Real-time 3D radiation risk assessment supporting simulation of work in nuclear environments

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Abstract
This paper describes the latest developments at the Institute for Energy Technology (IFE) in Norway, in the field of real-time 3D (three-dimensional) radiation risk assessment for the support of work simulation in nuclear environments. 3D computer simulation can greatly facilitate efficient work planning, briefing, and training of workers. It can also support communication within and between work teams, and with advisors, regulators, the media and public, at all the stages of a nuclear installation’s lifecycle. Furthermore, it is also a beneficial tool for reviewing current work practices in order to identify possible gaps in procedures, as well as to support the updating of international recommendations, dissemination of experience, and education of the current and future generation of workers.

IFE has been involved in research and development into the application of 3D computer simulation and virtual reality (VR) technology to support work in radiological environments in the nuclear sector since the mid 1990s. During this process, two significant software tools have been developed, the VRdose system and the Halden Planner, and a number of publications have been produced to contribute to improving the safety culture in the nuclear industry.

This paper describes the radiation risk assessment techniques applied in earlier versions of the VRdose system and the Halden Planner, for visualising radiation fields and calculating dose, and presents new developments towards implementing a flexible and up-to-date dosimetric package in these 3D software tools, based on new developments in the field of radiation protection. The latest
versions of these 3D tools are capable of more accurate risk estimation, permit more flexibility via a range of user choices, and are applicable to a wider range of irradiation situations than their predecessors.

Keywords: radiation risk, radiological protection, radiation transport, 3D modelling, point kernel approach

(Some figures may appear in colour only in the online journal)

1. Introduction

This paper describes recent developments at the Institute for Energy Technology in Norway, in the field of real-time 3D radiation risk assessment for the support of simulation of work in the nuclear sector. 3D computer simulations have been successfully applied to support work in environments where the risk of radiation exposure to workers, or the public and the surrounding environment, is elevated. 3D software tools can greatly facilitate efficient work planning, briefing and training of the workers, as well as communication within a team, between teams, with advisors, regulators and even the media and the public. Furthermore, 3D simulation can be of great benefit in all the stages of a nuclear installation’s lifecycle, beginning with the design phase, continuing with supporting maintenance and outage during operation, and concluding with the final decommissioning, remediation and documentation stages. In addition, it is also a valuable tool for applications such as reviewing actual practices to identify possible gaps in procedures, as well as to aid in the updating of international recommendations, dissemination of experience, and education of the current and future generations of workers in the nuclear industry.

The IFE has been involved in research and development into the application of 3D computer simulation and VR technologies to support work in nuclear environments since the mid 1990s. During this process two software tools have been developed, the VRdose system [27] and the Halden Planner [41], and, equally importantly, a number of reports and papers have been produced with substantially important results for the benefit of the scientific community, and as a contribution towards improving the training of future workers in the safety culture for the nuclear industry.

The Halden Planner and the VRdose system are sophisticated pieces of desktop 3D simulation software, suitable for planning and training for work involving possible exposure of humans to ionising radiation and other hazards. For a brief presentation of the Halden Planner and the VRdose system and their connection, the reader is referred to [42]. The technologies applied in the Halden Planner and VRdose are powerful tools for the support of work involving radiation risks, as they enable many possibilities, including the following:

- invisible radiation can be visualised in a virtual (or augmented reality) environment in real time;
- work scenarios can be taught and practised cost-effectively in a safe but realistic environment;
- communication between workers, regulators and other remote participants can be greatly enhanced.

In order to simulate a radiological environment, software tools, such as the Halden Planner and the VRdose system, have a number of specific data and processing requirements. Firstly, for calculating dose and visualising radiation, input data are required describing the environment
we wish to model. These data may describe the characteristics of the radiation sources in the environment, the entities affecting radiation transport (biological shields, walls, etc), the radiation field by means of measurements registered by various instruments, the movement of workers in the area of interest, etc. These data are then utilised by the algorithms to assess radiation risks (‘dosimetric approaches’). In the next step, the results calculated by the dosimetric approach are visualised for the user, ideally in a sufficiently sophisticated but user-friendly manner. The ultimate goal of these codes is to aid experts in making well-informed decisions, to reasonably minimise (and keep to acceptable levels) the exposures of humans involved, to support the ALARA (as low as reasonably achievable) dose planning principle.

The variety of techniques applied in radiation protection and shielding design is vast [24, 28, 33, 34, 39]. The selection of the dosimetric method most suitable for a specific situation is determined by (a) the radiological input data available (e.g. location of radiation sources or hot spots if any, geometry of the sources, available information on the energy spectrum or isotopic composition of the sources, existence of measurements, etc), (b) the exposure conditions, i.e. any parameters characterising the mode of the exposure (e.g. angular dependence of the radiation field, characteristics of the objects and media interacting with radiation, the route of the target, etc), (c) the output information required (e.g. dosimetric and/or radiometric quantities describing detector response), (d) the target of radiation that we wish to monitor, (e) the allowable inaccuracy of the results and (f) time constraints (e.g. emergency situation or non-stressful situation).

In this paper, we present the basic dosimetric approaches of the Halden Planner and the latest VRdose system, applicable when the sources of radiation and any objects interacting with the radiation field are ‘known’, i.e. there is information on the location, geometry and radiation emitted (energy spectrum or isotopic composition). In this fortunate situation, the most common techniques applied are Monte Carlo (MC) radiation transport simulations (e.g. MCNP, GEANT, etc), and point kernel (also called kernel integration) approaches [24, 26, 40] (e.g. MicroShield [11], QAD [5]). The first group of methods are, in general, very accurate in comparison with the point kernel methods, but are also relatively slow and difficult to use by non-experts, due to the very sophisticated description of radiation transport. In contrast, due to a much simpler theoretical approach, point kernel techniques are faster. However, as a consequence of the simplifications introduced, they are less precise and may break down in specific situations (e.g. very short distances between source and detector).

Earlier versions of the Halden Planner provided only one, relatively simple, dosimetric approach [43] for calculating radiation levels and visualising radiation fields generated by gamma sources. The underlying dosimetric technique is a basic point kernel approach, well known in the field of protection against gamma radiation. While this approach gives a good estimate in most simple irradiation situations (e.g. standard photon energy spectrum, shielding geometry and composition), recent developments in the field of radiation shielding design, subsequent changes to the standard radiation protection strategy, and the availability of additional standard input data have paved the way for a more up-to-date and comprehensive approach.

The new model, developed in this work, adopts the results of recent developments in the field of radiation shielding, and incorporates a larger variety of standard input data (e.g. dose conversion coefficients and buildup factors) stemming from modelling and experimental efforts of various research groups around the globe. The new model is in line with new recommendations issued by institutions with a significant impact on the evolution of the international radiation protection standards, such as the ICRP (International Commission on Radiological Protection) [20], ICRU (International Commission on Radiation Units and Measurements) [18], the Radiation Protection and Shielding Division of the American Nuclear
2. Methods

2.1. The earlier point kernel approach

The dosimetric unit implemented in earlier versions of Halden Planner is based on a basic point kernel approach, developed some time ago in the field of radiological protection. This approach quantifies the radiation burden originating from gamma sources based on the distance between the source and the detector, taking into account attenuation and buildup of photons in the objects (shields) intersecting the direct path of radiation to the detector. The core of the point kernel approach is a simple formula for calculating the detector response due to gamma photons emitted by a point isotropic source, at a specified distance in a homogeneous infinite medium,

\[ R = \int dE \times \frac{1}{4\pi r^2} \times S(E) \times \exp(-\Sigma r) \times R(E) \times B(6r, E) \]  

(1)

where \( R \) symbolises the detector response (dose), \( E \) is the photon energy, \( r \) is the distance between the source and the detector, \( S \) is the specific activity (the number of photons emitted per unit volume), \( \Sigma r = \mu \times r \) is the optical thickness from the source to the detector, \( \mu \) is the linear attenuation coefficient and \( B \) is the buildup factor. Integration over the source and the target volume have been omitted for simplicity. For more detail on the considerations underlying the point kernel approach the reader is referred to the literature [24, 33, 34, 39].

The calculation is based on the following consecutive steps.

First, the initial (unshielded) particle fluence rate is calculated, and then the attenuation of the uncollided beam is determined on the basis of the distance travelled by the photons in the shield on their way to the detector. In the next step, the radiation burden from the uncollided photons reaching the detector is characterised by calculating the transfer of energy from photons to absorber by simple methods (described later), utilising related mass-energy absorption coefficients.

To estimate the contribution of the scattered photons to the response of the detector, standard input data from the literature, so-called buildup factors [14, 35–37], are utilised. Calculation of the contribution of the collided photons, which may undergo multiple scattering in the shield, is a very complex task, and requires experimental results, or results based on sophisticated radiation transport models. Buildup of radiation in matter depends on a number of parameters, the most important of which are (a) the energy (or energy spectrum) of the radiation, (b) the atomic composition and (c) the density of the medium, (d) the distance travelled by the radiation before reaching the detector, as well as (e) the quantity (detector response) measured or calculated [33]. In more realistic situations we have a heterogeneous medium between the source and the detector, e.g. multi-layers of shielding and a variety of additional objects interacting with the radiation, which complicate things even further. However, standard data on photon buildup, for the most common photon energy range (15 keV–15 MeV), optical density (mfp) of the medium the photons traverse, engineering materials applied in shielding design, and simple (e.g. infinite) shielding configuration are provided in the literature for general use. These buildup factors are commonly applied in radiation protection practice to account for buildup of photons in real (more complex)
irradiation situations, in cases where the inaccuracies introduced by the differences between the standard and real exposure conditions are acceptable.

The final output quantity, referring to the biological effectiveness of the radiation, is obtained by multiplying the exposure rate (free in air), calculated as described above, by a constant based on simple assumptions (explained later).

Based on the above, the dosimetric model is based on the following equations, as reported in [43]:

$$\dot{X}_{\text{air}} = 5.26 \times 10^{-6} \frac{A}{d^2} \sum_i y_i E_i (\mu_{en}/\rho)_i \times \exp(-\mu/\rho) \times B_i(E_i, \rho, r)$$  \hspace{1cm} (2)

$$\text{Dose} = \dot{X}_{\text{air}} \times 9.7 \times 1$$  \hspace{1cm} (3)

where $\dot{X}_{\text{air}}$ is the exposure rate (R h$^{-1}$), $d$ is the distance to the detector (cm), $A$ is the source activity (Bq), $y_i$ is the yield of photons of energy $E_i$, $E_i$ is the photon energy (MeV), $(\mu_{en}/\rho)_i$ is the mass-energy absorption coefficient at energy $E_i$ (cm$^2$ g$^{-1}$), $(\mu/\rho)_i$ is the mass attenuation coefficient at energy $E_i$ (cm$^2$ g$^{-1}$), $\rho$ is the density of the shield (g cm$^{-3}$), $r$ is the distance travelled in the shield (cm), $B_i(E_i, \rho, r)$ is the exposure buildup factor at energy $E_i$, and Dose is the dose equivalent rate (mSv h$^{-1}$).

Equation (2) can be rewritten in the following form:

$$\dot{X}_{\text{air}} = 6.6 \times 10^{-5} \times \sum_i \left( \frac{A \cdot y_i}{4\pi d^2} \times \exp(-\Sigma r(E_i)) \times E_i (\mu_{en}/\rho)_i \times B_X(\Sigma r(E_i)) \right)$$  \hspace{1cm} (4)

where the integral in equation (1) has been replaced by a rougher summation, $\Sigma r(E_i) = \mu_i \times r$ is the optical thickness ($\mu_i$ is the linear attenuation coefficient) at energy $E_i$, $B_X(\Sigma r(E_i))$ is the exposure buildup factor, and the constant ($6.6 \times 10^{-5}$) is a conversion of units, which can be expressed as follows:

$$10^6 \left[ \frac{\text{eV}}{\text{MeV}} \right] \times \frac{1}{33.8 \left[ \text{eV} / \text{ion pair} \right]} \times \frac{1}{6.24 \times 10^{18} \left[ \text{C} / \text{ion pair} \right]}$$
$$\times 10^3 \left[ \frac{\text{g}}{\text{kg}} \right] \times 3600 \left[ \frac{\text{s}}{\text{h}} \right] \times \frac{1}{2.58 \times 10^{-2} \left[ \frac{\text{R}}{\text{kg}} \right] \left[ \frac{\text{kg}}{\text{C}} \right]}.$$  \hspace{1cm} (5)

Equation (4) nicely reveals the consecutive steps of the radiation transport calculation described above and that the technique is, in fact, an application of equation (1).

The equations above, for modelling radiation transport and energy deposition, are based on strong idealisations. In spite of the simplifications, this technique yields surprisingly accurate estimates for common situations, i.e. standard photon energies, simple irradiation situations (e.g. simple shielding composition), etc.

The first two terms in the summation in equation (4) calculate the initial and the attenuated fluence rate of the uncollided photons. The third term in the summation in equation (4) calculates the energy transferred to the absorbing medium by multiplying the fluence rate (on the detector side of the shield) by the mass-energy absorption coefficient and photon energy. The mass-energy absorption coefficient $(\mu_{en}/\rho)$ defines the ratio of photon energy transferred to the medium, excluding the energy retransferred to secondary photons through radiative processes. Hence, in equation (4) the energy transferred to charged particles from the uncollided photons, not including energy converted to secondary photons, is calculated. However, the overall fluence rate of photons, generated by a point isotropic source at a given detector position, includes Compton scattered photons, photons from emission of Bremsstrahlung, fluorescence x-ray photons and annihilation gamma photons [1, 26, 33].
generated by the medium surrounding the detector position. Since the photon fluence rate, calculated as described above, neglects these secondary photons, the calculated detector response will be underestimated. Calculation of the fluence including secondary photons is much more complicated than calculation of the primary fluence. Instead, to solve common radiation protection problems, a convenient simplification is usually applied.

A simple way of including the contribution of the scattered photons indirectly reaching the detector is multiplication by a suitable buildup factor (last term in the summation in equation (4)), applicable to the photon energy, absorber (shield) type and distance travelled through the absorber (shield thickness). This simple method is frequently applied in current radiological protection practice. If a more sophisticated characterisation of the energy spectrum of particle fluence is needed, then Monte Carlo radiation transport models are usually applied. While MC models provide a more detailed description of the radiation transport, they require a more detailed description of the exposure conditions, and are not generally suitable for non-expert users and real-time simulation.

In the following step, the rate of energy absorbed in matter at the detector is converted into the exposure rate (free in air) by application of a constant \( \times 10^{-5} \) multiplier. In general, the constant conversion from MeV g\(^{-1}\) s\(^{-1}\) to R h\(^{-1}\) slightly depends on energy. This is mainly due to the fact that the number of ion pairs produced by 1 eV depends on the circumstances of the exposure. On the other hand, 1 eV corresponds to 33.8 ion pairs with a ±0.2 eV uncertainty for photon energies between 20 keV and 3 MeV for 50% relative humidity and 22 °C temperature.

In the final steps of this dosimetric approach, the exposure rate (free in air) is multiplied by 9.7 to obtain the dose rate absorbed in tissue, expressed in mGy h\(^{-1}\), and then multiplied by the so-called quality factor of the radiation, which equals one for gamma radiation, to obtain the tissue dose equivalent rate (organ equivalent dose rate).

The constant conversion of 9.7 is determined by taking into account that 1 eV equals 1.6 × 10\(^{12}\) J (joules). According to the above (see equation (5)), 1 R can be converted into mJ kg\(^{-1}\) s\(^{-1}\) as follows:

\[
1 \text{ R} = 2.58 \times 10^{-4} \left( \frac{\text{C}}{\text{kg}} \right) \times 6.24 \times 10^{18} \left( \frac{\text{ion pair}}{\text{C}} \right) \times 33.8 \left( \frac{\text{eV}}{\text{ion pair}} \right) \times 1.6 \times 10^{12} \left( \frac{\text{J}}{\text{eV}} \right).
\]

Following from equation (6), we have 1 R h\(^{-1}\) = 8.7 mJ kg\(^{-1}\) h\(^{-1}\), which is a good approximation for energies ranging from 20 keV to 3 MeV. In terms of energy absorption, there is no significant difference between air and soft tissue. Indeed, in the range from about 15 keV to 3 MeV, the ratio of the mass-energy absorption coefficients for soft tissue and air is quite constant [16], and is equal to roughly 1.1, which is equivalent to the ratio of the constant 9.7 multiplier (see equation (3)) to the 8.7 conversion constant. Consequently, application of a 9.7 constant to convert exposure (free in air) in R h\(^{-1}\) to energy absorbed in soft tissue mJ kg\(^{-1}\) h\(^{-1}\) is a good approximation in most cases.

Absorbed dose is defined as the mean energy absorbed by matter per mass of the matter in which the energy has been absorbed \( (d\bar{e}/dm) \). From the definition, it can be seen that while the expression in equation (4) describes the energy transferred to charged particles per unit mass \( dm \), produced in \( dm \) but absorbed anywhere (partially outside \( dm \)), absorbed dose refers to energy absorbed only in \( dm \) (partially from charged particles from outside \( dm \)). Thus, absorbed dose can be calculated as in equation (4), if the energy carried outside \( dm \) is equal to the energy coming from outside \( dm \), i.e. charged particle equilibrium (CPE) exists, where energy transfer and energy absorption are equal [1, 12].
Equilibrium of charged particles requires (a) negligible radiation loss, (b) uniform radiation field and (c) homogeneous matter (in terms of both density and atomic number). In our case, absorption of energy in the human body, usually immersed in air, is of most interest. Based on [1], the first condition of CPE is fulfilled for photon energies up to $\sim 10$ MeV, above which radiation loss becomes increasingly important, due to the high energy of the generated primary electrons. The second condition of CPE (uniform radiation field) requires that the radiation be only negligibly attenuated in matter. By inspecting standard attenuation coefficients [15] for air and for soft tissue it is easy to see that both air and soft tissue have a low attenuating effect on high-energy gamma photons. However, soft tissue attenuates relatively strongly, compared to air, in the region of low photon energies, compromising CPE in this region. This means that at very low energies the skin absorbs most of the energy, and provides strong shielding for internal organs. The third criterion of CPE is that the absorber is homogeneous, in terms of both density and atomic number. The human body can be regarded as relatively homogeneous in terms of interaction with gamma radiation. Inhomogeneity is only considerable, affecting CPE, near bone surfaces and the skin–air interface.

As mentioned previously, the selection of appropriate buildup factors also depends on the quantity we wish to calculate. In the model described above, developed during earlier work, exposure buildup factors are applied, conforming to the international recommendations of that time. As explained earlier, the expression in equation (4) gives a good estimate of the exposure (free in air) for a wide photon energy range, and is especially accurate for photon energies between 0.2 and 3 MeV. However, the calculated exposure rate is then converted into organ equivalent dose rate (tissue dose equivalent rate), which, as explained above, has different behaviour (compared to exposure) at low photon energies. Since an exposure buildup factor $B(E_0)$, at a given energy $E_0$, accounts for the contribution of scattered photons to the calculated exposure [26],

$$B(E_0) = \int_0^{E_0} \frac{X(E) dE}{X(E_0)}, \tag{7}$$

its application is only justified for the calculation of detector response types that, compared to exposure, have a similar energy dependence.

The assumptions of the dosimetric model, described above, and the connection to air kerma ($K_{air}$) are summarised in figure 1.

Based on the above, we can thus conclude that the dosimetric approach described provides a good approximation in most common radiation protection cases, and considerable inaccuracy can only be expected in, from a nuclear industry perspective, special situations, e.g. exposure to very low photon energy (below 0.2 MeV), especially at bone surfaces and red marrow, and exposure to very high-energy photons (above $\sim 10$ MeV).

Below, we present how the input parameters required by the basic formulae described above are calculated, and how this computational model is applied for the simulation of radiation transport in more complex situations.

As can be seen from equation (4), one of the input parameters required is the energy spectrum ($E_i$, $A \times y_i$) of the radiation emitted by the sources in the modelled scene. This is calculated based on the activity and the isotopic composition, defined by the user for each radiation source. The characteristic energies and associated fractional yields are fetched from an internal database of the software tool, which was created based on standard data from the literature [43]. At the time, input data for the four most interesting isotopes, i.e. $^{58}$Co, $^{60}$Co, $^{110m}$Ag, $^{137}$Cs (and $^{137m}$Ba in secular equilibrium) were coded, with the possibility of adding supplementary isotopes for later applications.
Figure 1. Assumptions underlying the dosimetric model ($Q_{\text{air}}$: total charge of ions of one sign, $m$: mass, $\Psi^*_{\text{(att)}}$: direct energy fluence, $\bar{\varepsilon}_{\text{tissue}}$: mean energy absorbed in tissue).

The distance between the source and the ‘detector’ (see explanation later) ($d$) is simply calculated from the actual position of the source and the detector.

The next input parameter of equation (4) is the optical thickness ($\Sigma r$) of the medium between the source and the detector. This input is needed to calculate both the attenuation of the radiation along a straight pathway between a point source and the detector, and the contribution of photon buildup to the detector response. In principle, $r$ equals $d$; however, in typical situations, the radiation would travel through layers of different composition, most of the time including air, before reaching the detector. Air has little attenuating effect on gamma photons, so, as a convenient simplification, air is neglected altogether in this model, and $r$ refers to the optical thickness of the medium, disregarding the air on a straight pathway from the source to the detector. The linear attenuation coefficient ($\mu$) depends on the photon energy,
and the type and density of the medium through which the photons travel before reaching the detector. Since this parameter depends on the absorber material, the optical thickness has to be calculated separately for each layer the radiation traverses. The total optical thickness is then found by adding the optical thicknesses of all layers together ($\Sigma r = \Sigma_i \mu \times r_i$).

The earlier point kernel model was implemented with the application of standard linear attenuation coefficients for the four most important engineering materials (lead, iron, concrete and water) drawn from the literature of the time [43]. The linear attenuation coefficients applied in the model neglect coherent scattering, and consider free electrons in incoherent scattering.

The mass-energy absorption coefficient ($\mu_{en}/\rho$) is a function of the photon energy and the target absorber material. The target material in this model is air. The model applies energy-dependent values of this coefficient drawn from the literature of the time.

The last input of equation (4) is the buildup factor. As mentioned earlier, this parameter is also dependent on the material the photons travel through. Consequently, photon buildup is calculated in each consecutive layer of supported shielding material separately, similarly to the technique applied for calculating attenuation, based on the energy of the photons passing through, and the same optical thickness that is used to determine the attenuation (i.e. based on the intersection of the layer with a straight pathway from the source to the detector). Tabulated values for different photon energies, optical thicknesses and four engineering materials are utilised by the model from the standard literature of the time [43]. For transitional parameters, linear interpolation of the buildup factors corresponding to tabulated energy and optical thickness values is performed. The total buildup in all intersecting layers is computed by simply multiplying the buildup of photons in each layer.

As mentioned above, buildup factors are usually produced using more sophisticated radiation transport modelling results for standard irradiation situations (e.g. homogeneous infinite medium). Hence, application of these factors for more special situations (e.g. complex shielding configuration of multiple layers, where photons may traverse layers in very slanted pathways) introduces increasing error depending on the complexity. However, for radiation protection purposes, where conservatism is usually a requirement, this technique gives reliable results in practice, and is extensively utilised for solving standard radiation protection problems.

The technique described above is suitable for quick estimates of the risk of exposure to gamma sources that have negligible size in comparison with their distance from the detector, i.e. point sources. However, modelling of a real radiation source using just one point source results in increasing error as the size-to-distance ratio of the source increases. Consequently, extended sources are decomposed into point sources in the dosimetric model as follows. First, the detector response is calculated for an imaginary worst-case scenario, where all the activity is compressed to the point of the source that is closest to the detector. The procedure is repeated considering that all activity is at the furthest point. Then, based on the difference between these two results, the source is uniformly split into equivalent pieces, each piece represented by a point source, to calculate the total detector response.

In terms of this dosimetric model, ‘detector’ means a virtual point isotropic detector, which can either be attached to a manikin, modelling a personal dosimeter worn by a human participant, or can scan the whole scene in discrete steps, in order to map the scene in terms of exposure. While the first case is used to monitor the radiation risks to workers, the second is utilised to visualise the distribution of the exposure within the whole 3D scene (figure 10).
2.2. The new point kernel approach

As indicated earlier, since the implementation of the dosimetric approach described above, numerous new developments have been made by the authors of this paper to better support risk assessment associated with work in nuclear environments. In the frame of these developments, a new dosimetric model has been elaborated to supplement the one just described. The motivations for developing the new model are described below.

In the development of the earlier model, calculation speed and simplicity were of the utmost priority. Thus, the input data were integrated into the model implementation. This enables high computation speed, but prevents users from extending the list of supported isotopes and materials, or updating the applied standard radiological input data (attenuation and absorption coefficients, buildup factors, etc). Additionally, the integrated approach does not provide users with enough flexibility, for instance for selecting the desired output information from a list of currently applied radiometric and dosimetric quantities. Since the primary purpose of the software tool is to support work involving possible radiation exposure so that the radiation risk to humans is as low as reasonably achievable, it is important that the radiation burden calculated can be compared with the limits applicable to a particular situation.

It is quite a common situation to have a source shielded by some solid and/or liquid layers, with the detector being at some distance in air from the shield. Since air is not supported as a material, both attenuation and buildup of photons in air are neglected. Although, air has a low interaction with penetrating radiation, extensive layers of air may have a non-negligible effect, resulting in increasing inaccuracy at long distances from the radiation sources.

In the earlier model, extended sources are uniformly disaggregated into point sources based on the difference between a worst-case and best-case scenario. This uniform splitting technique results in either high inefficiency (if a very fine splitting is performed), or lower accuracy (if a rougher splitting is applied) when the detector is close to a large extended source.

Since the development of the earlier model, updates of earlier published buildup data [9], and buildup data for additional photon energies and shield thicknesses have been published, and the literature providing standard photon buildup data is continuously expanding. As users of the earlier dosimetric model are not able to account for these new findings, the database is inaccessible to them. For example, the largest shield thickness reported in [2] is 40 mfp (mean free path). In contrast, there are buildup factors associated with exposure and other quantities for up to 100 mfp available in more recent publications (e.g. [30–32, 44]).

All the issues listed above narrow the scope of applicability of the model. In order to extend the applicability, a new dosimetric approach has been implemented.

The international market for software tools that can be utilised for radiation transport simulations and dosimetric calculations has been evolving and expanding for many years now. Some of these models implement a complex radiation transport approach. These very accurate models are, in general, unsuitable for rapid and real-time assessments in dynamically changing environments, due to the time required for setting up and completing computations for realistic exposure situations. In contrast, there are also very simple models that are able to yield a reasonably reliable estimate much more quickly; but these are, of course, less accurate. In addition, some of the commercially available models have sophisticated visualisation tools capable of 3D realistic representation of the results. It has to be stressed that advanced visualisation of the results is just as important as state of the art simulation of radiation transport. Ultimately, not the computer but humans utilising a model and looking at a computer display will make decisions and perform jobs, based on their perception of the information provided by the model. 3D simulation tools for advanced planning and virtual reality tools for immersive training have undergone an immense evolution, due to the revolution of the
underlying computer technology. This area is in very rapid development, and has relatively recently gained focus in the nuclear sector. Hence, there is the room and a strong need for further development in this area. The new dosimetric model has been developed to supplement the earlier model, and provide an even more powerful tool for improving the safety culture in the nuclear industry.

The theoretical assumptions underlying the new dosimetric model are described below. This new model is based on a more up-to-date and flexible application of the basic point kernel approximation,

\[
R = \sum_i \left( \frac{A y_i}{4\pi r^2} \times \exp(-\Sigma r(E_i)) \times CF(E_i) \times B(\Sigma r(E_i)) \right)
\]  

where \(R\) is the detector (point isotropic) response (dose) (R h\(^{-1}\), Gy s\(^{-1}\), Sv s\(^{-1}\), ...), \(A\) is the activity of the source (point isotropic) (Bq), \(y_i\) is the yield of photons at photon energy \(E_i\), \(r\) is the distance from the source to the detector (cm), \(\Sigma r\) is the optical thickness of the medium between the detector and the source (including air), CF is the conversion coefficient at energy \(E_i\) (R \(\times\) cm\(^2\), Gy \(\times\) cm\(^2\), Sv \(\times\) cm\(^2\), ...), \(B(\Sigma r(E_i))\) is a suitable buildup factor.

Note that in this new implementation of the basic theory the radiological input constants and, as a consequence of this, the calculated detector response, are treated in a more flexible manner, permitting customisation of the dosimetric approach to a specific situation. The selection of standard radiological data, for application in deterministic radiation transport calculations, is subject to continuous development. In order to provide users with a wide range of selectable options, all the applied standard radiological input data (e.g. attenuation and absorption coefficients, conversion factors, buildup factors, etc) are fetched from a user-accessible database. The database contains a large selection of input parameters based on international standard publications, with the possibility to update and extend the data. This permits the user to revise the constants applied in the dosimetric modelling based on state of the art (latest literature) in future. For example, the earlier dosimetric model incorporates attenuation coefficients that neglect or mistreat Bremsstrahlung and disregard coherent scattering. The configurable input file, however, allows users to apply data that allow for these minor effects [2, 15] to calculate attenuation in shields more correctly.

Calculation of the energy absorbed by the target (detector) from the uncollided photons has also been generalised by the application of suitable conversion factors as opposed to the earlier model, where photon energy \(\times\) mass-energy absorption coefficient is used to calculate energy absorption. Such coefficients (per unit fluence rate) can be found in standard publications for various photon energies, detector response types and exposure conditions. This also allows the output quantity (detector response) to be formulated in a more flexible way and, depending on the conversion coefficients applied, express the results in the desired radiometric or dosimetric quantities. For example, by selecting appropriate conversion coefficients (per unit fluence or unit kerma) from the default database (based on [16, 18, 19]) or introducing new coefficients from other sources, it is possible to obtain the results in a series of desired radiometric and dosimetric (protection and operational) quantities (figure 2), it is possible to calculate organ equivalent doses (figure 3) and it is even possible to take into account the angular dependence of the irradiation conditions (figure 4). For a detailed description of the radiological quantities in figures 2–4, the reader is referred to the literature [3, 4, 17, 18, 20, 33].

As figures 2–4 clearly show, utilisation of more general (multipurpose) radiological input constants should result in only slight change to the results. However, the importance of irradiation situation and detector type specific constants increases with the relative contribution of low-energy photons to the radiation burden.
Figure 2. Exposure $X$, kerma (free in air) $K_a$, effective dose (anterior–posterior irradiation) $E$, ambient $H^*$, directional $H'$, and personal dose equivalents $H_p$ per unit fluence (conversion factors). Data were taken from [16, 18, 19].

Figure 3. Organ equivalent doses and the effective dose, for anterior–posterior (AP) irradiation geometry, per unit air kerma, against photon energy. Data were taken from [19]. Note that according to the latest recommendations of the ICRP [20], a different categorisation of the organs and tissues of the human body has been proposed.

In order to check the validity of the affirmation above, standard fluence to detector response conversion coefficients were compared with the conversion method applied in the earlier model. The earlier model applies $(\mu_{en}/\rho) \times E$ to convert gamma fluence to exposure, and then...
a constant multiplier 9.7 to obtain the dose equivalent. Figure 5 compares this conversion method to standard conversion coefficients for a more practical case of the dose equivalent, the effective dose, over the 1 keV–10 MeV photon energy range. Figure 5 demonstrates that the term $9.7 \times (\mu_{en}/\rho) \times E$ can be applied as an analytic conversion method from fluence to dose equivalent for higher photon energies and certain irradiation conditions (e.g. AP). However, for low photon energies and some irradiation conditions (e.g. ISO) the result may considerably deviate from results obtained with the help of standard conversion coefficients applied in today’s radiation protection practice.

As mentioned earlier, the literature of constants for application in deterministic radiation transport is continuously changing and extending. This is especially the case for the constants quantifying the contribution of scattered photons to a detector response, the buildup factors. In line with this, the buildup coefficient in equation (8) has been generalised, and is calculated from data in the above mentioned user-editable input database. Compared with the earlier model, which utilises hardwired buildup factors from [2], this allows the user to extend the range of supported shield thicknesses for the existing materials (figure 6), and enable new shielding materials by simply updating and extending the database using data from new literature. The user-accessible input database, as a default, contains shielding material specific data for standard engineering materials (now including air), up to 100 mfp, from standard publications [2, 44], with the possibility of extending the list by simply inserting appropriate parameters (densities, buildup factors, etc) for the new material into the database.

Figure 6 clearly shows that the range of shield thicknesses for which the model can apply a reliable buildup factor is considerably increased compared to the earlier model. However, depending on the photon energy, there are realistic shield thickness ranges (thick shields for low-energy photons and thin shields for high-energy photons) that are not covered by buildup data for 0.5–100 mfp. The earlier model applies a constant extrapolation (the closest available coefficient) for the calculation of buildup for input parameters outside the supported range, which may introduce error, especially for very thick shields. The ANS standard [2] recommends application of the geometric progression (GP) fitting formula for extrapolation
Figure 5. Comparison of effective dose conversion factors \( (E/\Phi) \) to mass-energy absorption coefficients \( \times \) photon energy \( (\mu_{en}/\rho) \times E \), for AP (left panel) and ISO (isotropic) irradiation (right panel). Data were taken from [19].

Figure 6. The range of additional supported shield thicknesses enabled by extending the initial buildup data for up to 40 mean free path (black line) by additional buildup coefficients for up to 100 mfp (red line) for water and ordinary concrete. Of buildup factors for optical thicknesses above 40 mfp. In order to assess the applicability of the GP fitting to extrapolate for values above the reported upper limit, buildup factors for up to 100 mfp, calculated by the GP formula based on coefficients reported in [2], have been compared to similar coefficients based on [44], for optical thicknesses up to 100 mfp (figure 7). Note that since the buildup coefficients based on [44] are applicable to effective dose, and those based on [2] are for exposure, direct comparison is not possible. However, based on
Figure 7. Buildup factors for concrete calculated by the GP fitting formula and the coefficients reported in [2] (left panel) and [44] (right panel) for 40, 50, 60, 80 and 100 mfp.

Figure 7, we can conclude that extrapolation with the help of the GP technique results in a more accurate output than a constant extrapolation in general, and the inaccuracy resulting from extrapolation by the GP fitting formula is low for a moderate (~15%) contravention of the upper limit in shield thickness. Extrapolation of buildup data by the GP fitting method, on the other hand, results in increasing error for shield thicknesses strongly exceeding the upper limit. Similar investigations show that extrapolation towards lower optical thicknesses using the GP fitting technique provides a better solution than applying a constant extrapolation. However, the additional accuracy is minimal.

In line with the above, the new dosimetric model applies the GP formula for the calculation of buildup factors based on user-editable coefficients, and the user can tune the extent to which violation of the upper and lower limits of the supported parameters is allowed.

The buildup factors in the new model are also connected to the desired detector response, and may vary depending on the quantity we wish to calculate. In accordance with the multiple adjustable output concept, buildup factor tables have to be provided not just for each material but also for each possible output quantity. In line with the earlier discussion, the buildup factors for a specific material can be equal for multiple output quantities, if these have a similar energy dependence. However, application of one set of buildup coefficients for all possible output quantities introduces considerable error in the calculations for certain energies (figure 8).

The standard buildup factors derived for infinite media, generally utilised in current radiation protection practice, guarantee a conservative estimate in most common situations [33]. However, it has been shown that in some special cases (e.g. heavy shield—tissue configuration [2, 33], stratified shielding configuration [38], strongly oblique incidence of photons onto slab shields [22, 23, 29], non-slab-like shielding composition) these standard buildup factors may become unreliable, and may underestimate detector response to the photons scattered by the medium. Some of these more special situations can be solved simply by the application of situation-specific buildup factors or situation-specific corrections. The literature dealing with these limiting cases is growing and is subject to constant change. The user-accessible input file of the new dosimetric model allows users to apply situation-specific buildup coefficients for more special irradiation conditions from the latest literature available.
The user-accessible input database also contains radiological input data for the isotopes supported by the model. As a default, the model offers a list of 117 gamma active radionuclides [21] with the possibility of updating the isotope-specific data and extending the list by editing the input database.

Note that if suitable standard unit fluence rate to quantity conversion coefficients and related buildup factors are applied, the more general equation (8) becomes identical to equation (4). Hence, from a dosimetric perspective, the earlier model can be regarded as a special case of the new model. However, the application of user-adjustable input requires a different programming technique, resulting in a somewhat lower computation speed.

Similarly to the earlier dosimetric approach, extended sources are also substituted by a number of point sources. In the new model, the process of converting the extended sources in the scene into point sources is based on the ‘definition’ of the point source. By convention, a source can be considered dimensionless if its largest diameter is negligible compared to its distance from the detector. This definition is flexible, and what can be considered negligible thus depends on the situation. A criterion limiting the maximum diameter to ten percent of the distance from the centre of the source (i.e. dist./diam. < 0.1 or 10%) is generally an acceptable constraint in radiation protection. Thus, in the new model extended sources are broken down into pieces that are 10% of their distance from the detector by default, and the criterion for source splitting can be user adjusted.

Note that, following from this definition, the maximum size of a ‘point source’ varies with its distance from the detector, thus extended sources are fragmented unevenly (adaptively). Figure 9 shows point sources created by the new dosimetric model for a line (i.e. an elongated source) and a rectangular plane (i.e. surface contamination) source. The figure clearly demonstrates how the number of point sources increases, i.e. the resolution of the fragmentation rises, with decreasing distance of the detector from the source. The adaptive conversion method ensures a better result in situations when the detector is close to extended sources, and point kernel simulation based on uniform meshing breaks down. The trade-off for this increased accuracy and applicability is decreased computation speed.

As can be seen from the earlier discussion, the more flexible implementation of the basic point kernel methodology, the user control over the radiological input constants and
the improved treatment of extended sources in the new model resulted in a series of very important benefits as compared to the earlier dosimetric approach. In addition, there are a series of developments in progress for further improvement of the accuracy and flexibility of the new dosimetric model. Some of these are described below.

The user-accessible input file contains mass attenuation coefficients (versus photon energy) for each supported shielding material. However, the same coefficients are utilised for calculating optical thickness both for calculating attenuation and for calculating photon buildup. On the other hand, it is important that when finding buildup factors, based on buildup factor tables, the right attenuation coefficients are utilised, i.e. coefficients that are based on the cross sections utilised by the authors reporting the buildup factors. Since many of the standardly applied buildup factors neglect minor effects, e.g. coherent scattering, and the electron binding effect in Compton scattering, application of these buildup data requires a different set of attenuation coefficients from those applied for calculating the attenuation. Note that neglecting the coherent scattering may cause very slight underestimation of photon buildup at some energies and low optical thicknesses [2, 13]; however, this usually leads to a higher level of conservatism [33].

As mentioned earlier, reliable calculation of photon buildup may not be ensured by application of infinite medium buildup factors for strongly oblique incidence of radiation onto slab shields. Possible underestimation or strong overestimation of buildup, based on the optical thickness corresponding to the slant penetration, can be prevented by using special buildup factors, or techniques that take into account incidence angles [7, 8, 22, 23, 29, 33, 34]. The user-configurable database of the new model allows users to utilise these buildup factors for special situations where certain incidence angles are dominant. However, more complex situations where oblique penetration plays a major role, but a large variety of incidence angles
are possible, require incidence angle dependent buildup calculations. Further development of the model to allow application of incidence angle specific input data is planned.

Similarly, the infinite homogeneous medium buildup factors are applied to stratified shielding configurations by simply multiplying the buildup in different layers. However, more sophisticated techniques, applicable to multilayer shielding configurations, have been elaborated in the literature [38], and research continues, aimed at improving buildup estimation in non-homogeneous media.

One additional weakness originating from estimating the contribution of scattered radiation only via simple buildup calculations based on buildup coefficients is that this methodology disregards radiation ‘reflected’ (scattered) by surrounding objects, e.g. large surfaces of heavy material (walls, ceiling, pavement). Radiation scatter is a highly complex phenomenon and, in general, sophisticated radiation transport modelling is required for its description, which is incompatible with real-time simulation. However, the effects of ‘back-scattered’ radiation can be treated in a simplified manner by application of the so-called ‘albedo’ method [6]. The albedo method offers a somewhat simplified approach to the problem which, due to its relative simplicity, could be used in a fast (real-time or semi-real-time) calculation. It is clear that the computing power required by this technique is significantly higher compared to the techniques described above. However, since in some real-world circumstances, for example a room having a strong source shielded, but not enclosed by, a massive slab-like shield, the detector response is mainly generated by scattered radiation, investigation of this issue is important.

The last issue that must be noted here is ray tracing. The optical thickness of the medium between a point source and the detector is calculated based on the intersection of a straight line between the source and the detector with the objects defined as shields in the scene (the rest is considered to be air). According to this, the medium between the source and the detector is treated as a series of layers (‘infinite’ slabs) of different materials when calculating attenuation and buildup. This hypothesis is reasonable in many cases. However, as previously explained, this approach is less applicable to highly complex shielding configurations, such as if the space between the source and the detector is filled with a series of compact heavy objects. This situation is particularly problematic for extended sources when different parts of the source may be shielded very differently. Firstly, in this case the standard buildup factors are only applied due to the lack of more appropriate data. Secondly, calculation of the optical thickness between the detector and an extended source, based on straight lines connecting the centres of the point sources representing the extended source with the detector, may involve significant error. Ray tracing is a very important and time-consuming part of the simulation and is strongly influenced by how realistically shaped sources are converted into idealised, dimensionless, point sources, to which the point kernels are applied.

All the completed improvements and those in progress significantly affect computation speed. Since the main aim is to develop real-time (or semi-real-time) solutions, the methodologies implemented are always carefully inspected and optimised for speed. In addition, programming techniques such as parallel computing and exploitation of the capabilities of modern graphics hardware in ray tracing are utilised for increased computational speed on high-end computers.

3. Results

The VRdose and Halden Planner systems are being developed to support real-world activities in nuclear environments by applying real-time 3D simulation and visualisation of work activities and associated radiation exposure (figure 10). In line with the issues pointed out
earlier, simple and quick dosimetric techniques, compatible with real-time 3D simulation, are not suitable for very accurate dose calculations in realistic environments. However, these techniques are very powerful for producing quick conservative dose estimates associated with work procedures that are subject to dynamic variation of the exposure conditions. Validation of the techniques with the help of a complex realistic work scenario is challenging for a number of reasons. Hence, as a first step, simple irradiation situations were chosen for validation; these are supported by internationally accepted alternative tools, and the error resulting from simplification of the radiation transport is expected to be low. In line with this, a sample problem package has been designed which includes a series of calculations with common parameters but varying parameter values for source geometry, source-to-detector distance, shield (slab shield) thickness and shielding material. Dosimetric calculations have been performed using MicroShield (versions 5 and 6) \cite{10, 11}, the simple online Rad Pro Calculator \cite{25} and the two dosimetric units described in this paper to calculate the absorbed dose rate in air. Note that the detector response type selected for the new dosimetric model, for comparison to the rate of dose absorbed in air calculated by MicroShield and the Rad Pro Calculator, was the kerma rate.

The irradiation setup is demonstrated in figure 11. The source is a multi-isotopic source containing 500 GBq of $^{60}$Co and 4 TBq of $^{137}$Cs ($^{137m}$Ba, the radioactive daughter generated by nuclear decay of the $^{137}$Cs, has naturally been included). A 5 m $\times$ 5 m slab shield is positioned, with its centre 0.5 m from the centre of the source, and perpendicular to the source-to-detector line. Calculations have been performed for various shield materials and thicknesses (including no shielding), source-to-detector distances and source geometries (point, line and plane). In the case of extended sources (line and rectangular plane), the source is always centred relative to the shield, and orthogonal to the source-to-detector line.

The results calculated using MicroShield v.5, the Rad Pro Calculator, the earlier (VRdose earlier) and the new (VRdose new) models described in this paper, are listed in table 1, sorted by input parameters. Note that the two last rows, labelled ‘combined’, give the sum of all preceding rows, corresponding to an irradiation condition combining all the simple situations investigated.

Table 1 nicely shows that for all shielding configurations, dose rate decreases when replacing the point source with a line and then a plane source. This follows from the activity being more and more spread, which (a) causes some of the activity to be at greater distance from the detector and (b) improves shielding, due to the slanted pathway of the radiation from the distal parts of the extended source. As expected, the decrease in dose rate is less evident if the detector is 10 m from the source.

Table 2 quantifies the deviation of the VRdose and Rad Pro Calculator results from those obtained by the MicroShield v.5. More specifically, the table applies the following formula to compare the results:

$$
\text{Deviance (VRdose)} = \begin{cases} 
\frac{\text{VRdose}}{\text{MicroShield}} - 1 \times 100 & \text{if VRdose} > \text{MicroShield} \\
1 - \frac{\text{MicroShield}}{\text{VRdose}} \times 100 & \text{if VRdose} < \text{MicroShield}
\end{cases}
$$

Investigation of table 2 reveals that there is very good agreement between the results obtained by the different calculation tools for low optical thicknesses. For strong shielding, however, the deviance strongly increases with shield thickness, and reaches great proportions for extreme optical thicknesses. Nevertheless, as mentioned earlier, the simplified methods applied in the deterministic tools utilised in this benchmark exercise are not designed for
Table 1. Absorbed dose rates in air (in mGy h⁻¹), calculated by the earlier (VRdose earlier) and the new (VRdose new) dosimetric models, MicroShield v.5 (MicroSh.) and the Rad Pro Calculator (RadPro), at 1 and 10 m from the source centre. (Note: the initial output of the VRdose earlier model (corresponding to dose rate absorbed in tissue) was converted to dose rate absorbed in air by multiplication by 8.7/9.7.)

| Point source | Line source | Plane source |
|--------------|-------------|--------------|
| 1 m from the source | | |
| VRdose earlier | VRdose new | MicroSh. | VRdose earlier | VRdose new | MicroSh. | VRdose earlier | VRdose new | MicroSh. |
| Unshielded | | |
| 0.5 m water | 4.55 x 10⁴ | 4.55 x 10⁴ | 4.61 x 10⁴ | 4.58 x 10⁴ | 3.57 x 10⁴ | 3.57 x 10⁴ | 3.62 x 10⁴ | 2.91 x 10¹ | 2.91 x 10¹ | 2.95 x 10¹ |
| 1.0 m water | 5.00 x 10⁵ | 5.16 x 10⁵ | 4.98 x 10⁵ | 5.95 x 10⁵ | 2.35 x 10⁰ | 2.42 x 10⁰ | 2.35 x 10⁰ | 1.13 x 10⁰ | 1.17 x 10⁰ | 1.14 x 10⁰ |
| 30 cm concrete | 2.59 x 10⁵ | 2.92 x 10⁵ | 3.13 x 10⁵ | 2.58 x 10⁵ | 1.41 x 10⁰ | 1.60 x 10⁰ | 1.72 x 10⁰ | 7.93 x 10⁰ | 9.04 x 10⁰ | 9.79 x 10⁰ |
| 0.5 m water | 2.09 x 10⁶ | 2.52 x 10⁶ | 2.84 x 10⁶ | 2.69 x 10⁶ | 9.45 x 10⁻¹ | 1.15 x 10⁰ | 1.30 x 10⁰ | 4.40 x 10⁻¹ | 5.40 x 10⁻¹ | 6.17 x 10⁻¹ |
| 1.0 m water | 3.78 x 10⁻³ | 5.25 x 10⁻³ | 6.48 x 10⁻³ | 4.79 x 10⁻³ | 1.27 x 10⁻¹ | 1.78 x 10⁻¹ | 2.25 x 10⁻¹ | 4.36 x 10⁻⁴ | 6.17 x 10⁻⁴ | 7.92 x 10⁻⁴ |
| 2 mm iron | 4.44 x 10² | 4.43 x 10² | 4.42 x 10² | 4.09 x 10² | 3.48 x 10² | 3.47 x 10² | 3.46 x 10² | 2.82 x 10² | 2.82 x 10² | 2.81 x 10² |
| 5 mm iron | 4.23 x 10² | 4.24 x 10² | 4.23 x 10² | 3.49 x 10² | 3.29 x 10² | 3.30 x 10² | 3.29 x 10² | 2.66 x 10² | 2.66 x 10² | 2.66 x 10² |
| 1 cm iron | 3.86 x 10² | 3.87 x 10² | 3.91 x 10² | 3.71 x 10² | 2.96 x 10² | 2.97 x 10² | 2.99 x 10² | 2.35 x 10² | 2.36 x 10² | 2.38 x 10² |
| 5 cm iron | 1.21 x 10² | 1.23 x 10² | 1.25 x 10² | 1.13 x 10² | 7.83 x 10¹ | 7.99 x 10¹ | 8.10 x 10¹ | 5.26 x 10¹ | 5.37 x 10¹ | 5.45 x 10¹ |
| 10 cm iron | 1.94 x 10¹ | 2.01 x 10¹ | 2.08 x 10¹ | 1.92 x 10¹ | 9.67 x 10⁰ | 1.08 x 10² | 1.13 x 10² | 5.01 x 10¹ | 6.06 x 10¹ | 6.31 x 10¹ |
| 0.5 m iron | 6.29 x 10⁻⁷ | 4.77 x 10⁻⁶ | 5.43 x 10⁻⁶ | 6.69 x 10⁻⁶ | 1.68 x 10⁻⁴ | 1.32 x 10⁻⁶ | 1.50 x 10⁻⁶ | 4.56 x 10⁻⁸ | 3.67 x 10⁻⁷ | 4.21 x 10⁻⁷ |
| 1.0 m iron | 7.02 x 10⁻⁶ | 1.41 x 10⁻¹⁴ | 1.80 x 10⁻¹⁴ | 2.24 x 10⁻¹³ | 1.35 x 10⁻¹⁵ | 2.77 x 10⁻¹⁵ | 3.57 x 10⁻¹⁵ | 2.62 x 10⁻¹⁷ | 5.45 x 10⁻¹⁶ | 7.13 x 10⁻¹⁶ |
| 1 mm lead | 4.22 x 10² | 4.25 x 10² | 4.24 x 10² | 4.09 x 10² | 3.29 x 10² | 3.31 x 10² | 3.30 x 10² | 2.65 x 10² | 2.67 x 10² | 2.66 x 10² |
| 2 mm lead | 3.92 x 10² | 3.95 x 10² | 3.96 x 10² | 3.68 x 10² | 3.02 x 10² | 3.05 x 10² | 3.06 x 10² | 2.42 x 10² | 2.44 x 10² | 2.45 x 10² |
| 5 mm lead | 3.12 x 10² | 3.15 x 10² | 3.22 x 10² | 3.01 x 10² | 2.33 x 10² | 2.36 x 10² | 2.42 x 10² | 1.81 x 10² | 1.84 x 10² | 1.89 x 10² |
| 1 cm lead | 2.08 x 10² | 2.13 x 10² | 2.23 x 10² | 2.20 x 10² | 1.49 x 10² | 1.53 x 10² | 1.61 x 10² | 1.11 x 10² | 1.14 x 10² | 1.20 x 10² |
| 5 cm lead | 1.19 x 10² | 1.26 x 10² | 1.43 x 10² | 1.22 x 10² | 6.79 x 10⁰ | 7.18 x 10⁰ | 8.17 x 10⁰ | 4.03 x 10⁰ | 4.27 x 10⁰ | 4.87 x 10⁰ |
| 10 cm lead | 5.63 x 10⁻¹ | 6.05 x 10⁻¹ | 7.40 x 10⁻¹ | 6.35 x 10⁻¹ | 2.50 x 10⁻¹ | 2.78 x 10⁻¹ | 3.44 x 10⁻¹ | 1.14 x 10⁻¹ | 1.32 x 10⁻¹ | 1.65 x 10⁻¹ |
| 0.5 m lead | 2.76 x 10⁻¹² | 1.06 x 10⁻¹¹ | 2.80 x 10⁻¹¹ | 1.27 x 10⁻¹¹ | 5.98 x 10⁻¹³ | 2.33 x 10⁻¹² | 6.25 x 10⁻¹² | 1.31 x 10⁻¹³ | 5.15 x 10⁻¹³ | 1.40 x 10⁻¹² |
Table 1. (Continued.)

| VRdose earlier | VRdose new | MicroSh. | RadPro | VRdose earlier | VRdose new | MicroSh. | VRdose earlier | VRdose new | MicroSh. | VRdose earlier | VRdose new | MicroSh. |
|----------------|------------|----------|--------|---------------|------------|----------|---------------|------------|----------|---------------|------------|----------|
| 1.0 m lead     | 0.30 × 10^{-26}a | 2.65 × 10^{-25}a | 3.18 × 10^{-23} | 1.0 × 10^{-24} | 4.66 × 10^{-27}a | 4.16 × 10^{-26} | 2.51 × 10^{-23} | 7.25 × 10^{-28}a | 6.55 × 10^{-27}a | 2.05 × 10^{-23} |
| Combinedb      | 3.31 × 10^3   | 3.33 × 10^3   | 3.36 × 10^3   | 3.14 × 10^3   | 2.50 × 10^3   | 2.52 × 10^3   | 2.54 × 10^3   | 1.97 × 10^3   | 1.99 × 10^3   | 2.01 × 10^3   |

10 m from the source

| VRdose earlier | VRdose new | MicroSh. | VRdose earlier | VRdose new | MicroSh. | VRdose earlier | VRdose new | MicroSh. | VRdose earlier | VRdose new | MicroSh. |
|----------------|------------|----------|---------------|------------|----------|---------------|------------|----------|---------------|------------|----------|
| Unshielded     | 4.55 × 10^0   | 4.50 × 10^0   | 4.55 × 10^0   | 4.42 × 10^0   | 4.53 × 10^0   | 4.49 × 10^0   | 4.54 × 10^0   | 4.52 × 10^0   | 4.474121    | 4.52 × 10^0   |
| 0.5 m water    | 7.94 × 10^{-1} | 7.96 × 10^{-1} | 7.72 × 10^{-1} | 7.65 × 10^{-1} | 7.88 × 10^{-1} | 7.91 × 10^{-1} | 7.66 × 10^{-1} | 7.82 × 10^{-1} | 7.87 × 10^{-1} | 7.60 × 10^{-1} |
| 1.0 m water    | 5.00 × 10^{-2} | 0.051031    | 4.80 × 10^{-2} | 5.76 × 10^{-2} | 4.94 × 10^{-2} | 0.050541    | 4.74 × 10^{-2} | 4.87 × 10^{-2} | 5.01 × 10^{-2} | 4.68 × 10^{-2} |
| 2.0 m water    | 1.56 × 10^{-4} | 1.62 × 10^{-4} | 1.51 × 10^{-4} | 1.73 × 10^{-4} | 1.52 × 10^{-4} | 1.59 × 10^{-4} | 1.48 × 10^{-4} | 1.49 × 10^{-4} | 1.57 × 10^{-4} | 1.45 × 10^{-4} |
| 5.0 m water    | 4.10 × 10^{-12} | 4.50 × 10^{-12} | 4.12 × 10^{-12} | 6.13 × 10^{-12} | 3.90 × 10^{-12} | 4.32 × 10^{-12} | 3.91 × 10^{-12} | 3.71 × 10^{-12} | 4.16 × 10^{-12} | 3.72 × 10^{-12} |
| 0.5 m concrete | 0.020904      | 0.024933    | 0.0269 × 10^{-2} | 0.261 × 10^{-2} | 0.0269 × 10^{-2} | 0.024679    | 0.265 × 10^{-2} | 0.203 × 10^{-2} | 0.024427    | 0.262 × 10^{-2} |
| 1.0 m concrete | 0.378 × 10^{-5} | 0.518 × 10^{-5} | 0.625 × 10^{-5} | 0.465 × 10^{-5} | 0.369 × 10^{-5} | 0.509 × 10^{-5} | 0.610 × 10^{-5} | 0.360 × 10^{-5} | 0.500 × 10^{-5} | 0.596 × 10^{-5} |
| 2.0 m concrete | 1.04 × 10^{-10}| 1.90 × 10^{-10}| 2.95 × 10^{-10}| 2.40 × 10^{-10}| 9.95 × 10^{-11}| 1.84 × 10^{-10}| 2.82 × 10^{-10}| 9.49 × 10^{-11}| 1.77 × 10^{-10}| 2.70 × 10^{-10}|
| 0.5 m iron     | 6.29 × 10^{-9} | 4.71 × 10^{-8} | 5.12 × 10^{-8} | 6.49 × 10^{-8} | 6.06 × 10^{-9} | 4.58 × 10^{-8} | 4.95 × 10^{-8} | 5.84 × 10^{-9} | 4.46 × 10^{-8} | 4.78 × 10^{-8} |
| 1.0 m iron     | 7.02 × 10^{-18}| 1.40 × 10^{-16}| 1.73 × 10^{-16}| 2.17 × 10^{-15}| 6.55 × 10^{-18}| 1.33 × 10^{-16}| 1.62 × 10^{-16}| 6.11 × 10^{-18}| 1.26 × 10^{-16}| 1.51 × 10^{-16}|
| 0.5 m lead     | 2.76 × 10^{-14} | 1.05 × 10^{-13} | 2.64 × 10^{-13} | 1.24 × 10^{-13} | 2.61 × 10^{-14} | 1.01 × 10^{-13} | 2.51 × 10^{-13} | 2.47 × 10^{-14} | 9.65 × 10^{-14} | 2.38 × 10^{-13} |
| 1.0 m lead     | 3.01 × 10^{-28}a | 2.62 × 10^{-27}a | 3.25 × 10^{-25} | 1.07 × 10^{-26} | 2.71 × 10^{-28}a | 3.24 × 10^{-27}a | 2.44 × 10^{-28}a | 2.23 × 10^{-27}a | 3.23 × 10^{-25} |
| Combinedb      | 5.41 × 10^0    | 5.37 × 10^0    | 5.40 × 10^0    | 5.27 × 10^0    | 5.39 × 10^0    | 5.35 × 10^0    | 5.38 × 10^0    | 5.37 × 10^0    | 5.34 × 10^0    | 5.35 × 10^0    |

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[a] Extrapolation of the adopted buildup factors above the upper limit of confidence reported.

[b] The sum of the preceding rows (separately for 1 m and 10 m distance).
Figure 10. Snapshots of the 3D interface of the VRdose and Halden Planner systems.

general application to extreme situations, where the radiation dose mainly arises from scattered photons. These situations usually call for more sophisticated Monte Carlo simulations, if an accurate answer is required. The accuracy of the simpler deterministic radiation transport models, however, is acceptable in many common situations, where dose (detector response) is mainly generated by direct exposure, and the contribution of single and multiple scattered
radiation is low. Indeed, in table 2 the two last rows, for the two detector distances, show
the discrepancy of the sum of the results for exposure to individual sources. These combined
cases correspond to exposure situations that combine exposure to multiple sources shielded
by different shields (including one unshielded case). Comparison of the combined results
shows that the two dosimetric models presented in this paper are in very good agreement with
MicroShield v.5.

The results also show that for more specialised situations, involving only shields of great
optical thickness, the variation of the results provided by deterministic models is high. In these
situations, the resulting detector response mainly depends on how the scatter of photons is
accounted for; that is, what kind of buildup factors, and how they are applied. Since the buildup
factors reported in the literature inherit a great uncertainty for high optical thicknesses, and the
values reported vary from report to report, the deviance between the results is neither surprising
nor unexpected. Users of these radiation transport tools must recognise the inaccuracy inherent
to the simplified methodology applied, and resort to more sophisticated Monte Carlo radiation
transport tools for more precise simulation.

Similar investigations have been performed using a newer version of MicroShield
(MicroShield v.6). Table 3 shows the deviance of our results from MicroShield v.6 calculated
with the same methodology as before:

\[
\text{Deviance (VRdose)} = \begin{cases} 
  \left( \frac{\text{VRdose}}{\text{MicroShield}} - 1 \right) \times 100 & \text{if VRdose} > \text{MicroShield} \\
  \left( \frac{\text{MicroShield}}{\text{VRdose}} + 1 \right) \times 100 & \text{if VRdose} < \text{MicroShield}.
\end{cases}
\]  

Comparing table 3 to table 2, its counterpart for the earlier MicroShield version, we can see
that the tendency of the deviance is very similar; the deviance strongly increases at high optical
thicknesses, but for more common shield thicknesses the agreement is, in general, good. It is
the difference for the combined case.

This can also be observed in apparent that at 1 m from the source, the deviance is substantially larger at lower optical thicknesses than it was in the comparison to MicroShield v.5. This can also be observed in the difference for the combined case.
### Table 3
Discrepancy (in %) from MicroShield v.6 results, at 1 and 10 m from the source (centre). A minus sign indicates underestimation.

| Point source | VRdose earlier | VRdose new RadPro | Line source | VRdose earlier | VRdose new | Plane source | VRdose earlier | VRdose new |
|--------------|----------------|-------------------|-------------|----------------|-------------|--------------|----------------|-------------|
| **1 m from the source** | | | | | | | | |
| Unshielded | −1 | −1 | −1 | −1 | −1 | −1 | −1 | −1 |
| 0.5 m water | −1 | 1 | −1 | 0 | 1 | 0 | 1 | 1 |
| 1.0 m water | 0 | 3 | 19 | 0 | 3 | 0 | 4 | 4 |
| 0.5 m concrete | −40 | −24 | −40 | −40 | −24 | −41 | −24 | −24 |
| 1.0 m concrete | −48 | −23 | −15 | −49 | −23 | −50 | −22 | −22 |
| 2 mm iron | −62 | −17 | −28 | −64 | −17 | −65 | −17 | −17 |
| 5 mm iron | −1 | −1 | −10 | −1 | −1 | −1 | −1 | −1 |
| 1 cm iron | −3 | −3 | −25 | −3 | −3 | −4 | −4 | −4 |
| 5 cm iron | −7 | −7 | −12 | −9 | −8 | −10 | −9 | −9 |
| 10 cm iron | −47 | −44 | −58 | −49 | −46 | −53 | −50 | −50 |
| 0.5 m iron | −71 | −65 | −73 | −86 | −67 | −102 | −67 | −67 |
| 1.0 m iron | −1182 | −69 | −21 | −1233 | −70 | −1275 | −71 | −71 |
| 1 mm lead | −3879 | −93 | 721 | −3878 | −94 | −3946 | −94 | −94 |
| 2 mm lead | −6 | −5 | −10 | −7 | −7 | −8 | −7 | −7 |
| 5 mm lead | −13 | −12 | −20 | −16 | −14 | −16 | −15 | −15 |
| 10 mm lead | −34 | −32 | −39 | −44 | −42 | −44 | −41 | −41 |
| 0.5 m lead | −76 | −72 | −67 | −106 | −101 | −95 | −89 | −89 |
| 1.0 m lead | −1182 | −69 | −21 | −1233 | −70 | −1275 | −71 | −71 |
| Combined | −97.7 × 10^6 | −1.1 × 10^6 | −2.6 × 10^5 | −4.9 × 10^7 | −5.5 × 10^6 | −2.6 × 10^8 | −2.8 × 10^7 | −2.8 × 10^7 |
| **10 m from the source** | | | | | | | | |
| Unshielded | 0 | −1 | −3 | 0 | −1 | 0 | −1 | −1 |
| 0.5 m water | 4 | 5 | 1 | 5 | 5 | 4 | 5 | 5 |
| 1.0 m water | 1 | 8 | 22 | 6 | 8 | 6 | 9 | 9 |
| 2.0 m water | 1 | 8 | 15 | 3 | 8 | 4 | 10 | 10 |
| 5.0 m water | 1 | 10 | 50 | 1 | 11 | 1 | 13 | 13 |
| 0.5 m concrete | −40 | −18 | −12 | −40 | −17 | −40 | −17 | −17 |
| 1.0 m concrete | −53 | −12 | −25 | −53 | −11 | −53 | −10 | −10 |
| 2.0 m concrete | −98 | −9 | 16 | −99 | −8 | −99 | −7 | −7 |
| 0.5 m iron | −1111 | −62 | −17 | −1115 | −61 | −1117 | −59 | −59 |
| 1.0 m iron | −3575 | −85 | 741 | −3579 | −81 | −3582 | −79 | −79 |
| 0.5 m lead | −8158 | −2074 | −1739 | −8214 | −2049 | −8240 | −2035 | −2035 |
| 1.0 m lead | −9.7 × 10^6 | −1.1 × 10^6 | −2.7 × 10^5 | −1.1 × 10^7 | −1.2 × 10^6 | −1.2 × 10^7 | −1.3 × 10^6 | −1.3 × 10^6 |
| Combined | 0.5 | −0.3 | −2.1 | 0.4 | −0.3 | 0.5 | −0.1 | −0.1 |
4. Conclusions

In the scope of the work described in this paper, a new real-time point kernel based radiation risk assessment algorithm has been developed and implemented in the VRdose and Halden Planner 3D real-time simulation systems, to supplement the point kernel based model implemented previously. The new dosimetric model is in line with recent international radiation protection recommendations, and offers several improvements over the earlier model. The drawback of the new model is a slower modelling speed in some situations and on some computers, which follows from the greater time requirement of the programming techniques that enable greater flexibility and provide more accurate results. Real-time 3D simulation enforces very strict speed restrictions on the risk assessment methodology, which places very high demands on the theoretical and programming solutions utilised. This paper describes in detail the theoretical assumptions underlying the VRdose and Halden Planner systems, and the challenges associated with past and future developments in minimising the shortcomings resulting from the speed–accuracy trade-off.

In the frame of a benchmarking exercise, results for simple irradiation situations were compared to MicroShield (versions 5 and 6) and the online Rad Pro Calculator. The results show that, for the majority of the common radiation protection tasks in nuclear installations (associated with maintenance, outage or decommissioning), these simple radiation transport models are able to provide reliable real-time radiation risk assessments. In addition, due to the high performance in speed and simplicity of the input required, these models are suitable for

• users with no experience in radiation transport theory or modelling,
• modelling dynamically changing exposure conditions,
• providing information rapidly for decision-making in stressful situations,
• real-time optimisation of work tasks from a radiation protection perspective,
• real-time simulation for education, training and briefing of personnel.

In many applications, these capabilities offer huge advantages over more sophisticated models based on detailed, high accuracy simulation of radiation transport.

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