Occupational Exposure in a CANDU NPP: Management of the Risk for Internal Alpha Contamination

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ISOE 2012

Prague 20 – 22 June, 2012

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INTRODUCTION

For a CANDU reactor tritiated heavy water (DTO) is the major contributor to the internal dose of professionally exposed workers –up to 40% of the total effective dose.

Internal contamination with other radionuclides is infrequent, usually having a negligible contribution to occupational exposure.

Recently after some contamination events, mainly during refurbishment of PHWR reactors, significant internal exposures to TRU were reported.

Contributing to these unplanned exposures was an inadequate radiological protection program, not able to recognize the potential for alpha contamination at the job site.

INTRODUCTION

Neutron activation of ^{238}U is the principal mode of transuranic radionuclides (TRU) production in a nuclear reactor. The activation of ^{238}U yields ^{239}U, and a wide distribution of both long and short-lived species of TRU. A fraction of those nuclides can be released due to fuel failures, and circulates through reactor components and systems.

As the reactors age, these radionuclides accumulate and can potentially leak from the components and systems of the reactor into various service buildings and work environments.

INTRODUCTION

The presence of alpha emitting radionuclides in nuclear reactors working areas is related to fuel cladding defects, allowing the fission products to migrate into the primary heat transfer – PHT circuit.

Once in the reactor coolant, this material may be distributed throughout other areas of the facility creating accumulations of TRU which could have significant dose implications for personnel involved in work activities such as routine operations, maintenance, repair and refurbishment of equipment and systems.

INTRODUCTION

The presence of TRU includes primarily longer-lived radionuclides: ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am, ^{243}Cm, ^{244}Cm.

Because alpha-emitting nuclides generally have high internal dose conversion factors for the inhalation pathway, they result in lower limits and action levels, compared to beta/gamma nuclides.

Unlike beta/gamma-emitting nuclides, alpha-emitting nuclides are generally more difficult to measure, and often must be inferred from the presence of surrogate radionuclides, or quantified using relatively expensive and time-consuming analytical methods.

ACTINIDES IN CANDU REACTORS

The radioactivity in the Primary Heat Transport System – PHT system is generated by fission in the core fuel and by activation of dissolved and suspended materials.

The majority of these activated materials are corrosion products.

The activity may be transported out of the reactor core in solution or as a suspended solid.

Outside the reactor core the radionuclides may remain in the heavy water or may deposit on PHT system surfaces.

ACTINIDES IN CANDU REACTORS

Smear samples from boilers, service rooms and equipment, crud samples from the PHT System and pressure tube scrape samples, from some CANDU plants have been characterized to assess potential internal contamination hazards.

The identified contamination is from both activation and fission products, mainly dominated by ^{60}Co and ^{137}Cs.

The results confirmed Fueling Machine maintenance rooms and Spent Fuel Bay as the most alpha contaminated locations in the plant.

The major contributors to the measured total alpha contamination were ^{241}Am, $^{239/240}Pu$, ^{238}Pu and $^{243/244}Cm$.

AREAS WITH POTENTIAL FOR ALPHA CONTAMINATION IN CERNAVODA NPP

First step in designing a program for the control of alpha contamination is the characterization of gross alpha activity levels in the plant areas where TRU may be present in order to assign a proper Area Action Level.

CNE Cernavoda started the characterization of potentially alpha contaminated areas in 2011.

Table 1 presents some significant results for alpha, beta activities and alpha/beta ratios for the locations where alpha contamination was identified: spent fuel discharge bay (Unit#2 R-001 room) and steam generator (Unit#1).

AREAS WITH POTENTIAL FOR ALPHA CONTAMINATION IN CANDU PLANTS

Sampling Area	Beta activity (Bq)	Alfa activity (Bq)	Beta/Alfa ratio
Spent Fuel Discharge Bay (U#2) sample #1	5.02E+04	2.3	2.22E+04
Spent Fuel Discharge Bay (U#2)sample #2	3.08E+04	0.6	5.08E+04
Magnetite SG#1 sample 1	1.35E+04	2.3	5.93E+03
Magnetite SG#1 sample 2	1.72E+04	2.7	6.41E+03

Table 1 Beta and alpha gross activity in some samples from Cernavoda NPP

AREAS WITH POTENTIAL FOR ALPHA CONTAMINATION IN CANDU PLANTS

Sample / Radionuclide	Spent Fuel Discharge Bay (U#2)	SG#1 Magnetite sample
^{51}Cr		4.02E+03
^{54}Mn	3.28E+04	5.06E+02
^{56}Ni		3.25E+01
^{59}Fe	1.64E+04	2.39E+03
^{60}Co	7.10E+04	1.83E+03
^{95}Nb	1.52E+06	2.04E+04
^{95}Zr	8.33E+05	8.97E+03
^{103}Ru		8.40E+03
^{113}Sn	2.72E+04	9.89E+01
^{124}Sb		6.78E+02
^{125}Sb	6.44E+04	
^{137}Cs	1.19E+05	
^{134}Cs	2.87E+04	
^{131}I	4.38E+03	
^{110m}Ag	1.17E+04	

Table 2 Gamma emitting radionuclides in two Cernavoda samples

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

Generally applied system of dose assessment from bioassay data relies first on the intake calculation of a radionuclide either from direct measurements (e.g. external monitoring of the whole body or of specific organs and tissues) or indirect measurements (e.g. urine, faeces or environmental samples).

Predicted values of the internal residual quantities for unit intake are recommended by ICRP and these values can be used to estimate the intake.

The effective dose resulting from a particular intake is then calculated from the dose coefficients recommended by ICRP, EU Basic Safety Standards and International Basic Safety Standards.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

The radioisotopes of plutonium and higher actinides present particular difficulties of measurement and interpretation. They are alpha- emitters with only low abundance, and low energy photon emissions.

In-vivo measurements do not have sufficient sensitivity for routine monitoring, but may be useful for special investigations. In vitro bioassay (urine and faeces) for actinides requires expensive and labour-intensive radiochemical techniques.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

In vivo whole body counting (WBC) for the detection of the major gamma emitters (e.g., Cs-137 and Co-60), that can be used as surrogate-nuclides, can be applied for the routine monitoring practices of TRU at CANDU facilities.

In this case, ^{60}Co could be a better indicator of potential intake of TRU at the studied work areas.

In the event that an intake of ^{60}Co is found by WBC, a follow-up faecal bioassay of actinides for dose assessment would be indicated depending on scaling factor.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

The principal impact of the presence of TRU at commercial nuclear power facilities correlates to the inhalation hazard

For the same activity ^{238}Pu dose is 3000 times greater than ^{60}Co dose. Removable ^{238}Pu contamination of 100 dpm/100 cm^2 (1.7 Bq/100 cm^2) would be equivalent to approximately 300,000 dpm / 100 cm^2 (5,000 Bq/100 cm2) of ^{60}Co removable contamination

Since all of the long-lived alpha emitting nuclides have similar ALIs, this numeric example is generally applicable to the TRUs of concern.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

Therefore, for radiation protection purposes we may want to consider the inclusion of appropriate action levels to identify and control TRU contamination.

Actions to address for personnel protection to TRU may include increased use of respirators and engineering controls, lower bioassay action levels, more frequent in vitro bioassays, increased air sampling, and analysis of several smears and air samples for alpha radioactivity.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

Instrumentation

The internal contamination with gamma nuclides at Cernavoda NPP is measured by using Whole Body Counters - WBC Fastscan, and Accuscan II Canberra USA.

FASTSCAN WBC is designed to quickly and accurately monitor people for internal contamination of radionuclides with energies between 300 keV to 2000 keV. The system includes two large sodium iodine detectors [NaI(Tl)] that provide a Lower Limit of Detection of 150 Bq for ^{60}Co with a count time of one minute for a normal person containing ^{40}K.

For a counting time of 3-5 minute range, the detection limits are about 50 Bq for ^{137}Cs and and about 60 Bq for ^{60}Co.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

The ACCUSCAN-II is a high resolution, stand-up WBC. It is designed to identify and quantify radionuclides with energies between 100 keV and 2000 keV in complicated combinations.

It also provides information on the location of the radioactive materials found in the body through its scanning mechanism.

The system includes a shield, scanning detector mechanism, a 25% two coaxial germanium detectors and cryostat assembly and a digital spectrum analyzer.

Count time is the 3-5 minute range. The detection limit for uncontaminated person (count time 30 minute) is about 81 Bq for ^{137}Cs and about 76 Bq for ^{60}Co.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

Estimates of Intake and Assessment of Dose

Investigative whole body counts provide a timely and direct measure of the deposition of gamma emitting radionuclides in the body.

An estimate of the intake [I] of gamma emitting radioactivity is determined from these whole body count measurements [A] and applicable intake retention fractions corresponding to the nuclide, solubility class, mode of intake, time after intake, and type of bioassay measurement [IRF] :

$$I = \frac{A(t)}{IRF(t)}$$

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

Estimates of Intake and Assessment of Dose

The committed effective dose (CED) from gamma emitters can be calculated using the estimates of the intake (I) and the Dose coefficients in the ICRP, e_{50}:

$$CED = e_{50} \times I \ (mSv)$$

The CED from alpha emitters will be calculated from scaling the alpha-emitter intakes from the beta-gamma emitter intakes and calculating the CED from the intake of each radionuclide.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

Estimates of Intake and Assessment of Dose

Since at this moment we don't have the means for quantitative determinations of individual alpha emitting radionuclides, the most conservative approach is to assume all alpha emitters are Americium-241; T1/2 = 432 y; inhalation class M.

In order to obtain a better estimate of the CED for ^{241}Am (and for ^{60}Co) the intake will be computed with the computer code LUDEP 2.0, thus being possible to take into consideration the contribution of all the daughter – nuclides in the decay chain of ^{241}Am.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

Scaling Factors

Since the ratio of alpha-to-gamma activity is variable, a job specific / area scaling factor should be determined (Table 1).

A representative air sample provides the most accurate measurement of the relative abundance of alpha and gamma emitters.

When specific job coverage air samples are not available, other air samples that are related to the same area and type of work may be used.

If no representative air samples are available, job specific loose surface contamination smears may be used.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

Retention Models

Inhalation and ingestion e(50) are based on biological retention models.

The larger inhaled particles are rapidly cleared from the upper respiratory track and cleared through the GI tract. As a result, the ingestion model more accurately describes the actual retention of radioactive material

In order to determine as accurate as possible the biokinetic retention model for the surrogate nuclide (^{60}Co) whole body counting must be performed daily after the intake.

The results will be compared with special monitoring predicted values (Bq per Bq intake) for inhalation and ingestion in ICRP.

INTERNAL DOSIMETRY PROGRAM FOR ACTINIDES

Retention Models

Transuranic alpha emitting nuclides are highly insoluble. Their retention is similar to other insoluble gamma emitters, e.g ^{58}Co and ^{60}Co which are well suited for determining retention fractions for alpha emitters because of their relative high abundance and solubility class.

As it was confirmed by calculation results based on ICRP models, ^{60}Co is a suitable “surrogate” nuclide for the evaluation of internal doses due to TRU alpha emitters for professionally exposed workers at CNE Cernavoda NPP.

Regardless of the beta-gamma to alpha ratio, all WBC results for ^{60}Co greater than MDA will be investigated. This approach ensures that all intakes are appropriately assessed.

CONCLUSIONS

A can be seen in Table 1 CNE Cernavoda contaminated areas are classified in present as Level III Areas (low alpha contamination) and in case of an internal contamination the alpha emitters' exposure is not likely to exceed 10% of the total internal dose.

Whole body counting represents an effective and inexpensive gamma bioassay tool that can be used initially to assess an intake. Whole body counters may differ in design, sensitivity, and function, but all designs are capable of detecting small quantities of ^{137}Cs or ^{60}Co deposited in the lung or GI tract.

Due to its consistent ratio to gross alpha activity over time ^{60}Co is the best surrogate for estimating internal dose from alpha emitting contaminants, if scaling factor can be determined.

CONCLUSIONS

Taking into consideration the limitations of our current program, the most conservative approach is to consider that every internal contamination with ^{60}Co, occurred in plant areas susceptible of alpha contamination, is accompanied by TRU intake, regardless the level of the ^{60}Co activity detected.

For an internal contamination near the ^{60}Co whole body counter detection limit, the committed effective dose due to TRU ranges from 6.1E-5 mSv to 5.3E-3 mSv (Table 3).

CONCLUSIONS

Surrogate Nuclide: Co-60* M class				Am-241; M class; $e_{inh(50)}$ = 3.7E-05					
Days after intake	IRF	Intake (Bq)	CED mSv	Scaling Factor: 2E-5			Scaling Factor: 1.7E-4		
				Bq	mSv	CED_{Am}/ CED_{Co}	Bq	mSv	CED_{Am}/ CED_{Co}
0.25	0.736	81.5	1.4E-3	1.6E-3	6.1E-5	4%	1.4E-2	5.2E-4	38%
1	0.483	124.3	2.1E-3	2.5E-3	9.3E-5	4%	2.1E-2	7.9E-4	38%
2	0.255	235.5	4.0E-3	4.7E-3	1.8E-4	4%	4.0E-2	1.5E-3	38%
3	0.151	397.8	6.8E-3	8.0E-3	3.0E-4	4%	6.8E-2	2.5E-3	38%
4	0.108	556.4	9.5E-3	1.1E-2	4.2E-4	4%	9.5E-2	3.5E-3	38%
5	0.090	663.4	1.1E-2	1.3E-2	5.0E-4	4%	1.1E-1	4.2E-3	38%
6	0.083	727.1	1.2E-2	1.5E-2	5.5E-4	4%	1.2E-1	4.6E-3	38%
7	0.078	766.3	1.3E-2	1.5E-2	5.7E-4	4%	1.3E-1	4.9E-3	38%
8	0.076	792.8	1.3E-2	1.6E-2	5.9E-4	4%	1.3E-1	5.1E-3	38%
9	0.074	814.5	1.4E-2	1.6E-2	6.1E-4	4%	1.4E-1	5.2E-3	38%
10	0.072	835.2	1.4E-2	1.7E-2	6.3E-4	4%	1.4E-1	5.3E-3	38%

Table 3 Alpha doses assigned for an MDA intake of ^{60}Co

Thank you for your attention!

Questions?