
- Neutron fields near large sources of UF_6 were easily measurable, but were less than 0.5 µSv/h in virtually every work area assessed.
- This means that doses to workers from neutrons are much less than 1 mSv/y.

Neutron Measurements at PHCF

- A neutron survey was conducted on December 14, 2000 with an ADM-300C Gamma/Neutron survey meter owned by Ontario Power Generation (OPG).
- The meter has energy compensated response to neutrons that measures in units of µSv. The ADM meter can operate as either a rate meter or in a dose integration mode.
- The unit had to be used in dose integration mode and kept still for 4 minutes during the measurement period.

Neutron Measurements at PHCF

Cameco

- Priority was given to locations with large UF_6 sources nearby because UF_6 is the strongest source of neutrons.
- Contact measurements were done on large uranium-fluorine sources (e.g., UF_6 cylinders) to determine the maximum neutron dose rates at the facility; these measurements were called engineering locations.

Neutron Measurements at PHCF

- High-occupancy work areas in the general vicinity of large UF_6 sources were also measured to give a realistic estimate of dose rates in these important locations; these measurements were called workplace locations.
- In addition, measurements were taken at some of the routine TLD stations along the fence line to determine if there is any measurable dose to the public; these measurements were called fence line locations.

Neutron Measurements at PHCF

		Neutron Dose Rate µSv/h		
Location Type	No.	Average	Minimum	Maximum
Engineering	7	1.51	0.05	3.00
Workplace	13	0.14	0.00	0.56
Fence line	8	0.03	0.00	0.05

Neutron Measurements at PHCF

- As anticipated, the engineering samples had the highest results and in the approximate range expected based on the literature review of the topic.
- Contact readings on UF_6 cylinders ranged between 1.5 to 3.0 µSv/h and one reading on top of the head of the primary filter in the second floor flame reactor area had a value of 0.2 µSv/h.
- Lying on a full UF_6 cylinder would give a neutron dose rate of about 2 µSv/h, while the dose rate for a worker standing between two UF_6 cylinders, which one would hope to be a more realistic exposure scenario, would be about 1 µSv/h.

Neutron Measurements at PHCF

Cameco

- In typical workplace locations the neutron dose rate ranged from non-detectable to about 0.6 µSv/h, with an average value of 0.1 µSv/h.
- Even close to UF_6 cylinders the dose rate was low; for example, 2 m from a row of 15 full cylinders the dose rate was only 0.4 µSv/h.
- None of the control rooms measured had any appreciable neutron fields. With maximum dose rates at 1 µSv/h and more typical locations having an average about 0.1 µSv/h, the survey shows that doses to workers from neutrons should be less than 1 mSv/y.

Neutron Measurements at PHCF

Cameco

- The fence line monitoring detected a low-level neutron field around the UF_6 plant. Neutron fields are less than the gamma fields at these locations, but possibly a substantial fraction of the gamma dose rate.
- Neutron Bubble Detector manufactured by Bubble Technologies Industries (BTI) used for environmental monitoring of neutrons has been recommended by Cameco's Health Physicist .

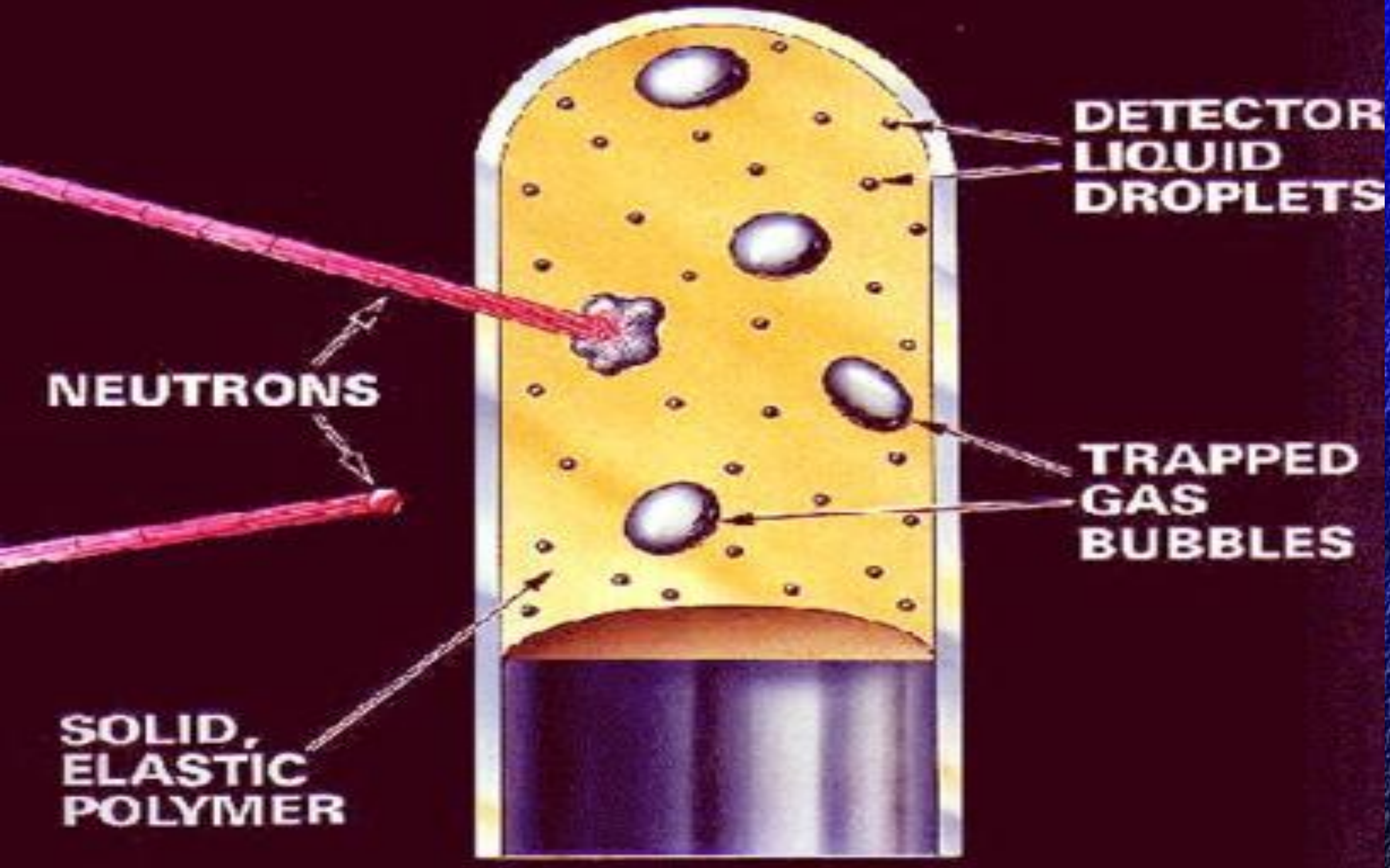
BTI – Bubble Detectors

Cameco

- Bubble tube dosimeters from (BTI) a dynamic company with world-renowned expertise in the field of radiation detection. These dosimeters have a detection limit of 0.01μSv.
- Neutron Bubble Detectors are reusable, integrating, passive dosimeters that allow instant visible detection of neutron radiation.
- Bubble Detectors consist of minute droplets of a superheated liquid dispersed throughout an elastic polymer. When neutrons strike these droplets they form small gas bubbles that remain fixed in the polymer, providing a real time, immediate visual record of the dose.

BTI – Bubble Detectors

HOW A BUBBLE DETECTOR WORKS



Neutron Measurements at PHCF

- Four fence line locations were selected for the Bubble technology trial in 2001.

Location	Number of trials	Highest results (μSv/h)
South of UF6 Plant	5	0.062
West of UF6 Plant	5	0.080
North of UF6 Plant	5	0.230
Critical Receptor at Mill Street	5	0.030

Neutron Measurements at PHCF

Cameco

- Neutron radiation studies have been conducted at the PHCF in 2005 to confirm that this source of radiation does not pose a health risk to employees. For these trials Cameco have used two different devices to measure neutron radiation.
- Neutron dosimeters from Landauer Inc. These dosimeters have a minimum detection level of 100 µSv.
- Bubble tube dosimeters from (BTI). These dosimeters have a detection limit of 0.01µSv.

Neutron Measurements at PHCF

Personal monitoring

- April and May 2005 - Employees from UF_6 plant, UO_2 plant and Materials handling wore a special Landauer CR-39 fast neutron dosimeter designed to neutron radiation exposure
- These three groups represent a broad cross section of employees at our site and will provide the data we need to assess the potential impact of neutron radiation on our entire work force.
- Results were below the detection level of 200 µSv.

Neutron Measurements at PHCF

Cameco

Neutron Measurements at PHCF

Cameco

- The Neutrak 144 detector is a CR-39 (allyl diglycol carbonate) based, solid-state nuclear track detector that is not sensitive to x, beta or gamma radiation, and can be packaged specifically for neutron detection only.
- The CR-39 is laser engraved for permanent identification, and wrapped with a 2-D bar code to assure efficient chain-of-custody.

Neutron Measurements at PHCF

- Track Etch Technology consists of etching the CR-39 for 15 hours in a chemical bath to enlarge exposure tracks.

- The fast neutron dose is measured by counting the tracks generated as a result of the proton recoil with the polyethylene radiator, while the thermal/intermediate dose is measured by counting the alpha tracks generated with the boron radiator.

Neutron Measurements at PHCF

- On April 1, 2005, thirty-three Landauer fast neutron dosimeters were deployed along Port Hope Conversion Facility fence line. After 30 days of exposure dosimeters show results below the minimum detection level of 100 μSv.
- A second trial has been initiated on December 1, 2006 and detectors are to be collected on December 1, 2007.

Landauer – Neutrak Detector

CR-39

Fast/Intermediate/

Thermal Neutron dosimeter

Landauer – Neutrak Detector

Cameco

BTI – Bubble Detectors

- 2005-2006 - An extensive study of neutron radiation was conducted at the UF_6 operation plant using BTI neutron bubble detectors.
- The average neutron radiation dose inside the UF_6 plant was less than 0.3 µSv/h.
- The operators in the cylinder lay down area logged the highest dose rate at 0.12 µSv/h.
- All other operators showed lower dose rates.

BTI – Bubble Detectors

- The study also looked at neutron dose to the public and the environment. The maximum neutron dose along the fence line was 0.28 µSv/h.
- The studies concluded that neutron radiation at the PHCF represents a small fraction of the total whole body dose received by employees and therefore does not pose a risk to employees.
- Cameco will continue to use these devices to demonstrate that neutron radiation levels at the facility remain constant.

The End

Hinton (2005)

Government of Canada Gouvernement du Canada

MEMORANDUM NOTE DE SERVICE

B: to 1253453

À	M. O'Brien c.c. J. Jaferi, R. Garg, P. Mirfakhraei
From De	A. Hinton
Security Classification - Classification de sécurité	
Our File - Notre référence	30-1-15 / 30-6-3 / 36-3-1-0
Your File - Votre référence	
Date	June 24, 2005
Tel. No. - Nº de tél.	613-996-7327

Subject
Objet

PACKAGING AND TRANSPORT LICENSING DIVISION TYPE II INSPECTION: NEUTRON DOSE RATE SURVEYS OF URANIUM HEXAFLUORIDE (TYPE H) PACKAGES AT CAMECO – PORT HOPE, ON

Licensee Name	Cameco Corporation – Fuel Services Division
Licensee Location	1 Eldorado Place Port Hope, ON L1A 3A1
Licence Number	07370-1-10.0
Date of Inspection	2005-05-12
CNSC Inspector	A. Hinton
Inspector Authority	- nuclear facilities and all places and vehicles where nuclear substances are or may be located in Canada - TDG Inspector #10887 (Class 7 Dangerous Goods)
Inspector Accompanied by	C. Green – Cameco R. Garg – CNSC-CCSN P. Mirfakhraei - CNSC-CCSN

Summary

As part of the ongoing CNSC Compliance Program and initiated by requests from members of the public, a series of neutron dose rate measurements were taken from full uranium hexafluoride ($UF_{6(natural)}$) packages at Cameco Corporation's Fuel Services Division plant in Port Hope, ON. These measurements were taken under the authority of the *Nuclear Safety and Control Act* as part of a Type II compliance inspection.

The scope of the inspection was limited to these measurements and the analysis of neutron contribution to the gamma dose rate for 'transport index' as per *Packaging and Transport of*

Nuclear Substances Regulations subsection 16(4) referencing the *Regulations for the Safe Transport of Radioactive Material* (TS-R-1) paragraph 533.

Radiation Detection Equipment

1. Neutron Dose Rate Meter – 'REM ball'

Make	Eberline
Model	ASP-1
Serial Number	3453
Calibration	2005-04-06 by CNSC to ± 20% true dose rate (Am-241:Be)

This instrument uses polyethylene to thermalize incident neutrons for capture by the Helium-3 proportional counter. The electronics provide an instantaneous dose rate in microSieverts per hour on the analog display. The instrument doesn't show incident neutron energy, but accounts for it within the geometry of the detector (i.e. neutrons with very high or low energies will not be thermalized sufficiently or are already too slow to be seen by the detector, and will not contribute to the dose displayed by the instrument).

2. Neutron Counting Meter – FieldSPEC Gamma Spectrometer

Make	ThermoElectron Corporation
Model	fieldSPEC-N
Version	He-3
Serial Number	03F3/1725
Calibration	2005-05-12 by CNSC with internal Cs-137 source

This instrument is primarily a field spectrometer for gamma dose rate measurements and source identification purposes based on incident photon energies. Gamma detection is provided by a sodium iodide scintillator. The neutron counting function is performed by a Helium-3 proportional counter, however there is no inherent hydrogenous material for neutron thermalization – this process is facilitated by the operator's anatomy. The electronics provide for a count rate in counts per minute on the digital display. Again, no incident neutron energy is determined.

Survey Results

Using the calibrated instruments, above, battery and functionality checks were successfully performed before the survey began. A visual inspection for damage showed nothing out of the ordinary. Background radiation measurements were: 0.15 μSv/h gamma, < 1 μSv/h neutron, and 0 cpm neutron.

1. The first location chosen was one where a measurable dose rate was expected due to the proximity of many full 'type 48X' UF_6 cylinders. These were arranged in an 11 x 2 array, and were adjacent to a similar sized array of full 'type 48Y' cylinders.

Measurements were taken 1 m from the surface of cylinder EL5984 (3rd cylinder from the north). The gamma dose rate was 5.3 μSv/h. Instantaneous neutron dose rates were fluctuating over a period of several minutes from 1 to 3 μSv/h; counting was 10 cpm.

2. The second location chosen was predicted to have the highest exposure rate. Again, a 4x8 array of full 'type 48Y' cylinders was chosen.

   Measurements were taken 1 m from the surface of a cylinder (4th cylinder from the north). The gamma dose rate was 5.0 μSv/h. Instantaneous neutron dose rates were fluctuating over a period of several minutes from 2 to 3 μSv/h; counting was 8 cpm.

3. The final location was that of a single, full, 'type 48Y' UF_6 cylinder loaded on board a flat-bed trailer in the west parking lot of the facility. There were no other sources of radiation within 25 metres of this cylinder.

   Measurements were taken at 1 m from the surface of the cylinder. The gamma dose rate was 2.1 μSv/h; neutron dose rate was indistinguishable from background (< 1 μSv/h), and counting was 2 cpm.

Conclusion

The survey of full Type H packages containing UF_6 was performed in order to verify that the neutron dose rate was negligible with respect to the gamma dose rate in its contribution to the transport index. The transport index is a number which is used to provide information with regards to the radiation exposure hazard from a package being transported. It is a unit-less number and is determined by dividing the maximum dose rate at one meter from the package (in microSv/h) by 10.

For situations in the proximity of many full UF_6 cylinders the neutron dose rate measured was approximately one half of the gamma dose rate.

In the instance of the single, full, UF_6 cylinder the neutron component of the dose rate at one meter was not determined. This is because of detector limitations in the equipment used. Only instantaneous dose rate was displayed, and was no different than that of background. In order to completely quantify the contribution to transport index from the neutron dose rate it is necessary to perform additional surveys with: (i) more sensitive detection instrumentation; or (ii) instrumentation which can integrate neutron dose over a longer period of time.

Regardless, the observed external radiation hazard from a solitary, full, UF_6 cylinder is minimal. The transport index would be 0.2 in this case which gives rise to package category II-YELLOW. This is an expected and reasonable value for these packages.

Regulatory Criteria

Packaging and Transport of Nuclear Substances Regulations, Nuclear Safety and Control Act, SOR/2003-405, December 2003, Canada Gazette Part II, Vol.137 No.13, Ottawa.

Regulations for the Safe Transport of Radioactive Material, 1996 (Revised), No. TS-R-1 (ST-1, Revised), International Atomic Energy Agency, Vienna.

Transportation of Dangerous Goods Regulations, Transportation of Dangerous Goods Act, 1992, SOR/2003-400, December 2003, Canada Gazette Part II, Vol.137 No.13, Ottawa.

Pereira (2005a)

Mireille Normandeau - Reply to your letter of April 8, 2005

From: Ken Pereira
To: irisblue16@hotmail.com
Date: 2005-05-03 11:30 AM
Subject: Reply to your letter of April 8, 2005
CC: Records Office-Bureau des Documents

File: 37-26-0-0

Dear Mr. Miller, Mrs. More,

The President has asked me to reply to your letter of April 8, 2005, to her, on the concerns you have raised about neutron radiation and gamma radiation in places accessible to the public, and your call for the establishment of an air monitoring program in Port Hope. I understand that Cameco is willing to provide you with information too, which would serve to supplement what is presented here.

Following are the CNSC staff's responses on each of these topics.

1. Neutron Radiation and Potential Doses in Public Areas at Cameco Site

The emission of neutron radiation from uranium hexafluoride (UF_6) is a well known phenomenon.

The neutron radiation fields measured from the UF6 cylinders are produced by interactions between alpha particles and fluorine atoms. Alpha particles are produced during the natural radioactive decay of uranium atoms. The neutron radiation fields produced in this manner are measurable but are small relative to the gamma radiation fields emitted by the UF6. Radiation levels measured at the surface of UF6 cylinders are typically less than 0.03 mSv/hr for gamma radiation and less than 0.003 mSv/h for neutron radiation.

The levels of all ionizing radiation (i.e., gamma, neutron or some combination thereof) emitted from packages containing nuclear substances, including UF6 cylinders, are regulated through the *Packaging and Transport of Nuclear Substances Regulations* promulgated under the *Nuclear Safety and Control Act*. These Regulations limit the dose rates at the surface and at one meter from packages to 2 mSv/h and 0.1 mSv/hr respectively. These limits are prescribed in subsection 16(4) of the *Packaging and Transport of Nuclear Substances Regulations*, which directly references paragraphs 526, 530, 531, and 533 of the IAEA *Regulations for the Safe Transport of Radioactive Material*, 1996 Edition (Revised) No. TS-R-1 (ST-1, Revised).

The maximum radiation level measured in mSv/h at a distance of 1 metre from the package when multiplied by 100 gives a Transportation Index (TI). The TI is inscribed on the labels affixed to the package, but does not display the units. The maximum TI allowed under the regulations for the UF6 cylinders is 10, which results from a maximum dose rate of 0.1 mSv/h at one meter from the package (i.e. 0.1 mSv/h X 100 = 10). In comparison, a full UF6 cylinder when shipped from Cameco's site typically has a TI less than 1, which means the maximum dose rate measured 1 meter from the cylinder's surface is less than 0.01 mSv/hr.

Considering that the neutron radiation is less than 10 percent of the total radiation emitted from UF6 cylinders and the actual TI measurements posted on UF6 cylinders are less than 10 percent of the regulatory limit, CNSC staff is satisfied that the transport of UF6 cylinders from and to Cameco site does not pose an undue risk to members of the public in Port Hope.

Finally, relative to this topic, CNSC staff would be interested in seeing the results from the measurements that the Uranium Medical Research Centre made on December 11, 2004, and details about the type of instrument(s) used in measuring the neutron radiation levels and their calibration including neutron energy response curve, if these could be made available to us. This would enable the staff to understand the UMRC findings and compare them to what would be expected based on theoretical models.

2 Elevated Gamma Radiation in Public Areas Near Cameco

The CNSC's principal actions to protect the health and safety of the public, and workers who are or may be exposed, against the hazards from nuclear facilities or the use of radioactive materials, are to establish and enforce compliance with limits on radiation doses and releases of hazardous substances. The radiation dose limits are prescribed in the *Radiation Protection Regulations*; these are supplemented by additional requirements to keep doses As Low As Reasonably Achievable (ALARA) in the conditions of the licence issued for the operation of each facility or use of radioactive material. Regulatory limits are based on values established to protect human health and the environment.

In the case of Cameco's facility, for the protection of Port Hope residents and visitors, the licence sets a limit on the maximum annual radiation dose to a member of the public that is lower than the generally applicable value of 1 mSv/y prescribed in the Regulations; this is 0.3 mSv/y. Compliance is determined from the results of a suite of performance monitoring programs, within and outside the facility. The monitoring parameters comprise: emissions to the atmosphere, liquid effluent discharges to the harbour and radiation levels around the perimeters of the facility main site, the Centre Pier property and the storage buildings on Dorset Street East, together with uranium and fluoride depositions and sampling for environmental effects in a number of locations in Port Hope. The results from these programs are compiled quarterly and annually in Cameco's environmental performance/compliance reports and submitted to the CNSC. These reports are available in the Port Hope Public Library and at the municipal office. Also, Cameco staff presents the quarterly compliance reports to the P3 Committee of Port Hope Municipal Council.

Cameco's monitoring program includes 17 perimeter gamma monitoring stations around its main site and Centre Pier property. The gamma exposure readings at these locations typically range from 1 to 100 μR/h. One of these stations, identified as station no. 14, which is located on east side fence of the Centre Pier property, is treated differently from the others because it serves a special function: to assure that the radiation dose to people who live in the vicinity of the intersection of Mill and Madison Streets, does not exceed the 0.3 mSv/y limit referred to above. For that reason, this station, and only this one, is subject to a specific limit of 14 μR/h in the facility operating licence. The value of this limit is predicated on the assumption that those residents may be in their homes 24 hours each day, for 365 days/y. Thus, theoretically speaking, they have the potential to receive the maximum exposure from all sources of radiation from the operation of Cameco's facility.

Due in part to historic contamination, the readings on the western shore of Port Hope Harbor are routinely the highest recorded. However, they would not represent a significant radiation exposure to a member of the public who walks around the perimeter of the Port Hope harbour due to their relatively short exposure timeframe. For example, a person who walks around the Port Hope harbor one hour per day, five days per week and 26 weeks per year (130 hours total), receiving a gamma exposure rate of 100 μR/h, would receive an estimated annual dose of 0.087 mSv. This value is far below the annual licensed and regulatory public dose limits of 0.3 mSv and 1 mSv respectively. Please note that this estimate is conservative as most gamma fields around the Port Hope harbour are well below 100μR/h.

3. Establish Air Monitoring Program

As noted in relation to topic 2, the Port Hope facility has an air monitoring program that measures radiation and particulate levels around the facility on an ongoing basis. The monitoring includes provisions for detecting radiation as well as uranium and fluoride releases. Stack emissions are also monitored on an ongoing basis for uranium and fluorides, and controlled below limits specified in the site licence. Uranium isotope assays are not required to be taken in the stack emissions, as this value is known from the raw materials used and does not change during processing.

The CNSC could not agree to these monitoring programs being put under the control of the Municipality of Port Hope, for policy reasons. This is because Cameco is required by the CNSC Regulations and certain licence conditions to report on its performance and it does so using data gathered by means of the sampling and analytical protocols defined in those programs. Were the equipment to be operated and the monitoring to be carried out by the municipality, the validity or reliability of the results could be challenged if they were used as the basis for regulatory action or legal proceedings against Cameco. The CNSC has no jurisdiction to regulate any activities carried out by the Municipality of Port Hope; hence, it would have no control over the quality of data produced by a municipally implemented monitoring program.

In your letter you suggest that a local weather station be established in Port Hope and issue public smog alerts as required. Because the Cameco operation dose not contribute significantly to smog formation which is mainly due to emissions from motor vehicles, it would be inappropriate for the CNSC to request that the licensee establish a local weather station to monitor smog levels in Port Hope.

As also mentioned above, the results from the air monitoring program are presented in Cameco's quarterly environmental performance/compliance reports, which are already available to the public at the municipal office or at the Port Hope Public Library. Providing minute-by-minute readings would not be possible as ambient air monitoring samples are collected and analyzed on a daily, weekly or monthly basis. The monitoring results are not currently available on-line but we agree that would be beneficial to enhance access to the information.

I trust these responses satisfy your needs.

Sincerely,

J.K. Pereira
Executive Vice-President
Operations

Pereira (2005b)

From: Ken Pereira
To: irisblue16@hotmail.com (Mrs. Faye More)
Date: June 21, 2005
File #: 36-3-1-0
Subject: Reply to your letter dated June 1, 2005 concerning Cameco's Fourth Quarter Report – Port Hope

Dear Mrs. More:

This is in response to your letter to Ms. Keen dated June 1, 2005 related to the analysis of Cameco's fourth quarter report. You have raised questions related to lack of reporting on neutron radiation from UF_6 cylinders, gamma radiation monitoring around site 1, background gamma level and safe levels of air pollution.

CNSC staff provides the following responses to these questions:

On the issue pertaining to neutron radiation, CNSC staff visited Cameco recently to measure the activity resulting from uranium hexafluoride (UF_6) cylinders. The CNSC staff's measurement determined the doses resulting from the neutron radiation fields are extremely low and are also small relative to the gamma radiation fields emitted by the UF_6 cylinders. Radiation levels measured at the surface of these UF_6 cylinders are typically less than 0.03 mSv/h for gamma radiation and less than 0.003 mSv/h for neutron radiation. These values are well below the limit of dose rates at the surface and at one metre from packages of 2 mSv/h and 0.1 mSv/h, respectively as prescribed in the *Packaging and Transport of Nuclear Substances Regulations of the Nuclear Safety and Control Act (NSCA)*. UF_6 cylinders from Cameco have typically II-yellow label meaning that the maximum dose rates at the surface and at one metre from the cylinder are less than 0.5 mSv/h and 0.01 mSv/h, respectively. Cameco is required to verify the dose rate of 100% of the UF_6 cylinders before the label is affixed on the cylinder. Finally, the CNSC regulatory limits for radiation levels from the surface and at one metre from the package are based on the "effective dose" which takes into account all types of radiation including neutron radiation.

With respect to ionizing radiation, in general, although different types of ionizing radiation may have different efficiencies in causing effects on the human body, these differences are all taken into account in reporting doses in mSv; that is, a person receiving 1 mSv from neutrons will be exposed to the same risk as a person exposed to 1 mSv of gamma radiation.

Considering the extremely low level of neutron radiation produced by UF_6 cylinders, separate monitoring of neutron radiation levels is not warranted and has not been required by the CNSC.

With respect to gamma radiation monitoring around site 1, Cameco maintains 17 perimeter gamma monitoring stations around its main site and the Centre Pier property. The gamma exposure readings at these locations typically range from 1 to 100 µR/h. One of these stations, identified as station no. 14, which is located on the east side fence of the Centre Pier property, is treated restrictively compared to the other stations. This is to ensure that the radiation doses to people residing in the vicinity of the intersection of Mill and Madison Streets do not exceed the licence limit of 0.3 mSv per year (a limit more restrictive than the 1 mSv per year specified in the *Radiation Protection Regulations of the NSCA*). For this reason, this station, i.e., station no. 14 and only this one is subjected to a specific limit of 14 µR/h in the facility operating licence. The average value provided in the 2004 4th quarter report was 1.2 µR/h at station no. 14, significantly less than the restrictive limit of 14 µR/h. In addition, due in part to historical contamination, the readings on the western shore of Port Hope harbour are routinely the highest recorded. However, they would not represent a significant radiation exposure to a member of the public who walks around the perimeter of the Port Hope harbour due to their relatively short exposure timeframe. For example, a person who walks around the Port Hope harbour one hour per day, 5 days per week and 26 weeks per year, i.e., 130 hours in total in a year, receiving a gamma exposure rate of 100 µR/h, would receive an estimated annual dose of 0.087 mSv. This value is far below the annual licensed and regulatory public dose limits of 0.3 mSv and 1 mSv, respectively. It should be noted that in the Cameco's fourth quarter report, there was not a single average gamma field result that exceeded the assumed value of 100 µR/h and, therefore, a conservative estimate is presented above. With respect to the background gamma value of 8 µR/h for the Port Hope area, this is deducted to account for cosmic radiation, natural background from soil and radon progeny. This local background gamma level was established by Health Canada in the 1980s for areas not affected by radiological contamination.

Regarding the air pollution, the Canadian Council of Ministers of the Environment (CCME) publishes the current Canadian environmental quality guidelines for air and the values of the principal air pollutants are presented in a tabular form below.

Pollutant	**Values, µg/m^3**	**Averaging Period**
Total Suspended Particulate	120-400 60-70	24-hour Annual
Particulate Matter-10 µm	25	24-hour
Particulate Matter-2.5 µm (Fine Particles)	15	24-hour
Sulphur Dioxide	450-900 150-800 30-60	1-hour 24-hour Annual
Nitrogen Dioxide	400-1000 200-300 60-100	1-hour 24-hour Annual
Carbon Monoxide	15000-35000 6000-20000	1-hour 8-hour

Takala (2001)

Interoffice Memo

Cameco

Date: February 14, 2001

To: H. Carisse

From: J.M. Takala

Re: **Neutron Measurements at Port Hope**

1. Summary

Neutrons are generated at a low rate from spontaneous fission of uranium and at significantly higher rates from uranium-fluorine compounds through the reaction ${}^{19}F+\alpha\rightarrow n+{}^{22}Na$[1]. Laboratory measurements and theoretical calculations indicate that the dose rate due to neutrons around UF_6 cylinders could be in the range of 0.5 to 2 μSv/h[2,3]. These values suggest the doses to workers in the UF_6 plant should be less than 1 mSv/y because only a small fraction of a typical work year is spent in close proximity to UF_6 cylinders.

To determine the validity of these studies, as they apply to the situation at Port Hope, measurements were conducted with a neutron survey meter. Neutron fields near large sources of UF_6 were easily measurable, but were less than 0.5 μSv/h in virtually every work area assessed. This means that doses to workers from neutrons are much less than 1 mSv/y. Measurements conducted along the fenceline indicated the possibility of a low, but measurable a neutron dose. Follow-up measurements are recommended to better document the potential doses.

2. Monitoring Methodology

The neutron survey was conducted on December 14, 2000 with an ADM-300C Gamma/Neutron survey meter (S/N 001801, calibration due 05-04-01) by J.M. Takala. The meter was loaned to Cameco by Ontario Power Generation (OPG) for the purposes of the survey and to help evaluate its performance. The meter can switch between measuring neutron and gamma radiation. It has energy compensated response to neutrons that measures in units of μSv. Kris Szornel of OPG said that the meter would tend to slightly overestimate the neutron dose in some physical situations, but should give a reasonable estimate of the dose rate from neutrons.

The ADM meter can operate as either a rate meter or in a dose integration mode. The unit had to be used in dose integration mode and kept still during the measurement period because of a spurious signal caused by a loose cable between the neutron probe and the meter. When the meter was being used in the rate mode, it was noticed that sometimes when the probe and cable were moved, the neutron dose rate would increase dramatically and then start to drop at a very regular rate. Kris Szornel was aware of the problem and had concluded there was some sort of loose connection in the cable that occasionally introduced electrical noise to the meter from the probe.

To avoid this problem the meter and probe was placed in a stationary position and used in the dose integration mode. Typically, a period of four minutes was used as the dose integration period. This strategy was successful, however, while measuring outside along the fenceline it was only possible to place the probe down and the meter had to be held by hand. Although every effort was taken to keep the cable between the probe and meter still, it is possible that this loose connection biased these measurements high. The only other problem with this approach was that it limited the number of measurements that could be taken.

The ADM meter also can measure gamma radiation and was used along with a Bicron µSv meter to assess gamma radiation at the same locations as the neutron measurements.

Several different types of locations were measured. Priority was given to locations with large UF_6 sources nearby because UF_6 is the strongest source of neutrons. Contact measurements were done on large uranium-fluorine sources (e.g., UF_6 cylinders) to determine the maximum neutron dose rates at the facility; these measurements were called engineering locations. High-occupancy work areas in the general vicinity of large UF_6 sources were also measured to give a realistic estimate of dose rates in these important locations; these measurements were called workplace locations. In addition, measurements were taken at some of the routine TLD stations along the fenceline to determine if there is any measurable dose to the public; these measurements were called fenceline locations.

3. Monitoring Results

Table 1 lists a summary of the results by location type and Table 2 lists the detailed monitoring results.

Table 1: Summary of Neutron Dose Rate Measurements

Location Type	No.	Neutron Dose Rate µSv/h		
		Average	Minimum	Maximum
Engineering	7	1.51	0.05	3.00
Workplace	13	0.14	0.00	0.56
Fenceline	8	0.03	0.00	0.05

As anticipated, the engineering samples had the highest results and in the approximate range expected based on the literature review of the topic. Contact readings on UF_6 cylinders ranged between 1.5 to 3.0 µSv/h and one reading on top of the head of the primary filter in the second floor flame reactor area had a value of 0.2 µSv/h. Lying (e.g, sleeping) on a full UF_6 cylinder would give a neutron dose rate of about 2 µSv/h, while the dose rate for a worker standing between two UF_6 cylinders, which one would hope to be a more realistic exposure scenario, would be about 1 µSv/h. On this basis, no special restrictions or monitoring are warranted for work in close proximity to UF_6 cylinders.

In typical workplace locations the neutron dose rate ranged from non-detectable to about 0.6 μSv/h, with and average value of 0.1 μSv/h. Even close to UF_6 cylinders the dose rate was low; for example, 2 m from a row of 15 full cylinders the dose rate was only 0.4 μSv/h. None of the control rooms measured had any appreciable neutron fields. With maximum dose rates at 1 μSv/h and more typical locations having an average about 0.1 μSv/h, the survey shows that doses to workers from neutrons should be less than 1 mSv/y. On this basis no additional dosimetry is needed for the workforce.

The fenceline monitoring apparently detected a low-level neutron field around the WUF_6 plant. The term "apparently detected" is used because of the problem with the loose cable and having to hold the meter by hand for these locations. Assuming the measured values are accurate, neutron fields are less than the gamma fields at these locations, but possibly a substantial fraction of the gamma dose rate. For example, a neutron field of 0.04 μSv/h would be the equivalent of about 6 to 7 μR/h. Given that the incremental gamma field from the facility at these locations is typically 10 to 20 μR/h, if the measured neutron field is accurate, it may require some consideration in the environmental monitoring program. To clarify this issue I recommend that more reliable neutron measurements be done around the fenceline.

I recommend that a device called a Bubble Detector manufactured by Bubble Technologies Industries by used for environmental monitoring of neutrons. The device has excellent sensitivity to neutrons, can be reused, and costs $75. The most significant operational constraint on this unit is that it is somewhat temperature sensitive and the measurements would need to be conducted during the summer. To achieve a detection limit of 0.01 μSv/h the detector would need to be placed in the field for about one week (using a model BD100R at 3.3 bubbles per μSv sensitivity). To achieve this level of sensitivity with a TLD would take around one year. If this course of action is acceptable, I can supply the details of the supplier to Port Hope.

J. M. Takala

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	B. Steane	J.P. Jarrell
	A. Oliver	M.B. Wittrup
	F. Dobri	K. Toews

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[1] T.E. Sampson, Neutron yields from uranium isotopes in uranium hexafluoride. *Nucl. Sci. Eng.* **54**, 470 (1974)

[2] Y. Boneh *et al*, Neutron yields from uranium-fluorine compounds. *Nucl. Sci. Eng.* 86, 106 (1984)

[3] NIYS Corporation , *Determination of Neutron to Gamma Dose Ratios for the UF_6 Cylinder Yards*. Calculation Number 1073-018-C001.August 25, 2000