Presidential Report on Radiation Protection Advice for the Pulsed Fast Neutron System Used in Security Surveillance:

Part III. Methods for the Determination of Effective Dose to Inadvertently Exposed Individuals

A Report Prepared by the National Council on Radiation Protection and Measurements

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Preface

This Presidential Report from the National Council on Radiation Protection and Measurements (NCRP) has been prepared at the request of Sensor Concepts and Applications, Inc. (SCA) of Phoenix, Maryland. SCA, working with the U.S. Department of Defense (DoD) and federal agencies with the responsibility for control of commerce between the United States, Mexico and Canada, asked NCRP for advice regarding a Pulsed Fast Neutron Analysis (PFNA) system. This is the third NCRP Presidential Report prepared for SCA concerning the PFNA system.

The first NCRP Presidential Report was completed in September 2002, entitled *Radiation Protection Advice for Pulsed Fast Neutron Analysis System Used in Security Surveillance*. It covered: (1) the appropriate dose limit for persons inadvertently irradiated by the PFNA system, (2) the proper methods to determine the dose received, and (3) an opinion on whether the use of the PFNA system could result in levels of activation products in pharmaceuticals and medical devices that might be of concern to public health.

The second NCRP Presidential Report was completed in February 2003, entitled *Radiation Protection Advice for the Pulsed Fast Neutron System Used in Security Surveillance: Part II. The ALARA Principle and Related Issues*. It covered: (1) a description of the relevant concepts of radiation protection that should be applied to the PFNA system; (2) a critique, in the form of advice on the necessary content of the draft System Safety Specifications and the draft Radiation Safety Plan for the PFNA system; and (3) the application of the “as low as reasonably achievable” (ALARA) principle to the PFNA system.

In this third NCRP Presidential Report, SCA requested NCRP to describe in more detail the specific methods and instruments recommended for the measurement of and the determination of the radiation dose (*i.e.*, the effective dose) that an individual would receive by inadvertent exposure to radiation from the PFNA system.

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Thomas S. Tenforde
President

1. Summary

The Pulsed Fast Neutron Analysis (PFNA) system is being evaluated by the responsible agencies of the United States government as a security surveillance device for analyzing the contents of large cargo containers such as truck trailers. The specific application is a six month test of the PFNA system at a port of entry to evaluate its effectiveness.

The first Presidential Report prepared for Sensor Concepts and Applications, Inc. (SCA) (NCRP, 2002a) proposed that the effective dose (E) delivered by the PFNA system to an inadvertently exposed person (*e.g.*, a stowaway) under various irradiation scenarios should be determined by mathematical simulation and confirmed by experimental measurement. With the information for different scenarios, it should be possible to make an adequate estimate of E to the individual by monitoring appropriate performance characteristics of the PFNA system. This Report presents in greater detail the specific methods recommended for the determination of E to these individuals. The explanation of the quantity E and the manner in which it is applied to this complex radiation field is presented in detail in Section 2.3 and Appendix B.

1.1 Approach to Determination of Effective Dose

Dose assessment for inadvertently exposed individuals requires the use of radiation transport calculations normalized to a unit fluence of the fast neutron beam incident at a selected reference site (*e.g.,* incident on the container or emitted from the PFNA system components). This will require a series of calculations for likely scenarios, employing the irradiation characteristics of the incident fast neutron beam, the characteristics of the cargo container and its contents, the radiation scattering characteristics of the shielding and equipment surrounding the container, the likely locations of an individual in the container, and the simulation of a human body with simulated organs and tissues at these locations.

Because of the large number of variables involved, it is not feasible to evaluate E after exposures have occurred. Instead, it is recommended that the specific combination of load composition and the location within the load that yields the maximum credible effective dose (designated E^*) to an inadvertently exposed individual be determined. The actual value of E received by an inadvertently exposed person would have substantial variation, but would not exceed E^*.

Section 7 provides a succinct list of conclusions, presenting the rationale for determining the values of E and E^* in the event that an individual is inadvertently exposed in the process of scanning cargo containers with the PFNA system.

1.2 Determination of Effective Dose

The buildup of secondary radiations, the modification of the incident neutron energy spectrum, and the backscatter of neutrons as well as neutron attenuation all play significant roles in determining E to an individual at a specific location within a scanned container. The geometry and composition of the collimator, shielding tunnel, and the container itself all contribute to these factors, but they are relatively constant from one exposure to the next. The contents of the container may vary dramatically, from hydrogenous materials such as foodstuffs and paper to higher atomic number materials such as fabricated steel parts or raw materials. These differences in the load can result in very different values of E.

Also, the distribution of the load within the cargo container can vary, from low-density materials such as potato chips that completely fill the cargo space to high-density materials such as steel bar that fills only a small fraction of the volume of a container when the maximum allowed mass is present. The atomic composition, and particularly the hydrogen content, of the load are particularly important in determining the attenuation and backscattering characteristics, and therefore E as a function of location within the load. Thus it is important to explore loadings of realistic density and representative compositions, both as homogeneously distributed cargo, and as high density slabs in various configurations when evaluating E^*.

As the PFNA system is used, potential improvements to its design may be identified. If these improvements result in changes to the shielding, collimator, source, or relative positions of these components with respect to the scanned container, they may result in significant changes to the values of E, and E^* should be reevaluated.

In order to perform the neutron and gamma-ray calculations at various locations in a truck, it is necessary to construct reasonably accurate models of a number of aspects of the radiation environment, including the neutron source term, the speed of the tow vehicle, the collimator system, the tunnel, the truck and cab, the location of the stowaway in the cargo container, and the truck cargo. Each of these aspects is discussed in turn in Sections 4.1.1 through 4.1.7.

While it is anticipated that E^* would occur in locations nearest the neutron source with little intervening material, E^* could conceivably occur behind some amount of "buildup" material or at greater distances from the source where neutron albedo and capture gamma-rays from the tunnel walls and structures contribute most strongly.

The neutron transport code used to perform the calculations should be capable of determining the neutron and gamma-ray spectra in three dimensions as well as performing the conversions from spectral fluence to absorbed dose at a point. The quality factor (Q) relationship as a function of linear energy transfer (L) [*i.e.*, $Q(L)$] given in Equation B.5 in Appendix B (ICRP, 1991; ICRU, 1993a; NCRP, 1993) is recommended for conversion of absorbed dose at a point to dose equivalent at the point. The quantity E should be calculated using the recommended formalism in NCRP (2000) and NCRP (2002b), and using a MIRD-like adult anthropometric model (*e.g.*, Snyder *et al.* 1978; Kramer *et al.*, 1982; Cristy and Eckerman, 1987) (see footnote [5]), to obtain the organ dose equivalents (ICRU, 1993a) needed to compute E (see Appendix B).

A suggested approach to the calculations to search for the conditions that yield E^* is detailed in Sections 4.1.8 and 4.2. It involves performing calculations at the nine locations noted in Figure 4.1 for: (1) an empty truck; (2) a representative sampling of commonly transported homogeneous loadings (*e.g.*, steel, water, aluminum and concrete) that cover a range of atomic weights, mixtures and densities to simulate realistic packing and loading arrangements; and (3) some inhomogeneous cargo configurations with high albedo and capture gamma-ray production.

1.3 Validation Measurements and Uncertainties

Due to the inherent uncertainties in the calculation of E for the complex radiation field of neutrons and gamma rays, validation is required prior to routine use of the PFNA system for scanning cargo containers. Although E is not a measurable quantity due to its dependence on nominal values of tissue weighting and radiation weighting factors, there is a closely related quantity [*i.e.*, the Q(L)-weighted lineal energy density] that can be derived from both measurements and calculations.

The Q(L)-weighted lineal energy densities at depths of 1 and 10 cm in a hydrogenous physical phantom can be calculated by radiation transport codes and can be measured by a method that uses a tissue-equivalent proportional counter (TEPC) (see Section 4.3) in such a way as to determine a reasonable estimate of the uncertainty of the calculated value of E using the MIRD-like model. The TEPC structure itself should be included in a special calculation for the hydrogenous physical phantom so that the discrepancy between the calculated and measured Q(L)-weighted lineal energy densities will be as low as possible.

The limiting condition for operation of the PFNA system is that $E^*(1 + 2\,r_c)$ should be less than the recommended E limit of 1 mSv (or 5 mSv in the case of compelling national security requirements) (NCRP, 2002a). The quantity r_c is the combined relative standard uncertainty for the ratio of the calculated to experimental values, and is taken to be a valid estimate of the combined relative standard uncertainty for the ratio of the MIRD-like model calculations of E^* to the true value of E^*. The factor of two multiplying r_c provides a 95 percent confidence interval for the assurance that the E^* received by a stowaway will be less than the designated limit.

In addition to the TEPC measurements, incident (*i.e.*, without a phantom) spectral fluence rate measurements for both neutrons and gamma rays should be carried out for the configuration that yields E^* to verify that the major features of the observed spectra are qualitatively in agreement with the calculations.

1.4 Application of the Results

The results of the radiation transport code calculations described in Sections 3 and 4 can be expected to provide an estimate of E to an individual located at the position in the cargo container where E is the highest (*i.e.*, E^*). Since it is possible for an inadvertently exposed individual to be at the location of E^*, the calculated value of E^* should provide the basis for operation of the PFNA system. However, if it is known that there are no individuals in the cargo container or truck cab (by prescreening or manned in-container inspections), this limitation does not apply.

1.5 Quality Control of PFNA System Performance

Any change in the design of the shielding (composition or dimensions), the accelerator, the target, or the inspection tunnel from those used in the radiation transport calculations could result in invalidating the calculated values of E. In order to ensure against any unrecognized impact from such changes, a document listing those items that could affect the reliability of the calculated values should be available to designers, operators, maintenance personnel, and facility management.

Specific recommendations for quality control are given in Section 6 for the accelerator fluence rate, the target assembly rotation rate, and the rate of container travel, and are summarized below.

- The fluence rate is the most appropriate measure of the output of the accelerator. Therefore, a live-time display of the fluence rate should be available to the operators whenever the accelerator is in operation, and the data should be recorded.
- The rotation rate of the target assembly provides the vertical scanning rate. As a result, the rotation of the target assembly should occur at or above the specified rate, and procedures to establish and maintain the rotation rate values are essential to ensure that the calculated values of E continue to be applicable.
- The horizontal scanning rate is determined by the speed of the cargo container. As a result, the tow vehicle, the truck, and the cargo container should pass through the scanning beam at or above the specified rate and with a high degree of reliability. Vehicle travel rate should be tied to accelerator operation.

2. Introduction

Sensor Concepts and Applications, Inc. (SCA) of Phoenix, Maryland, working with the U.S. Department of Defense (DoD), has asked the National Council on Radiation Protection and Measurements (NCRP) to provide additional recommendations concerning the radiation protection aspects of the Pulsed Fast Neutron Analysis (PFNA) system (Brown *et al*., 2001).

In the first report for SCA (NCRP, 2002a), it was proposed that the values for effective dose (E) delivered by the PFNA system under various irradiation scenarios should be determined by mathematical simulation and confirmed by experimental measurement. NCRP (2002a) stated that it should be possible to evaluate, prior to the routine use of a PFNA system, the potential unintended values of E to individuals associated with a range of irradiation conditions likely to be encountered during implementation of the PFNA system. NCRP (2002a) also stated that it should be possible to monitor appropriate performance characteristics during routine use of the PFNA system that would enable an adequate estimate of E to an individual who is actually exposed to be made, using the data obtained for the range of irradiation conditions.

This Report presents in greater detail the specific methods and instruments recommended for the measurement of and the determination of the E that an individual would receive by inadvertent exposure to radiation from the PFNA system. As background for the discussion on determining E, a brief description of the PFNA facility, a recap of related recommendations in the first two reports, and a review of the quantity E, are included in the Introduction.

2.1 Description of PFNA Facility

The objective of the PFNA system is to produce a three-dimensional image of the distribution of atomic composition for the materials in the contents of large cargo containers such as truck trailers. The three-dimensional image of atomic composition makes it possible to identify certain materials, or classes of materials, and allows discrimination between materials even though they may have essentially identical electron density and are, therefore, indistinguishable by conventional x-ray imaging.

This is accomplished by irradiation with a fast neutron beam, and measurement of the energy spectra of prompt gamma rays produced by fast neutron absorption in the contents of the container. The measured gamma-ray spectra are unique to specific elements and can be used to determine atomic composition. The neutrons are produced by a deuteron beam from an accelerator that strikes a deuterium target, resulting in the generation of neutrons (*i.e.*, the neutron beam) and the production of tritium (a radioactive side product that needs to be properly managed; see Section A.3 in Appendix A).

In order to resolve the positions and volumes of materials of different atomic composition, the neutron beam is collimated and scanned in the vertical plane, the container is moved through the plane of the beam, and the time of flight of the neutron from when it was produced to when it was absorbed is used to determine the position along the direction of the beam. Thus, the spectrum of gamma rays received at a specific time after the neutron pulse gives the atomic composition at the corresponding point in the container. The radiation dose to the contents, and the activity of activation products in scanned containers, is directly related to the neutron fluence provided all other irradiation conditions are invariant.

The typical PFNA inspection facility would consist of a building (approximately 70 m by 20 m) housing the PFNA equipment and several smaller structures for electronic equipment and operating staff.

When the PFNA system is in routine use, vehicles are selected for inspection from the stream of commerce and are directed to the corridor-like entrance of the test facility. The driver leaves the vehicle and walks to a designated waiting area located at the other side of the PFNA building. A self-powered towing machine slowly pulls the unoccupied vehicle through the facility and past the scanning device located in the tunnel. Once all safety checks are verified, the vehicle is scanned with the neutrons. The pulsed beam moves up and down while the vehicle slowly passes by to ensure that all of the contents are inspected.

2.2 Related Recommendations from Previous Presidential Reports

2.2.1 *The Appropriate Dose Limit for Inadvertently Exposed Persons*

NCRP (2002a) recommended that the PFNA system be designed and operated in a manner that ensures that an inadvertently exposed person (*e.g.*, a stowaway) will receive an effective dose of less than 1 mSv (millisievert)[1]. NCRP (2002a) further recommended that this

[1] Throughout this Report, the International System (SI) of units, specifically the units of millisievert (mSv) and milligray (mGy), are used. The relationships between the SI units and the previous units are: 1 mSv = 100 mrem (millirem); and 1 mGy = 100 mrad (millirad).

limit can be raised to 5 mSv, if necessary, to achieve national security objectives. In all cases, the PFNA system should be designed and operated in accordance with the principle of keeping exposures "as low as reasonably achievable, taking into account economic and social factors" (*i.e.*, the ALARA principle).

NCRP (2002a) also recommended that the law enforcement authority responsible for the PFNA system should provide information about the exposure to individuals known to have been inadvertently exposed. The information should be easy to understand and presented in a manner that is accessible to the individual.

2.2.2 *Determination of Effective Dose*

NCRP (2002) stated that the radiation protection quantity of interest for an exposed individual is the effective dose (E). The quantity E can be evaluated for a series of likely PFNA system scenarios that describe the irradiation conditions for cargo containers, using a combination of radiation transport calculations and supporting dosimetry measurements.

Practical implementation can be accomplished by comparing the actual PFNA system characteristics against the planned PFNA system characteristics (see Section 4.3). Values of E for individuals who have been exposed can be estimated from the E values obtained for the likely irradiation scenarios noted above.

2.2.3 *The ALARA Principle and its Application to Inadvertently Exposed Individuals*

In NCRP (2003), advice on application of the ALARA principle was provided on four aspects of the operation of the PFNA: tritium production and use, inadvertently exposed persons, neutron activation of foodstuffs, and radiation levels outside the facility. The advice concerning inadvertently exposed persons stated that the dose to these individuals is directly related to the neutron fluence they receive. The only ways to reduce neutron fluence are to increase the sensitivity of gamma-ray detection, or to accept reduced resolution in the measurements, which may compromise the ability of the PFNA system to determine the composition of the cargo. The remaining three aspects are covered in Section A.3 of Appendix A.

2.3 Review of the Quantity Effective Dose

The use of fast neutrons for cargo inspections results in a complex distribution of photons (gamma rays) and neutrons (fast, epithermal and thermal) inside the cargo container. The radiation spectrum (*i.e.*, type, energy and direction of the radiation) that reaches an individual inside the cargo container is dependent on the materials that make up the container, the contents of the container, and the location of the individual inside the container.

In less complicated situations, when only x and gamma rays interact with the human body, the mean absorbed dose in an organ or tissue [D_T (*i.e.*, the total amount of the energy deposited in the organ or tissue divided by its mass)] is the basic quantity in radiation protection. For the PFNA system, D_T needs to be modified to reflect the greater biological effectiveness of neutrons. Also, the radiation field incident on the container is significantly altered, as noted above, before it reaches an individual, which makes this irradiation case more complex than usual. The quantity D_T is usually modified in two ways. First, to reflect the increased biological effectiveness of neutrons compared with x and gamma rays, a modifying factor for radiation type called the radiation weighting factor (w_R) is applied and yields the quantity that represents the equivalent dose (H_T) to an organ or tissue. The w_R values apply to the type of radiation incident on the body. Second, to reflect the variations in radiation risk among different organs or tissues in the body, a different modifying factor for tissue type called the tissue weighting factor (w_T) is applied. The w_T values are the same for all radiations. The sum of the products of H_T and w_T yields the quantity effective dose (E), which is the quantity used to express the radiation dose received by an exposed individual (NCRP, 1993).

Dose assessment for individuals located inside a cargo container being irradiated by a high-energy neutron beam requires the use of radiation transport calculations normalized to a unit fluence of the fast neutron beam incident at a selected reference site (*e.g.*, incident on the container or emitted from the PFNA system components). This will require a series of calculations for likely scenarios, employing the irradiation characteristics of the fast neutron beam, the characteristics of the cargo container and its contents, and the likely locations of an individual or individuals in the container. This approach (*i.e.,* for each selected scenario) allows one to obtain the energy imparted at each interaction site in the interior of the container and permits calculation of the dose equivalent (H) at points in the cargo container by utilizing the current quality factor relationship [$Q(L)$][2] (ICRP, 1991; ICRU, 1993a; NCRP,

[2] $Q(L)$ is the quality factor relationship [*i.e.*, as a function of linear energy transfer (L)] by which absorbed dose [$D(L)$] at a point is modified to obtain the dose equivalent (H) at the point, in order to express the effectiveness of an absorbed dose on a common scale for all types of ionizing radiation.

1993). The energy imparted and the mean quality factor at all interaction sites in an organ or tissue can be used to obtain a quantity called organ dose equivalent (ICRU, 1993a), given the symbol $\overline{H}_{\mathrm{T}}$ by NCRP (NCRP, 2002b) (see Section B.2). The quantity $\overline{H}_{\mathrm{T}}$ was adopted by NCRP as an acceptable approximation for H_{T} for situations where complex internal spectra are generated by high-linear energy-transfer radiation (NCRP, 2000; 2002b).

Therefore, if a simulated person with simulated organs and tissues is located inside the container, the resulting value of $\overline{H}_{\mathrm{T}}$ for each internal organ or tissue of the individual can be calculated utilizing the $Q(L)$ relationship, and the quantity E can be estimated by substituting $\overline{H}_{\mathrm{T}}$ for H_{T} (*i.e.*, for $w_{\mathrm{R}} D_{\mathrm{T}}$). The evaluation of $\overline{H}_{\mathrm{T}}$ can be performed with computerized anthropometric models in conjunction with an appropriate radiation transport code. In this way, a value of a conversion coefficient (*i.e.*, E divided by the fluence) for the irradiation conditions encountered with the PFNA system in the specific scenario can be obtained. Appendix B presents the specific dosimetry formulations pertinent to the dosimetry approach outlined above.

2.4 Scope of Work for Current Report

The scope of work in the SCA request was:

"What are the specific methods and instruments recommended for the measurement of and the determination of the radiation dose that an individual would receive by inadvertent exposure to radiation from the Pulsed Fast Neutron Analysis (PFNA) system?"

NCRP (2002a), outlined the main features of the proposed dose assessment approach for *E* (see the summary of the approach at the beginning of Section 2), as follows:

- Identify the neutron beam parameters, container and content characteristics, scanning rate, and surrounding cargo environment for a number of potential inspection scenarios.
- Estimate *E* values for individuals, using radiation transport codes and appropriate anthropometric models to calculate the internal radiation environments in the cargo container for selected scenarios (*i.e.*, beam parameters, container and content characteristics, and location of individuals inside the containers).
- Conduct a sensitivity analysis of the *E* values for PFNA systems (*i.e.*, evaluate the amount of change in the values of *E* over the range of selected scenarios).
- Validate the calculations with physical measurements (*i.e.*, with appropriate tissue-equivalent phantoms and appropriate radiation detectors to measure the photon and various neutron components) for scenarios that represent the maximum contribution from buildup, albedo, backscatter, and capture gamma rays to the dose in the cargo container.
- Evaluate the practicality of determining the value of *E* that results from an inadvertent exposure using the actual beam characteristics, container and cargo characteristics, scan time, and the values of *E* from the pre-established database for the likely scenarios.

NCRP (2002a) also noted the methods for monitoring the PFNA system in practice, as follows:

- Monitor the beam characteristics (*i.e.*, scan rate, fluence rate, beam energy, and beam current) of the PFNA system to confirm that the actual characteristics conform to the characteristics planned for the application of the PFNA system.
- Conduct periodic (*e.g.*, monthly) quality control checks of the fluence rate incident on the cargo container for fixed neutron beam conditions.

In this Report, these aspects of the method and associated details are further developed.

3. Approach to Determination of Effective Dose

Because of the size and complexity of the PFNA system, including the neutron source, collimator, cargo container, and shielding, the evaluation of the effective dose (E) to individuals in the cargo container is complex. Each of the major components of the system contributes scattered neutrons that may reach the individual. Furthermore, the exposed person is likely to be embedded in a relatively large volume of cargo that may have any of a wide variety of atomic compositions. As a result, the radiation spectrum will include a variable component of low-energy scattered neutrons and gamma rays. The scattered radiation is particularly important because many of the scattered neutrons will have higher quality factors (radiation weighting factors) than the incident neutrons. Experimental measurement of the quantities needed to evaluate dose equivalent (radiation spectrum as well as fluence) requires multiple detection methods and generally results in significant uncertainties. Consequently, evaluation of E at the locations where individuals may be irradiated, and for the range of load configurations that may be present, is most easily accomplished by careful calculation of the components of the radiation field at those locations. To assure the accuracy of the calculations, experimental validation using quantities that can be directly measured is also required.

3.1 Use of Maximum Credible Values of Effective Dose

Because of the large number of variables involved, and the resulting effort required to fully evaluate E to an individual in a specific container that has been scanned by the PFNA system, it is not feasible to plan to evaluate E after exposures have occurred. Instead it is recommended that the specific combination of load composition and location within the load that will lead to the maximum credible effective dose for an inadvertently exposed individual be identified. This quantity, designated E^*, can be directly compared with specified E limits. The dose received by inadvertently exposed individuals will have substantial variation, but will not exceed E^*. There is no compelling radiation protection reason to record the actual value of E received by an exposed individual. Reporting E^* as the value of E received by each individual is acceptable. However, it is likely that the actual average E over the population of inadvertently exposed individuals is substantially lower than E^*. If there is an administrative reason for obtaining the average value, this could be accomplished by evaluating the position of exposed individuals within a load and the approximate composition of the load from the three-dimensional data produced by the PFNA scan. The value of E to the individual at that position could then be determined by comparison with a catalog of values listed as functions of position and load composition, or by a detailed retrospective dose evaluation. For this reason, it is recommended that all available data (scan results, neutron attenuation, actual content of container, etc.) be recorded for any container that is found to include an inadvertently exposed individual. Development of data of this type for all inadvertently exposed individuals would make it possible to determine the actual distribution of E for the exposed population.

3.2 Purpose of Radiation Transport Calculations

Because of the dependence of E on the absorbed dose and the spectrum of the radiation at defined positions (the critical organs), and because of the difficulty in measuring these quantities, E can be more precisely determined by radiation transport calculations. However, all significant sources of scattered radiation and secondary radiations should be included. This requires inclusion of the radiation source geometry, collimator, shielding structure (tunnel), and cargo composition, as well as the location of the exposed individual in the transport calculation. By varying the descriptions of these components in the calculations it is possible to determine the sensitivity of E to changes in each parameter, and thus determine the load composition and distribution that will maximize E as well as how precisely the actual facility geometry and load configuration should be modeled to achieve an acceptable level of uncertainty in the results.

3.3 Purpose of Measurements

Although recent advances in computer power and memory have made it possible to perform radiation transport calculations for very complex systems, many errors and omissions can occur in setting up these calculations. Furthermore, some of the radiation interaction cross sections needed to accurately calculate the absorbed dose and radiation spectrum at specific locations are known with limited accuracy. As a result, it is essential that the results of radiation transport calculations be validated by experimental measurement. Because of the use of administrative values such as tissue weighting factors and quality factors in the definition of E, it is not appropriate to use a comparison of measured and calculated values of E for validation. Instead, physical quantities that can be calculated and measured should be used. The nature of the evaluation of E by transport calculations is such that if the physical quantities are properly evaluated, the values of E can be confirmed. Physical quantities that can be used for validation of the calculations include the radiation spectrum, that is, the fluence as a function of energy for the different types of radiation present at each point, energy imparted in micrometer-scale volumes by individual events, and the absorbed dose. The radiation spectrum is an intermediate step in calculating E in typical Monte Carlo codes, so that absorbed dose, energy imparted, and radiation spectrum can be obtained easily.

3.4 Discussion of Factors that Define Irradiation Scenarios

The buildup of secondary radiations, the modification of the neutron energy spectrum, and the backscatter of neutrons as well as neutron attenuation all play significant roles in determining E to an individual at a specific location within a scanned container. The geometry and composition of the collimator, shielding tunnel, and the container itself all contribute to these factors, but they are relatively constant from one exposure to the next. The collimator and shielding change only when the PFNA system is modified, and the amount of material used in the construction of most containers and truck trailers is too small to have much effect. However, the contents of the container can vary dramatically, from hydrogenous materials such as foodstuffs and paper to relatively high atomic number materials such as fabricated steel parts or raw materials, and these differences in the load can result in very different values of E. Also, the distribution of the load within the container can vary, from low-density materials such as potato chips that completely fill the cargo space to high-density materials such as steel bar that fills only a small fraction of the volume of a container when the maximum allowed mass is present. The atomic composition, and particularly the hydrogen content, of the load are particularly important in determining the attenuation and backscattering characteristics, and therefore E as a function of location within the load. Thus it is important to explore loadings of realistic density and representative compositions, both as homogeneously distributed cargo, and as high density slabs in various configurations when evaluating the maximum credible effective dose (E^*).

3.5 Re-evaluation of Effective Dose When Irradiation Conditions Change

The geometry and composition of the source, collimator and shielding all contribute to the radiation spectrum and fluence rate at the locations where an individual may be inadvertently exposed. These factors will normally be constant from one exposure to the next, but as the PFNA system is used, potential improvements to its design may be identified. If these improvements result in changes to the shielding, collimator, source, or relative positions of these components with respect to the scanned container, they may result in significant changes to E, and E^* should be reevaluated.

4. Determination of Effective Dose

4.1 Modeling the Characteristics of the Radiation Environment

Determination of the effective dose (E) delivered to a potential stowaway in a truck scanned by the PFNA system is a difficult task due to the complexities of the irradiation system and wide variety of possible cargos. In principle, neutron transport calculations or measurements could be conducted at many possible locations in a truck assuming many different types of materials in the truck cargos to obtain the range of values for E that may be delivered in a stowaway scenario. However, as discussed in Section 3, the scope of the calculations or measurements can be greatly reduced by considering only the limiting case of the maximum credible value of the effective dose (designated by E^*). This conservatively safe approach ensures that stowaways would receive an E less than or equal to E^*. The following subsections detail the minimum requirements for dose calculations and experimental verifications that would be required to determine E^*. More detailed calculations or measurements could be performed for other purposes such as the determination of more realistic doses. It is important to note that the required methodology is the same in any case and that this Report proposes an approach that would result in fewer calculations or measurements while ensuring that no individual receives an E in excess of that calculated.

Calculations of E require determination of the neutron and gamma-ray fluence spectra at given points and transformation from fluences to E values using standard formalisms (NCRP, 2002). In order to perform the neutron and gamma-ray calculations at various locations in a truck, it is necessary to construct reasonably accurate models of a number of aspects of the radiation environment, as discussed in Sections 4.1.1 through 4.1.7.

4.1.1 *The Neutron Source Term*

The neutron source term should be fully described in terms of source strength (neutrons per second) and energy spectrum as a function of angle subtended by the collimator for a given spread of accelerator beam energies and target thickness. As discussed in Section 6, some method of quality control is needed to ensure that the system operates within these stated parameters.

4.1.2 *The Speed of the Tow Vehicle*

The speed of the tow vehicle should be given in order to determine the total dose received at a given point in the truck.

4.1.3 *The Collimator System*

The collimator system should be fully described in terms of the components, dimensions, and angular rotation or scan range.

4.1.4 *The Tunnel*

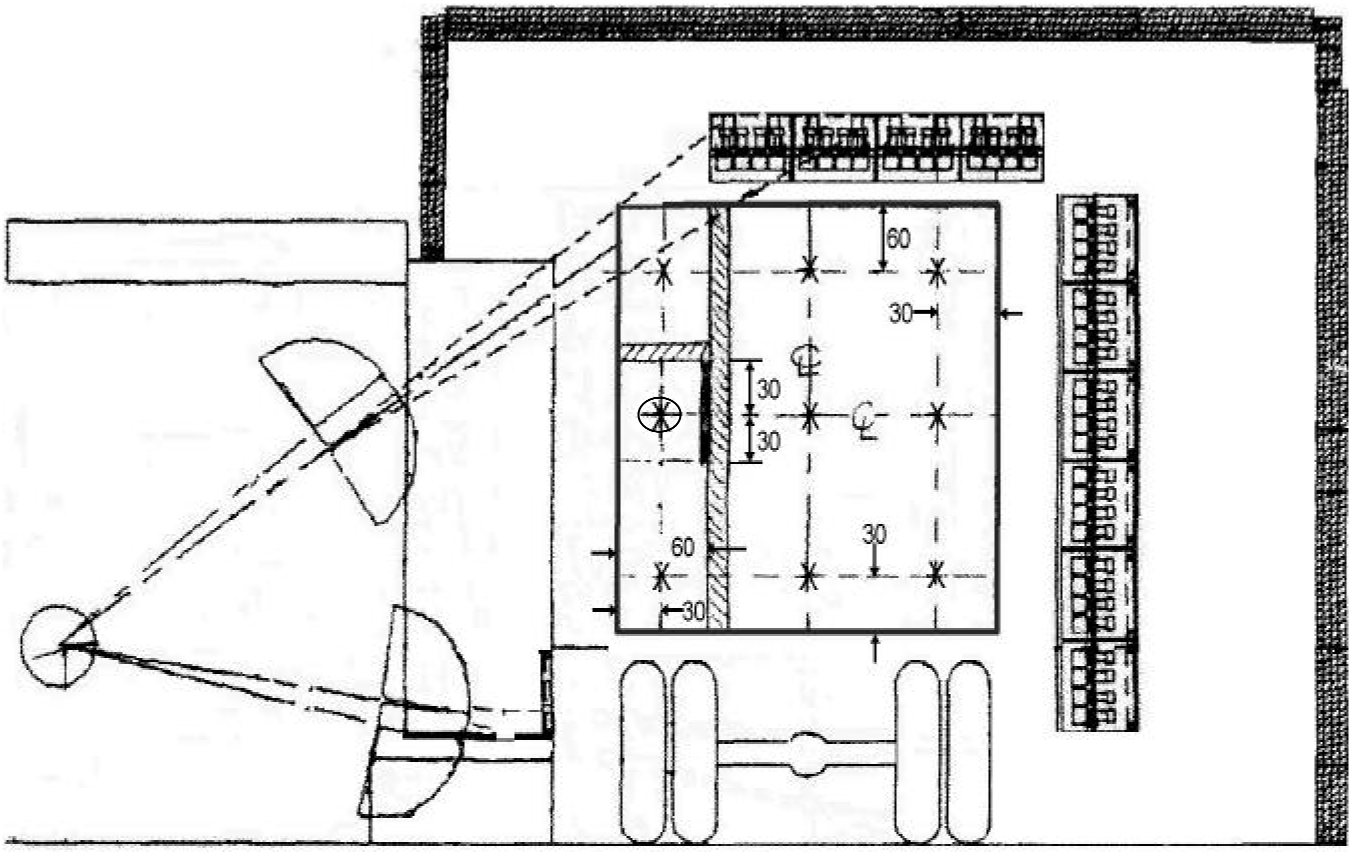
The tunnel, shielding walls, detector arrays, and floor should be described in terms of the composition, thickness, and major penetrations in order to adequately account for reflected and absorbed neutrons, and prompt gamma rays that may contribute to E.

4.1.5 *The Truck and Cab*

A model of the truck, including the cab, should be included. The model should be described in terms of the major components that present sufficient mass and representative composition that would have any significant effects on the neutron transport or gamma-ray production or shielding. Due to the general similarity among the wide variety of possible truck configurations, a simplified generic model that takes the major components noted above into account is sufficient.

4.1.6 *The Location of the Stowaway in the Cargo Container*

It is suggested that nine locations be considered as possible places where a stowaway might be located in a truck's cargo. These consist of locations near the neutron beam entry point, in the middle of the truck (in the transverse direction), and on the opposite side from the neutron beam entry point for locations on the centerline of the cargo (in the vertical direction), near the top of the cargo, and near the bottom of the cargo. Evaluation of E to a person at each of these nine locations (see Figure 4.1), which are evaluated one at a time,



Source and collimator assembly and shield

Roof and wall shielding in the tunnel

Detector arrays

X
Nine locations for effective dose determination (by measurement or calculation)

Ⓧ
Sample location for effective dose determination (showing albedo and and cadmium plates)

Albedo Plate

Cadmium Plate

℄
Vertical and horizontal centerlines of truck

Fig. 4.1. Location of the nine test positions. Reflector slabs and head location shown for one case. All dimensions are in cm; drawing not to scale. Adapted from Slater *et al.* (2001).

should provide data that span the range of possible values of E, and should thus be sufficient for the determination of E^*.

4.1.7 *The Truck Cargo*

Truck cargos may consist of an enormous variety of materials that may prove difficult to model. However, the scope of the problem can be greatly reduced by consideration of material properties that would be significant relative to neutron absorption and reflection. Without consideration of the cargo (empty truck or no truck case), E will vary inversely as the square of the distance of the stowaway from the neutron source, as a first approximation. However, overlapping of beam sweeps and neutron scattering from or gamma-ray production in the tunnel walls and floor may prove to contribute significantly to the value of E with variation with distance different from inverse-square.

As various types of cargos are added to the truck, E may increase or decrease significantly from the empty truck case due to the effects of neutron scattering and absorption as well as gamma-ray production. Consequently, it should be sufficient to categorize cargos into several different classes based on their neutron scattering and absorption properties. Suggested materials include those of higher atomic weight (such as steel), medium atomic weight (such as aluminum), and high moderation characteristics (such as water or concrete). It is anticipated that consideration of these cases as well as the empty truck case and certain high-albedo configurations discussed below should provide sufficient information to give bounding values for E at the nine locations mentioned above.

4.1.8 *Summary of Suggested Maximum Credible Irradiation Conditions*

Identification of the maximum credible irradiation conditions requires a survey of configurations that are representative of credible cargo loadings. Although it is anticipated that the highest value of E will occur in positions nearest the neutron source with little intervening material, the maximum credible condition could conceivably occur behind some amount of "buildup" material or at greater distances from the source where neutron albedo and capture gamma-rays from the tunnel walls and structures contribute most strongly.

The search for the maximum credible condition should therefore include calculations at the nine points in Figure 4.1 for at least the following conditions: (1) the empty truck, (2) a representative sampling of homogeneous loadings, and (3) some inhomogeneous cargo configurations with high albedo and capture gamma-ray production. For the homogeneous loadings, commonly transported substances such as steel, water, aluminum and concrete should be included to cover a range of atomic weights and mixtures, using artificial densities to simulate realistic packing and loading arrangements.

Since the search is for the maximum credible effective dose (E*) condition only, the homogeneous loading calculations should begin with the lowest densities and continue to higher densities until the value for E begins to decrease. This approach should identify any positive buildup configuration without looking at many denser loadings for which attenuation dominates. It should suffice to begin with an artificial density of 0.02 g cm^{-3}; even at this low density, attenuation may already dominate buildup in the computation of E. If positive buildup configurations are discovered, the density and composition of the most positive of these should be taken as the filling material in the otherwise-empty space in the heterogeneous loading configurations discussed below.

The nine positions of interest for the anthropometric model are shown in Figure 4.1. To achieve the maximum value of E at a given position, the model should be assumed to be lying parallel to the direction of motion, facing the side of the truck from which the beam enters (*i.e.*, the beam is incident on the front of the model), with the center of the model's head at one of the X marks shown in Figure 4.1.

The specifications for the inhomogeneous cargo loadings are as follows: A vertical floor-to-ceiling albedo wall is positioned 30 cm behind the center of the anthropometric model. On this wall a sheet of cadmium, 0.5 mm thick and 60 cm high is positioned behind the model. At 30 cm above the model, a second albedo slab of 60 cm width is positioned. In the calculations, these reflector slabs would run the full length of the cargo space. The two albedo slabs are both made of the same material, either steel, aluminum, concrete or water. For the case in which the slab material is water, the slab thickness is 30 cm; for steel, aluminum or concrete, the slab thickness is 10 cm.

In experimental tests, the albedo slabs need to be only long enough to extend 1 m beyond the largest test instrument in both the fore and aft directions. Figure 4.1 shows the arrangements of the albedo slabs for one of the nine test points. The void spaces in the drawing are filled with the homogeneous density and composition that was found to give the most positive buildup increase in the calculation of E. Also in the experimental tests, the void spaces in front of the anthropometric model are filled with material of the same areal density (g cm^{-2}) and composition corresponding to the most positive buildup condition. This added buildup material should extend at least 1 m beyond the largest test instrument in both the fore and aft directions.

4.2 Neutron Transport Codes

The neutron transport code used to perform the calculations should be capable of determining the neutron and gamma-ray spectra in three dimensions as well as performing the conversions from fluence-rate spectra to dose rate using the recommended formalism for *E* (NCRP, 2002b). The neutron and photon cross-section data should be based on the Evaluated Nuclear Data File, Part B, Version VI (ENDF/B-VI)[3], and this Report recommends using a minimum of 47 energy groups. The calculations should be performed in sufficient detail to give accurate photon production; in particular, scattering kernels for chemically-bound hydrogen [such as the light-water cross-section files in the Monte Carlo N-Particle code (MCNP4C)[4]] should be used to predict thermalization and capture gamma-ray production realistically. MCNP4C is an example of a well-supported and well-documented computer code that would be appropriate for these calculations.

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[3] ENDF/B-VI is available from the National Nuclear Data Center, Brookhaven National Laboratory; see www.nndc.bnl.gov/nndc/endf (Click ENDF).

[4] MCNP4C is available from the Radiation Safety Information Computation Center at Oak Ridge National Laboratory, see www-rsicc.ornl.gov/rsicc.html (search on MCNP4C).

The quantity E should be calculated using a MIRD-like adult anthropometric model[5] (*e.g.*, Snyder *et al.*, 1978; Kramer *et al.*, 1982; Cristy and Eckerman, 1987). The quality factor relationship [$Q(L)$] given in Equation B.5 in Appendix B (ICRP, 1991; ICRU, 1993a; and NCRP, 1993), is recommended for calculations of organ dose equivalents $\left(\overline{H}_{\mathrm{T}}\right)$ in the MIRD-like model. These calculations should be carried out in such a way as to correspond to a single scan of the truck or cargo container, so that all aspects of the beam and raster overlaps and the scatter of neutrons in the direction of motion of the container are weighted properly.

4.3 Validation Measurements and Uncertainties

Due to the inherent uncertainties in the neutron and gamma-ray calculations for E, validation is required in the finished facility prior to testing with cargo containers that may contain stowaways. These tests should be carried out in such a way as to correspond to a single scan of the truck or cargo container, so that all aspects of the beam and raster overlaps are weighted properly.

Although E is not a measurable quantity due to its dependence on tissue weighting factors and radiation weighting factors, there is a closely related quantity that can be derived from both measurements and calculations. The $Q(L)$-weighted lineal energy

[5] "MIRD-like" refers to the anthropometric mathematical model of a reference male/female first published by the Medical Internal Radiation Dose Committee of the Society of Nuclear Medicine (Snyder *et al.*, 1978). There are also in the literature voxel-type models derived from computed tomography data for specific individuals that present a more realistic simulation of the human anatomy for that individual, but they do not represent a reference person.

densities[6] at depths of 1 and 10 cm in a hydrogenous physical phantom can be calculated by transport codes and can be measured by the tissue-equivalent proportional counter (TEPC) method (ICRU, 1983), in such a way as to determine a reasonable estimate of the uncertainty of the calculated value of E using the MIRD-like anthropometric model.

The hydrogenous physical phantom should have cross sectional dimensions of roughly human torso size and a length of at least 50 cm in the truck's direction of motion, with the center of the TEPC cavity at the center of that length. The penetration for the TEPC electrical connections should run along the direction of motion, so that there is no direct streaming into the penetration from either the neutron beam or the albedo walls.

The TEPC structure itself should be included in a special calculational model for the hydrogenous physical phantom so that the discrepancy between the calculated and measured $Q(L)$-weighted lineal energy densities will be as low as possible. A spherical cavity of approximately 1 cm diameter is recommended for modeling the TEPC in order to provide adequate counting statistics without producing a large perturbation in the simulated physical phantom.

The transport codes, cross-section data, and $Q(L)$-weighting for the calculation for the hydrogenous physical phantom calculations should be the same as that employed in the calculations for the MIRD-like anthropometric model. Because the calculations for the hydrogenous physical phantom parallel the calculations for the MIRD-like model very closely, any biases in these two calculations can be assumed to be essentially the same, and

[6] The $Q(L)$-weighted lineal energy density is the probability density of absorbed dose in lineal energy [$d(y)$] multiplied by $Q(L)$, substituting y for L.

the comparison of the $Q(L)$-weighted lineal energy densities from the calculations[7] and the TEPC measurements should be a nearly direct validation of the MIRD-like phantom calculations.

The combined relative standard uncertainty (u_C) of the experimental value (E_{exp}) should be determined following the International Organization for Standardization (ISO) *Guide to the Expression of Uncertainty in Measurement* (ISO, 1993). The combined relative standard uncertainty (U_C) of the calculated value (E_{cal}) for the physical phantom should be determined in like manner. Assuming that the experimental uncertainties are independent of the uncertainties in the calculation, then the combined relative standard uncertainty (r_C) in the ratio of the calculated to experimental values is given by $r_C = \sqrt{\left(u_C{}^2 + U_C{}^2\right)}$, for the larger of the two values of $\sqrt{\left(u_C{}^2 + U_C{}^2\right)}$ determined at the 1 and 10 cm depths in the phantom. However, if the ratio of the calculated to experimental values differs from unity by more than r_C, then the absolute value of that difference from unity $|1 - E_{cal} / E_{exp}|$ is taken as a revised estimate of r_C. The resulting value of r_C is taken to be a valid estimate of the combined relative standard uncertainty for the ratio of the MIRD-like model calculations of E^* to the true value of E^*. The limiting condition for the inadvertent exposure of stowaways is that $E^* (1 + 2\, r_C)$ should be less than 1 mSv (or 5 mSv in the case of compelling national security requirements). The factor of two multiplying r_C provides a 95 percent confidence

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[7] Calculation of the $Q(L)$-weighted lineal energy requires calculation of charged-particle track interactions in the proportional counter, rather than the local deposition of energy transferred to charged particles, which is sufficient for evaluation of E. Since calculation of the charged-particle tracks requires a large amount of computer time, it may be efficient to separate the calculation into evaluation of the charged-particle spectrum and a separate evaluation of energy deposition in a detector cavity by charged particles.

interval for the assurance that the E^* received by a stowaway will be less than the designated limit.

In addition to the TEPC measurements, incident (*i.e.*, without a phantom) spectral fluence rate measurements for both neutrons and gamma-rays should be carried out for the E^* configuration to verify that the major features of the observed spectra are qualitatively in agreement with the calculations. Proton recoil proportional counters, liquid scintillation counters, and Bonner Sphere techniques have been highly developed for neutron spectrometry measurements by the Department of Energy (DOE) Environmental Measurements Laboratory,[8] the British National Physics Laboratory[9] and the German Physikalisch-Technische Bundesanstalt.[10] A review of recent work in this field and contact information for active investigators can be found in the Proceedings of the International Workshop on Neutron Field Spectrometry in Science, Technology and Radiation Protection.[11] The lowest neutron energies of radiation protection concern are seen only by the gas counters, but the liquid scintillation counters are needed above 3 MeV. Gamma-ray spectrometry is an integral part of the PFNA system, so that the system may be self-validating in regard to gamma-ray spectra. Conventional gamma-ray spectrometry should be possible except in the direct neutron beam.

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[8] Goldhagen, P., DOE, Environmental Measurements Laboratory, New York, goldhagen@eml.doe.gov.

[9] Thomas, D.J., National Physics Laboratory, Teddington, United Kingdom; *Neutron Spectrometry for Radiation Protection*, http://www.npl.co.uk/npl/rad/services/rn0601.html.

[10] Klein, H., Physikalisch-Technische Bundesanstalt, Braunschweig, Germany; *Deconvolution of Measurement Data in Neutron Spectrometry*, http://www.ptb.de/en/org/6/64/642/entfaltung.htm.

[11] Available in Nuclear Instruments and Methods in Physics Research, Section A: Accelerators, Spectrometers, Detectors and Associated Equipment, Vol. 476 (2002).

5. Application of the Results

The results of the radiation transport code calculations described in Sections 3 and 4 can be expected to provide an estimate of *E* to an individual located at the position in the cargo container where *E* is the highest (*i.e.*, the maximum credible effective dose, denoted by *E**). Since it is possible for an inadvertently exposed individual to be at the location of *E**, the calculated value of *E** should provide the basis for operation of the PFNA system. However, if it is known that there are no individuals in the cargo container or truck cab (by prescreening or manned in-container inspections), this limitation does not apply.

The calculations also provide additional information on estimates of *E* to individuals located at nine locations distributed horizontally and vertically within the cargo container and for a variety of load compositions. This information will allow for the selection of *E* at the actual location of an inadvertently exposed individual, and for the actual load composition. As a result, for a retrospective estimate of *E* when the location of an individual is known, and/or the load compositions are known, the location and load dependent values of *E* may be used to assign a more appropriate value of *E* to this individual.

Further refinements in estimating *E* will require significantly greater modeling of container contents together with developments in techniques for characterizing cargo contents for each container scanned. Decisions on the desirability of such refinements will depend on the values of *E* calculated as given above, decisions by regulatory authorities on the need for more realistic values, and a clear understanding of the uncertainties in the calculated values.

6. Quality Control of PFNA System Performance

Confidence in the application of the calculated values of *E* is dependent on the stability of several physical and administrative conditions, as discussed below.

6.1 Configuration Management

Any change in the design of the shielding (composition or dimensions), the accelerator, the target, or the inspection tunnel from those used in the development of the codes could result in invalidating the calculated values of *E*. In order to ensure against any unrecognized impact from such changes, a configuration-management document listing those items that could affect the reliability of the calculated values should be available to designers, operators, maintenance personnel, and facility management.

6.2 Specific Quality Control Issues

6.2.1 *Fluence Rate*

Because the fluence rate measured at the collimator, or at some other specified location in the primary beam, is a function of beam current and target composition, both subject to change, the fluence rate becomes the most appropriate measure of the output of the accelerator. In addition, *E* at any point is dependent directly on the fluence rate and exposure duration. In view of the importance of this quantity, a number of quality-control actions are required.

A live-time display of the fluence rate should be available to the operators whenever the accelerator is in operation. The data should be recorded so that in the event of an inadvertent exposure, an estimate of the maximum credible effective dose (E^*) can be obtained by direct proportionality with the calculated value of E^* for the fluence (fluence rate times exposure duration) used in the calculation. Furthermore, if the intent is to operate the facility to meet the calculated E^* criteria, the fluence should be maintained below the fluence value used in the radiation transport calculation. The procedures for calibrating and maintaining this device, computer software for alerting the operators to changing values of fluence rate and operator training should be subject to, and part of, a quality control program.

6.2.2 *Target Assembly Rotation Rate*

The calculations of E depend not only on the fluence rate, but also on the scanning rate both vertically and horizontally. The rotation of the target assembly provides the vertical scanning rate. As a result, the rotation of the target assembly should occur at or above the specified rates, and procedures to establish and maintain the rotation rate values are essential to ensure that the calculated values of E continue to be applicable. Consideration should be given to tying rotation rates to accelerator operation. The procedures for calibrating and maintaining this device, computer software for alerting the operators to changes in the rotation rate, and operator training should be subject to, and part of, a quality-control program.

6.2.3 *Rate of Container Travel*

As noted in Section 6.2.2, the calculations of E depend not only on the fluence rate, but also on the scanning rate both vertically and horizontally. The horizontal scanning rate is determined by the speed of the cargo container. As a result, the tow vehicle, the truck and the cargo container should pass through the scanning beam at or above the specified rate and with a high degree of reliability. Vehicle travel rates should be tied to accelerator operation. The procedures for calibrating and maintaining this system, computer software for alerting the operators to changes in the travel rate and operator training should be subject to, and part of, a quality-control program.

7. Conclusions

Evaluation of the effective dose (E) received by each inadvertently exposed individual would require detailed evaluation of the composition and distribution of the load within the container where the individual was exposed, as well as the location of the individual within the load. Although such an analysis is possible, it is not needed to establish that the limit for E has not been exceeded. Instead, the maximum credible effective dose (E^*) for specified PFNA operating conditions should be determined in advance, and operating conditions should be set so that E^* does not exceed the limit for E.

A quality assurance program that ensures the operating parameters do not exceed specifications and that E^* has been properly re-evaluated after any significant modification of the system will ensure that actual values of E do not exceed the recommended limit.

It should be noted that if direct observation, prior to scanning the cargo container, has shown that no one is present in a given cargo container (including the truck and cab), the scanning does not need to be limited by the limit for E for inadvertently exposed individuals, and operating parameters that would result in a higher value of E could be selected.

It is also noted that E^* should be evaluated for each specific PFNA facility, and after any significant modification of a PFNA facility. In the latter case, the calculations will require validation if the modifications are such that the neutron spectrum or the backscatter conditions change significantly.

Evaluation of E^* requires calculation of E for load configurations and locations in the cargo container that are expected to result in the maximum value of E for any realistic load composition. The calculation should be conducted for sufficient additional configurations and compositions in order to unambiguously identify the value of E^*.

Validation of the calculated values of E requires identification of quantities which can be directly measured and which are sensitive to the factors that determine E (*i.e.*, absorbed dose and radiation spectrum). Measured values of these quantities are then compared with calculated values obtained using the same radiation transport code that is used to evaluate E^*. The limiting condition for operation of the PFNA system is that $E^* (1 + 2\, r_c)$ should be less than the recommended limit for E.

Appendix A.

Additional Recommendations in Previous Presidential Reports

A.1 Neutron Activation in Pharmaceuticals and Medical Devices

NCRP (2002a) concluded that activation of pharmaceuticals and medical devices by the PFNA system would not result in effective dose (E) values of concern for public health. Absorbed doses calculated for the elements expected to produce the highest values of absorbed dose, using the specified neutron fluence for the PFNA system, will result in values for E that are far less than the limit for E (*i.e.*, 1 mSv for the general public, from all sources, excluding medical and natural background; see Section A.2). This conclusion is based on the findings that: (1) the maximum whole-body absorbed dose, due to consumption of activated pharmaceuticals, is less than 10^{-6} mGy; and (2) the maximum absorbed dose rate due to activated medical devices is less than 10^{-9} mGy h^{-1} at 5 cm, and such values occur only near the device.

A.2 Applicable Radiation Protection Dose Limits for Occupational Exposures and for Members of the Public

The following recommendations were referred to in NCRP (2003). For occupational exposures to radiation workers, NCRP (1993) recommends that the cumulative lifetime effective dose not exceed the age of the individual in years times 10 mSv. NCRP (1993) also recommends the use of 50 mSv as the limit on the annual effective dose, and that new facilities be designed to prevent the annual effective dose from exceeding 10 mSv.

NCRP (1993) recommends that continuous or frequent exposure of members of the public be limited to an annual effective dose of 1 mSv. This limit excludes exposures from natural background radiation and radiation exposure associated with medical diagnosis and treatment. The limit applies to the sum of exposures from all other man-made sources, not to each source individually. Generally, the exposure from a single source or set of sources controlled by any single entity should be constrained to 0.25 mSv annually (NCRP, 1993).

A.3 The ALARA Principle and its Application to the PFNA Facility

The ALARA principle (*i.e.*, keeping exposures as low as reasonably achievable, taking into account economic and social factors) has been introduced into radiation protection programs because of the prudent assumption that potential deleterious effects might occur at any level of exposure, while recognizing that as the doses become smaller and smaller, the likelihood of a deleterious effect becomes less and less.

The ALARA principle should not be misinterpreted as simply a requirement for dose reductions irrespective of the dose level; sound judgment is essential in its proper application. Nevertheless, even at very low exposure levels, if simple and low-cost means would result in still lower exposures while retaining the beneficial outcome, ALARA considerations would indicate that such means should be implemented.

In NCRP (2003), advice on application of the ALARA principle was provided on four aspects of the operation of the PFNA. Application of the ALARA principle to inadvertently exposed persons was noted in Section 2.2.3; the other three aspects are listed here.

- For tritium production and release: (1) ensure there is proper equipment to manage tritium that is produced in the beam line, (2) minimize the risk that the target assembly where tritium accumulates over time will rupture during operation, and (3) decide how to manage tritium produced in the target *via* venting it to the atmosphere or capturing it as a solid or liquid radioactive waste.
- For neutron activation of foodstuffs: the production of trace amounts of radioisotopes *via* neutron activation in foodstuffs would be at a level similar to that for pharmaceuticals and medical devices (NCRP, 2002a). Significant effort or cost to reduce dose from foodstuffs is not warranted.
- For radiation levels outside the facility: dose rate to individuals outside the facility is directly related to the neutron fluence rate used to detect contraband. There are standard radiation protection measures to reduce the radiation levels reaching individuals outside the facility. These methods are: (1) increasing the shielding thickness, (2) increasing the distance between the source and those individuals, or (3) reducing the amount of time during which the individuals are exposed. Decisions to implement these measures should be based on evaluation of the overall costs and benefits, which is the usual approach to implementing the ALARA principle.

A.4 Advice on the System Safety Specifications and the Radiation Safety Plan

The system safety specifications need to set forth the basic requirements for radiological safety of the PFNA system. These specifications should be consistent with applicable federal and state regulations, the recommendations of the NCRP, the ALARA principle, and Customs Service policy. Maximum acceptable dose rates at specific locations can be determined based on the applicable annual limit, the maximum source operating time per year, and the maximum time any individual would be present in the area per year.

The radiation safety plan is the detailed policy and procedures for the implementation of the system safety specifications for the PFNA system, and needs to be specific to each installation. The radiation safety plan should incorporate both engineered and administrative procedures for ensuring that the requirements of the system safety specifications are met.

For a PFNA facility, a key part of the radiation safety plan is a system for controlling access to areas where elevated radiation exposures can occur. Experience has shown that this is most easily accomplished by defining a set of nested areas that are characterized by differences in maximum exposure, access requirements, and training requirements. In a typical PFNA facility there would be four classifications: (1) uncontrolled access, (2) controlled access, (3) restricted access, and (4) radiation-generating-device area.

A summary of the characteristics that define the four types of areas is provided in Table A.1 (NCRP, 2003).

Table A.1 — *Summary of the characteristics that define the four types of areas (1, 2, 3 and 4), giving the area classification, applicable annual effective dose limit, access control, and individuals who have access (including location of inadvertently exposed persons) (NCRP, 2003).*

Area Classification	Annual Effective Dose Limit	Access Control	Training	Individuals with Access
Area 1 (uncontrolled)	0.25 mSv	none	none	general public
Area 2 (controlled)	0.25 mSv	authorization	GERT[a]	non-radiation workers; escorted visitors
Area 3 (restricted) no neutron beam	0.25 mSv	authorization	PFNAT[b]	PFNA operators; escorted visitors
neutron beam	NA	no access	NA	NA
neutron beam	1 mSv[c]	NA	NA	inadvertently exposed persons
Area 4 (RGD room) no high voltage	0.25 mSv	authorization	RGDT[d]	RGD operators; escorted visitors
high voltage[e]	50 mSv[f]	limited to radiation workers	RWT[g]	radiation workers
neutron beam	NA	no access	NA	NA

NA, not applicable
[a] GERT, general employee radiation training
[a] PFNAT, PFNA system operator training
[c] The effective dose limit can be raised to 5 mSv, if necessary, for national security objectives (NCRP, 2002a)
[d] RGDT, radiation-generating device operator training
[e] Usually only during RGD maintenance
[f] New facilities should be designed to not exceed the 10 mSv per year limit implied by the lifetime limit (NCRP, 1993)
[g] RWT, radiation worker training

Additional advice in NCRP (2003), was that the radiation safety plan should also address: (1) radiation monitoring, including methods for personnel monitoring, active area monitoring locations, and passive area monitoring for long-term compliance with the appropriate dose limits; (2) the placement of active radiation monitors in Areas 3 and 4 in order to provide an alarm if radiation levels exceed normal operating parameters; (3) meeting the access-control requirements for Areas 3 and 4, such as the requirement that any interlock that stops the progress of the container through the PFNA scanner should also stop neutron beam production; (4) designing shielding walls enclosing Area 4 and the boundaries of Area 3 to maintain the dose rate at any point in Area 2 below a value required to prevent individual effective doses (*i.e.*, to nonradiation workers and escorted visitors) in excess of 0.25 mSv y^{-1}; and (5) the appropriate responses for radiological emergencies such as accidental exposures or accidental release of radioactive material (*e.g.,* rupture of a deuterium target containing significant tritium), and for other types of emergencies (*e.g.,* such as a fire or natural disaster) that may be exacerbated by the radiation generated.

A.5 Potential Effects on Nuclear Weapons in Scanned Cargo

The maximum neutron fluence rate that can be produced by the PFNA accelerator is minuscule compared to that needed to cause a significant yield of secondary neutrons or thermal energy from a clandestine nuclear weapon or cache of nuclear material located in a cargo container. Thus, neutron irradiation from the PFNA system cannot cause a clandestine nuclear weapon to detonate by direct action.

Appendix B.

Effective Dose Formulations

This Appendix presents the dosimetry formulations pertinent to the determination of effective dose, as outlined in Section 2.3.

B.1 General Formulation

The dose limits for delayed stochastic effects are expressed in effective dose (E), where:

$$E = \sum_{\mathrm{T}} w_{\mathrm{T}} H_{\mathrm{T}} \quad , \tag{B.1}$$

H_{T} is the equivalent dose in an organ or tissue T and w_{T} is the tissue weighting factor (ICRP, 1991; NCRP, 1993).

The quantity H_{T}, for stochastic effects, is usually obtained as:

$$H_{\mathrm{T}} = \sum_{\mathrm{R}} w_{\mathrm{R}} D_{\mathrm{T,R}} \quad , \tag{B.2}$$

where $D_{\mathrm{T,\,R}}$ is the mean absorbed dose in an organ or tissue T for a given type of radiation R, and w_{R} is a nominal radiation weighting factor used in most radiation protection situations that accounts for the biological effectiveness of radiation type R. The radiation weighting factor (w_R) applies to the type of radiation incident on the body.

For complex radiation spectra, the radiation transport approach allows one to obtain point values of absorbed dose (D) and dose equivalent (H) [*i.e.*, using D and the appropriate $Q(L)$ relationship, rather than w_R, to obtain H]. Therefore, this Report recommends that the quantity H and the currently recommended $Q(L)$ relationship be used to evaluate the complex radiation distribution inside the body resulting from a PFNA scan.

The dose equivalent (H) is defined at a point (ICRU, 1993a), and also can be evaluated based on appropriate measurements. The quantity H is given by:

$$\mathrm{H} = \int_{\mathrm{L}} Q(L)\, D(L)\, \mathrm{d}L \tag{B.3}$$

where $Q(L)$ is the quality factor for radiation with linear energy transfer L and $D(L)$ is the spectral distribution in terms of L of the absorbed dose at the point.

When an average value over an organ or tissue is required, it can be obtained by means of computational models or measurements using anthropomorphic phantoms and defined sites for organs or tissues. In this case, the point quantity H at multiple locations in an organ or tissue can be used to obtain the organ dose equivalent (ICRU, 1993a), given the symbol $\overline{H}_T$ by NCRP (NCRP, 2002b). The quantity $\overline{H}_T$ was adopted by NCRP as an acceptable approximation for H_T (NCRP, 2000; 2002b).

Therefore, in general terms:

$$\overline{H}_{\mathrm{T}} = \frac{1}{\mathrm{M}_{\mathrm{T}}} \int_{\mathrm{x}} \int_{\mathrm{L}} Q(L)\, D(L)\, \mathrm{d}L\, \rho\, \mathrm{dx}, \tag{B.4}$$

where there is a second integration over the points x in tissue T with tissue density $r(x)$ and total mass M_{T}. The special name of the unit for the quantities E, H, H_{T} and $\overline{H}_{\mathrm{T}}$ is the sievert (Sv).

The quality factor relationship as a function of linear energy transfer [$Q(L)$] is given in ICRP (1991), ICRU (1993a), and NCRP (1993), where:

$$\begin{aligned} &Q(L) = 1 && \text{for } L < 10 \text{ keV } \mu\text{m}^{-1} \\ &Q(L) = 0.32\, L - 2.2 && \text{for } L = 10 \text{ to } 100 \text{ keV } \mu\text{m}^{-1} \\ &Q(L) = 300\, L^{-1/2} && \text{for } L > 100 \text{ keV } \mu\text{m}^{-1} \end{aligned} \tag{B.5}$$

For example, for a value of L equal to 20 keV μm^{-1}, the value of $Q(L)$ would be 4.2.

Since $\overline{H}_{\mathrm{T}}$ is an acceptable approximation for H_{T}, E can be obtained from:

$$E = \sum_{\mathrm{T}} w_{\mathrm{T}} H_{\mathrm{T}} \approx \sum_{\mathrm{T}} w_{\mathrm{T}} \overline{H}_{\mathrm{T}} \tag{B.6}$$

B.2 Specific Application to Monte Carlo Calculations

When using Monte Carlo methods to evaluate $\overline{H}_{\mathrm{T}}$, Equation B.4 can be applied assuming that energy transferred from an indirectly ionizing particle (*e.g.*, a neutron) to a directly ionizing charged particle (*e.g.*, a proton) is deposited locally, that is, at the interaction site. This is justified because the range of the charged particle is much less than the dimensions of the tissue or organ. Since Monte Carlo calculations tabulate events in terms of the type of the resulting charged particle (m) and its kinetic energy (T_{m}), it is convenient to reformulate Equation B.4 as a summation over all n events of each type and energy within the boundary of the tissue. This can be accomplished by recognizing that for each type of charged particle m there is a unique relationship [*i.e.*, the stopping power formula (ICRU, 1993b)] between L and T_{m}. The mean quality factor relationship, as a function of T_{m} [*i.e.*, $\overline{Q}(T_{\mathrm{m}})$] can be defined for each energy T_{m} of each type of particle m. The quantity $\overline{Q}(T_m)$ is evaluated for the track of a particle with initial energy T_{m}. Thus,

$$\overline{H}_{\mathrm{T}} = \frac{1}{M_{\mathrm{T}}}\left[\sum_{\mathrm{m}}\sum_{T_{\mathrm{m}}}\sum_{\mathrm{n}} \overline{Q}(T_{\mathrm{m}})T_{\mathrm{m}}\right], \qquad \text{(B.7)}$$

where n are the interaction sites as indicated above in tissue T and M_{T} is the total mass of the tissue of interest.

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