NEUTRON DOSIMETRY AT ONTARIO POWER GENERATION

NEUTRON DOSIMETRY AT ONTARIO POWER GENERATION: CALIBRATION FACTORS FOR THE SNOOPY – NP-100 NEUTRON REM-METER

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Abstract

Within the CANDU workplace only a small fraction of workers are exposed to neutron radiation. For these individuals, roughly 4.5% of the total radiation equivalent dose is the result of exposure to neutrons. When this value is considered across all workers within the CANDU workplace only 0.25% of the total radiation equivalent dose is the result of exposure to neutrons. Neutron dosimetry at Ontario Power Generation (OPG) is governed by the Canadian Nuclear Safety Commission (CNSC) through Regulatory Standard S-106, an Atomic Energy Control Board (AECB) document. The dosimetry program includes both direct and indirect dosimetry methods. For direct dosimetry, a moderator-based neutron rem-meter is used to measure both ambient dose equivalent and ambient dose equivalent rates. One method of indirect dosimetry employs maps of neutron dose rates, measured using a moderator-based neutron rem-meter, along with the time spent in a particular area to calculate the equivalent dose. The current neutron rem-meter employed is the NP-100, previously the NP-2, manufactured by Canberra Industries Incorporated. These detectors are both known as "SNOOPY". The rem-meters used at Ontario Power Generation are calibrated by the National Research Council of Canada (NRCC), Institute for National Measurement Standards. The result of the calibration is a factor which relates the neutron count rate to the ambient dose equivalent rate, using a standard Am-Be neutron source. Using the measurements presented in a CANDU Owner's Group Inc. Technical Note, "Capability maintenance in Neutron Dosimetry 2003/04 – Performance-testing a Neutron Survey Meter" (Nunes and Surette, 2004) readings from the rem-meter for six different neutron fields—in six source-detector orientations—were used, to determine a calibration factor for each of these sources. The calibration factor is dependent on the fluence-to-dose conversion coefficients. These coefficients rely on the radiation weighting factor to link neutron fluence and the resulting equivalent dose. Although the neutron energy spectra measured in the CANDU workplace cannot be approximated by the calibration source's neutron energy spectrum, the calibration factor remains constant—within acceptable limits—regardless of the neutron source used in calibration; for the specified calibration orientation and current radiation weighting factors. However, changing the value of the radiation weighting factors would result in changes to the calibration factor. OPG should evaluate the effect of any such modifications to determine whether a change to the calibration process or resulting calibration factor is warranted.

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Chapter 1

Introduction

Ontario Power Generation (OPG) owns and operates two nuclear power stations. The Pickering Nuclear Generating Station (PNGS) consists of two facilities—Pickering A and Pickering B—with each facility housing four separate CANDU (Canadian Deuterium Uranium) reactor units. Pickering A comprises units 1 through 4, while Pickering B comprises reactor units 5 through 8. The Darlington Nuclear Generating Station (DNGS) houses four CANDU reactor units, numbered 1 through 4. Currently there are nine reactor units which are operational: Unit 4 (PNGS-A), Units 5-8 (PNGS-B) and Units 1-4 (DNGS).

At OPG, as with all CANDU reactors, workers are exposed to radiation sources outside the body (external exposure) as well as radioactive materials which enter the body (internal exposure). Although the risk associated with exposure to radiation is small, it is presumed that even low radiation doses may produce some deleterious health effects. Therefore, as part of a radiation protection program, the radiation dose incurred by workers is measured and recorded. This is an important way to assess their probability of experiencing any negative effects.

A large component of a radiation protection program is the dosimetry program; a program which measures and records the radiation doses received by employees in the course of their work. Currently OPG operates its dosimetry program under a license from the Canadian Nuclear Safety Commission (CNSC). As a part of this license, OPG must fulfill the requirements of Regulatory Standard S-106, Technical and Quality Assurance Standards for Dosimetry Services in Canada (S-106), a document produced by the Atomic

Energy Control Board (AECB), the predecessor to the CNSC (AECB, 1998). This document outlines the specifications associated with the measurement of radiation dose from sources internal and external to the body. For those external sources which contribute to the worker radiation doses with greater significance—specifically γ-ray and β-particle radiation sources—the requirements of the dosimetry program are more stringent. In these instances the technical and quality assurance standards ensure that these doses are measured, recorded and maintained, both accurately and consistently. On the other hand, for radiation sources which do not significantly contribute to the total radiation dose received by worker, the requirements of Regulatory Standard S-106 are less specific. Currently, there are no specifications for neutron dosimetry. However proposed revisions to S-106 contain more detailed requirements for neutron dosimetry programs (CNSC, 2005). As a result, it will be necessary to document information and those procedures relevant to neutron dosimetry. It will also be necessary to understand the effect of possible modifications to the dosimetric quantities involved in the calculation of neutron radiation dose. How these quantities are incorporated into both dosimeter calibration and dose calculation will affect how the neutron dosimetry program evolves in response to future modifications.

Driven by the impending revisions to Regulatory Standard S-106, documentation of the neutron dosimetry program in place within OPG has become important. This need has also been reinforced by an incident at the DNGS where neutron dose rates ranging from 0.15 to 0.80 mSv/hr (15 to 80 mrem/hr) were discovered at Airlock #2 of Unit 3 (OPG, 2004). To address these concerns, this document focusses on the NP-100 neutron rem-meter used at OPG for neutron surveys and dosimetry. The NP-100 uses a slow neutron proportional counter surrounded by a neutron moderator. It is commonly known as SNOOPY. The aim is to determine the calibration factor for the NP-100 and decide whether it should be modified to account for the neutron fields present within the OPG-CANDU workplace.

Chapter 2

Fundamentals of Radiation Dosimetry

Ionizing radiation and radioactive materials (subsequently called "radiation") have had a constant presence throughout the environment. With its discovery in the late 19th century, research centred around developing beneficial uses for radiation. However, scientists failed to foresee the associated dangers, and it was not until the negative repercussions became more evident that focus shifted. It was necessary to evaluate both the effects of beneficial practices and the corresponding hazards of radiation. To address concerns regarding negative effects, the International X ray and Radium Protection Committee was established in 1928 and was later restructured and renamed the International Commission on Radiological Protection (ICRP). The ICRP is an advisory body, providing recommendations to regulatory agencies. These recommendations, based both on the observable effects of radiation at high doses and dose rates, are used to assess the probability of detriment at lower doses and dose rates. They are also used to establish relevant legislation, regulation and policies regarding practices and standards of protection.

Consistent with these recommendations and appropriate regulation, radiological protection programs are implemented by individual organizations for their specific operations. The ICRP states that "the primary aim of radiological protection is to provide an appropriate standard of protection for man without unduly limiting the beneficial practices giving rise to radiation exposure" (ICRP, 1991). Acknowledging this definition of protection, and the principle of maintaining doses "as low as reasonable achievable (ALARA), economic and social factors being taken into account", provides a foundation for establishing any radiological protection program.

2.1 Dosimetry

The radiation dosimetry program is an integral part of the radiation protection program. Radiation dosimetry is the complete process of measuring or estimating the dose received by an individual as a result of exposure to radiation. The primary purpose of the dosimetry program is to ensure that the dose received by an individual is accurately determined. However, rather than just simply quantifying the amount of radioactivity in a person's body, the measurements must be interpreted to obtain an estimate of the biological effect of the radiation dose. This provides a measure of the potential for harm so that exposures can be limited to minimize the probability of future effects (ICRP, 1991).

The dosimetry program is composed of three essential elements. The first element is an assessment of the radiation sources in the workplace, in terms of the types of hazard and the hazard level present or potentially present. The next element is the design of the dosimetry monitoring program. This includes selection of the monitoring methods, and specification of the required sensitivity and frequency of use. The third major element of the dosimetry program comprises obtaining a relevant dosimetric measurement (such as the radioactivity level in a biological sample or the amount of radiation received by a dosimeter), interpreting and converting the the measurement into radiation dose, recording and retaining the result, and assigning it to the person. Other activities necessary for the successful operation of the dosimetry program include distribution and collection of dosimeters or biological samples, calibration of equipment, and quality control and quality assurance of the measurement process.

2.1.1 Considerations for Neutron Dosimetry

The components of a dosimetry program depend on the type and size of the hazards. The sources present in each environment will vary according to the types of practices being performed within it, and perhaps even those in the surrounding areas. Only a small fraction of the dose received by individuals within the CANDU workplace is due to neutron exposure. Nonetheless, determining the dose due to neutrons is difficult because of the challenges associated with its measurement; challenges associated with the physics of the neutron and neutron sources, and the associated biological effects. Understanding the variations specific to neutrons provides a basis for accounting these differences with regards to neutron detection and dosimetry.

Neutron Physics

The neutron is an uncharged, sub-atomic particle. Its mass is given as 1.008665 atomic mass units, or 939.56563 MeV/c^2; a mass approximately 0.1% larger than that of the proton. Within the nucleus, the neutron serves to bind together the protons, which would otherwise repel one another so strongly the nucleus would not remain intact. The number of neutrons present within a nucleus is variable; this number can be determined from the difference between the mass number and the atomic number. Nuclides with the same atomic number but varying numbers of neutrons, are called isotopes.

A free neutron is unstable and will undergo β-decay, decaying into a proton and electron, releasing approximately 1 MeV, with a half-life of 10.24 minutes (Firestone et al., 1996).

The energy range of neutrons varies from thermal energies of approximately 0.025 eV, epithermal energies of approximately 1 eV, and fast neutron energies between 100 keV and 10 MeV. However, the energy range observed in a particular case is directly dependent on the neutron source.

Sources of Neutrons

There are a number of neutron sources including: α-Beryllium sources, photoneutron sources, spontaneous fission reactions, accelerator reactions, as well as reactor sources.

α-Beryllium Sources α-Beryllium sources are composed of a mixture of an α-emitter and an isotope of beryllium, Beryllium-9; an isotope which has a loosely bound neutron which can be released through interaction with an α-particle.

$$^{4}He + ^{9}Be \longrightarrow ^{12}C + n$$

This reaction, denoted (α, n), produces a spectrum of neutrons. The neutrons are not monoenergetic for a number of reasons, such as the α-particle slowing down through collision within the material and the possibility of leaving the ^{12}C atom in an excited state. Generally, the α-emitter consists of americium, plutonium, and polonium as they emit an α-particle with limited production of other radiations.

Photoneutron Sources Photoneutron sources are governed by the same general principles as α-Beryllium sources, except that a γ-ray, or photon, is used to provide the energy required to release the neutron. This process is described as a (γ, n) reaction such that

$$\gamma + {}^{2}H \longrightarrow {}^{1}H + n.$$

If the source of photons is of approximately the same energy, the resulting neutrons will also have similar energies. This reaction is also relevant in the case of neutrons produced during normal operations of CANDU reactor systems.

Spontaneous Fission Sources Spontaneous fission sources will emit neutrons with a distribution of energies when the parent nuclide fissions, a process which occurs only in specific isotopes. Roughly four neutrons will be produced directly in the fission process. A typical spontaneous fission neutron source is Californium-252, which yields approximately 2.3×10^{12} neutrons per second per gram of Californium-252.

Accelerator Sources Accelerator sources use a beam of particles from an accelerator to initiate a neutron producing reaction. The advantage of this process is that by carefully selecting the incident energy and the angle at which we observe the emitted neutron, a reasonably monoenergetic beam of neutrons can be produced. However, these sources are inconvenient, relying on the particle beam produced by the accelerator to yield neutrons.

Nuclear Reactor Sources Nuclear reactors are a major source of neutrons which are primarily produced by the reaction of Uranium-235 with a thermal neutron. This reaction creates two or more fission fragments, several neutrons, and γ-rays, as well as releasing energy. Fission induced neutrons have a wide range of energies, spanning eight orders of magnitude.

Interactions Between Neutrons and Biological Tissue

There are a number of neutron interactions observed in biological material, resulting from both the varying composition of tissues and the wide energy range available to neutrons. The probability of each individual interaction mechanism is dependent on both these factors. While the radiation dose due to neutron radiation comes largely from elastic scattering interactions, both the inelastic scattering interactions and neutron capture reactions also play an important role.

Elastic Scattering Collisions between the incident neutron and a nucleus, where both components retain their identity, is described as elastic scattering, or an elastic collision. In this interaction, the neutron collides with a nucleus in the target material. During the collision, kinetic energy is transferred from the neutron to the nucleus. Finally the neutron is scattered or deflected from the nucleus at an angle θ, and can continue to undergo another interaction. The amount of energy the neutron loses, which is transferred to the nucleus and deposited in the material, is a function of the mass of the two particles and the neutron scattering angle. Specifically

$$\Delta E_n = \frac{4MmE_0}{(M+m)^2}\cos^2\psi \tag{2.1}$$

where M is the mass of the nucleus, m is the mass of the neutron, E_0 is the initial energy of the neutron and $\psi = (\pi - \theta)/2$ (NRCP, 1971). This process is often referred to as neutron moderation. As the neutron undergoes elastic scattering, it continually loses energy until it no longer has more energy than the surrounding material. At this point, the neutron is said to be a thermal neutron, and can be readily captured by a number of other nuclei.

In biological material, the interaction occurs primarily with hydrogen nuclei, or protons. This is due to both the abundance of protons in tissue and the high interaction probability associated with the reaction. Because neutrons and protons are of similar mass, there is a large energy transfer associated with this interaction, corresponding to a large amount of energy deposited in the tissue. Therefore, the dose due to exposure to neutron radiation comes largely from the energy deposited in tissue through elastic collisions with protons.

Inelastic Scattering For higher energy neutrons, another interaction pathway is predominant. Inelastic scattering interactions are those in which the neutron is temporarily captured by the nucleus, and then re-emitted. The capture leaves the nucleus in an excited state. In returning to a more stable configuration a photon is also emitted. The energy of the neutron must be greater than the energy associated with the excited state—which is also the energy of the emitted photon—in order for inelastic scattering to occur. This threshold energy is expressed as

$$E_{threshold} = E_{\gamma}\left[\frac{M+m}{M}\right] \tag{2.2}$$

where E_{γ} is the energy of the photon (NRCP, 1971). Typically the threshold energy is approximately 6 MeV (Hall, 1993).

In biological material, both the photon and the neutron contribute to the amount of energy absorbed. The neutron, once re-emitted from the excited nucleus, continues through the tissue, depositing more and more of its energy in the course of further interactions. Like the neutron, the photon may be absorbed in the body following its emission, contributing to total radiation dose.

Capture Processes As the neutrons lose energy through scattering interactions, there is an increased probability of neutron capture. Here, the nucleus absorbs the neutron and emits a photon, or another particle.

With thermal neutrons, there are two capture reactions which are particularly important. The first, is the capture of a neutron by nitrogen such that $^{14}N(n,p)^{14}C$ where $E_p = 0.6$ MeV. The second is the capture of a neutron by a hydrogen nucleus described by $^{1}H(n,\gamma)^{2}H$ with $E_{\gamma} = 2.2$ MeV (NRCP, 1971). These secondary radiations contribute significantly to the amount of radiation deposited, or absorbed within tissue. As a result, they also contribute significantly to the total radiation dose.

2.2 Dosimetric Quantities

The type of biological effects and the probability of their occurrence depends on the amount of radiation to which an individual has been exposed. Consequently, quantifying the amount of ionizing radiation becomes relevant in estimating the potential for damage.

For non-stochastic, or deterministic effects—effects caused by cell killing, leading to clinically detectable impairment of the function of a tissue or organ—there is an associated threshold of exposure. Below this level, the effect will not occur. In the case of stochastic effects—cancer induction in the individual exposed and hereditary disorders in the progeny of the irradiated individual—there is an observed dose response relationship. Dosimetric quantities are used to express the relationship between absorbed dose and the resulting effects.

2.2.1 Protection Quantities

Protection quantities provide a standard measure of the amount of ionizing radiation, which is correlated with both the potential and actual effects (ICRU, 1993). All recommendations and regulatory limits are based on values of the protection quantities.

Biological Damage and Linear Energy Transfer (LET)

The biological effects of radiation are the result of damage to critical targets within the material. In the case of tissue, the relevant critical target is the DNA. When any form of radiation is absorbed in biological material, there is a probability for interaction with the critical target via one of two routes: direct interaction and indirect interaction (Hall, 1993). With direct interaction, the absorption of radiation occurs in the DNA itself. This causes the atoms or molecules of the target to become excited or ionized. Changes in the target, as a result of the absorption of radiation, may lead to the permanent damage associated with radiation exposure. On the other hand, indirect interaction occurs when the initial absorption of radiation occurs in other targets. As cells are mainly composed of water, the likely interaction is the absorption of radiation by a water molecule. This causes the water molecule to become excited or ionized, leading to the production of a free radical.[1] The free radicals produced in the indirect interaction are free to diffuse a distance about twice the diameter of the DNA helix during their lifespan. Therefore, a free radical produced near the DNA, may interact with it, causing it to become excited or ionized itself. The final result is the same type of damage which results from the direct interaction process.

[1] Free radicals are highly reactive species which contain an unpaired electron in the outer orbital shell.

Similarly, the radiation energy absorbed by biological material can also be described as the energy deposited by the radiation in the material. As the radiation deposits its energy, it leaves tracks of ionization and excitation events. The amount of energy deposited in each track is described by the linear energy transfer (LET). Measured in units of kiloelectron volts per micrometer (keV/μm) of unit density material, the LET was defined by the International Commission of Radiation Units and Measurements (ICRU) in 1962 as follows:

> The unrestricted linear energy transfer (L) of charged particles in a medium is the quotient of dE/dl, where dE is the average energy lost by a charged particle traversing a distance dl (Hall, 1993).

While not defined as a protection quantity, the LET has been used in the derivation and definition of a number of relevant dosimetric quantities.

Absorbed Dose

The most fundamental dosimetric quantity used in protection is absorbed dose, D. Bearing the units joule per kilogram, and given the name gray (Gy), absorbed dose is a measure of the energy absorbed in a medium, per unit mass. However, the potential for negative effects is not solely dependent on the amount of energy absorbed in the material; rather, it also depends on both the type of radiation and its energy.

Relative Biological Effectiveness (RBE)

The variation in types of radiation leads to variation in the amount of radiation required to produce a specific effect. It may be the case that the same effect can result from two different absorbed doses through the use of two different types of radiation. To account for this variation it was necessary to develop a way to compare radiation to one another. This was achieved using x-rays as a radiation standard and defining the relative biological effectiveness (RBE) as follows:

> The RBE of some test radiation (r) compared with x-rays is defined by the radio D_{250}/D_r where D_{250} and D_r are, respectively, the doses of 250 kV x-rays and the test radiation required for equal biological effect (Hall, 1993).

Like the linear energy transfer, the RBE is not in itself a dosimetric quantity. It too, has been used in the definition and determination of a number of other quantities relevant to the calculation of radiation dose.

Quality Factors and Radiation Weighting Factors

It has always been apparent that the individual characteristics of a type of radiation need to be considered when calculating the dose resulting from radiation exposure. Time has seen the method of accounting for radiation type change. The ICRP introduced the quality factor, $Q(L)$, a dimensionless quantity. The quality factor is a judgement of the appropriate way to weight the absorbed dose for different types of radiation for the biological effects important in radiation protection. It considers the probability of stochastic effects associated with radiation exposure and a radiation's RBE. The quality factor is specified as a function of unrestricted linear energy transfer in water such that the quality factor can take the values outlined in following table. Other values of $Q(L)$ were to be interpolated as required.

Unrestricted linear energy transfer, L in water $[keV/\mu m]$	$Q(L)$
3.5	1
7	2
23	5
53	10
175	20

Table 2.1: Relationship between Unrestricted Linear Energy Transfer and Quality Factor (ICRP, 1977).

In 1990, the ICRP redefined a number of the dosimetric quantities based on more recent data. It was decided that the previous relationship between quality factor and unrestricted linear energy was too precise. Based on the uncertainties inherent in the radiobiological information available, the ICRP considered such precision to be both inappropriate and unjustifiable. As a result the ICRP recommended another factor to account for the differences in various types of radiation, and their ability to cause biological damage. A radiation weighting factor, w_R, is currently assigned to each type of radiation, and in the case of neutrons, for specific energy ranges as well. At their introduction, it was implied that the radiation weighting factors were determined from radiobiological studies. These factors are included in ICRP Publication 60, "1990 Recommendations of the ICRP" and the following table.

Type and energy range	Radiation weighting factor w_R
Photons, all energies	1
Electrons and muons, all energies	1
Neutrons, energy > 10 keV	5
$\geq$ 10 keV to 100 keV	10
$\geq$ 100 keV to 2 MeV	20
$\geq$ 2 MeV to 20 MeV	10
$\geq$ 20 MeV	5
Protons, other than recoil protons, energy > 2 MeV	5
Alpha particles, fission fragments, heavy nuclei	20

Table 2.2: Radiation Weighting Factors (ICRP, 1991).

The values of the quality factor and radiation weighting factors cannot be derived from one another. They account for the effects of radiation differently and are not interchangeable. The quality factor is related to the radiation field inside a tissue equivalent material, while the radiation weighting factors are related to the external radiation field. Currently there is great debate on which of these factors yields a more accurate description of the radiation effects, particularly for neutron radiation where these factors vary quite widely. The current sentiment is that, particularly for lower energy neutrons, the radiation weighting factor over-estimates the biological effects associated with exposure; an over-estimate that did not exist with the use of the quality factor. Introducing a formal inter-relation between the quantities of $Q(L)$ and w_R would alleviate these concerns and simplify the calculation of neutron doses.

Currently, the ICRP is attempting to link the values of $Q(L)$ and w_R. In ICRP Publication 92, the ICRP clarifies the origin of each quantity (ICRP, 2003). The hope is that by making explicit the details as to how the values of each quantity was determined, an appropriate relationship between the quantities can be determined. Three options for a modified convention are presented in ICRP Publication 92. The first is a radical simplification of the radiation weighting factor such that it takes only two or three numerical values. The second would make w_R coherent with $Q(L)$, causing a significant reduction in the effective dose resulting from neutron exposure. The third would link w_R to an LET-dependent internal weighting factor without reducing the magnitude of effective dose from neutrons significantly, in most cases (ICRP, 2003).

ICRP 92 has proposed the use of option three; a compromise of sorts with respect to the first two options. This would correct the large factors currently attributed to low energy neutrons while not drastically reducing the values for neutron of high energy. Figure 2.1, taken from ICRP 92, shows the current radiation weighting factor w_R, the proposed modification to the values of the radiation weighting factor, and the effective quality factor, for neutron radiations. The effective quality factor is defined as the ratio of the effective dose equivalent, H_E, divided by the organ-weighted absorbed dose, D. The proposed numerical convention for w_R preserves the values of w_R outlined in ICRP 60 for neutron energies above 1 MeV. However, for neutron energies below 1 MeV, the proposed values of w_R are substantially smaller. The substantial decrease at low neutron energies and the slight increase at high neutron energies are consistent with available radiobiological data which describes the biological effectiveness of neutrons. As well, the modified dependence of the radiation weighting factor on neutron energy represents a dependence on LET; a similar dependence exhibited by the quality factor. The consequences of modifying the radiation weighting factor are detailed in Chapter 6.

Dose Equivalent and Equivalent Dose

Along with the quality factor, the ICRP expressed the quantity of dose equivalent in its 1977 Recommendations (ICRP, 1977). At this time, the dose equivalent at point in tissue, was defined as the product of absorbed dose D, and the quality factor, Q. However, with its subsequent recommendations in 1990, the name dose equivalent was changed to equivalent dose, indicating the use of the newly defined radiation weighting factors.

The product of absorbed dose and current radiation weighting factor, summed over all

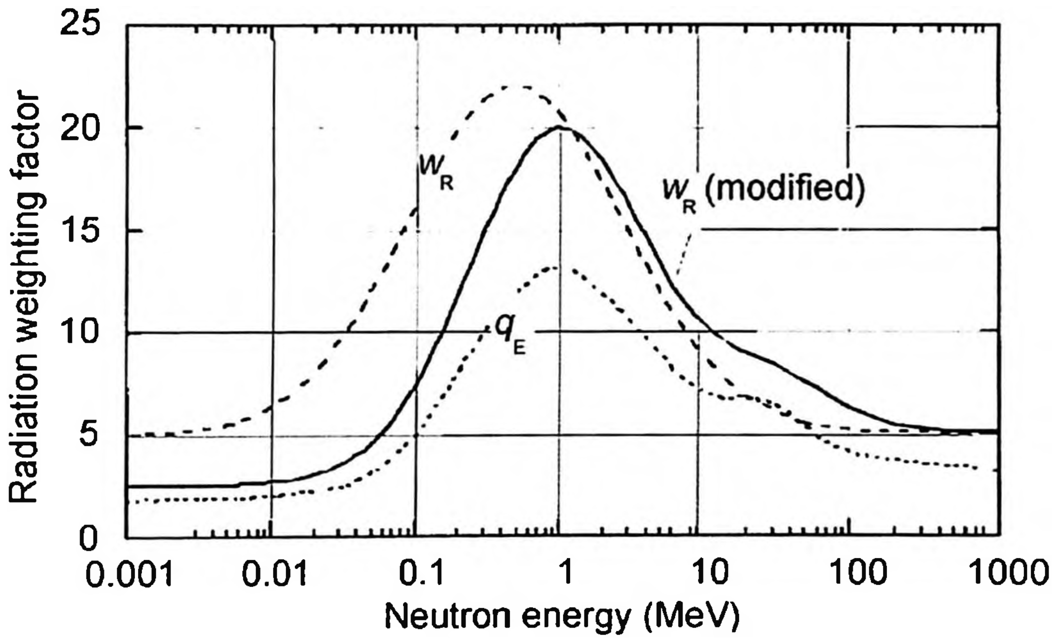

Figure 2.1: The current radiation weighting factor w_R (ICRP 60), the proposed modification w_R (modified) (ICRP 92), and the effective quality factor q_E for neutron radiation (ICRP, 2003).

contributing radiations, yields the equivalent dose. Expressed as

$$H_T = \sum_R w_R \cdot D_{T,R} \tag{2.3}$$

where $D_{T,R}$ is the absorbed dose averaged over the tissue or organ, T, from radiation R. The equivalent dose has units of joule per kilogram, but it is given the special name sievert (Sv) to indicate the weighting for radiation type. However, the probability of stochastic effects is also dependent on the organ or tissue irradiated. As a result, the equivalent dose must be further weighted by another weighting factor to account for this dependence.

Tissue Weighting Factors

The tissue weighting factor, w_T, represents the relative contribution of that organ or tissue to the total detriment[2] due to these effects resulting from a uniform—whole body—irradiation. The tissue weighting factors of ICRP 26 are used to weight the dose equivalent and the tissue weighting factors of ICRP 60 are used to weight equivalent dose, to account for the variation of the probability of stochastic effects with organ or tissue. The values of tissue weighting factors are chosen such that a uniform, whole-body equivalent dose gives an effective dose which is equal to the uniform equivalent dose. In other words, the sum of all tissue weighting factors is 1. The values of individual tissue weighting factor have been modified in to reflect the changes in information regarding the incidence of stochastic effects following radiation exposure. The current values of tissue weighting factors are published in ICRP Publication 60.

Effective Dose Equivalent and Effective Dose

The effective dose, E, is defined as the product of equivalent dose and appropriate tissue weighting factors, summed over all exposed tissues and organs.

$$E = \sum_T w_T \cdot H_T \tag{2.4}$$

Like the equivalent dose, the effective dose also has units of joules per kilogram bearing the name sievert.

[2]detriment - a measure of the total harm that would eventually be experienced by an exposed group of and its descendants as a result of the group's exposure to a radiation source (ICRP, 1991)

Historically, the value of effective dose was termed effective dose equivalent. However, with the publication of ICRP 60, this rather complicated name was simplified to effective dose.

2.2.2 Operational Quantities

The absorbed dose, the equivalent dose and the effective dose are fundamental dosimetric quantities related to radiation protection. However, the measurement of radiation fields for protection purposes employs the use of operational quantities; quantities which have been defined by the International Commission on Radiation Units and Measurements (ICRU). Operational quantities serve to standardize absorbed dose measurements, which yield effective dose. There are two types of monitoring which have separate operational quantities: area monitoring and individual monitoring.

Area Monitoring

There are two quantities of particular interest to area monitoring. The first, used for fields of strongly penetrating radiations such as neutrons, is the ambient dose equivalent $H^*(d)$. The second, used for fields of weakly penetrating radiation, is the directional dose equivalent $H'(d,\Omega)$. Both the ambient dose equivalent and the directional dose equivalent, like the primary dosimetric quantities, are measured in joules per kilogram and also bear the unit sievert. ICRU Report 51 outlines the following definitions (ICRU, 1993):

> The ambient dose equivalent $H^*(d)$, at a point in the radiation field, is the dose equivalent that would be produced by the corresponding expanded and aligned field, in the ICRU sphere at a depth, d, on the radius opposing the direction of the aligned field.
>
> The directional dose equivalent, $H'(d,\Omega)$, at a point in the radiation field, is the dose equivalent that would be produced by the corresponding expanded field, in the ICRU sphere at a depth, d, on a radius in a specified direction, Ω.

These quantities are measured in a phantom approximating the human body. The ICRU sphere is a phantom; a 30 centimeter diameter tissue equivalent sphere with a density of

1 gram per cubic centimeter. By mass it is composed of 76.2% oxygen, 11.1% carbon, 10.1% hydrogen, and 2.6% nitrogen. Both measurements require the specification of the reference depth; a depth expressed in millimeters, which for strongly penetrating radiations is 10 mm. For weakly penetrating radiation the reference depth is 0.07 mm for skin and 3 mm for the eye. In area monitoring the radiation fields are derived from the actual field such that they can be characterized as expanded and/or aligned. In the expanded field, the fluence and its angular and energy distribution have the same values throughout volume of interest as in the actual field at a point of reference (ICRU, 1993). In the case of an aligned field the fluence has no angular dependence; it is isotropic.

Individual Monitoring

For individual monitoring, a single operational quantity is used. The personal dose equivalent, $H_p(d)$, is used for both strongly and weakly penetrating radiations; the variation occurring in the reference depth employed in the measurement. It is defined in ICRU Report 51 such that

> The personal dose equivalent, $H_p(d)$, is the dose equivalent in soft tissue, at an appropriate depth, d, below a specified point on the body.

The personal dose equivalent is measured in joules per kilogram and bears the unit sievert. The reference depths are also specified in millimeters such that for weakly penetrating radiations depths of 0.07 mm for the skin and 3 mm for the eye are employed while for strongly penetrating radiations a depth of 10 mm is used. This value is measured by wearing a detector at the surface of the body. By covering it with an appropriate amount of tissue-equivalent material, the specific reference depth can be simulated.

Determining Dosimetric Quantities

The relevant quantity which is required to perform dosimetric calculations will directly influence the type of detection system which can be employed. As there are a variety of detectors and monitors, measurement techniques and practices, which can used to yield relevant data, understanding these tools and the information they provide, ensures that an appropriate system is in place when required.

2.3 Determining Dose

Measuring quantities directly, rather than determining them through calculation, has always been preferred. However, as ideal as measurement is, it is often difficult to measure operational quantities because of the requirements associated with their definitions. For example, the ambient dose equivalent is defined as a measurement of the equivalent dose in an expanded and aligned field; conditions which are rare in normal operations. The energy dependence of the radiation weighting factor—the factor used to calculate the equivalent dose—introduces another complication; a difficulty particularly relevant in the case of neutrons. Therefore it is often necessary to measure basic physical quantities from which operational quantities can be calculated.

The basic physical quantity for neutrons is the neutron fluence. Defined as the number of neutrons that pass through a unit cross-sectional area, the neutron fluence can also be distributed in energy and direction. Using the fluence-to-dose conversion coefficients, the neutron fluence, and its various distributions, can be related to the defined operational quantities.

2.3.1 Fluence-to-Dose Conversion Coefficients

The absorbed dose, equivalent dose and effective dose are measures of the amount of energy deposited within a medium. Through a variety of interactions, the incident radiation transfers energy to the material. In the case of neutrons and ICRU tissue, the primary interaction, at lower neutron energy, is elastic scattering between the neutrons and hydrogen. There are two other interactions which also contribute to the deposition of energy within the ICRU tissue material; the probability of these interactions increases as the neutron energy decreases. Within the large volume of material, higher energy neutrons become moderated making them available for further interaction via one of the following two paths. The first is the $^{14}N(n,p)^{14}C$ reaction in which neutrons interact with nitrogen atoms to produce 0.6 MeV protons, while the second is the $^{1}H(n,\gamma)$ reaction which produces 2.2 MeV photons. These secondary radiations contribute significantly to the amount of energy deposited; a contribution which raises the value of the fluence-to-dose coefficient.

The fluence-to-dose conversion coefficients are of great importance in calibrations; particularly the calibration of dosimeters and survey meters. For continuous neutron spectra, the energy dependent, fluence-to-dose conversion coefficients must be averaged over the entire energy range to yield a single conversion coefficient. The spectrum averaging can be evaluated using the following formula (ICRU, 2001):

$$h_{\Phi} = \frac{\int h_{\Phi}(E)\Phi_E(E)dE}{\int \Phi_E(E)dE}. \tag{2.5}$$

From this value, the calibration coefficient can be determined such that the monitoring method, measuring basic physical quantities, now yields relevant operational quantities.

2.3.2 Relating Operational Quantities to Protection Quantities

The International Commission on Radiation Protection has defined protection quantities such as the equivalent dose and the effective dose. These quantities are essential for dose control and dose limitation. However, the operational quantities established by the International Commission on Radiation Units and Measurements are essential in dose measurement, and serve to directly estimate the value of specific protection quantities. The ambient dose equivalent and the personal dose equivalent, measured at a tissue depth of 10 mm, approximate the value of effective dose. On the other hand, the value of personal dose equivalent, at a depth of 0.07 mm, can be used as the skin equivalent dose; a dosimetric quantity implemented to protect the skin from non-stochastic effects.

As well as defining protection quantities, the ICRP has recommended limits on radiation exposure. These limits, adopted by individual nations and incorporated into legislature, protect against the possibility of harm resulting from radiation exposure.

2.4 Dose Limits

In 1990, the ICRP revised previous dose limits, publishing updated recommendations in "1990 Recommendations of the International Commission on Radiological Protection" (ICRP, 1991). It has been shown that there are two types of effects resulting from radiation exposure: stochastic effects and deterministic effects. As a result, limits on the effective dose serve to minimize the likelihood of stochastic effects, while limits on equivalent dose eliminate the possibility deterministic effects.

2.4.1 Effective Dose Limits

The effective dose limits are based on limiting the total detriment to an exposed population; detriment being a measure of the total harm that would eventually be experienced by an exposed group of individuals and its descendants, as a result of the group's exposure to a radiation source (ICRP, 1991). Although there is little direct evidence of stochastic effects due to low levels of radiation exposure, radiation protection is guided by the Linear No-Threshold (LNT) model. This model extrapolates the observed dose response relationship in highly exposed individuals to provide estimates of the risk of harmful effects at lower levels of exposure.

The current ICRP effective dose limits include specifications for individuals who are exposed in the course of their occupation, as well as a separate limit for individuals in the public. The occupational effective dose limit is specified to be 20 mSv per year, averaged over a defined period of 5 years. A further stipulation states that the effective dose should not exceed 50 mSv in any single year. The public effective dose limit is much lower; 1 mSv per year.

2.4.2 Equivalent Dose Limits

The equivalent dose limits are guided by the threshold dose for non-stochastic effects. Equivalent doses below the threshold dose for a particular non-stochastic effect will not result in the appearance of that effect. Currently, the ICRP specifies annual equivalent dose limits for three area of the body: the lens of the eye, the skin, and the extremities.

To prevent the formation of cataracts in the eyes, the equivalent dose limit for the eyes is 150 mSv per year. For the skin, the equivalent dose limit is 500 mSv per year, averaged over any 1 cm^2, regardless of the area that has been exposed. The equivalent dose limit for the extremities—the hands and feet—is also 500 mSv per year. In the case of the skin and the extremities, the limits prevent erythema—the reddening of the skin—and severe damage of the skin tissue.

Chapter 3

Special Considerations for Neutrons

The properties of the neutron provides challenges with respect to neutron detection and dosimetry. Because the neutron is uncharged it cannot be detected using more traditional methods of radiation detection which rely on electric or magnetic fields. However, neutron interactions with matter create tracks of ionization that can be detected. In this case, the detector relates the number of secondary charges created during interaction to the number of neutrons incident on the material by directly measuring the secondary particles.

The wide range of neutron energy spectra—as much as eight orders of magnitude—affects the detection method. In one method, detection over a wide range involves modification to the typical detection system to include a moderator. This modification results in the loss of energy spectral information; limiting detection to quantifying the presence of neutrons without attempting to determine the energies.

Calculating the equivalent dose resulting from neutron exposure is dependent on the physical quantity measured through detection. While a number of detection systems yield relevant dosimetric quantities directly, others measure quantities which need to be appropriately weighted. Oftentimes this weighting accounts for radiation type; a radiation weighting factor which, in the case of neutrons, is dependent on energy. As a result, dosimetric quantities related to neutrons are also energy dependent. Therefore, in detection systems which do not quantitatively determine neutron energy, it is important to characterize the neutron fields so that the dose due to neutrons can be determined accurately.

The challenges associated with neutron dosimetry arise due to a number of factors. The properties of the neutron, as well as the wide range of energies available to them, affect neutron detection methods. Energy dependent weighting factors used to calculate dosimetric quantities require that the energy spectrum of a particular field be known. Otherwise, the detection system itself must be calibrated to yield the relevant dosimetric quantities directly at the expense of spectral information. Neutrons constitute the most important radiation for which dosimetry considerations must take into account both the radiation quantity and the energy dependent weighting factors. Although doses resulting from neutron exposures are only a small fraction of the total dose within the CANDU workplace, the energy dependent relationship between neutron exposure and the potential for harm must be considered in the development of any dosimetry program.

3.1 Neutron Monitoring Systems, Measurement Techniques and Practices

Monitoring is the measurement of radiation, or activity, in order to estimate or control the exposure to radiation or radioactive materials by relating measured quantities to relevant dosimetric quantities. It is the basis of any dosimetry program, providing the data required for calculating individual worker doses. Not limited to the detection and measurement of radiation, monitoring methods also include aspects such as the operation of the detector, calibration, measurement techniques and operational practices.

Choosing an appropriate monitoring method is very difficult; each method having its own advantages and disadvantages. However, this is a choice which also depends on the specific application and environment for which monitoring is required. Therefore when selecting a monitoring method it is necessary to consider a number of factors, not only those related to the performance characteristics of the detection system, also but factors dependent on the radiation environment. As outlined in ICRU Report 66 (ICRU, 2001), examples of these considerations include the following:

- operational quantity to be measured;
- radiation field characteristics;

- required detection limit;
- need for immediate reading;
- length of monitoring period;
- environmental conditions;
- size and weight of device;
- user acceptance; and
- cost.

The frequency of monitoring will be determined by the conditions of the radiation field; when fluctuations are expected in the radiation field, monitoring will be more frequent. Direct monitoring—the direct measurement of the radiation field through area or personal monitoring—is essential in environments where there is an increased dose. However, indirect monitoring—the use of maps and tables of previously measured dose rates, or the use of the ratio of neutron-to-gamma ray dose rates, to calculate neutron equivalent doses—can be used in area where the radiation fields are known to be stable. The specific methods of monitoring associated with direct and indirect monitoring will vary. As a result, each case must be addressed individually.

3.1.1 Direct Monitoring

Direct monitoring is necessary in those environments where radiation exposures are high or where there is an increased probability for the field to fluctuate to hazardous levels. Including area monitoring and personal monitoring, direct monitoring provides accurate measurement of the radiation field at the time of measurement.

Area Monitoring

Area monitoring is the measurement of radiation fields using portable survey meters or fixed area monitors. As the dose resulting from exposure to neutrons generally contributes only a small fraction of the total dose, it is often not necessary to make individual measurements of this contribution. Rather, a measurement of the neutron radiation field in a particular area is sufficient enough to yield relevant dosimetric information to calculate the neutron dose.

Moderator-based survey meters—also known as rem-meters—are the most common, active, portable neutron survey meters. Although there are a variety of designs, these instruments are constructed so the shape of their fluence response as a function of energy approximates the fluence-to-ambient dose equivalent conversion coefficient. As a result of this, and their calibration, the measurement of neutron fluence provides the ambient dose equivalent. Although limited by their cumbersome size and weight, the use of neutron rem-meters is the sole method of active neutron monitoring in place within OPG.

Less commonly used than moderator-based survey meters, other survey meters such as tissue-equivalent proportional counters (TEPC's) have also shown to be effective. However, even these systems are not without their own disadvantages. The principle of their operation is similar to that of the moderator-based instruments; each is calibrated to relate some physical measurement to yield a dosimetric quantity. In the case of the TEPC the measured pulse height spectrum is used to provide the absorbed dose as a function of lineal energy.

Personal Monitoring

Personal monitoring is used to obtain an estimate of the mean effective and equivalent doses in significantly exposed tissues. The monitors are generally worn—by each individual—over the part of the body for which the dose is to be determined. Unlike area monitors, which tend to be active devices, personal monitors are generally passive, making them suitable for long-term monitoring at low dose rates.

Albedo thermoluminescence dosimeters (TLD's), superheated-emulsion detectors and track-etch detectors are the most commonly used personal neutron dosimeters. The relevant dosimetric quantity with respect to personal monitoring is the personal dose equivalent. Albedo TLD's are calibrated such that the light emitted by the thermoluminescent material is related to this dose, superheated-emulsion detectors relate the number of bubbles formed within the detector with absorbed dose and track-etch detectors are calibrated to relate the number of tracks formed to the number of particles and their energy from which personal dose equivalent can be calculated. However, as useful as these dosimeters have proven to be in a number of radiation environments, the use of personal monitoring for neutrons is generally limited.

Currently, the OPG neutron dosimetry program does not include personal monitoring.

3.1.2 Indirect Monitoring

In environments where the radiation field is known to be stable, indirect monitoring provides enough information for the calculation of equivalent dose. Neutron fields in specific areas of the CANDU workplace vary only according to the power at which the reactor operates. When the reactor is initially started, each area is surveyed—using an active survey meter—to determine the neutron ambient dose equivalent and neutron ambient dose equivalent rate present. Recorded on area maps, this information is used to calculate the individual equivalent dose incurred by each worker present within the area.

In addition to mapping existing neutron radiation fields, it is also possible to use the neutron-to-γ-ray dose rate ratio as a method of indirect monitoring. In each area, the ratio of the neutron ambient dose equivalent rate to the γ-ray dose rate would be measured an recorded using the ratio. The neutron dose would be determined from the γ-ray dose as measured by a Dose Control Device (DCD) such as an Electron Personal Dosimeter (EPD).

The neutron dosimetry program within OPG includes the use of indirect monitoring within those areas whose neutron fields are known, and well documented or where the neutron-to-γ-ray dose rate ratio has been determined.

3.2 Neutron Dosimetry

For neutron dosimetry, the current requirements are less defined as a result of two factors. The first is due to the challenges presented by the neutron. Unlike photons and β-particles, for which detection and dosimetry is relatively straightforward, neutron dosimetry is complicated by the properties of the neutron. The second, is a factor which arises due to the nature of neutron doses. Generally, the radiation dose resulting from neutrons is small; making it neither practical nor relevant to establish structured requirements.

However, there are impending revisions to S-106 which include specifications regarding the requirements of neutron dosimetry (CNSC, 2005). While remaining less stringent that those for photon and β- particle dosimetry, the new requirements will dictate the standards regarding documentation, calibration and accuracy for neutron dosimetry services. Until the revised version of S-106 is formally issued, the precise requirements remain unavailable.

Chapter 4

Dosimetry Practices at Ontario Power Generation

At Ontario Power Generation CANDU nuclear reactors, the radiation doses of workers arise from both external and internal sources. On average for those workers exposed to neutrons, 70% of the total committed effective dose is received externally; of that, the greatest fraction is due to β- and γ- radiations, with a small contribution resulting from neutrons. The following table summarize the results of external dosimetry measurements at OPG over the past three years.

Location Site	Year	Total Whole Body Dose μSv (mrem)	Total Neutron Dose μSv (mrem)	% Contribution due to Neutrons
DNGS	2001	2673130 (267313)	4760 (476)	0.18
DGNS	2002	2520800 (252080)	3830 (383)	0.15
DNGS	2003	3075000 (307500)	2570 (257)	0.08
PNGS	2001	4315790 (431579)	9720 (972)	0.22
PNGS	2002	4130430 (413043)	11400 (1140)	0.28
PNGS	2003	4171460 (417146)	24170 (2417)	0.58
TOTAL		20886610 (2088661)	56450 (5645)	0.27

Table 4.1: Summary of all external dosimetry measurements for Darlington and Pickering Nuclear Generating Stations from 2001 to 2003.

For the entire site, not all workers who receive radiation dose will be exposed to neutron radiations. As a result, the site average percent contribution appears very small because of the large population of workers who received external radiation dose, but who did not receive any neutron dose.

Internal exposure accounts for the remaining 30% of the total committed effective dose; of this 95% is due to the intake of tritiated water with the additional component arising from the contribution of the intake of 14-Carbon, mixed fission products and actinides.

During routine operations at CANDU nuclear generating stations the effective doses due to neutrons received by workers are very small; on average the contribution of neutrons to the total effective dose, for those who receive neutron exposures is roughly 4.5%.

Radiation dosimetry for external exposures—particularly exposures due to β- and γ-radiations—is relatively straightforward. Thermoluminescent dosimeters (TLD's) are widely used in nuclear power stations. In comparison, radiation dosimetry for neutron radiation is much more difficult; due to both their physical properties and the wide range of energies available to neutrons. For internal exposures, dosimetry includes routine whole body counting and the analysis of urine bioassay samples through liquid scintillation counting (LSC).

4.1 Neutron Dosimetry Practices

There are a variety of techniques available for neutron dosimetry; techniques which can be classified as direct or indirect dosimetry methods. The neutron dosimetry program within OPG employs the use of both direct and indirect methods.

4.1.1 Direct Neutron Dosimetry

Ontario Power Generation has outlined the conditions for the use of direct neutron dosimetry (Lamothe, 2003). Neutron doses are measured directly when there is an increased probability for exposure. Such circumstances include work areas where neutron doses are known to be elevated and areas where indirect dosimetry is unavailable.

Moderator Based Survey Meters - "Neutron Rem Meters"

Moderator based survey meters are the most common instruments used in neutron dosimetry. Consisting of a thermal neutron detector surrounded by a moderating material, typically polyethylene, rem-meters are active, hand-held instruments calibrated to relate the number of neutrons to the ambient dose equivalent. Although the moderating material makes the response of the meter independent of energy and increases the sensitivity to neutrons, it also contributes to the short-comings of the detector; increasing the meter's size and weight. There are two main types of neutron rem-meters—the cylindrical Andersson-Braun type and the spherical Bonner sphere type—it is the Andersson-Braun rem meter which is used at OPG.

The development of the Andersson-Braun (A-B) type neutron rem-meter, also called SNOOPY, was first described by Andersson and Braun in 1963 at the International Atomic Energy Agency symposium on neutron dosimetry (Andersson and Braun, 1963). This type of rem-meter is composed of a cylindrical BF_3 proportional counter surrounded by a moderator assembly. The moderator assembly is formed of two layers of polyethylene plastic separated by a perforated, boron-loaded plastic layer. The thickness of the two polyethylene layers as well as the number of holes in the boron layer is experimentally determined to provide appropriate amounts of moderation and absorption. The result is that the shape of the detector response as a function of energy approximates that of the fluence to ambient dose equivalent conversion coefficient.

It is the calibration which converts the neutron count rate to the ambient dose equivalent rate; enabling the SNOOPY to provide measurements of both ambient dose equivalent and ambient dose equivalent rate. Currently, the SNOOPY's used at OPG are calibrated at the National Research Council Canada (NRCC), Institute for National Measurement Standards. The calibration begins by determining a spectrum-averaged, fluence-to-dose conversion coefficient. The NRCC employs a computer code written by Dr. Len Van der Zwan which folds a neutron energy spectrum with the tabulated vales of $h^*_\phi(E)$, where

$$h^*_\phi = \frac{H^*(10)}{\phi}. \qquad (4.1)$$

Using the standard ^{241}Am-Be spectrum and the values of $h^*_\phi(E)$ published in ICRU Report 57 and shown in Figure 4.1, a single fluence-to-dose conversion factor, $\hat{h}^*_\phi = 3.9 \times$

$10^{-10} Sv\ cm^2$, is determined. This factor, while independent of the particular detector being calibrated, depends on the calibration spectrum and will vary between neutron sources. The total neutron fluence rate at any distance is determined from the emission rate of a source as well as its anisotropy. Multiplying the total neutron fluence rate and $\hat{h}^*_\phi$ provides the "true" ambient dose equivalent at that point. The response of the SNOOPY is known to be directionally dependent. The NRCC places the instrument with its axis normal to the cylindrical neutron source such that the neutrons are incident upon the flat face of the cylinder. The ambient dose equivalent is measured at distances ranging from 30 cm to 150 cm; distances measured between the centre of the source and the centre of the moderating cylinder. Comparing the measured ambient dose to the expected "true" ambient dose results in the calibration factor for the instrument, applicable when the survey meter is exposed to neutrons in the calibration geometry. In order to eliminate effects of both room and air scatter, a number of geometry corrections are applied to make the calibration factor independent of the characteristics of the calibration facility.

The response characteristics of A-B type rem-meters vary with respect to a number of factors. First, as a result of its cylindrical shape, the detector response varies with its orientation. When the neutrons are incident on the side of the cylinder, perpendicular to the axis of the detector, the response is 25% higher than when the neutrons are incident on the flat face of the cylinder (Saull, 2003). As outlined, the calibration factor is determined through the measurement of the total neutron fluence at a point for neutrons incident on the flat face of the cylinder. In this orientation the response of the detector is lower than for the orientation which sees neutrons incident on the curved sides. Therefore, measurements made with neutrons incident on this curved face, will overestimate the effective dose due to neutrons by approximately 25%. OPG's operating procedure for the SNOOPY is to rotate the device to find the maximum dose reading. The maximum dose reading, occurring when the neutrons are incident on the side of the detector, will therefore be an overestimate of the ambient dose equivalent; recorded doses will be conservative, overestimating the dose resulting from neutron exposure by approximately 25%. Second, the construction of the SNOOPY is such that its response, as a function of neutron energy, approximates the shape of the fluence-to-dose conversion coefficient as a function of energy. The response, as provided by the manufacturer, is shown in Figure 4.2. However, this energy-dependent

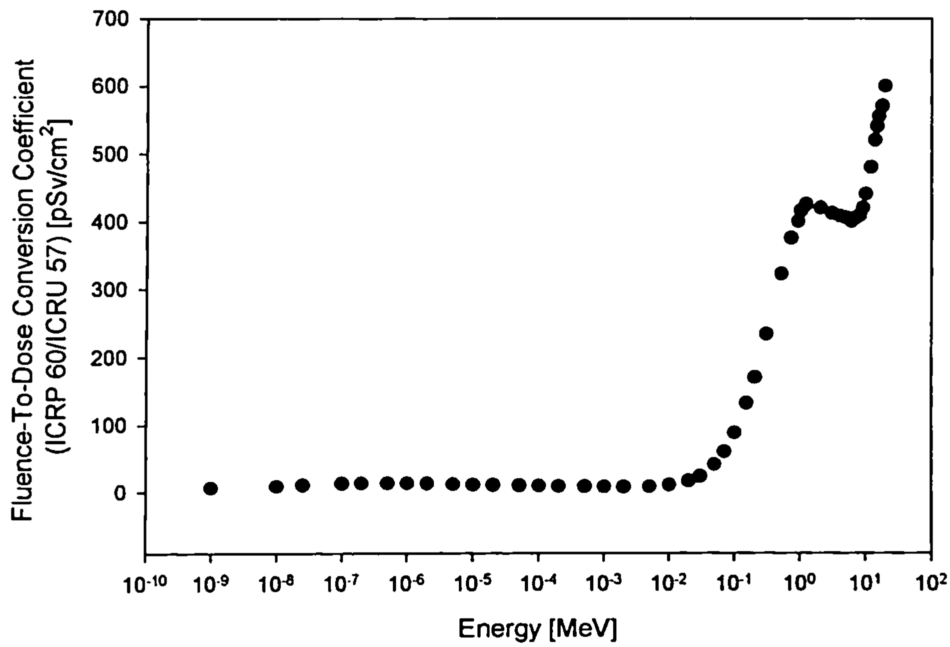

Figure 4.1: The Fluence-To-Dose Conversion Coefficients (ICPR 60/ICRU 57) (ICRP, 1991).

response, is still only an approximation and as a result there remain known inaccuracies. At low energies, below 20 keV, there is an over response of roughly a factor of 3, while at energies above 500 keV there is an approximately 25% under response (Rogers, 1979). These values represent the response of the SNOOPY when using the fluence-to-dose conversion coefficient, and hence the quality factors, is use at the time of measurement (1979). The effect of the energy dependence will vary according to the neutron field in which the survey meter is used. In low energy fields, such as those generally encountered in the CANDU workplace, the energy dependent response results in an overestimate of the ambient dose equivalent. However, in the case of high energy fields, the ambient dose equivalent will be underestimated.

Despite its effectiveness as a neutron survey meter, there is a clear disadvantage associated with the SNOOPY. With a length and diameter of approximately 30 cm and 25 cm respectively, and weighing roughly 10 kg, the SNOOPY can be cumbersome and challenging to use. The majority of concerns related to the use of the SNOOPY, are based on its size and weight. However, these concerns aside, OPG's practice of using the highest SNOOPY reading and NRCC's calibration procedure result in a 25% over-response when SNOOPY is exposed to an unmoderated ^{241}Am-Be neutron source.

4.1.2 Indirect Neutron Dosimetry

Within the CANDU workplace, the neutron radiation fields are relatively stable over time, varying only with the reactor operating power. As a result, indirect neutron dosimetry can be used to simplify neutron dosimetry. Indirect dosimetry consists of measuring quantities which are easier to measure, but which are related to the quantities of neutron dosimetry. OPG employs two types of indirect neutron dosimetry. The first uses maps or tables of measured neutron ambient dose equivalent rates and the time an individual occupied the area. The second uses the γ-ray dose rate and the measured ratio of neutron-to-γ ray dose rates to calculate the ambient dose equivalent resulting from neutron exposure.

The neutron radiation fields have been mapped for the Pickering Nuclear Generating Station, making indirect neutron dosimetry available. Using the SNOOPY, the neutron ambient dose equivalent rate has been measured for a number of locations throughout the station. Tables and maps of these values, and the amount of time an individual occupies each area, are used to calculate the ambient dose equivalent from the product of occupation time and ambient dose equivalent rate.

The Darlington Nuclear Generating Station is implementing the use of neutron-to-γ-ray dose rate ratio as method of indirect neutron dosimetry. Here the neutron dose would be determined directly from the γ-ray dose, as measured by a DCD. This method incorporates the ease of use associated with neutron dose rate tables and maps, but also adds a direct measurement component which accounts for possible variations in the neutron radiation field as a result of the changing γ-ray radiation field.

The process of establishing a system of indirect neutron dosimetry, although straightforward, is time consuming. First, areas where there is the potential for both the presence of workers and neutron hazards must be identified and marked on workplace maps. Then, using a γ-meter to assure safe γ-ray doses and a SNOOPY to measure the ambient dose equivalent rate, the relevant readings are taken at each location. Once completed, these measurements can be summarized on both workplace maps and in tabular form.

There are a number of advantages with regards to indirect neutron dosimetry. First, it is much simpler to use, requiring only the amount of time within each area or the γ-ray dose rate in order to calculate dose. This makes the continued use of the SNOOPY unnecessary and the process of neutron dosimetry much faster. However, indirect neutron dosimetry may not be able to account for variations in the neutron radiation field. Therefore, if the neutron ambient dose equivalent rates were to change, unexpectedly, it would be impossible to provide warning of the increased dose and dose rate. This is particularly true when using maps or tables of measured neutron ambient dose equivalent rates.

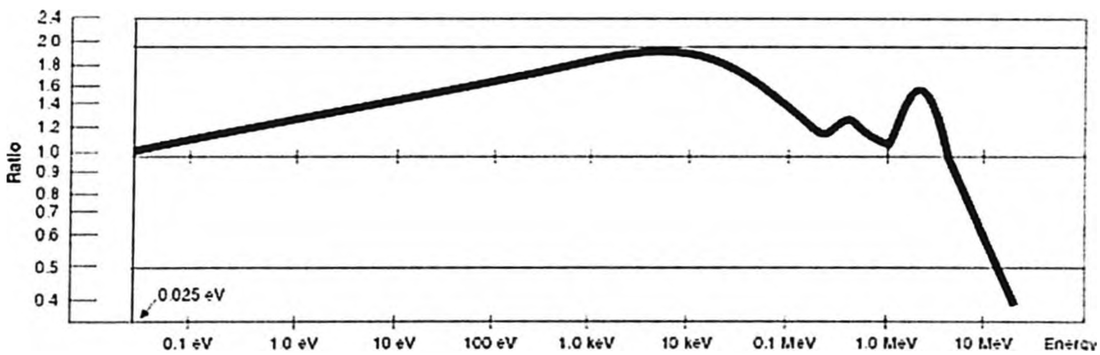

Figure 4.2: The energy dependent response of the SNOOPY (Canberra Instruments Inc., 2003).

Chapter 5

Analysis of Calibration

The SNOOPY rem-meters used at OPG are calibrated by the NRCC, Institute for National Measurement Standards. The result of the calibration is a factor which is used to relate the neutron count rate to the ambient dose equivalent rate, using a standard Am-Be neutron source. However, the calibration factor should not be dependent on the neutron source used in calibration. In fact using the neutron dose conversion coefficients recommended in ICRP 60, the calibration factor can be shown to remain constant—within acceptable limits—regardless of the neutron source used to perform the calibration.

As part of a CANDU Owner's Group Inc. (COG) sponsored initiative, performance testing was completed for the SNOOPY neutron survey meter. Readings from the SNOOPY for six different neutron fields—in six source-detector orientations—were recorded to determine the dose-equivalent response. The six source-detector orientations are shown in Figure 5.1. Using these measurements, and following a procedure similar to that employed by the NRCC in detector calibration, a calibration factor was determined for each of these sources.

5.1 Isotopic Neutron Sources

Measurements were performed using isotopic four neutron sources in the following configurations: Bare ^{241}Am-Be at 50 cm, Bare ^{252}Cf at 50 cm, Bare ^{252}Cf at 100 cm, and D_2O Moderated ^{252}Cf at 100 cm. No correction was made for room-scatter; a particularly

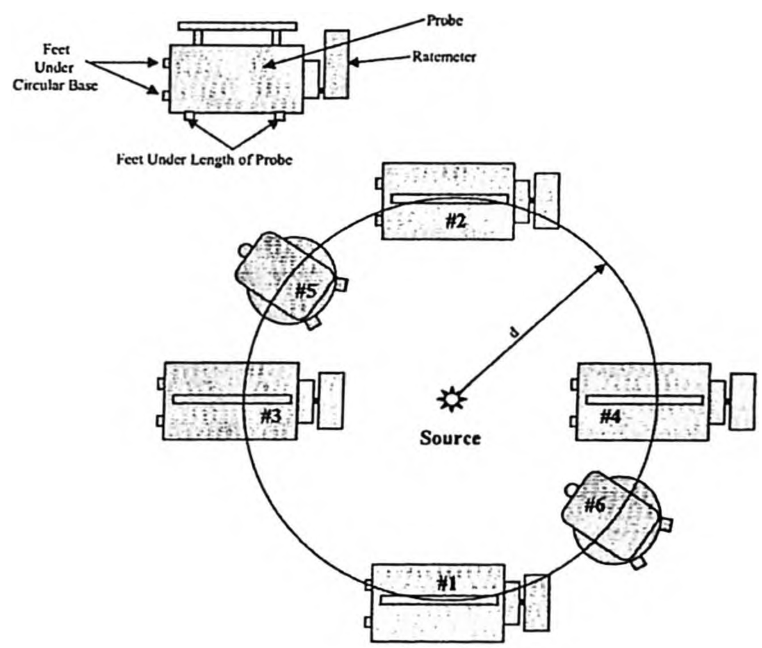

Figure 5.1: The six source-detector orientations (Nunes and Surette, 2004).

important contribution for both Bare ^{252}Cf at 100 cm, and D_2O Moderated ^{252}Cf at 100 cm.

5.1.1 Calibration Procedure

As outlined previously, the NRCC calibration of the SNOOPY begins with the determination of a fluence-to-dose conversion coefficient for the neutron source to be used in the calibration. Then, knowing the characteristics of the neutron source, the "true" ambient dose equivalent at a point can be calculated; the "true" ambient dose equivalent is the product of the fluence-to-dose conversion coefficient and the expected neutron fluence at the measurement point. Using the SNOOPY, the ambient dose equivalent is measured at the same point and factors correcting for room and air scattering may be applied. These factors ensure that the resulting calibration factor is independent of the calibration facility and therefore applicable in any environment. The calibration factor is the ratio of "true" ambient dose equivalent to measured ambient dose equivalent.

The procedure for determining the calibration factors, without access to both the SNOOPY and the relevant neutron source, is based more on calculation using available data than actual measurement. First, using the computer code written by Dr. Len Van der Zwan which folds a neutron energy spectrum with the tabulated vales of $h^*_\phi(E)$, the fluence-to-dose conversion coefficient was determined for each of the six neutron sources. The characteristics of these sources were outlined in the COG Technical Note "Capability Maintenance in Neutron Dosimetry 2003/04 - Performance-testing a Neutron Survey Meter" (Nunes and Surette, 2004). The "true" ambient dose equivalent was calculated from the physical characteristics of the neutron source and is given by:

$$H_{true}(d) = C\frac{\dot{S}}{4\pi d^2}\hat{h}^*_\phi \cdot t \tag{5.1}$$

where $H_{true}(d)$ is the "true" ambient dose equivalent in μSv at the source-detector distance d, C is a unit conversion factor with a value $C = 3.6 \times 10^{-3}$, $\dot{S}$ is the source neutron emission-rate $[s^{-1}]$, $\hat{h}^*_\phi$ is the spectrum-averaged fluence-to-dose conversion coefficient per unit fluence $[pSv\ cm^2]$ and t is the irradiation period in hours. Using the values of SNOOPY-measured ambient dose equivalent included in the COG Technical Note and the calculated "true" ambient dose equivalent, the calibration factor for each neutron source was determined. However, no corrections for room and air scattering were applied.

5.1.2 Bare ^{241}Am-Be at 50 cm

The $^{241}Am-Be$ source was housed in a cylindrical, aluminum can having a 2.5 cm diameter and a 5 cm height. The neutron emission rate, measured in April 2001, was found to be $(5.3\pm0.3)\times10^{6}\ s^{-1}$. With a half-life of 432 years, the neutron emission rate measured in April 2001 must be corrected for decay to the date of measurement, March 2004. The associated neutron spectrum is included in Figure 5.2.

The fluence-to-dose conversion coefficient was calculated using the computer code provided by the NRCC, and the $^{241}Am-Be$ neutron spectrum. The calculated value is

$$\hat{h}^{*}_{\Phi}=3.91\times10^{-10}\ Sv\,cm^{2}=391\ \ pSv\,cm^{2}.$$

The "true" ambient dose equivalent is calculated from the neutron emission rate $(5.275\pm0.29)\times10^{6}\ s^{-1}$ and Equation 5.1, such that $H_{true}(d)=79\pm6\ \mu Sv$.

There are a number of source-detector orientations available for the calibration process. While the response of the detector is greater when the neutrons are incident on the curved surface, the NRCC calibration is performed such that the neutrons are incident on the flat face of the cylinder. In this orientation, the SNOOPY was used to measure the ambient dose equivalent; it was found that $H_{measured}=83.7\pm0.8\ \mu Sv$.

The calibration factor is calculated by taking the ratio of the "true" ambient dose equivalent to the measured ambient dose equivalent, such that

$$\begin{aligned}calibration\ factor&=\frac{H_{true}}{H_{measured}}\\&=\frac{79\pm6\ \mu Sv}{83.7\pm0.3\ \mu Sv}\\&=0.941\pm0.075\end{aligned}$$

5.1.3 Bare ^{252}Cf at 50 cm

The ^{252}Cf source was purchased by Chalk River Laboratories in 2002. Documentation provided with the source outlined its physical characteristics, geometry and encapsulation. On 2002 March 12, the neutron emission rate was found to be $(3.6\pm0.1)\times10^{8}\ s^{-1}$. The half-life of the source is 2.645 years; correcting for decay between 2002 March 12, and the date of measurement 2004 February 10 the neutron emission rate is $(2.18\pm0.06)\times10^{8}\ s^{-1}$. The neutron spectrum of ^{252}Cf is included in Figure 5.3.

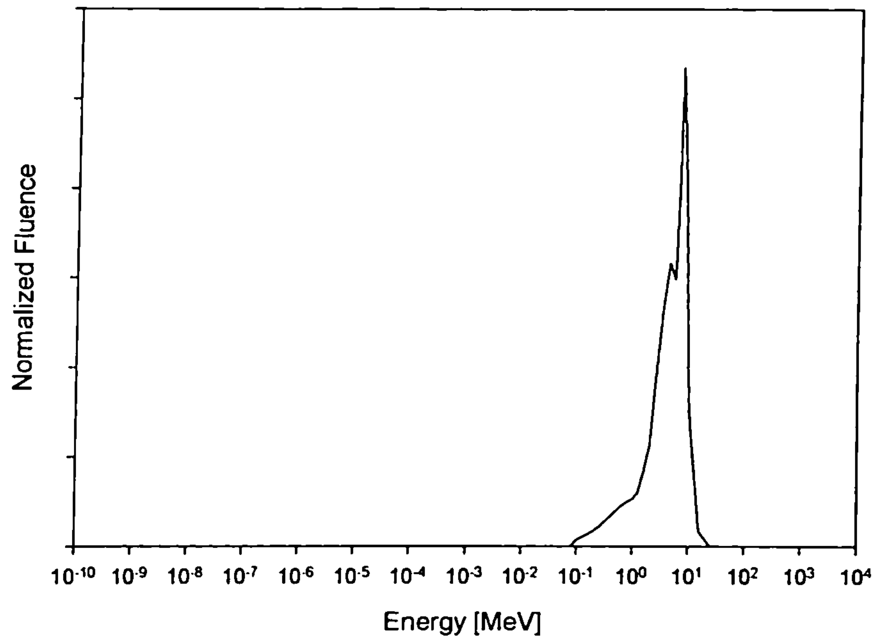

Figure 5.2: The ^{241}Am-Be neutron spectrum (International Atomic Energy Agency, 2001).

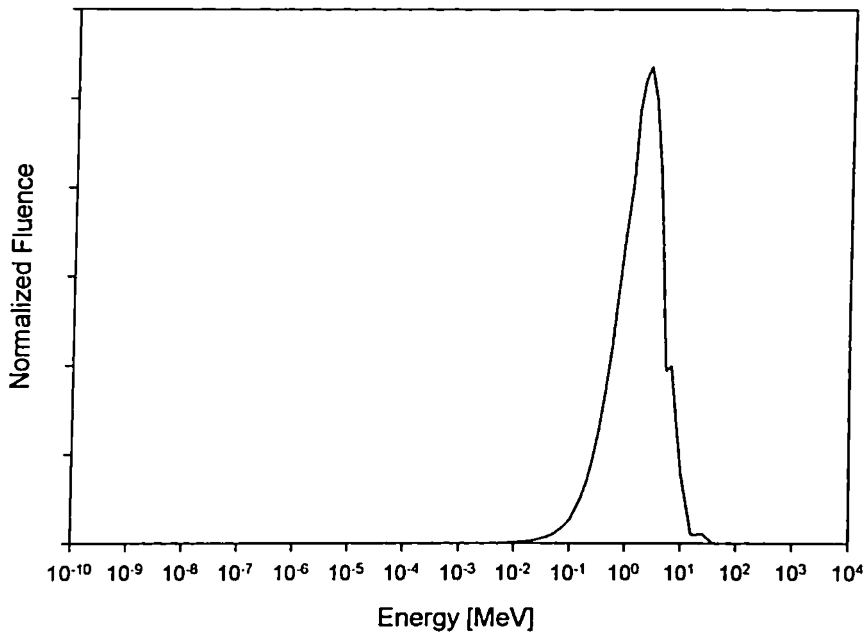

Figure 5.3: The ^{252}Cf neutron spectrum (International Atomic Energy Agency, 2001).

The fluence-to-dose conversion coefficient was calculated using the computer code provided by the NRCC, and the ^{252}Cf neutron spectrum. The calculated value is

$$\hat{h}^*_\phi = 3.85 \times 10^{-10}\ Sv\,cm^2 = 385\ pSv\,cm^2.$$

The "true" ambient dose equivalent is calculated from the neutron emission rate $(2.18 \pm 0.06) \times 10^8\ s^{-1}$ and Equation 5.1, such that $H_{true}(d) = 801 \pm 40\ \mu Sv$.

The ambient dose equivalent—in the orientation such that neutrons were incident on the flat face of the cylindrical detector—was measured with the SNOOPY; it was found that $H_{measured} = 859 \pm 3\ \mu Sv$.

The calibration factor is calculated by taking the ratio of the "true" ambient dose equivalent to the measured ambient dose equivalent, such that

$$\begin{aligned} calibration\ factor &= \frac{H_{true}}{H_{measured}} \\ &= \frac{801 \pm 40\ \mu Sv}{859 \pm 3\ \mu Sv} \\ &= 0.936 \pm 0.050 \end{aligned}$$

5.1.4 Bare ^{252}Cf at 100 cm

The ^{252}Cf source described in 5.1.3 was also measured at a source-detector distance of 100 cm. Using the neutron emission rate of $(2.19 \pm 0.06) \times 10^8\ s^{-1}$ and the calculated value

$$\hat{h}^*_\phi = 3.85 \times 10^{-10}\ Sv\,cm^2 = 385\ pSv\,cm^2.$$

the "true" ambient dose equivalent was re-calculated for this orientation. Following from Equation 5.1 the "true" ambient dose equivalent is given by $H_{true}(d) = 201 \pm 18\ \mu Sv$.

The SNOOPY was used to measure the ambient dose equivalent such that $H_{measured} = 231 \pm 2\ \mu Sv$. As such, the calibration factor is given by

$$\begin{aligned} calibration\ factor &= \frac{H_{true}}{H_{measured}} \\ &= \frac{201 \pm 18\ \mu Sv}{231 \pm 2\ \mu Sv} \\ &= 0.870 \pm 0.086 \end{aligned}$$

The bare ^{252}Cf at 100 cm is not a standard isotopic source. As a result, the "true" ambient dose equivalent does not account for the additional contribution due to room and air scatter or due to variation of ambient equivalent dose with source distance. The calibration factor presented in the COG Technical Note is roughly 10% higher than that which has been calculated here (Nunes and Surette, 2004). By underestimating the value of the "true" ambient dose equivalent, as a result of not including the additional contribution due to room-scatter, the corresponding calibration factor is also underestimated. While there is agreement between the values, inclusion of the room and air scatter in the calculation of the "true" ambient dose equivalent would lead to a better agreement between the values calculated here and those presented in the COG Technical Note.

5.1.5 D_2O Moderated ^{252}Cf at 100 cm

By placing the ^{252}Cf neutron source, described previously, in a stainless-steel spherical shell—with a diameter of 30 cm—filled with heavy water, the neutron field is moderated such that it better resembles those encountered in the CANDU workplace. Again, the ^{252}Cf source has a decay-corrected neutron emission rate of $(2.18 \pm 0.06) \times 10^8\ s^{-1}$. The neutron spectrum for D_2O-moderated ^{252}Cf is shown in Figure 5.4

The fluence-to-dose conversion coefficient was calculated using the computer code provided by the NRC, and the D_2O-moderated ^{252}Cf neutron spectrum. The calculated value is

$$\hat{h}^*_\Phi = 1.11 \times 10^{-10}\ Sv\,cm^2 = 111\ \ pSv\,cm^2.$$

The "true" ambient dose equivalent is calculated from the neutron emission rate $(2.18 \pm 0.06) \times 10^8\ s^{-1}$ and Equation 5.1, such that $H_{true}(d) = 116 \pm 3\ \ \mu Sv$.

Measurement of the ambient dose equivalent was made using the SNOOPY; the resulting value of $H_{measured} = 121 \pm 1\ \ \mu Sv$. Therefore, the calibration factor is given by

$$\begin{aligned} calibration\ factor &= \frac{H_{true}}{H_{measured}} \\ &= \frac{116 \pm 3\ \ \mu Sv}{121 \pm 1\ \ \mu Sv} \\ &= 0.959 \pm 0.033 \end{aligned}$$

The D_2O moderated ^{252}Cf at 100 cm is not a standard isotopic source. As a result,

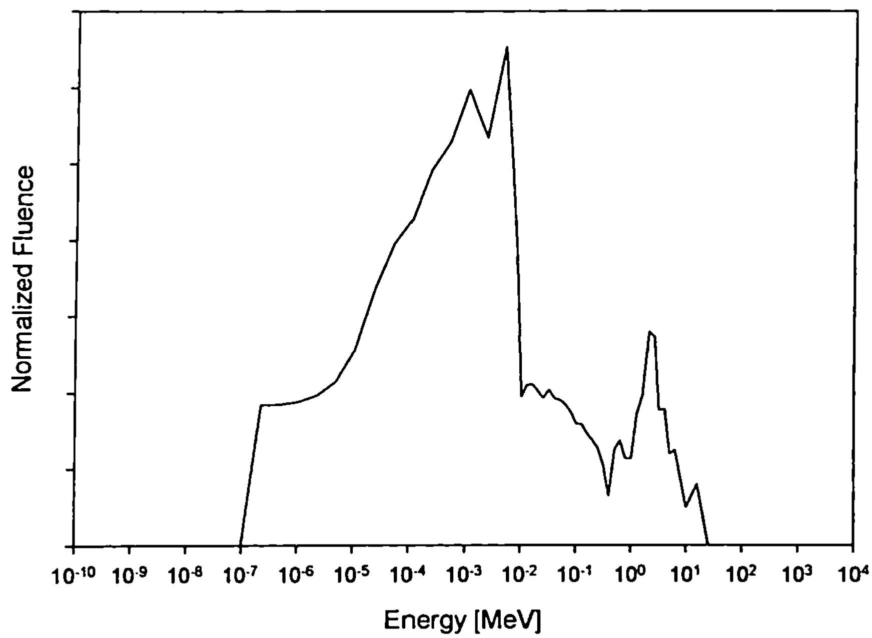

Figure 5.4: The D_2O moderated ^{252}Cf neutron spectrum (International Atomic Energy Agency, 2001).

the "true" ambient dose equivalent does not account for the additional contribution due to room and air scatter or due to variation of ambient equivalent dose with source distance. Again, the calibration factor presented in the COG Technical Note is roughly 10% higher than that which has been calculated here (Nunes and Surette, 2004). By underestimating the value of the "true" ambient dose equivalent, as a result of not including the additional contribution due to room and air scatter, the corresponding calibration factor is also underestimated. Therefore, inclusion of the room-scatter in the calculation of the "true" ambient dose equivalent would lead to a better agreement between the values calculated here and those presented in the COG Technical Note.

5.2 Accelerator Neutron Sources

Following the characterization of neutron fields at several Canadian nuclear generating stations, it became evident that there were significant differences between typical neutron calibration fields and those encountered in the CANDU workplace. By developing neutron fields which better simulate those experienced in reality, more relevant studies and calibration can be completed for dosimeters which would eventually be used in CANDU nuclear generating stations.

There are two CANDU-like neutron fields; CANDU-like I and CANDU-like II, produced at the Neutron Irradiation Facility at Chalk River Laboratories. Deuterium ions were accelerated through a potential of 150 keV, onto deuterium targets, to yield 2.8 MeV neutrons. A custom-made moderator assembly was used to moderate the neutrons in energy and angle to create the CANDU-like I field at a specified point after exiting the assembly. The CANDU-like II field was created by placing a 1 mm thick, cadmium sheet in front of the moderator assembly. This modification reduced the thermal neutron field at the point of measurement.

The CANDU-like neutron fields are not like traditional neutron sources with a known fluence rate; measuring the absolute neutron fluence, while the fields are being used is not possible. Therefore, the calibration procedure varies as compared to that for isotopic neutron sources.

5.2.1 Calibration Procedure

Typically, the calibration of the SNOOPY begins with the calculation of the fluence-to-dose conversion coefficient using the computer code provided by the NRCC. However, the energy spectra of the CANDU-like fields shows that there is a component of neutrons with energies below 0.025 eV; the lower limiting value for which the computer code used to determine the fluence-to-dose conversion coefficient. As a result there is a portion of the CANDU-like neutron spectra which cannot be evaluated using the computer code provided by the NRCC—a challenge which introduces significant errors in the value of the fluence-to-dose conversion coefficient. To overcome this, the value of the fluence-to-dose conversion coefficient, determined in the characterization of the CANDU-like fields was used in place of that value determined by the NRCC computer code.

The characteristics of the neutron source are used to calculated the "true" ambient dose equivalent. Because it is not possible to measure the absolute neutron fluence while the neutron fields are being used, calculating the "true" ambient dose equivalent using Equation 5.1 is also not possible. An external, ^{3}He-filled proportional counter surrounded by polyethylene, is used to measure a 'monitor count rate'. This value is used to normalize spectroscopic and dosimetric measurements in the absence of absolute neutron emission rates. The characterization process also determined the total fluence rate for each CANDU-like field, with respect to the external 'monitor count rate'. From this the "true" ambient dose equivalent rate—in units of $\mu Sv\ h^{-1}$—was found using

$$\dot{H}_{true}(D) = C \cdot \dot{\Phi}_{total} \cdot \hat{h}^{*}_{\phi} \tag{5.2}$$

where, $\dot{H}_{true}(d)$ is the "true" ambient dose equivalent—in units of $\mu Sv\ h^{-1}$—at the source-detector distance d, C is a unit conversion factor with a value $C = 3.6 \times 10^{-3}$, $\dot{\Phi}_{total}$ is the total fluence rate $[cm^{-2}\ s^{-1}]$, and $\hat{h}^{*}_{\phi}$ is the spectrum-averaged fluence-to-dose conversion coefficient per unit fluence $[pSv\ cm^{2}]$. Because the 'monitor count rate' is not constant—rather varying between experiments and measurements—the "true" ambient dose equivalent rate is also normalized to the external 'monitor count rate' yielding a value of ambient dose equivalent rate per monitor count rate. Therefore, the "true" ambient dose equivalent is determined from the 'monitor count rate', the irradiation time and the ambient dose equivalent rate per monitor count rate.

Following this, the procedure was similar to that regarding calibration using isotopic neutron sources. The values of ambient dose equivalent was measured using the SNOOPY; the ratio of "true" ambient dose equivalent to measured ambient dose equivalent provided the calibration factor.

Measurements using this procedure were performed using two accelerator neutron source: CANDU-like Neutron Field I at 100 cm, and CANDU-like Neutron Field II at 100 cm.

5.2.2 CANDU-like Neutron Field I at 100 cm

Produced in the method outlined above, the CANDU-like I Neutron Field can be described by its neutron spectrum shown in Figure 5.5.

In the development of the field, characterization measurements and calculations found the total fluence rate to be $\dot{\Phi}_{total} = 350 \pm 70\ cm^{-2}\, s^{-1}$ and the fluence-to-dose conversion coefficient to be $\hat{h}^*_{\Phi} = (0.94 \pm 0.27) \times 10^{-10}\ Sv\, cm^2 = 94 \pm 27\ pSv\, cm^2$ (Nunes and Faught, 2001). From Equation 5.2 the "true" ambient dose equivalent rate was calculated to be

$$\dot{H}_{true} = 119 \pm 24\ \mu Sv\, h^{-1}.$$

At the time of measurement, the 'monitor count rate' was given by $5381 \pm 73\ counts\, s^{-1}$ giving the ambient dose equivalent rate per monitor count rate a value of $(2.19 \pm 0.22)^{-2}$ $\mu Sv\, h^{-1}$.

Two measurements were made using both the SNOOPY—in the orientation for which the neutrons incident upon the flat-face of the cylinder—and the external monitor. In both measurements the irradiation time was 10 minutes. For the first measurement, the background and dead-time corrected monitor rate was $2066 \pm 6\ s^{-1}$. The measured ambient dose equivalent was $H_{measured} = 6.91 \pm 0.12\ \mu Sv$. The "true" ambient dose equivalent is calculated from the product of the ambient dose equivalent rate per monitor count rate, the monitor count rate and the irradiation time in hours, such that $H_{true} = 7.54 \pm 0.78\ \mu Sv$. The calibration factor for the first experiment is calculated by taking the ratio of the "true"

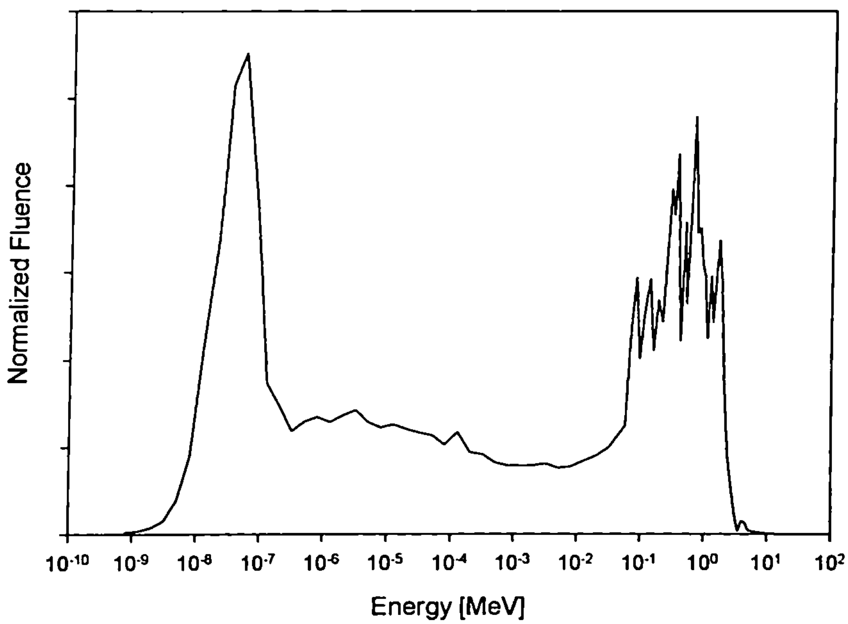

Figure 5.5: The CANDU-like I Neutron Field neutron spectrum (Nunes and Faught, 2001).

ambient dose equivalent to the measured ambient dose equivalent, such that

$$\begin{aligned} calibration\ factor &= \frac{H_{true}}{H_{measured}} \\ &= \frac{7.54 \pm 0.78\ \mu Sv}{6.91 \pm 0.12\ \mu Sv} \\ &= 1.091 \pm 0.132 \end{aligned}$$

For the second measurement, the background and dead-time corrected monitor rate was $3353.4 \pm 7.9\ s^{-1}$. The measured ambient dose equivalent was $H_{measured} = 10.92 \pm 0.16\ \mu Sv$. The "true" ambient dose equivalent is calculated from the product of the ambient dose equivalent rate per monitor count rate, the monitor count rate and the irradiation time in hours, such that $H_{true} = 12.24 \pm 1.41\ \mu Sv$. The calibration factor for the second experiment is calculated by taking the ratio of the "true" ambient dose equivalent to the measured ambient dose equivalent, such that

$$\begin{aligned} calibration\ factor &= \frac{H_{true}}{H_{measured}} \\ &= \frac{12.24 \pm 1.41\ \mu Sv}{10.92 \pm 0.16\ \mu Sv} \\ &= 1.121 \pm 0.146 \end{aligned}$$

The average calibration factor is

$$calibration\ factor = 1.106 \pm 0.015$$

5.2.3 CANDU-like Neutron Field II at 100 cm

The CANDU-like II neutron field can be described by the following neutron spectrum, shown in Figure 5.6.

Characterization measurements and calculations, performed during the development of the neutron field, found the total fluence to be $\dot{\Phi}_{total} = 288 \pm 58\ cm^{-2}\, s^{-1}$ and the fluence-to-dose conversion coefficient to be $\hat{h}^*_{\Phi} = (1.12 \pm 0.32) \times 10^{-10}\ Sv\, cm^2 = 112 \pm 32\ \ pSv\, cm^2$ (Nunes and Faught, 2001). Using Equation 5.2 the "true" ambient dose equivalent rate was calculated to be

$$\dot{H}_{true} = 116 \pm 23\ \mu Sv\, h^{-1}.$$

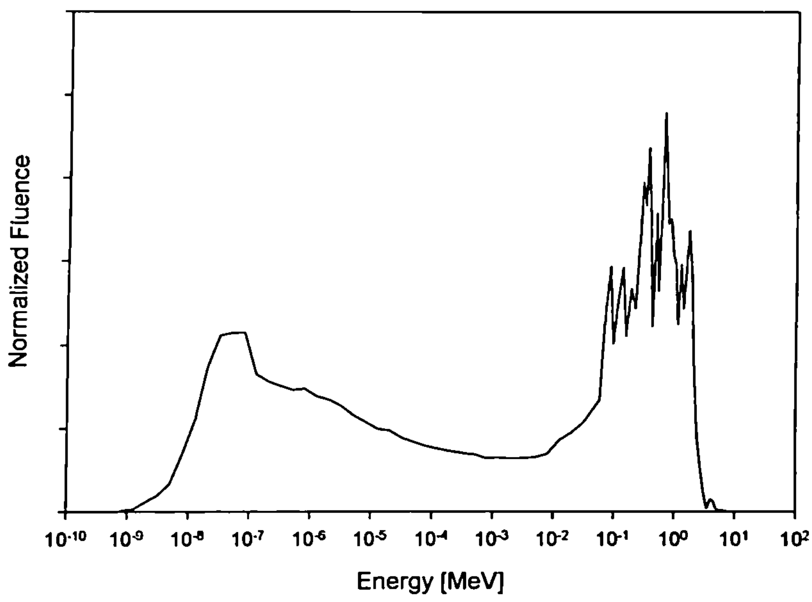

Figure 5.6: The CANDU-like II Neutron Field neutron spectrum (Nunes and Faught, 2001).

At the time of measurement, the 'monitor count rate' was given by 5381 ± 73 $counts\, s^{-1}$ giving the ambient dose equivalent rate per monitor count rate a value of $(2.19 \pm 0.22)^{-2}$ $\mu Sv\, h^{-1}$.

Like the CANDU-like I neutron field, the CANDU-like II neutron field was used to complete two measurements in the relevant orientation. For both measurements the irradiation time was 10 minutes. For the first measurement, the background and dead-time corrected monitor rate was 3675.5 ± 8.4 s^{-1}. The measured ambient dose equivalent was $H_{measured} = 11.89 \pm 0.19$ μSv. The "true" ambient dose equivalent is calculated from the product of the ambient dose equivalent rate per monitor count rate, the monitor count rate and the irradiation time in hours, such that $H_{true} = 13.42 \pm 1.38$ μSv. The calibration factor for the first experiment is calculated by taking the ratio of the "true" ambient dose equivalent to the measured ambient dose equivalent, such that

$$\begin{aligned} calibration\ factor &= \frac{H_{true}}{H_{measured}} \\ &= \frac{13.42 \pm 1.38\ \mu Sv}{11.89 \pm 0.19\ \mu Sv} \\ &= 1.129 \pm 0.134 \end{aligned}$$

For the second measurement, the background and dead-time corrected monitor rate was 3830.5 ± 8.6 s^{-1}. The measured ambient dose equivalent was $H_{measured} = 13.06 \pm 0.18$ μSv. The "true" ambient dose equivalent is calculated from the product of the ambient dose equivalent rate per monitor count rate, the monitor count rate and the irradiation time in hours, such that $H_{true} = 13.98 \pm 1.59$ μSv. The calibration factor for the second experiment is calculated by taking the ratio of the "true" ambient dose equivalent to the measured ambient dose equivalent, such that

$$\begin{aligned} calibration\ factor &= \frac{H_{true}}{H_{measured}} \\ &= \frac{13.98 \pm 1.59\ \mu Sv}{13.06 \pm 0.18\ \mu Sv} \\ &= 1.071 \pm 0.137 \end{aligned}$$

The average calibration factor is

$$calibration\ factor = 1.10 \pm 0.029$$

5.2.4 Summary

Neutron Source	Calibration Factor	Relative Response
Bare ^{241}Am-Be at 50 cm	0.944 ± 0.075	1.06 ± 0.08
Bare ^{252}Cf at 50 cm	0.936 ± 0.050	1.07 ± 0.05
Bare ^{252}Cf at 100 cm	0.935 ± 0.086	1.07 ± 0.09
D_2O Moderated ^{252}Cf at 100 cm	0.959 ± 0.033	1.04 ± 0.04
CANDU-like Neutron Field I at 100 cm	1.108 ± 0.393	0.90 ± 0.09
CANDU-like Neutron Field II at 100 cm	1.10 ± 0.33	0.91 ± 0.09

Table 5.1: Summary of calculated calibration factors, and relative response (inverse calibration factor) for various neutron sources and source-detector distances.

The calibration factors for each neutron source, for the detector in the orientation where neutrons are incident on the flat face of the SNOOPY, are summarized in Table 5.1

5.3 Other Source-Detector Orientations

As a result of its cylindrical shape, the response of the SNOOPY is known to vary with orientation. In particular, the response of the SNOOPY is higher when neutrons are incident on the side of the cylinder, perpendicular to the axis of the detector. It is operating procedure, in the use of the SNOOPY, to rotate the detector to find the maximum dose reading; the maximum dose reading occurring when neutrons are incident on this side. However, the calibration procedure has shown that the calibration factor is determined for the case of neutrons incident on the flat face of the detector. Therefore, measurements made with neutron incident on the curved face will overestimate the effective dose due to neutrons. By determining the calibration factor which would result from neutrons incident on the curved face, this overestimation can be quantified. These results are summarized in Table 5.2.

These orientations can be considered equal; neutrons are incident on the curved surface of the cylinder. The results of the average of the calibration factors for each source, in

Source	Calibration Factor Source-Detector Orientation 1	2	5	6
^{241}Am-Be at 50 cm	0.77 ± 0.06	0.78 ± 0.06	0.79 ± 0.06	0.83 ± 0.07
^{252}Cf at 50 cm	0.72 ± 0.04	0.72 ± 0.04	0.76 ± 0.04	0.79 ± 0.05
^{252}Cf at 100 cm	0.70 ± 0.06	0.70 ± 0.06	0.73 ± 0.06	0.76 ± 0.07
D_2O Moderated ^{252}Cf at 100 cm	0.67 ± 0.02	0.69 ± 0.02	0.68 ± 0.02	0.71 ± 0.02
CANDU-like I at 100 cm	0.79 ± 0.09	0.81 ± 0.08	0.83 ± 0.08	0.86 ± 0.09
CANDU-like II at 100 cm	0.83 ± 0.08	0.84 ± 0.09	0.85 ± 0.09	0.87 ± 0.09

Table 5.2: Summary of calibration factor for each of the calibration sources in the remaining source-detector orientations as shown in Figure 5.1.

this source-detector orientation are summarized in Table 5.3. Orientation 3, for which the neutrons are incident on the electronic components, has been omitted.

Source	Average Calibration Factor
^{241}Am-Be at 50 cm	0.79 ± 0.04
^{252}Cf at 50 cm	0.74 ± 0.05
^{252}Cf at 100 cm	0.73 ± 0.04
D_2O Moderated ^{252}Cf at 100 cm	0.69 ± 0.03
CANDU-like I at 100 cm	0.82 ± 0.05
CANDU-like II at 100 cm	0.85 ± 0.06

Table 5.3: Average calibration factor for each neutron source for neutrons incident on the curved surface of the SNOOPY.

5.4 Discussion

5.4.1 Results for the Calibration Orientation – Neutrons Incident on the Flat Face

The measurements performed as part of the performance testing of the SNOOPY rem meters—a COG sponsored initiative—were used to determine calibration factors as though the SNOOPY would have been calibrated in each of the fields. These results are summarized in Table 5.1.

From this, it is evident that regardless of the neutron source, the calibration factors will remain constant—within the uncertainty for this calibration orientation using current radiation weighting factors. Therefore, which neutron source is used in the calibration of the SNOOPY rem-meter is irrelevant.

However, it is important to note the variation in the calibration factor when measurements were completed in the CANDU-like neutron fields. These values are significantly larger than the calibration factors arising from measurements in the fields associated with isotopic neutron sources. For the calibration factors to be larger, it signifies that the SNOOPY underestimates the ambient dose equivalent in the CANDU-like neutron fields while overestimating the ambient dose equivalent in the isotopic sources.

The larger-than-expected calibration factors for measurements made in the CANDU-like neutron fields is likely due to the overestimate of the "true" ambient dose equivalent. The calibration factor is the ratio of "true" ambient equivalent dose and the measured ambient equivalent dose; the measured ambient equivalent dose depending only on the SNOOPY itself. Therefore, overestimating the "true" ambient equivalent dose would lead to a corresponding overestimate of the calibration factor rather than an under response of the detector. Calculated as the product of the ambient dose equivalent rate per monitor count rate, the monitor count rate and the irradiation time in hours, overestimating any of these values would lead to a larger value of the "true" ambient dose equivalent. However, the values of monitor count rate and irradiation time are less likely to be inaccurate than the value of ambient dose equivalent rate per monitor count rate; a value which depends on $\hat{h}^*_\phi$, the spectrum-averaged fluence-to-dose conversion coefficient.

It has been shown that the source of neutrons used in the calibration of the SNOOPY rem-meter is irrelevant; all sources yield calibration factors which are in agreement for this calibration orientation and current radiation weighting factors.

5.4.2 Results for Other Orientations – Neutrons Incident on the Curved Surface

For neutron incident on the curved surface of the detector, there is a known overestimate of the associated ambient dose equivalent. This overestimate can been seen in the significantly lower calibration factors resulting from measurements in this orientation. The lower calibration factor indicates that the ambient dose equivalent measured by the SNOOPY is much higher than the "true" ambient dose equivalent that was expected. This over response is corrected by applying any calibration factor, as it will lower the measured values to the expected "true" values.

From Table 5.3 we can see that the applying the calibration factors would lower the measured ambient dose equivalent by approximately 25%. This calculated overestimate is consistent with those previously established (Rogers, 1979; Saull, 2003).

Chapter 6

Effect of Modifying Radiation Weighting Factors

The proposed changes to the value of the radiation weighting factor will also have an effect on neutron dosimetry. Currently, the radiation weighting factors are incorporated in the fluence-to-dose conversion coefficients. These coefficients are used to determine the calibration factors of detectors survey meters. By calculating the calibration factor which results from using the previous values of quality factor when determining the fluence-to-dose conversion coefficient, the effects of such modifications can be observed.

6.1 Effect of Modifying Radiation Weighting Factors on SNOOPY Calibration Factor

Using the computer code, the fluence-to-dose conversion coefficients—based on the values of quality factor rather than radiation weighting factor—were calculated for each isotopic neutron source. The variation in the fluence-to-dose conversion coefficient as a result of the change from values of quality factor to radiation weighting factor is shown in Figure 6.1. Following the previous structure, the calibration factor was calculated for the calibration orientation with neutrons incident on the flat face of the detector. These results are summarized in Table 6.1.

As shown, there are some significant differences in the resulting calibration factors

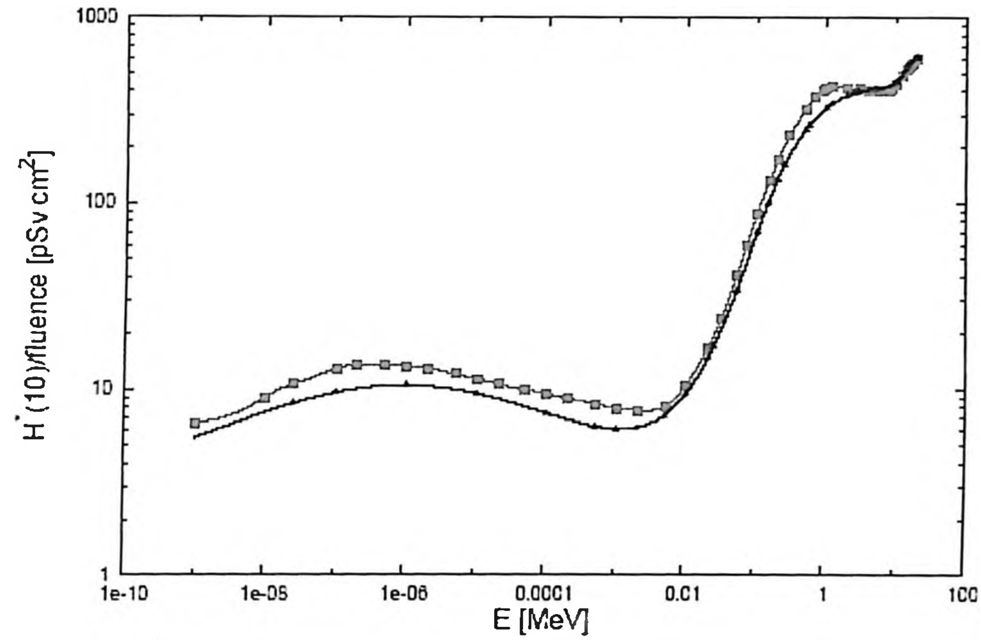

Figure 6.1: The neutron fluence to ambient dose conversion coefficients as a function of energy using values of quality factor (ICRP 26) as indicated by the triangle markings, and values of radiation weighting factor (ICRP 60) as indicated by the square markings (Saull and Ross, 2004).

Source	Calibration Factor	Response Factor
Bare ^{241}Am-Be at 50 cm	0.92 ± 0.05	1.09 ± 0.06
Bare ^{252}Cf at 50 cm	1.19 ± 0.03	0.84 ± 0.02
Bare ^{252}Cf at 100 cm	1.10 ± 0.03	0.91 ± 0.02
D_2O Moderated ^{252}Cf at 100 cm	0.54 ± 0.02	1.85 ± 0.07

Table 6.1: Summary of calibration factors calculated using fluence-to-dose conversion coefficients determined using previous values of radiation quality factor.

when using the previous quality factors to determine the fluence-to-dose conversion coefficient. Also, the neutron source used for calibration also appears to have a greater effect; this is evidenced by the significant difference in calibration factor between the unmoderated and D_2O moderated 252-californium sources. The likely results from the current overestimate of the effect of low energy neutrons.

The implementation of the modifications suggested by ICRP 92 would see the values of the calibration factor return to values closer to those presented in Table 6.1, as these recommendations suggest that the values of w_R become more similar to the previous values of quality factor.

The ultimate result would indicate that measures of neutron dose—at those energies encountered in the CANDU workplace—are overestimated by a factor approximately equal to 2.

6.2 Effect of Modifying Radiation Weighting Factor on Calculated Neutron Dose

The effect of modifying the radiation weighting factor on the SNOOPY calibration factor is shown in Table 6.1. Again this is the for the detector orientation which has the neutrons incident on the flat surface of the detector. Considering the practice of rotating the detector to find the maximum ambient dose equivalent reading, the doses calculated using the previous radiation weighting factors were overestimated by a factor greater than 2, resulting from the low-energy overestimate and the over response for neutrons incident on the curved surface.

As the implementation of the radiation weighting factors suggested by ICRP 92 would see a return values similar to those recommended in ICRP 26, it can be concluded that there would be a similar overestimate of neutron dose; an overestimate by a factor greater than 2.

Chapter 7

Conclusions

Between 2001 and 2003, less than 0.3% of the total external dose received at OPG CANDU stations was from exposure to neutrons. However, for those individuals who received neutron dose, roughly 4.5% of the total radiation equivalent dose was the result of neutron exposure. Neutron dosimetry at Ontario Power Generation—governed by the Canadian Nuclear Safety Commission through Regulatory Standard S-106—includes both direct and indirect dosimetry methods. For direct dosimetry, a moderator-based neutron rem meter known as SNOOPY is used to measure both ambient dose equivalent and ambient dose equivalent rates. The indirect dosimetry component uses maps and tables of measured neutron equivalent dose rates, or a neutron-to-γ-ray dose rate ratio at various locations, with the neutron equivalent dose and neutron equivalent dose rates measured by the SNOOPY. Individual dose is calculated from these rates and the time of exposure within the specified location.

The SNOOPY rem meters used at Ontario Power Generation are calibrated by the National Research Council Canada, Institute for National Measurement Standards. The result of the calibration is a factor which relates the neutron count rate to the ambient dose equivalent rate, using a standard ^{241}Am-Be neutron source. Using the measurements presented in a CANDU Owner's Group Inc. Technical Note (Nunes and Surette, 2004), readings from the SNOOPY for six different neutron fields—in six source-detector orientations—were used, to determine a calibration factor for each of these sources. Although the neutron energy spectra measured in the CANDU workplace is not approximated by the calibration source's neutron energy spectrum, the calibration factor remains constant—with the use of present fluence-to-dose conversion coefficients and within acceptable limits—regardless of the neutron source used in calibration. As a result, the precise source used to calibrate the SNOOPY, is not important. The modification of radiation weighting factors, and hence fluence-to-dose conversion coefficients, may introduce a dependence on the neutron energy spectrum as the current overestimated contribution of low energy neutrons is corrected. OPG should evaluate the effect of any such modifications and determine whether a change to the calibration process or resulting calibration factor is warranted.

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