VALIDATION OF A DOSE ASSESSMENT METHOD TO BE USED IN LOOSE CONTAMINATION EXERCISES

A Thesis

by

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ABSTRACT

Emergency responders could be exposed to loose radioactive material during a mission. As part of a research project at Texas A&M University, ^{18}F was sprayed in a small area where an Exercise Participant (in protective gear) conducted simulated search activities. A dose assessment tool developed by the researchers was used to estimate doses to the Radiation Worker (mixer and sprayer) and Exercise Participant. The current project aimed to validate the assessment methodology by comparing actual and estimated doses of the two personnel. In the scenario, the Radiation Worker injected and mixed 200 MBq Fludeoxyglucose ^{18}F (FDG) with 470 ml H_2O in a commercial weed sprayer. The solution was distributed evenly over a 3 m x 3 m region in 5 min. After 36 min of evaporation, the Exercise Participant entered the area for a total of 22 min. Actual whole body (WB) doses from optically stimulated luminescence (OSL) were 10 ± 2 μSv for both the Radiation Worker and Exercise Participant. WB digital personal dosimeter readings were 4.3 ± 0.4 μSv and 3.3 ± 1.0 μSv for the Radiation Worker and Exercise Participant, respectively. Actual extremity doses to Radiation Worker's finger dosimeters were < 100 μSv (minimum detectable limit), and to exercise participant's leg OSL was < 10 μSv.

Preliminary dose assessment method was conservative for the Radiation Worker and conservatively accurate for the Exercise Participant. The predicted Radiation Worker doses were 90 μSv to the whole body (WB) and 744 μSv to the hand, both $\gg 2\sigma$ above the actual exposures. The Exercise Participant's estimated doses were 7 μSv to the

WB and 15 μSv to the knee area, which were in the same order of magnitude as the actual.

Refined dose assessment aimed to predict personnel exposure more exactly and was shown to be accurate. The predicted Radiation Worker doses were 2.8 ± 0.8 μSv to the WB and 21.8 ± 7.5 μSv to the hand. The Exercise Participant's estimated doses were 5.2 ± 0.5 μSv to the WB and 13.4 ± 1.2 μSv to the knee area. Estimated whole body doses were in the same order of magnitude as the actual doses for both the Radiation Worker and the Exercise Participant. Comparing estimated extremity dose to the actual value was difficult, due to exposures having been below detectable limits, however, there were no obvious inconsistencies.

DEDICATION

This thesis is dedicated to my Creator, Lord, and Savior Jesus Christ.

ACKNOWLEDGEMENTS

I owe the possibility to pursue this degree to our Creator and Provider Jesus Christ our Lord. For without Him, nothing would exist and I would not have the strength nor ability to be where I am, “For in Him all things were created: things in heaven and on earth, visible and invisible, whether thrones or powers or rulers or authorities; all things have been created through Him and for Him” (Colossians 1:16). Not only that, despite my many faults and sins, He has promised that those who repent from our wickedness and “persevere to the end will be saved” because of His sacrifice for the sins of mine and the whole world (Matthew 24:13, John 3:16). So I should be grateful to Him more than anyone else for the grace and hope that help me to go on.

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NOMENCLATURE

ALARA	As Low as Reasonably Achievable
CF	Correction Factor
EHSD	Environmental Health and Safety Department
ENSDF	Evaluated Nuclear Structure Data Files
FDG	Fludeoxyglucose ^{18}F
GM	Geiger-Mueller
GUI	Graphic User Interphase
HDPE	High-Density Polyethylene
HPGe	High Purity Germanium
HVAC	Heating, Ventilation, and Air Conditioning
IACUS	Institutional Animal Care and Use Committee
IAEA	International Atomic Energy Agency
IRB	Institutional Review Board
LOD	Limit of Detection
MCNP	Monte Carlo N-Particle
MIRD	Committee on Medical Internal Radiation Dose
NCRP	National Council on Radiation Protection and Measurements
NNSS	Nevada National Security Site
OSL	Optically Stimulated Luminescence
PET	Positron Emission Tomography

PIMAL	Phantom with Moving Arms and Legs
Q	Quality Factor
RAM	Radioactive Material
RDD	Radiological Dispersal Device
REMM	Radiation Emergency Medical Management
RIIDS	Radioisotope Identification Devices
RSC	Radiation Safety Committee
SDE	Shallow-Dose Equivalent
SHARC	Specialized Hazard Assessment Response Capability
SI	International System of Units
TEDE	Total Effective Dose Equivalent
TEEX	Texas A&M Engineering Extension Service
TLD	Thermoluminescent Dosimeter
TODE	Total Organ Dose Equivalent
TRACER	Testing Radiation and Contamination in Emergency Response
TRIGA	Test, Research, Isotopes, General Atomics
WB	Whole Body
w/o	Weight Percent
w_r	Radiation Weighting Factor

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1. INTRODUCTION

1.1. Motivation

The long-term goal of this research is to aid in the development of safe exercise scenarios that involve unsealed radioactive material (RAM). A dose assessment tool designed by Lainy Cochran for this purpose is able to estimate the dose to personnel training with unsealed sources [1]. The objective of the current project was to validate this tool.

1.1.1. Emergency Response Training

The need to properly prepare responders for radiological dispersal incidents is of paramount importance. Ideally, training exercises would provide personnel the most realistic scenarios in order to instill practical skills for actual incidents; but only with the limiting condition that there are benefits in exposing responders to radiation fields during training. This is in accordance with the “as low as reasonably achievable” (ALARA) philosophy of radiation protection [2]. The training objectives for radiation emergency responders, according to the National Council on Radiation Protection and Measurements (NCRP) Commentary No. 19, “Key Elements of Preparing Emergency Responders for Nuclear and Radiological Terrorism” are the following [3] [1].

1. Enhance their ability to take appropriate measures to protect themselves and the public.
2. Increase their confidence about effectively managing an emergency involving radiation or radioactive materials.

It would be beneficial for responders to have the ability to train in areas with dispersed radioactive material. This would allow trainees to acquire the most accurate assessment of instrument response in a realistic environment. It would allow experience with decontamination. It would also help law enforcement understand how to collect, maintain chain-of-custody, and transport radioactively contaminated evidence. However, since this raises the potential of health hazards to exposed training personnel, careful selection of radioisotope and dispersion methods is required.

Most current radiation dispersion trainings involve table-top scenarios and sealed sources; which preclude trainees from hands-on experience in the actual detector behaviors and the contamination challenges in an environment with loose RAM [4]. In a setting with dispersed radioisotopes, acquiring detection skills such as locating dispersed hot-spots or delineating exclusion zones are essential. Providing feedback on the effectiveness of contamination avoidance and decontamination are also of upmost importance. Sealed-sources and table-top exercises cannot realistically provide such training.

Recognizing this deficiency, a handful of agencies have conducted limited field exercises with unsealed sources [1] [5]. Savanna River National Laboratory and Idaho National Laboratory performed field exercises with loose sources in 2010 [6] [7]. Nevada National Security Site (NNSS) Testing Radiation and Contamination in Emergency Response (TRACER) program also conducted a radiation dispersal exercise in 2012. In this field training, Tc-99m was dissolved in water and sprayed on target areas at the T-1 site. A pre-exercise dose assessment was performed by NNSS but was not published [1]. Texas A&M University was able to acquire a copy of the document; and the current research is based heavily on their report.

1.1.2. Support of Disaster City

The Texas A&M Engineering Extension Service (TEEX) in College Station, Texas is an organization which provides emergency responders “support to disasters across the state and nation and develops training and practical workforce solutions to fire and rescue, infrastructure and safety, law enforcement, economic and workforce development, and homeland security personnel”. It operates Disaster City®, which is a mock community that “features full-scale, collapsible structures designed to simulate various levels of disaster and wreckage which can be customized for the specific trainings.” [8] The Department of Nuclear Engineering at Texas A&M University has been a long-time partner with TEEX and supported various radiation exercises at Disaster City® using sealed sources. The long-term research aim is to design more realistic but safe response training using dispersed radioactive material at Disaster City®.

1.2. Literature Review

Pre-exercise dose assessments are essential in the design of a practical and safe exercise using unsealed sources. As mentioned above, the NNSS's TRACER conducted a dose assessment on the use of Tc-99m in a dispersion exercise. Various other research groups have developed and performed validation of models for estimating public dose during accidents involving the dispersal of radioactive material. Most simulations focused on releases from facilities in the nuclear fuel cycle, as listed in Table 1.1. At least one tool is available for estimating personnel dose specific to a Radiation Dispersal Device incident (Table 1.2); however, validation studies were not found.

Table 1.1 Example Model Validations of Nuclear Facility Accidents.

Modeling Method	Incident Description	Dose Type	Author
Analytical Equation	Atomic Test Fallout	Thyroid	Simon, 1988 [9]
Deterministic JSP5 Model	Chernobyl Environmental Contamination	External	Golikov, 1999 [10]
Various	Chernobyl I-131 Release	Various	EMRAS, 2007 [11]
NCAR Mesoscale, CG-MATHEW/ ADPIC Atmosphere Dispersion	Kr-85 Dispersion from Fuel Reprocessing	Gamma	Abe, 2015 [12]
WSPEEDI-II Atmosphere Dispersion	Fukushima Accident	Internal	Kim, 2015 [13]

Table 1.2 Example Dose Assessment Modeling of Radiation Dispersal Device.

Modeling Method	Incident Description	Output	Author
ERMIN/ARGOS Atmosphere Dispersion	Dirty Bomb Explosion in Urban Area	Internal and External Dose	Andersson, 2009 [14]

Several incident response tools are also available. For example, the Argonne National Laboratory developed the RESRAD-RDD "to evaluate human radiation exposures during the early, intermediate, or late phase of response after a radiological dispersal device (RDD) incident." [15]. It is a useful tool to calculate stay time, determine "Early-Phase Protection Action" (evacuation or sheltering), etc. However, RESRAD-RDD was designed to assess radioisotopes with significant human health risk, such as Am-241, Cf-252, Cm-244, Co-60, Cs-137, Ir-192, Po-210, Pu-238, Pu-239, Ra-226, and Sr-90. These isotopes are important and correlate with the most likely material to be used in RDDs. However, the preferred candidates for an exercise design should pose minimal risk to the trainees. Furthermore, RESRAD-RDD was not intended to be used for estimating dose on responders, though results can be extrapolated from the output with some manipulation.

Other software programs are available to assist responders in a radiation event, however, they are more applicable for actual incidents than for designing training exercises. TURBO-FRMAC is a response software created by Sandia which is designed to assist "incident commanders" make critical decisions during a radiation incident. However, the program uses "values generated by field samples, instrument readings, or computer dispersion models". The usability of this software for the current research is limited because TURBO-FRMAC depends on actual measurements to estimate projected dose [16]. Instead, the goal of the project is to predict dose prior to dispersion. SHARC (Specialized Hazard Assessment Response Capability) is a software that simulates the "release of radioactivity from a nuclear weapon via either conventional detonation or by non-explosive techniques". Similar to RESRAD-RDD, the primary isotopes are hazardous radioactive material more likely to be used in an actual attack [17]. Other software packages such as DC_PAK, AcuteDose Calculator, and RiskTab are tools available for estimating dose and health risks [18]. These mainly provide quick access to dose and risk coefficients of radiation exposures; therefore, they are of some but limited use for modeling dispersion scenarios.

In summary, various tools are available for estimating doses to the public when radioisotopes are dispersed into the environment. However, major modifications of these models would be required to be used for designing exercises where less hazardous material are dispersed, and where dose prediction will focus on the responders and those dispersing the source, rather than on the public.

1.2.1. Studies Performed at Texas A&M University

1.2.1.1. Radionuclide Candidates

A study published in the 2018, "Radionuclide Selection for Emergency Response Exercise at Disaster City® Using Unsealed Radioactive Contamination", Lainy Cochran and Dr. Marianno researched short-lived radioisotopes that could be used for radiation dispersion response training. The investigation included compiling information on isotopes used during the few publicly known unsealed source exercises and those which were readily available or producible by Texas A&M University [5]. The seven radionuclides selected as candidates for dispersal training at TEEX Disaster City® were ^{99m}Tc, ^{18}F, ^{24}Na, ^{56}Mn, ^{64}Cu, ^{82}Br, and ^{140}La. These were chosen because the radiopharmaceuticals ^{99m}Tc and ^{18}F were available for purchase from nearby vendors, while ^{24}Na, ^{56}Mn, ^{64}Cu, ^{82}Br, and ^{140}La could be produced by the Texas A&M Nuclear Science Center Test, Research, Isotopes, General Atomics (TRIGA) reactor via thermal neutron activation.

1.2.1.2. Preliminary Dose Assessment

Following the selection of radioisotopes, Cochran and Marianno performed preliminary dose assessments following the NNSS TRACER design using the point-kernel simulation software MicroShield® to determine the safe levels of source activities for training. The assessment method took place in the daytime with little-to-no wind nor precipitation. For the dose estimation, personnel were assumed to have no personnel protective equipment. Events analyzed included the injection of the radioactive source into a container, dissolution in about 3,800 ml of water, then dispersion onto the intended surface. The source was allowed to settle before responders were permitted into the contaminated area. Unplanned events including hypothetical spills and intrusions by members of the public were studied. Accidental exposure due to a drop of the radioactive solution on the skin was also analyzed. The study estimated external and internal exposures for these planned and unplanned scenarios, then compared with dose limits posed by federal and local agencies.

The current investigation was a validation study of this dose assessment method focusing on external exposure. The assessment tool was used to estimate personnel exposure to an actual radiation dispersion exercise, then a comparison was made between the estimated and actual doses in order to infer the reliability of the methodology.

1.2.1.3. Recommended Isotope Activities

In the same investigation by Cochran, recommended activities of the isotopes were found by restricting the exposures to below the following cut-offs on dose limits. For the Radiation Worker that dissolved and distributed the source, the analysis used the administrative dose limits set by the Texas A&M Environmental Health and Safety Department (EHSD) radiation safety office [19], which was 10% of the dose limits set by Texas Department of the State in 25TAC 289.202 and federal occupational dose limits in 10 CFR 20.1201 [20] [21]. This equated to 5 mSv total effective dose equivalent (TEDE) and 50 mSv total organ dose equivalent (TODE). The doses to Exercise Participants (responders) were held under a more restrictive threshold of 1% of the state and federal limits. This equated to 0.5 mSv total effective dose equivalent (TEDE) and 5 mSv total organ dose equivalent (TODE). Cut-off for skin exposure was the EHSD administrative dose limit for shallow-dose equivalent (SDE), which was 10% of state and federal annual occupational limit. This equated to 50 mSv. The limiting dose to the public was the state and federal annual dose limit (10 CFR 20.1301) for individual members of the public, i.e., 1 mSv TEDE [20] [22]. The identified activities which can be safely used are shown in Table 1.3 [1].

Table 1.3. Recommended Maximum Activity Levels of Candidate Radioisotopes from Preliminary Dose Assessment [1].

Radionuclide	Maximum activity, MBq (mCi)	Maximum dose rate at 100 cm during exercise, $\mu Sv\ h^{-1}$ ($mrem\ h^{-1}$)
^{18}F	740 (20) [a]	16.8 (1.68)
^{24}Na	370 (10)	22.4 (2.24)
^{56}Mn	740 (20)	20.9 (2.09)
^{64}Cu	1,480 (40)	7.1 (0.71)
^{82}Br	370 (10)	19.1 (1.91)
^{99m}Tc	740 (20) [a]	3.3 (0.33)
^{140}La	37 (1)	1.4 (0.14)

2. METHODOLOGY

2.1. Contamination Scenario

Fludeoxyglucose fluorine-18 (FDG) was used for this research. FDG is a positron emitting radiopharmaceutical containing no-carrier added radioactive 2-deoxy-2-[18F]fluoro-D-g1ucose, which is used for diagnostic purposes in conjunction with Positron Emission Tomography (PET) [23]. Its health effects having been well characterized by the Committee on Medical Internal Radiation Dose (MIRD) for human use made it a desirable candidate. FDG is isotonic, sterile, pyrogen-free, and water soluble [24]. The site where the dispersal was planned at also routinely handled FDG for its PET studies on animals. Lastly, F-18 decays with a half-life of 110 minutes to stable oxygen-18, which meant that radiation of contaminated surfaces was expected to return to background levels within 48 hours. Detailed nuclear data is included in APPENDIX E. The amount of F-18 planned for dispersion was 185 MBq (5 mCi). The typical dose injected in human patients is in the range of 185-370 MBq, therefore 185 MBq was expected to pose minimal external and internal hazards [25] [23]. The experiment was approved by Texas A&M University's Institutional Review Board (IRB), Institutional Animal Care and Use Committee (IACUC), and Radiation Safety Committee (RSC).

The designated dispersion site was a post-procedure room for animals which underwent tests involving medical radioisotopes. Thus, it was designed for containing F-18 excretions. This room had a flat, nonporous floor, no windows, concrete walls, and single entry with lockable metal doors. It was a negative pressure room with adjustable Heating, Ventilation, and Air conditioning (HVAC) systems. All drains could be plugged to prevent leakage to the sewage system. The building could be fully secured, and the room was inside a locked corridor (Figure 2.1). This allowed the room to be isolated until it returned to background levels [26]. The area was to be prepped to simulate a disaster area. Items were to be placed to model a rubble pile and corn starch was to be applied to the surface to create a dusty environment. A photo of the 3 m x 3 x taped area in the room to be contaminated is shown in Figure 2.2.

The planned dispersion event consisted of a Radiation Worker who mixed and dispersed the FDG and an Exercise Participant who performed response activities in the contaminated area. The Radiation Worker was to inject and dissolve FDG into water. He would then disperse the solution using a weed sprayer onto a 3 m x 3 m surface inside a post PET scan animal holding room. After adequate evaporation of the source, the Exercise Participant was to enter the scene and performed simulated search activities.

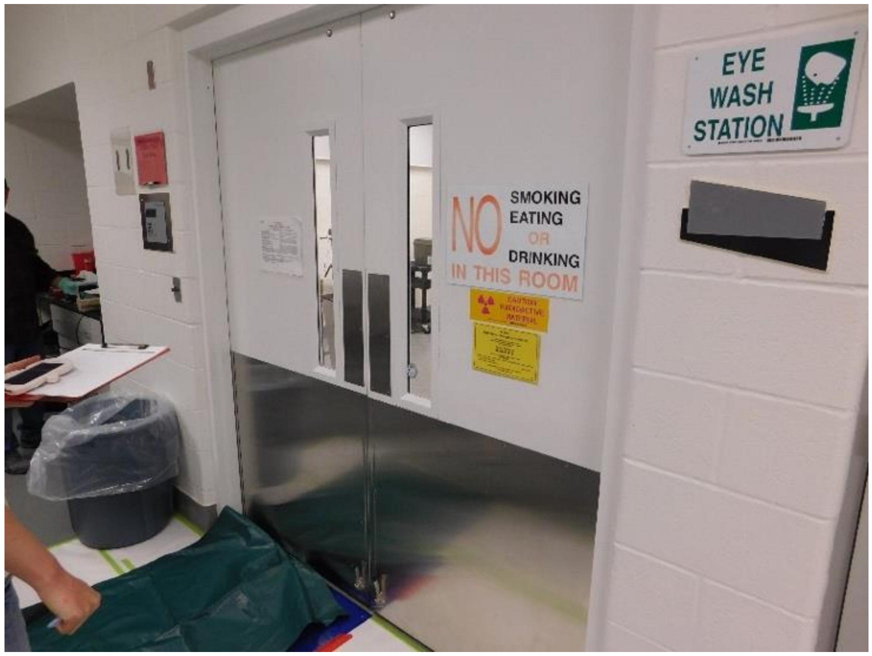

Figure 2.1 Room to be Contaminated Could be Fully Secured.

Figure 2.2 Area to be Contaminated with Rubble Pile and Dusty Environment.

2.2. Preliminary Dose Assessment Tool in Detail

Using the preliminary dose assessment tool and its default conservative parameters set by Lainy Cochran, the doses were estimated for the anticipated experimental scenario. The dose assessment method parameters were then revised to use the actual source characteristics, exposure distances, and exposure times observed during the dispersion event. The latter, refined dose prediction allowed further validation of the methodology.

The assessment tool was an MS Excel workbook which performed dose calculations based on user input of exposure parameters and calculations from MicroShield®. Cells expecting user inputs are highlighted orange in the Excel spreadsheet and shown in Figure 2.3, though there were some exceptions.

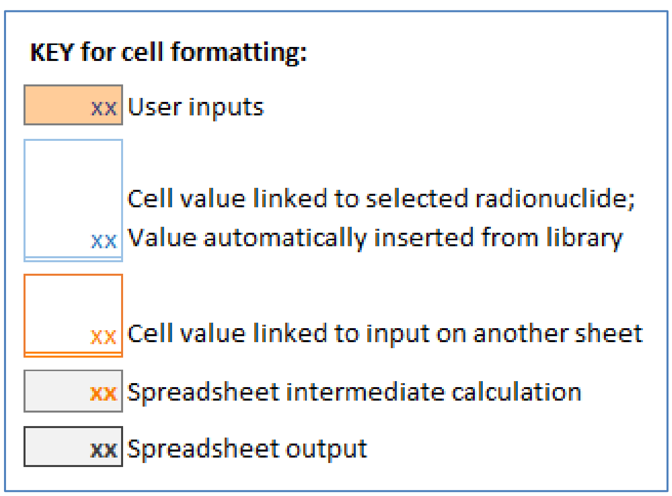

Figure 2.3 Assessment Tool Cell Formatting Legend.

2.2.1. Over-View of the Model

Two personnel were analyzed: 1) The mixer and sprayer, aka Radiation Worker, and 2) Exercise Participant. The first step in using the assessment tool was to select the F-18 radioisotope and the 185 MBq (5 mCi) activity, as shown boxed in Figure 2.4. In this report, only external dose was assessed.

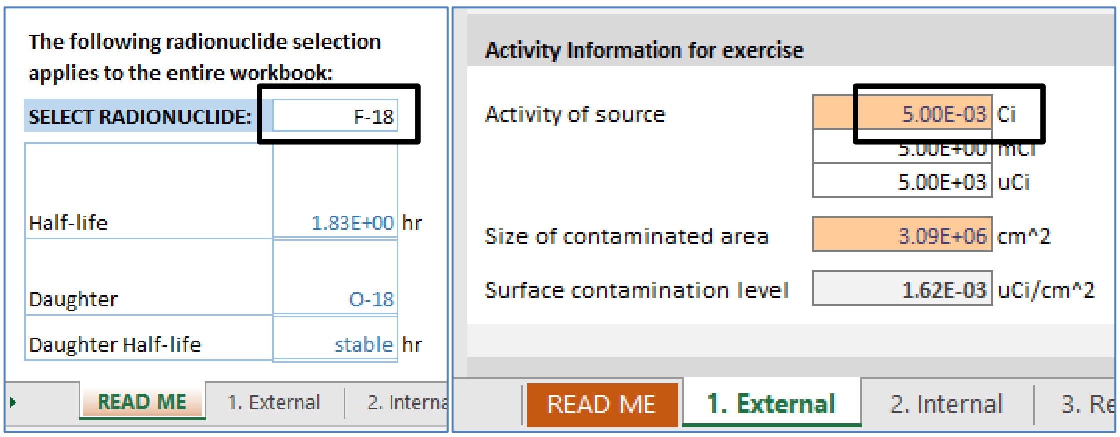

Figure 2.4 Source Characteristics Entry on Dose Assessment Method Spreadsheet.

2.2.2. Radiation Worker

Total external dose the Radiation Worker received was the sum of exposure from mixing and spraying. The external exposure for the Radiation Worker during mixing was assumed by default in the assessment tool to be as shown in Figure 2.5 and Table 2.1. The syringe containing the source was taken to be a point source. The extremity was the hand, which was 1 cm from the source volume. The whole body was the chest, 30 cm from the source. The mixing was assumed to take 1 min. The calculated dose was performed with the following gamma constant for F-18 [27]. Data entry to the assessment tool is shown in Figure 2.6.

$$\Gamma = 1.851 \times 10^{-4} \frac{mSv \cdot m^2}{hr \cdot MBq} \left(6.85 \times 10^{-1} \frac{rem \cdot m^2}{hr \cdot Ci}\right) \qquad \text{Equation 1}$$

Figure 2.5 External Exposure Description During Mixing (Photo Taken During Dry-Run).

Table 2.1 Preliminary External Exposure Parameters Used for Mixing.

	Whole Body	**Extremity**
Source Type	Syringe	Syringe
Source Distance	30 cm	1 cm
Exposure Time	1 min	1 min
Calculation Method	Gamma Constant (Point Source)	Gamma Constant (Point Source)

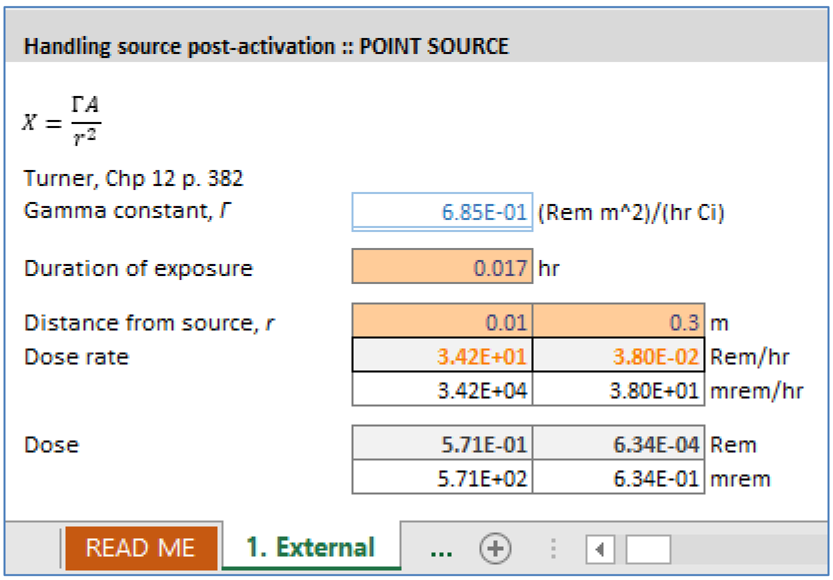

Figure 2.6 Assessment Tool Entry on External Dose during Mixing.

During spraying, external dose to the Radiation Worker was assumed to be only from the weed sprayer. The dose rates in the tool were determined using a cylinder source in MicroShield®. Ignoring the F-18 dispersed on the ground was based on that contribution from the radionuclide on the floor was minimal relative to the cylinder source. In the default case, the exposure rate from the cylinder was > 70 X that from the source plane. By assuming that exposure during the entire spraying event was from the cylinder source with the full volume of F-18, the estimated dose was expected to be more conservative than, for example, splitting half of the source between the cylinder and the plane source. The source and exposure distances are shown in Figure 2.7 and Table 2.2.

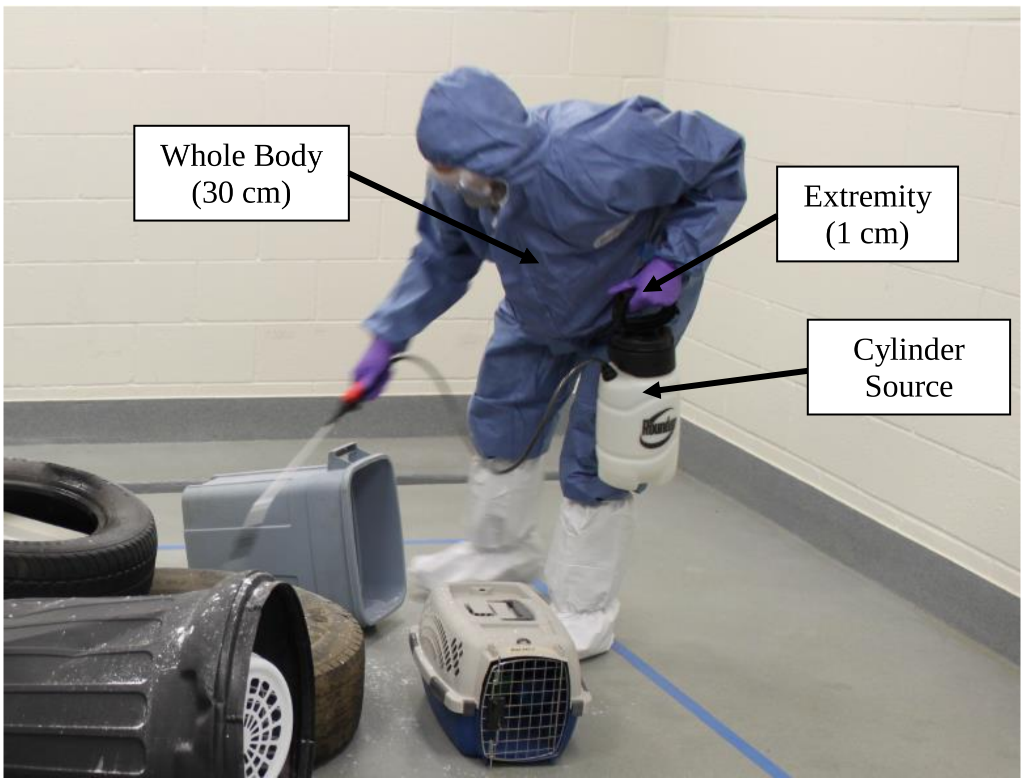

Figure 2.7 External Exposure Illustration during Spraying (Photo Taken During Dry-Run).

Table 2.2 Preliminary External Exposure Parameters While Spraying.

	Whole Body	**Extremity**
Source Type	3706.7 cm^3 Fluid	3706.7 cm^3 Fluid
Source Distance	30 cm	1 cm
Exposure Time	30 min	30 min
Calculation Method	MicroShield® 3706.7 cm^3 Cylinder	MicroShield® 3706.7 cm^3 Cylinder

The MicroShield® model assumed the weed sprayer was a container with 7.62 cm radius and 20.32 cm height. This was equivalent to about 3706.7 cm^3 (~ 1 gallon) of water. The extremity and chest exposed were 1 cm and 30 cm above the top of the cylinder, respectively. All other required inputs used program defaults, e.g., air density, water density, and energy grouping. The MicroShield® input is shown in Figure 2.8. The output used for analysis was "Exposure Rate mR/hr With Buildup", where the exposure to equivalent dose conversion was one, in the British unit system.

$$1\, roentgen \;\cong 1\, rem$$

The output from MicroShield® used for entry to the assessment tool is shown boxed in Figure 2.9. The input to the assessment tool is shown in Figure 2.10.

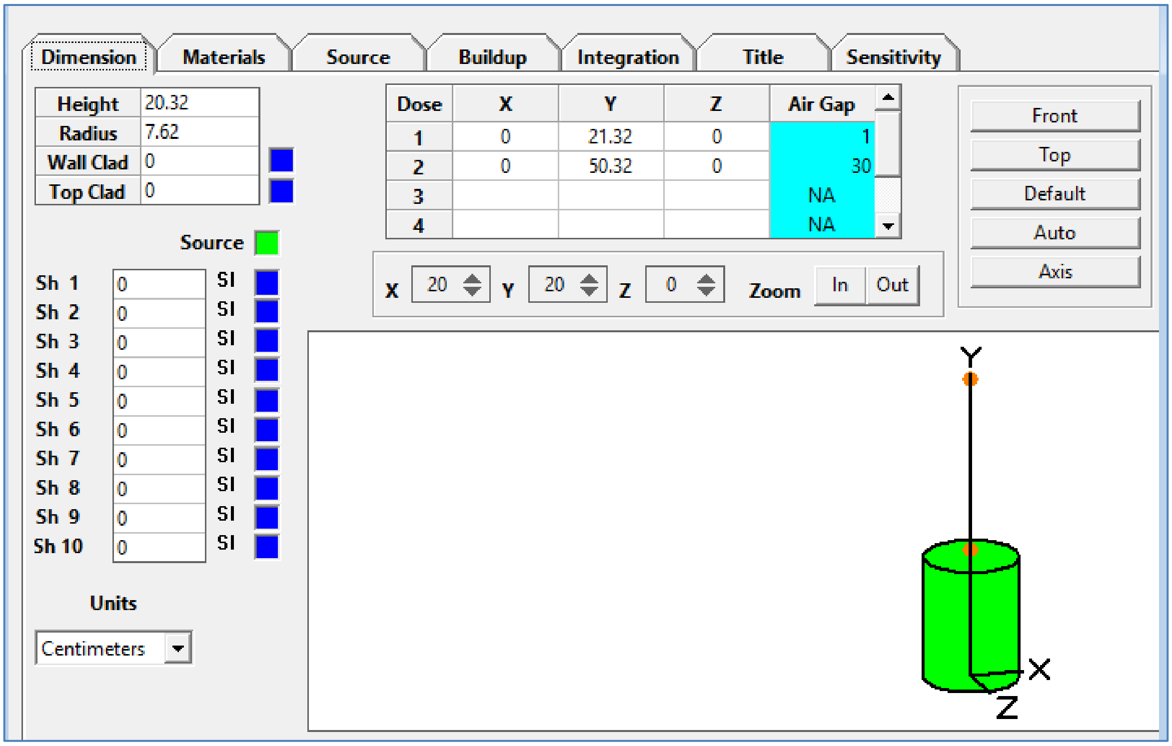

Figure 2.8 MicroShield® Modeling Input of Weed Sprayer as a Cylinder.

34			Results - Dose Point No. 1 - (X = 0, Y = 21.32, Z = 0) cm					
35–37	Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorb… N…
38	5.25E-04	3.31E+04	1.27E-03	1.47E-03	7.42E-03	8.56E-03	
39	5.11E-01	3.58E+08	1.16E+05	1.76E+05	2.28E+02	3.46E+02	
40	Total	3.58E+08	1.16E+05	1.76E+05	2.28E+02	345.8	1…
41							
42			Results - Dose Point No. 2 - (X = 0, Y = 50.32, Z = 0) cm				
43–45	Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorb… N…
46	5.25E-04	3.31E+04	4.16E-05	4.83E-05	2.43E-04	2.82E-04	
47	5.11E-01	3.58E+08	4.79E+03	8.50E+03	9.40E+00	1.67E+01	
48	Total	3.58E+08	4.79E+03	8.50E+03	9.40E+00	16.68	8…

Figure 2.9 Weed Sprayer MicroShield® Modeling Dose Rate Output of Spraying.

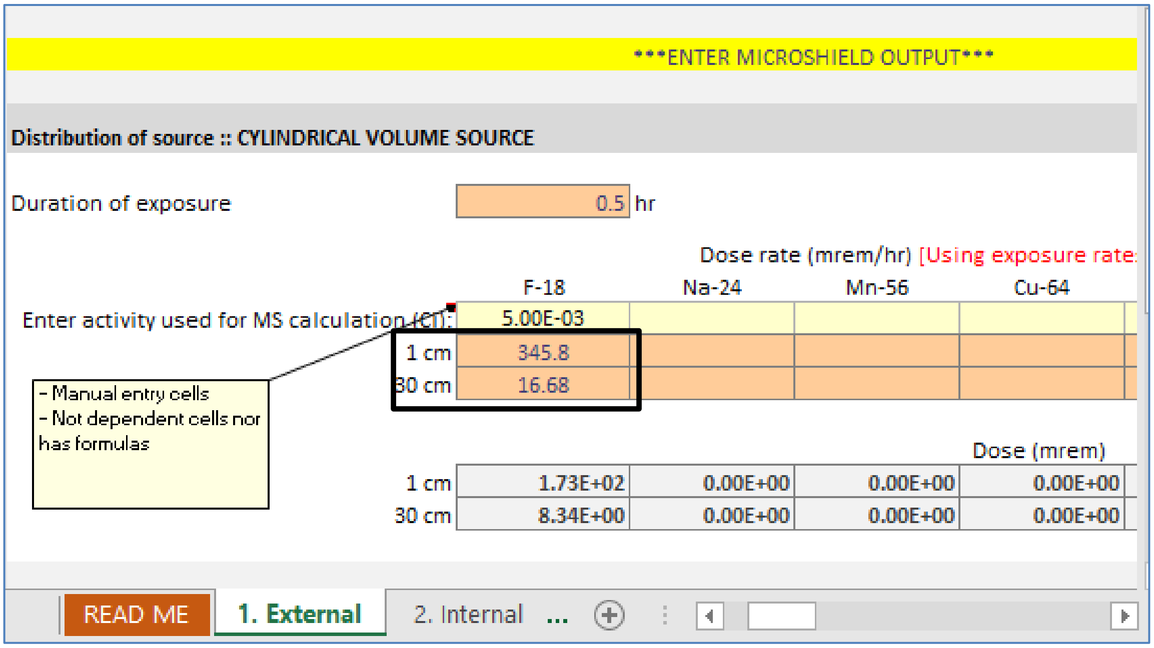

Figure 2.10 Dose Assessment Tool Input for Weed Sprayer.

2.2.3. Exercise Participant

External dose received by the Exercise Participant was from the source plane shown in Figure 2.11. The source was assumed have settled on the ground and was modeled as an infinite plane for the most conservative estimate. The extremity was the foot 1 cm from the ground. Whole body was 100 cm from the floor to be conservative. The total time of exposure was assumed to be 3 hours (Table 2.3).

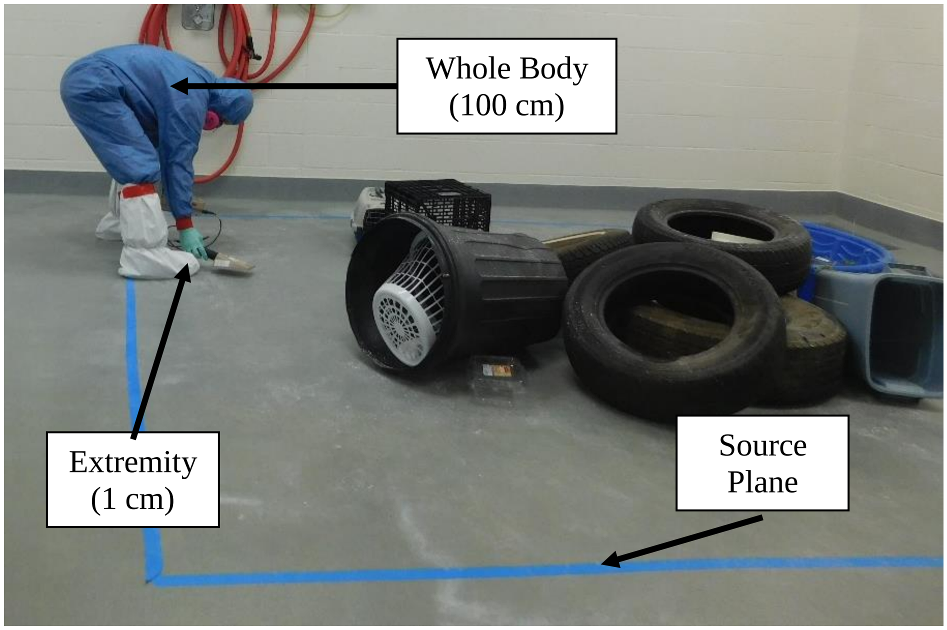

Figure 2.11 External Exposure Illustration of Exercise Participant

Table 2.3 Preliminary Exercise Participant External Exposure Parameters.

	Whole Body	**Extremity**
Source Type	$3.09 \times 10^6\ cm^2$ Plane	$3.09 \times 10^6\ cm^2$ Plane
Source Distance	100 cm	1 cm
Exposure Time	3 hours	3 hours
Calculation Method	MicroShield® Infinite Plane	MicroShield® Infinite Plane

Exposure rate was calculated using an infinite plane source in MicroShield®, shown in Figure 2.12. Source distances were 100 cm to the whole body and 1 cm to the extremity. The additional detector at 30 cm, boxed in Figure 2.12, was not used for calculating TEDE in the default analysis but was utilized in final modeling, to be discussed later. The areal density required for modeling was calculated using the assessment tool, which assumed the spray area to be 308.8 m^2 (3.09×10^6 cm^2). This areal density stemmed from the tool having been built to correlate with the NNSS TRACER exercise, which estimated that 3706.7 cm^3 (~ 1 gallon) of water would cover that amount of surface area [1] [28]. This is also the planned dispersion area for Disaster City®. The areal density output from the tool is shown boxed in Figure 2.13. The MicroShield® areal density input is circled in Figure 2.14. And, as before, the "Exposure Rate with Buildup" was used to estimate the equivalent dose rate, as shown boxed in Figure 2.15. Assessment tool input is shown boxed in Figure 2.16.

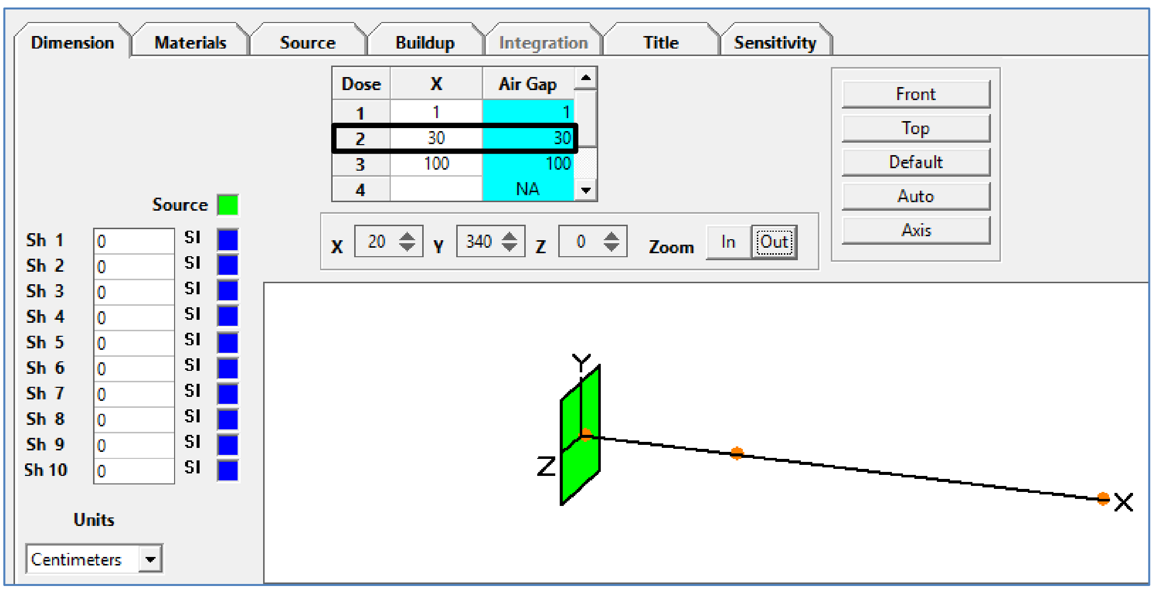

Figure 2.12 MicroShield® Infinite Plane Modeling of Exercise Participant External Exposure.

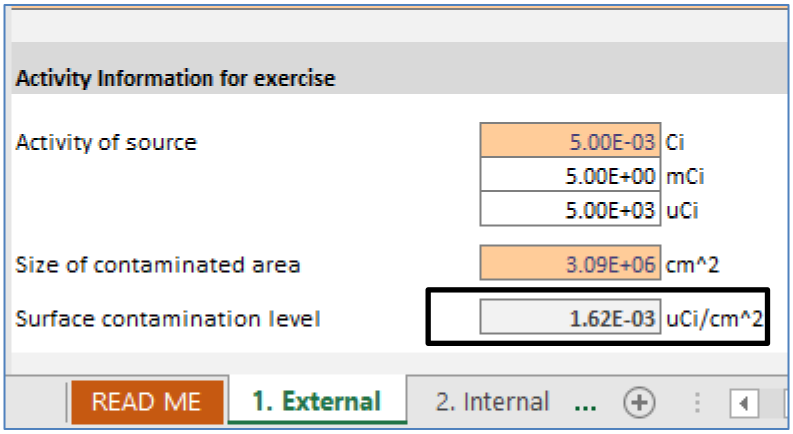

Figure 2.13 Areal Density Calculation with Assessment Method of the Dispersed Plane.

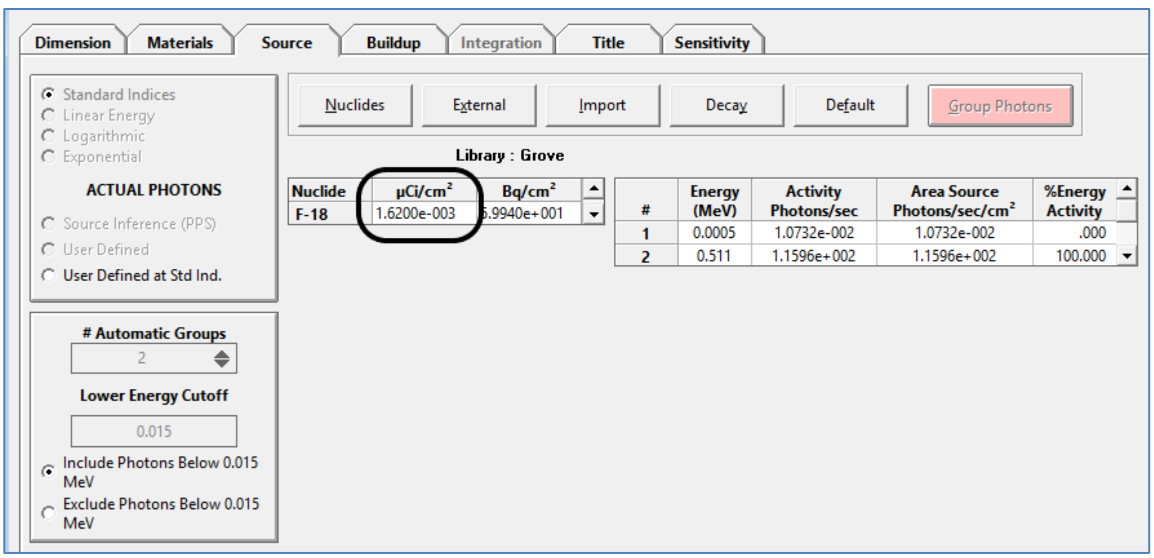

Figure 2.14 Areal Density Input to MicroShield® for the Dispersed Plane.

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
Results - Dose Point No. 1 - (X = 1, Y = 0, Z = 0) cm					
5.25E-04	1.07E-02	1.61E-05	1.65E-05	9.39E-05	9.60E-05
5.11E-01	1.16E+02	2.54E+02	2.54E+02	4.99E-01	4.99E-01
Total	**1.16E+02**	**2.54E+02**	**2.54E+02**	**4.99E-01**	**0.4992**
Results - Dose Point No. 2 - (X = 30, Y = 0, Z = 0) cm					
5.25E-04	1.07E-02	6.66E-06	7.00E-06	3.89E-05	4.09E-05
5.11E-01	1.16E+02	1.54E+02	1.54E+02	3.02E-01	3.02E-01
Total	**1.16E+02**	**1.54E+02**	**1.54E+02**	**3.02E-01**	**0.3016**
Results - Dose Point No. 3 - (X = 100, Y = 0, Z = 0) cm					
5.25E-04	1.07E-02	3.61E-06	3.91E-06	2.11E-05	2.28E-05
5.11E-01	1.16E+02	1.18E+02	1.18E+02	2.32E-01	2.32E-01
Total	**1.16E+02**	**1.18E+02**	**1.18E+02**	**2.32E-01**	**0.232**

Figure 2.15 MicroShield® Exposure Rate Output from Infinite Plane.

Exercise :: INFINITE PLANE SOURCE [Contributes to dose for participants only; not to sprayer]

Duration of exposure: 3.0 hr

Dose rate (mrem/hr) [Using exposu

	F-18	Na-24	Mn-56	Cu-64
Enter activity used for MS calculation (uCi/cm^2):	1.62E-03			
1 cm	0.4992			
30 cm	0.3016			
100 cm	0.232			

- Manual entry cells
- Not dependent cells nor has formulas

Dose (mre

1 cm	1.50E+00	0.00E+00	0.00E+00	0.00
30 cm	9.05E-01	0.00E+00	0.00E+00	0.00
100 cm	6.96E-01	0.00E+00	0.00E+00	0.00

READ ME | 1. External | 2. Internal | 3. Residual Rad | 4. A(...

Figure 2.16 Assessment Tool Input for Exercise Participant's External Exposure.

2.2.4. Sample of External Dose Output

The assessment tool calculated total external exposure for both the Radiation Worker and Exercise Participant as summarized in the “Nuclide Results” tab, is in Figure 2.17 and Figure 2.18, boxed in black. This data was used to calculate the TEDE, which was compared with the dose limit cut-offs to determine the safe amount of radioisotopes that can be used, as discussed in “1.2.1.3 Recommended Isotope Activities”.

				Duration (hr)	Intake (uCi)	Dose rate (mrem/hr)	Dose (mrem)
Radworker	External	Post-activation handling	1 cm	0.02		3.42E+04	5.71E+02
			30 cm			3.80E+01	6.34E-01
		Distribution	1 cm	0.5		3.46E+02	1.73E+02
			30 cm			1.67E+01	8.34E+00
	Internal	Ingestion	CEDE		5.00E+01		5.00E+00
			CDE		5.00E+01		5.31E+01
		Inhalation	CEDE		5.00E+01		3.57E+00
			CDE		5.00E+01		2.02E+01

		Limit (mrem)	10% admin limit (mrem)	Dose (mrem)	% of admin limit
Sum external at 30 cm =	DDE	5000	500	9	1.79E+00
Sum external at 1 cm =	SDE, EXT	50000	5000	744	1.49E+01

5. Nuclide Results | Sheet1 | 6. Library

Figure 2.17 Sample External Exposure from Dose Assessment Method to Radiation Worker.

				Duration (hr)	Intake (uCi)	Dose rate (mrem/hr)	Dose (mrem)
Exercise participant	External	Exercise	1 cm	3		4.99E-01	1.50E+00
			30 cm			3.02E-01	9.05E-01
			100 cm			2.32E-01	6.96E-01
	Internal	Ingestion	CEDE		1.00E+00		1.00E-01
			CDE		1.00E+00		1.06E+00
		Inhalation	CEDE		1.00E+00		7.14E-02
			CDE		1.00E+00		4.03E-01

		Limit (mrem)	10% of 10% admin limit (mrem)	Dose (mrem)	% of 10% of admin limit
External at 100 cm =	DDE	5000	50	0.7	1.39E+00
External at 1 cm =	SDE, EXT	50000	500	1.5	3.00E-01

5. Nuclide Results | Sheet1 | 6. Library

Figure 2.18 Sample External Exposure Results to the Exercise Participant.

2.3. Dispersion Exercise

The actual dispersion event took place on 22 May 2018. Pre-experimental preparation is described in APPENDIX F. The most pertinent events for the assessment are listed below, including several direct readings with ion chamber detectors. A summary of the exposure times is shown in Table 2.4. Actual amount of FDG injected was 200.5 MBq (5.419 mCi). The Exercise Participant was estimated to be in the contaminated area from 20 - 24 min. This range was used to calculate the minimum and maximum estimated doses.

0936 Radiation Worker Entered room

0937 Opened FDG lead casing

0937 Injected into sprayer and returned FDG to casing

Point source 1 – 3 s exposure (actual handling of syringe before injection)

0937 Swirled mixture

0938 Ion chamber 0.3 m from weed sprayer: 240 μSv/hr (24 mR/hr)

0938 Pumping

Ion chamber outside door 3 m from weed sprayer 70 μSv/hr (0.7 mR/hr)

0939 Started spraying

0941 Finished spraying

0942 Radiation Worker Exited room (Total time exposed to cylinder < 5 min)

0942-1018 Dispersion allowed to settle and dry (36 min elapsed)

1018 Exercise Participant entered room

1029 Exercise Participant surveyed room and took swipe samples

1041 Exercise Participant exited room

Table 2.4 Summary of Exposure Times during Experiment.

Exposure time	**Radiation Worker**	**Exercise Participant**
Syringe	1 – 3 s	-
Cylinder Source	< 5 min	-
Plane Source	~ 2 min (not for calculations)	22 min

2.3.1. Measurements

Monitoring of external exposure was accomplished with optically stimulated luminescence (OSL) badges and thermoluminescent dosimeter (TLD) rings, both from Landauer. OSL badges were Luxel®+, which had a γ and β detection range of 10 μSv – 10 Sv and 100 μSv – 10 Sv, respectively; uncertainty was ± 15%. At the lower limit of

10 μSv, the uncertainty was much higher at ± 20 μSv. Per Landauer datasheets, "Luxel®+ has Deep Dose (Hp 10) accuracy of ± 15% at the 95% confidence interval for photons above 20 keV" and "minimum reporting as low as 1 mrem, with a precision of ± 2 mrem" [29] [30]. TLDs were Saturn® Rings with γ and β detection range of 100 μSv – 10 Sv [31]; uncertainty was ± 20% (APPENDIX H).

Supplemental personal digital alarming dosimeters and direct reading instruments were also used. The first digital dosimeter was a Ludlum Model 25-1 Geiger-Mueller (GM) detector (Figure 2.19), which had a display range of 0.1 μSv/hr to 9.99 Sv/hr (0.01 mR/hr to 999 R/hr) and max cumulated dose of 9.99 Sv (999 R). Gamma response was 1800 cpm per mSv/h (18 cpm per mR/hr), β response was < .001 mSv/h (<0.10 mR/hr), and uncertainty was ±10% [32]. The second was a Canberra/Mirion UltraRadiac™-Plus Geiger-Mueller (GM) gamma detector (Figure 2.20), which had a measurement range for γ of 0.01 μSv/h to 2 Sv/h (1.0 μR/hr to 200 R/hr) and 0.001 μSv to 9.99 Sv (0.1 μR to 999 R); uncertain was ± 30% for 1 μSv/hr to 2.0 Sv/hr (100 μR/hr to 200 R/hr) and response time was 1 s [33]. Direct readings were taken with Fluke 451P gas ion chambers which were responsive to γ > 25 keV and β > 1 MeV, and had a response time of 1.8 s for dose rates from 0 to 12.9 μC/kg (0 to 50 mR/hr); uncertainty was ±10% [34].

Figure 2.19 Ludlum Model 25-1 Geiger-Mueller (GM) detector [32].

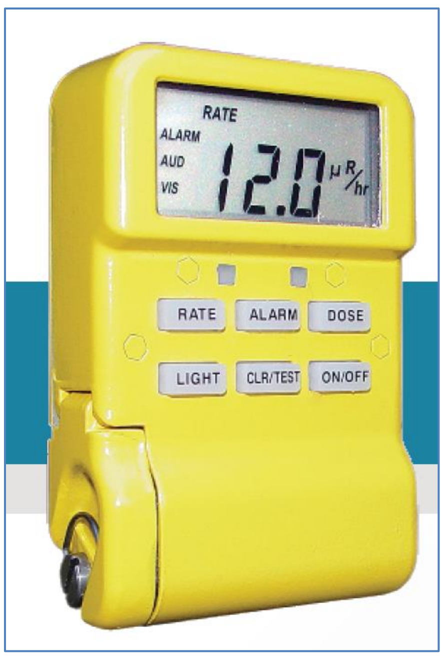

Figure 2.20 Canberra/Mirion UltraRadiac™-Plus Geiger-Mueller (GM) Detector [33].

2.3.1.1. Radiation Worker

Monitoring devices for the Mixer/Sprayer were placed as shown in Figure 2.21. OSL badges were worn under Tyvek® protective clothing at chest, waist, and one knee. TLD finger rings were fitted on both hands under two layers of 4 mil nitrile gloves. A Ludlum M25-1 Personal Radiation Monitor (alarm dosimeter) was worn on the chest inside the Tyvek®.

Figure 2.21 Dosimeters worn by Radiation Worker used for Validation.

2.3.1.2. Exercise Participant

Monitoring devices for the Exercise Participant were located as shown in Figure 2.22. OSL badges were worn under Tyvek® protective clothing, at chest and one knee. TLD finger rings were fitted on both hands under one layer of 4 mil nitrile gloves. A Canberra/Mirion UltraRadiac™-Plus Personal Radiation Monitor (alarm dosimeter) and a Ludlum M25-1 were worn on the chest inside the Tyvek®.

Figure 2.22 Dosimeters Worn by Exercise Participant used for Validation.

Other readings were taken to assess actual exposure. Two Fluke 451P gas ion chambers were placed near the door; one just inside, the other outside to monitor possible radiation leakage to the hallway. It also provided exposure rate estimations during the mixing process, which took place near the entryway. Though the digital personal radiation alarming dosimeters were factory calibrated, to verify the accuracy of the readings the meters were checked using a calibration standard after the experiment. Detailed procedure is discussed in APPENDIX B.

2.4. Refined Dose Estimation

After the dispersion exercise, more exact estimations of radiation doses were performed using the same dose assessment tool but with parameters from the actual event, rather than the defaults. This provided evidence on the accuracy of the dose assessment methodology.

2.4.1. Mixing and Spraying

A summary of the actual parameters for the Radiation Worker during mixing are shown in Figure 2.23 and Table 2.5. During mixing, the TLD on the hand holding the syringe was 5 cm from the source, while the chest OSL was about 30 cm from the syringe. Preliminary (default) parameters are shown also in Table 2.5, for comparison. While mixing the Radiation Worker injected the source then swirled the content. The exposure period for the refined estimate for mixing included only the time to inject the radioisotope into the container. Adding the swirling time over-estimated doses due to the high exposure rate of the point source (syringe), which was not applicable during swirling. Instead, swirling time was accounted for in the spraying period. During spraying, the TLD on the hand holding the weed sprayer was 37 cm above the top of the source volume, and the OSL on the chest was 90 cm from the source. This is shown in Figure 2.24 and Table 2.6. Preliminary (default) parameters are shown also in Table 2.6, for comparison.

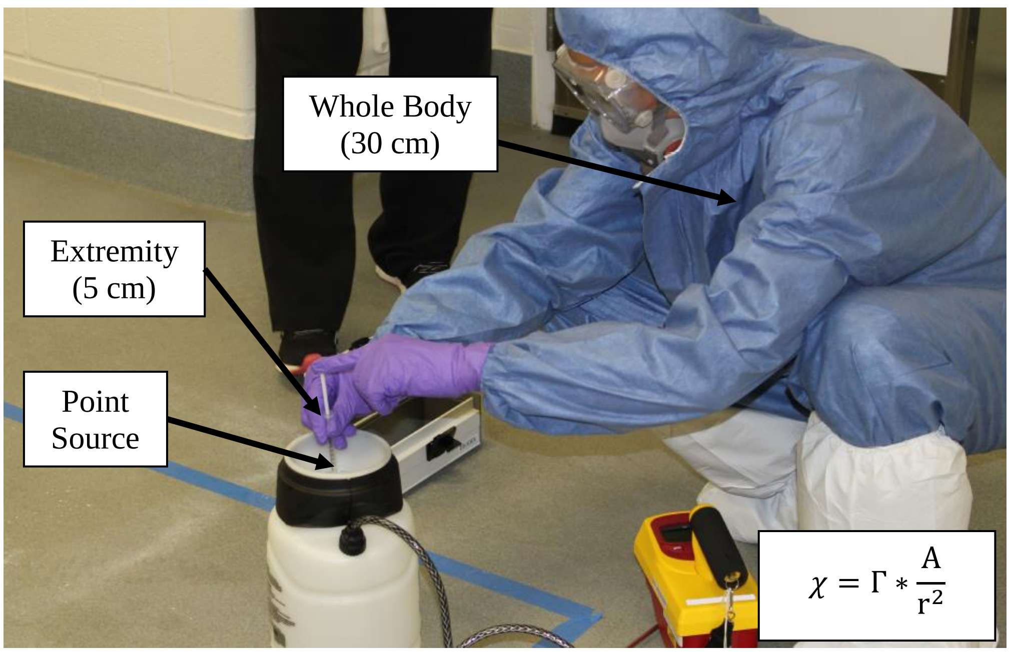

Figure 2.23 Calculation Method and Distances to Sources for Radiation Worker during Actual Mixing (Photo Taken during Dry Run).

Table 2.5 Preliminary and Final Parameters for Dose Predictions on Mixing Event.

	Preliminary Parameters		**Final Parameters**	
	Whole Body	**Extremity**	**Whole Body**	**Extremity**
Source Type	Syringe	Syringe	Syringe	Syringe
Source Distance	30 cm	1 cm	30 cm	5 cm
Exposure Time	1 min	1 min	1 – 3 s	1 – 3 s
Calculation Method	Gamma Constant (Point Source)	Gamma Constant (Point Source)	Gamma Constant (Point Source)	Gamma Constant (Point Source)

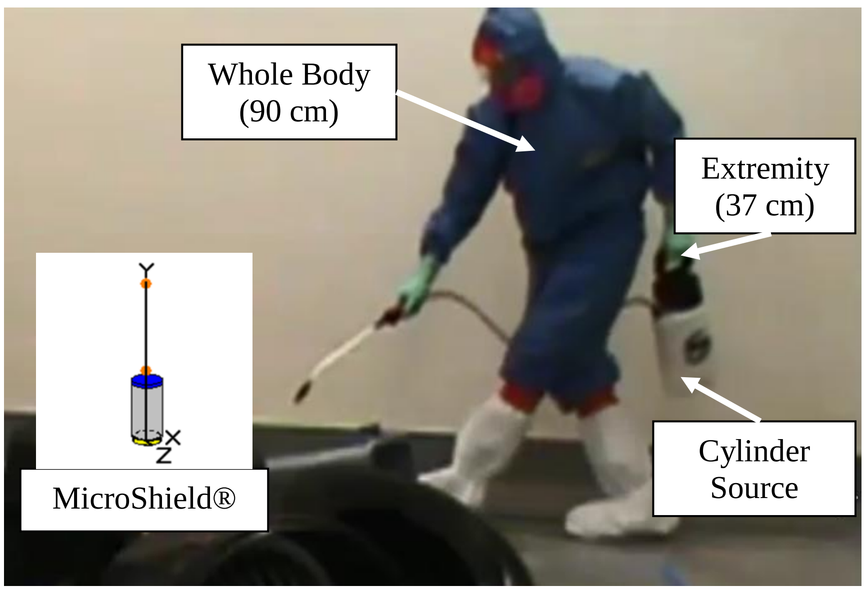

Figure 2.24 Calculation Method and Distances to Sources for Radiation Worker during Actual Spraying.

Table 2.6 Preliminary and Final Parameters for Dose Predictions on Spraying Event.

	Preliminary Parameters		**Final Parameters**	
	Whole Body	**Extremity**	**Whole Body**	**Extremity**
Source Type	3706.7 cm^3 Fluid	3706.7 cm^3 Fluid	462 cm^3 Fluid	462 cm^3 Fluid
Source Distance	30 cm	1 cm	90 cm	37 cm
Exposure Time	30 min	30 min	5 min	5 min
Calculation Method	MicroShield® 3706.7 cm^3 Cylinder	MicroShield® 3706.7 cm^3 Cylinder	MicroShield® 462 cm^3 Cylinder	MicroShield® 462 cm^3 Cylinder

Similar modeling methods implemented in the preliminary dose assessments were used to model the refined dose rates. To calculate dose rates during mixing, the F-18 gamma constant was used. For spraying, MicroShield® modeling was implemented to simulate exposure, shown in Figure 2.25. The source was modeled by a cylinder with 7 cm radius and 3 cm height for a volume of $462\ cm^3$. Shielding with non-borated high-density polyethylene (HDPE) was added around the source (Wall Clad boxed in figure) and just below the hand (Sh 2 boxed in figure) with 0.5 cm and 2 cm in thickness, respectively. This was to simulate the weed sprayer's plastic housing and the cap. The HDPE had a density of 0.944 g/cm^3 with 0.14372 w/o hydrogen and 0.85628 w/o carbon [35]. The area (Sh 1 boxed in figure) between the source and top shield was air.

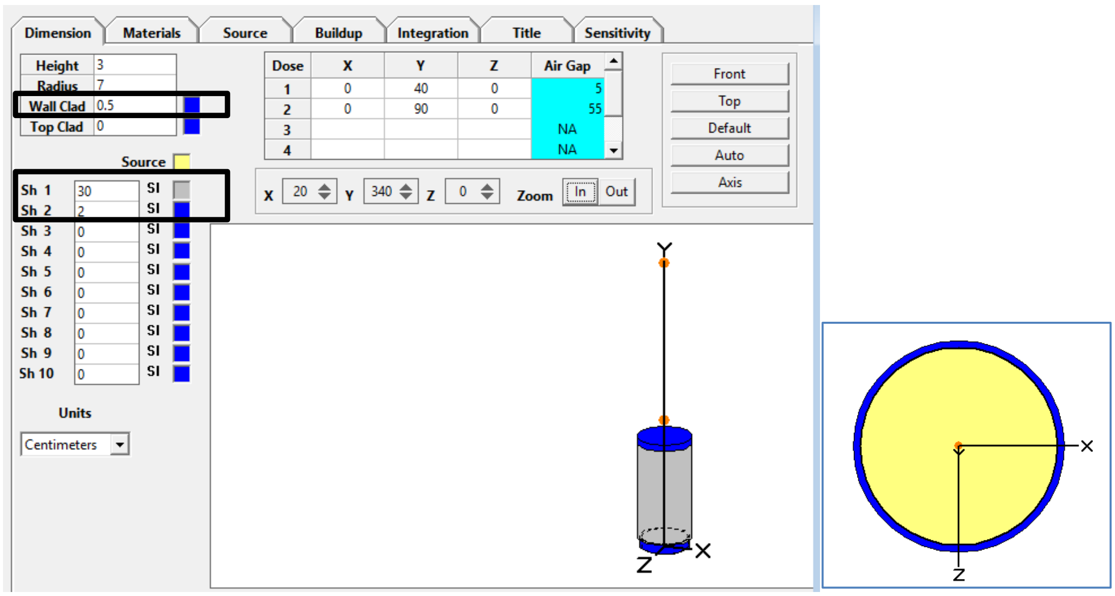

Figure 2.25 MicroShield® Modeling of Weed Sprayer Cylinder; Right figure is bottom view of source.

A sample output of the final adjusted model is shown in Figure 2.26. Note, due to uncertainties in the exposure times and distances, a range of results were calculated. For example, the mixer's exposure to the syringe containing the source could have ranged from 1 – 3 s. Due to the camera angle, the actual start and end of injection was not easily delineated from the video recording. This resulted in the estimated dose ranges of 2.0 – 3.5 µSv and 14.3 – 29.3 µSv for the dose to the whole body and extremity, respectively.

				Duration (hr)	Intake (uCi)	Dose rate (mrem/hr)	Dose (mrem)
Radwrkr (Mixer/Spayer)	External	Post-activation handling	5 cm	0.00		1.48E+03	1.24E+00
			30 cm			4.12E+01	3.44E-02
		Distribution	40 cm	0.0833333		2.03E+01	1.69E+00
			90 cm			3.83E+00	3.19E-01
	Internal	Ingestion	CEDE		5.42E+01		5.42E+00
			CDE		5.42E+01		5.75E+01
		Inhalation	CEDE		5.42E+01		3.87E+00
			CDE		5.42E+01		2.19E+01

		Limit (mrem)	10% admin limit (mrem)	Dose (mrem)	% of admin limit
Sum external Whole Body =	DDE	5000	500	0.35	7.07E-02
Sum external Hands =	SDE, EXT	50000	5000	2.93	5.86E-02
Sum ing and inh CEDE =	CEDE			9.29	
Sum ing and inh CDE =	CDE			79.40	
DDE + CEDE =	TEDE	5000	500	9.64	1.93E+00
DDE + CDE =	TODE	50000	5000	79.75	1.60E+00

Figure 2.26 Sample Final Adjusted Dose Modeling Results for Radiation Worker.

2.4.2. Exercise Participant

For the Exercise Participant, the parameters are shown in Figure 2.27 and Table 2.7. The OSL on the knee was used to measure and model extremity dose, which was 30 cm from the ground. The OSL on the chest for whole body dose was 100 cm from the floor. MicroShield® was used to estimate exposure rate, using a 304.8 cm x 304.8 cm (10 ft x 10 ft) or $9.29 \times 10^4\ cm^2$ source plane as shown in Figure 2.28. Preliminary (default) parameters are shown also in Table 2.7, for comparison.

Figure 2.27 Distances of Dosimeters to the Plane Source for Exercise Participant.

Table 2.7 Preliminary and Final Parameters for Dose Predictions on Exercise Participant.

	Preliminary Parameters		**Final Parameters**	
	Whole Body	**Extremity**	**Whole Body**	**Extremity**
Source Type	3.09×10^6 cm^2 Plane	3.09×10^6 cm^2 Plane	$9.29 \times 10^4\ cm^2$ Plane	$9.29 \times 10^4\ cm^2$ Plane
Source Distance	100 cm	1 cm	100 cm	30 cm
Exposure Time	3 hours	3 hours	20 – 24 min	20 – 24 min
Calculation Method	MicroShield® Infinite Plane	MicroShield® Infinite Plane	MicroShield® 304.8 cm x 304.8 cm Plane	MicroShield® 304.8 cm x 304.8 cm Plane

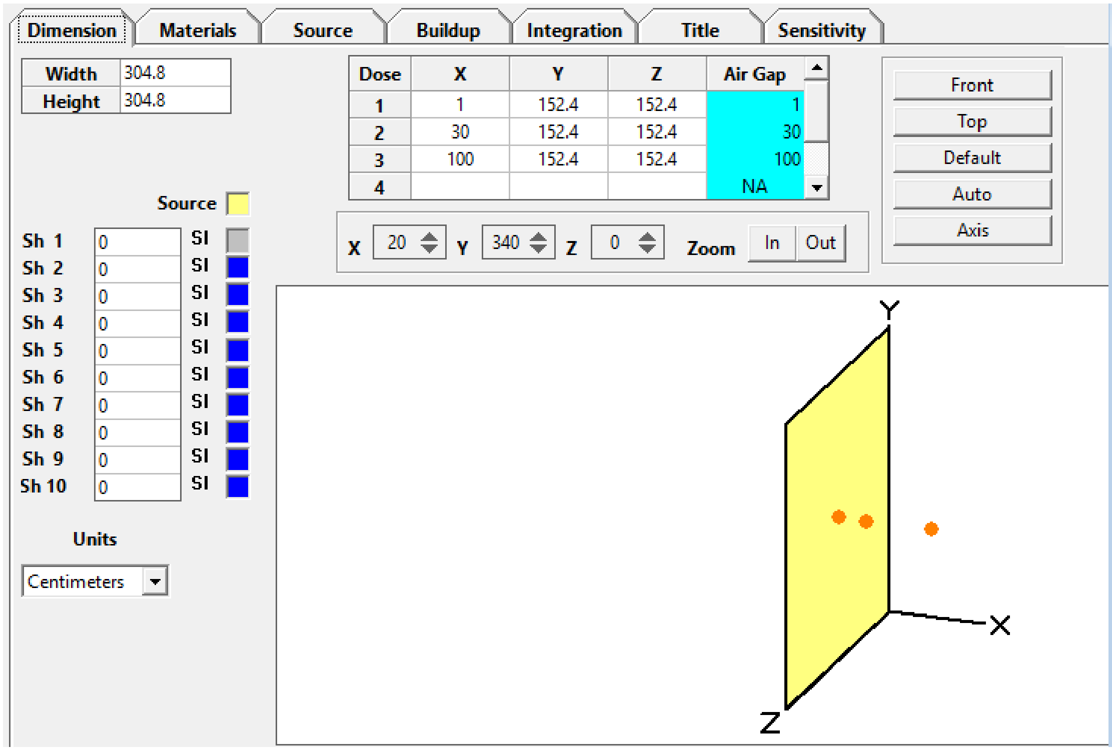

Figure 2.28 MicroShield® Modeling for Exercise Participant in Actual Event.

A sample result of the final dose modeling is shown in Figure 2.29. Similar to the Radiation Worker, ranges of results were obtained due to uncertainties in the exposures. In this case, the Exercise Participant left the contaminated area several times to retrieve instruments, etc.; therefore, his exposure time was estimated to have ranged from 20 – 24 min. These values were used to calculate the minimum and maximum estimates, which led to a dose range of 4.75 – 5.70 µSv and 12.2 – 14.66 µSv for the whole body and extremity, respectively. Furthermore, instead of using the MicroShield® detector at 1 cm in the preliminary assessment for extremity dose, the detector at 30 cm was used (Figure 2.28). This was due to not actually having worn a dosimeter 1 cm from the ground. The Exercise Participant, however, had an OSL worn at the knee, about 30 cm from the source plane.

				Duration (hr)	Intake (uCi)	Dose rate (mrem/hr)	Dose (mrem)	
Exercise participant (Dog Handler)	External	Exercise	1 cm	0.4		1.06E+01	4.25E+00	
			30 cm			3.67E+00	1.466	
			100 cm			1.43E+00	5.70E-01	
	Internal	Ingestion	CEDE		1.00E+00		1.00E-01	
			CDE		1.00E+00		1.06E+00	
		Inhalation	CEDE		1.00E+00		7.14E-02	
			CDE		1.00E+00		4.03E-01	
				Limit (mrem)	10% of 10% admin limit (mrem)		Dose (mrem)	% of 10% of admin limit
		External at 100 cm =	DDE	5000	50		0.570	1.14E+00
		External at 1 cm =	SDE, EXT	50000	500		4.248	8.50E-01
		Sum ing and inh CEDE =	CEDE				0.171	
		Sum ing and inh CDE =	CDE				1.465	
		DDE + CEDE =	TEDE	5000	50		0.742	1.48E+00
		DDE + CDE =	TODE	50000	500		2.036	4.07E-01

Figure 2.29 Sample Final Adjusted Modeling Results for Exercise Participant.

3. RESULTS AND DISCUSSION

3.1. Radiation Worker

3.1.1. Analysis of Preliminary Modeling Predictions

For the Radiation Worker, preliminary dose assessment modeling values were well above the actual dosimeter results, with the predictions at $\gg 2\sigma$ for the whole body and extremity values (Table 3.1). This provided assurance that the preliminary dose assessment methodology was a conservative method for evaluating loose contamination exercise doses and ensured that Radiation Worker exposure will be less than the dose limit cut-offs discussed in “1.2.1.3 Recommended Isotope Activities” and shown in Table 3.1.

Table 3.1 Radiation Worker Dose Results and Occupational Dose Limit Cut-Offs.

	Whole Body (µSv)	**Extremity (µSv)**
Preliminary Modeling	90	744
Final Adjusted Modeling	2.8 ± 0.8	21.8 ± 7.5
Actual Dosimeter	10 ± 2 (OSL)	< 100 (TLD)
Actual Digital Personal Radiation Monitor	4.3 ± 0.4	-
10% of Occupational Dose Limit	5,000 (TEDE)	50,000 (TODE)

Comparing the default preliminary modeling results versus the final modeling data provided some insight on how sensitive the parameters were on the predictions. Notably, the preliminary dose assessment predicted much higher whole body and extremity doses, at 90 μSv and 744 μSv, respectively. The source activity was actually lower for the preliminary model, at 185 MBq versus 200.503 MBq for the final model. The larger source volume in the preliminary model also decreased the initial predicted dose. Therefore, these factors did not contribute to the higher predicted dose.

Looking at the whole body dose, the process that contributed most to the Radiation Worker's exposure was spraying (distribution). As shown in the modeling results for the preliminary and final simulations in Figure 3.1 and Figure 3.2, the doses to the Radiation Worker during spraying ("Distribution", dark-colored boxes) were an order of magnitude greater than during mixing ("Post-Activation handling", light-colored boxes). Therefore, understanding how the parameters differed during spraying would explain why the preliminary model estimated higher whole body dose.

Two factors varied between the final and preliminary assessments during spraying--whole body distance from the source and exposure time. As shown in Table 3.2, preliminary parameters were more conservative by calculating with 1/3 the distance and 6X exposure time. (Note, the 5 min used to model actual spraying time was in the conservative end.) Both closer distance and longer exposure time increased the preliminary whole body dose predictions.

				Duration (hr)	Intake (uCi)	Dose rate (mrem/hr)	Dose (mrem)
orker	External	Post-activation handling	1 cm	1.67E-02		3.42E+04	5.71E+02
			30 cm			3.80E+01	6.34E-01
		Distribution	1 cm	5.00E-01		3.46E+02	1.73E+02
			30 cm			1.67E+01	8.34E+00

Figure 3.1 Preliminary Dose Predictions for Radiation Worker.

				Duration (hr)	Intake (uCi)	Dose rate (mrem/hr)	Dose (mrem)
ixer/Spayer)	External	Post-activation handling	5 cm	8.33E-04		1.48E+03	1.24E+00
			30 cm			4.12E+01	3.44E-02
		Distribution	40 cm	8.33E-02		2.03E+01	1.69E+00
			90 cm			3.83E+00	3.19E-01

Figure 3.2 Final Dose Model Estimates for Radiation Worker Comparing Dose Between Mixing and Spraying.

Table 3.2 Preliminary and Final Parameters for Dose Predictions on Spraying Event.

	Preliminary Parameters		**Final Parameters**	
	Whole Body	**Extremity**	**Whole Body**	**Extremity**
Source Type	3706.7 cm^3 Fluid	3706.7 cm^3 Fluid	462 cm^3 Fluid	462 cm^3 Fluid
Source Distance	30 cm	1 cm	90 cm	37 cm
Exposure Time	30 min	30 min	5 min	5 min
Calculation Method	MicroShield® 3706.7 cm^3 Cylinder	MicroShield® 3706.7 cm^3 Cylinder	MicroShield® 462 cm^3 Cylinder	MicroShield® 462 cm^3 Cylinder

Closer distance and longer exposure time also contributed to the higher extremity dose estimates in the preliminary model. For the extremity, total dose due to mixing and spraying were similar; however, the exposure rates were much higher during mixing (lighter boxes in Figure 3.3 and Figure 3.4). Therefore, understanding how the parameters differed during mixing instead, would explain why the preliminary model estimated higher extremity dose. The parameters in the preliminary and final modeling for mixing are shown in Table 3.3. The variations were in extremity exposure distance and time, with the preliminary parameters having 1/5 the distance and 20x exposure time. The above observations that closer distance and longer exposure times increased dose were intuitive. However, they helped to both validate the accuracy of the modeling tool and highlight that during the actual exercise, these factors greatly influence dose; especially when handing the syringe. Controlling these factors will greatly reduce personnel exposure in future experiments and exercises.

				Duration (hr)	Intake (uCi)	Dose rate (mrem/hr)	Dose (mrem)
orker	External	Post-activation handling	1 cm	1.67E-02		3.42E+04	5.71E+02
			30 cm			3.80E+01	6.34E-01
		Distribution	1 cm	5.00E-01		3.46E+02	1.73E+02
			30 cm			1.67E+01	8.34E+00

Figure 3.3 Preliminary Dose Predictions for Radiation Worker Comparing Dose Rates Between Mixing and Spraying.

				Duration (hr)	Intake (uCi)	Dose rate (mrem/hr)	Dose (mrem)
xer/Sprayer)	External	Post-activation handling	5 cm	8.33E-04		1.48E+03	1.24E+00
			30 cm			4.12E+01	3.44E-02
		Distribution	40 cm	8.33E-02		2.03E+01	1.69E+00
			90 cm			3.83E+00	3.19E-01

Figure 3.4 Final Dose Model Estimates for Radiation Worker Comparing Dose Rates Between Mixing and Spraying.

Table 3.3 Preliminary and Final Parameters for Dose Predictions on Mixing Event.

	Preliminary Parameters		**Final Parameters**	
	Whole Body	**Extremity**	**Whole Body**	**Extremity**
Source Type	Syringe	Syringe	Syringe	Syringe
Source Distance	30 cm	1 cm	30 cm	5 cm
Exposure Time	1 min	1 min	1 – 3 s	1 – 3 s
Calculation Method	Gamma Constant (Point Source)	Gamma Constant (Point Source)	Gamma Constant (Point Source)	Gamma Constant (Point Source)

3.1.2. Comparing Final Modeling Estimates with Actual Dosage

Comparing the final modeling results with the actual measurements provided insights on the accuracy of the assessment methodology. The preliminary dose estimates were designed to be conservative therefore they had a larger margin of error. Actual doses should be lower than the preliminary dose assessment for the method to be acceptable. The final refined model utilized parameters of the actual experiment. The assessment methodology can be further validated by studying how close the refined predicted results were to reality. The results are shown again below in Table 3.4 and compared with dose limit cut-offs discussed in “1.2.1.3 Recommended Isotope Activities”. In this experiment, the refined model accurately predicted that the dose would be well below the dose limits.

Table 3.4 Radiation Worker Dose Results and Occupational Dose Limit Cut-Offs.

	Whole Body (μSv)	**Extremity (μSv)**
Preliminary Modeling	90	744
Final Adjusted Modeling	2.8 ± 0.8	21.8± 7.5
Actual Dosimeter	10 ± 2 (OSL)	< 100 (TLD)
Actual Digital Personal Radiation Monitor	4.3 ± 0.4	-
10% of Occupational Dose Limit	5,000 (TEDE)	50,000 (TODE)

3.1.2.1. Whole Body

Final modeling result for the whole body was consistent with dosimetry. The whole body OSL measurement of 10 ± 2 μSv (20% uncertainty for 95% Confidence Level) was in the same order of magnitude as the final model estimate, after subtracting the background reading of 30 μSv. OSL results were also reported in increments of 10 μSv, therefore, the 95% confidence interval may have ranged from 4 – 18 μSv when ± 20% uncertainty was accounted for. The absolute range for a reported value of 10 μSv may have ranged from 5 – 15 μSv due to rounding. Subtracting 20% from 5 μSv provided the estimated minimum value of 4 μSv. Adding 20% to 15 μSv gave the maximum value of 18 μSv. With the actual exposure range of 4 – 18 μSv, the predicted dose was 0.4 μSv under and outside 2σ of the OSL reading (4 μSv – 3.6 μSv = 0.4 μSv).

The modeling result was also in the same order of magnitude and 0.3 μSv within 2σ from the digital dosimeter reading (3.9 μSv – 3.6 μSv = 0.3 μSv). Background readings were OSL results from unexposed members in the building. The consistency between the refined model and experimental values suggested that the methodology was accurate.

Arriving to the actual whole body OSL measurement of 10 μSv was not as straight-forward as expected and the experience served as a learning lesson for future studies. The analytical lab (Landauer) in-fact reported a dose of 40 μSv (4 mrem), about 10X higher than the final model estimate and the digital dosimeter reading. Further investigations revealed that the laboratory arrived at the net dose not by subtracting the background reading of a control dosimeter, but by subtracting a "Historical Customer Average Control Dose". This "Historical Customer Average Control Dose" was the typical dose of control OSLs sent from Texas A&M University in the past. Therefore, it was questionable whether the "Historical Customer Average Control Dose" was representative of the true background during the experiment.

By having made the further observation that whole body dosimetry results for all non-exposed members were 30 μSv, the conclusion was that the actual dose to the Radiation Worker was 30 μSv lower. This was consistent with other detector readings. The ion chamber having measured 240 μSv/hr at 0.3 m from the source implied that the maximum dose from the 5 min of total exposure would have been 20 μSv, half of the reported 40 μSv OSL dose. The readings from the digital personal radiation monitor, which were zeroed before spraying and verified with post-experimental testing, were consistent with the ion chamber reading and the modeling results. Therefore, the conclusion was that the background dose was 30 μSv higher than the "Historical Customer Average Control Dose", and the net dose to the Radiation Worker was actually 10 μSv. The lesson-learned was the need to carefully scrutinize dosimetry reports and any experimental results, and that inconsistencies may be indications of erroneous interpretation of data. (All dosimeter results are shown in APPENDIX C.)

3.1.2.2. Extremity Dose

The finger TLD result was less than the detection limit; and, the strength of validating the dose assessment method would be greater if TLD readings were above detection. The reported dose was < 100 μSv, consistent with the model estimate of 21.8 ± 7.5 μSv. However, the lab analysis of TLDs by Landauer had uncertainties of ± 20% (APPENDIX H) and most likely a wider range at the detection limit, similar to the OSLs. So, the actual dose to the extremity may have been as high as 120 μSv. Hence, inference on the validity of the model from a dose below the detection limit is not conclusive. In future studies, improvements can include implementing more sensitive dosimeters, which is discussed in "4.2 Low Resultant Dosage".

3.2. Exercise Participant

3.2.1. Analyzing Preliminary Modeling Predictions

For the Exercise Participant, preliminary dose assessment was in the same order of magnitude as the actual dosimeter results, as shown in Table 3.5. OSL results were reported in increments of 10 μSv, therefore, the 95% confidence interval ranged from 4 – 18 μSv when ± 20% uncertainty was accounted for, as discussed in "3.1.2.1 Whole Body". In that case, the predicted dose was within 2σ of the OSL reading. This finding implied that preliminary was accurate. One possible means of improving the preliminary dose assessment was discovered when the preliminary dose assessment results were compared with the final model estimates.

Table 3.5 Exercise Participant Dose Results and Occupational Dose Limit Cut-Offs.

	Whole Body (μSv)	**Extremity (μSv)**
Preliminary Modeling	7	15
Final Adjusted Modeling	5.2 ± 0.5	13.4 ± 1.2 (30 cm detector)
Actual Dosimeter (OSL)	10 ± 2	< 10
Actual Digital Personal Radiation Monitor	3.3 ± 1.0	-
1% of Occupational Dose Limit	500 (TEDE)	5,000 (TODE)

The difference between the preliminary and final estimates were expected to be much greater, considering the seemingly large variations in the parameters used. Because the preliminary analysis used an infinite plane rather than the actual size of the dispersion and an 8X longer exposure time, the dose prediction was expected to be greater than the final model. The parameters were shown in Table 2.7, and repeated below. The reason the preliminary estimate was not higher than the final modeling results was that, in the preliminary model the amount of source was dispersed on >30 X the area of the final model, reducing the areal density by that proportion. For the preliminary model, areal density was calculated by dividing the total activity over the estimated dispersion area of $3.09 \times 10^6\ cm^2$. In the final model, the areal density was calculated with a similar activity but over only 9.29×10^4 cm^2. Therefore, although the final assessment modeled with an infinite plane and a longer exposure time, the 1/30 reduction of areal density made the estimate similar to the final model.

Table 2.7 Preliminary and Final Parameters for Dose Predictions on Exercise Participant.

	Preliminary Parameters		**Final Parameters**	
	Whole Body	**Extremity**	**Whole Body**	**Extremity**
Source Type	3.09×10^6 cm^2 Plane	3.09×10^6 cm^2 Plane	$9.29 \times 10^4\ cm^2$ Plane	$9.29 \times 10^4\ cm^2$ Plane
Source Distance	100 cm	1 cm	100 cm	30 cm
Exposure Time	3 hours	3 hours	20 – 24 min	20 – 24 min
Calculation Method	MicroShield® Infinite Plane	MicroShield® Infinite Plane	MicroShield® 304.8 cm x 304.8 cm Plane	MicroShield® 304.8 cm x 304.8 cm Plane

The observation that the preliminary dose assessment was in the same order of magnitude as the dosimetry results and refined model suggested that the methodology is conservatively accurate. One potential method of improving the preliminary assessment methodology is to ensure that the areal density in the assessment will not be lower than in the actual dispersion. As discussed in 3.2.1, the reason the preliminary estimate was lower than expected was the over-estimation in the size of dispersion area. As shown in Table 3.6, if the areal density were modified to match the actual dispersion, the preliminary dose assessment results would have been > 10 X the dosimetry results. This would have provided a more conservative estimate.

Table 3.6 Results of Preliminary Dose Assessment with Areal Density Matching Experimental Dispersion.

	Whole Body (µSv)	Extremity (µSv)
Modified Preliminary Modeling	250	539
Final Adjusted Modeling	5.2 ± 0.5	13.4 ± 1.2
Actual Dosimeter (OSL)	10 ± 2	< 10
Actual Digital Personal Radiation Monitor	3.3 ± 1.0	-

3.2.2. Comparing Final Modeling Estimates with Actual Dosage

3.2.2.1. Whole Body

Similar to the preliminary dose assessment, the final simulation results for the whole body were in the same order of magnitude as the OSL readings and the digital personal radiation monitor (Table 3.7). Therefore, the final modeling estimate for the whole body was consistent with the actual exposure, which suggested that the methodology was accurate for the Exercise Participant.

Table 3.7 Exercise Participant Dose Results and Occupational Dose Limit Cut-Offs.

	Whole Body (μSv)	**Extremity (μSv)**
Preliminary Modeling	7	15
Final Adjusted Modeling	5.2 ± 0.5	13.4 ± 1.2 (30 cm detector)
Actual Dosimeter (OSL)	10 ± 2	< 10
Actual Digital Personal Radiation Monitor	3.3 ± 1.0	-
1% of Occupational Dose Limit	500 (TEDE)	5,000 (TODE)

3.2.2.2. Extremity Dose

Similar to the Radiation Worker, the experimental dose to the Exercise Participant's extremity (knee) was less than the OSL detection limit of 10 μSv. This was consistent with the model estimate when accounting for ±15% uncertainty and that doses were reported in increments of 10 μSv. This meant that the actual dose may have been as high as 17 μSv ($15\ \mu Sv \times 1.15 = 17.25\ \mu Sv$), which overlapped with the predicted dose range. Nonetheless, the impreciseness of an experimental dose that is below the detection limit made inference on the validity of the model weak. In future studies, improvements can include implementing more sensitive dosimeters, which is discussed in "4.2 Low Resultant Dosage".

4. LIMITATIONS

4.1. Sample Size

The first limitation of this study was the limited sample size. Although the modeling results were consistent with the dosimetry results, more experimental data would make statistical analysis possible--average doses and standard error could be calculated. In future experiments, having multiple Exercise Participants enter the contaminated area would be beneficial. Their dosimetry results could help predict the varying doses that trainees would receive due to their differing duties during an incident response. Having more than one Radiation Worker perform a small amount of spraying may not be practical. On the other hand, if the amount of the radioisotope will be high enough such that the Radiation Worker may be exposed to doses near the dose limit cut-offs, multiple mixers or sprayers may be possible and necessary.

4.2. Low Resultant Dosage

Another limitation was that several dosimeters received exposures similar or below background levels. One method to achieve dosage above detection limits would be to utilize dosimeters with lower limit of detection (LOD). While the LODs of OSL and TLDs were 100 μSv and 10 μSv, respectively, digital dosimeters worn in the current study had dose limits down to 0.001 μSv. More of digital detectors can be used to measure whole body dose in the future. Due to the bulkiness of digital dosimeters, using them to measure the extremity dose of hands will be more challenging. Unfortunately, the doses which were below dosimeter detection levels were for the extremity, therefore

OSLs and TLDs may be the only options. Post-experimental inquiries by the authors with the analytical laboratory (Landauer) and radiation safety office determined that special requests for analysis outside the default ranges can be arranged [30] [31]. It is possible to detect doses down to 0.01 μSv for an additional fee. Since the typical natural background radiation in the United States is about 3,200 μSv/year or 0.4 μSv/hr, 5 min exposure of the Radiation Worker and 20 min exposure of the Exercise Participant to background radiation would lead to about 0.02 μSv and 0.08 μSv of dose, respectively [36]. Detecting down to the lowest analytical capability of the lab may be useful in the future.

5. CONCLUSIONS

The need to properly prepare emergency workers for radiological dispersal incidents is of paramount importance. The conventional practice of using sealed-sources and table-top exercises cannot be expected to provide realistic training of detection skills (such as locating dispersed hot-spots or delineating exclusion zones), contamination avoidance, and decontamination. The long-term goal of this research is to aid in the development of safe exercise scenarios that involve unsealed radioactive material. A dose assessment tool based on the NNSS TRACER program was designed by Texas A&M University for this purpose in order to estimate the dose to personnel involved in the training.

The objective of the current project was to validate this tool using the dosimetry results of an actual dispersion event. The preliminary dose assessment was designed to conservatively estimate the dose to the personnel involved in the dispersion training, to ensure exposures were below the cut-off limits. In the tool, the cut-offs were 10% and 1% of federal occupational limit for the Radiation Worker and Exercise Participant, respectively. This validation study examined the accuracy of the assessment tool and ensured that preliminary modeled doses were still below set limits. A summary of the results is shown in Table 5.1 and Table 5.2.

Table 5.1 Radiation Worker Dose Results and Occupational Dose Limit Cut-Offs.

	Whole Body (µSv)	**Extremity (µSv)**
Preliminary Modeling	90	744
Final Adjusted Modeling	2.8 ± 0.8	21.8± 7.5
Actual Dosimeter	10 ± 2 (OSL)	< 100 (TLD)
Actual Digital Personal Radiation Monitor	4.3 ± 0.4	-
10% of Occupational Dose Limit	5,000 (TEDE)	50,000 (TODE)

Table 5.2 Exercise Participant Dose Results and Occupational Dose Limit Cut-Offs.

	Whole Body (µSv)	**Extremity (µSv)**
Preliminary Modeling	7	15
Final Adjusted Modeling	5.2 ± 0.5	13.4 ± 1.2 (30 cm detector)
Actual Dosimeter (OSL)	10 ± 2	< 10
Actual Digital Personal Radiation Monitor	3.3 ± 1.0	-
1% of Occupational Dose Limit	500 (TEDE)	5,000 (TODE)

In the actual dispersion experiment used to validate the dose assessment tool the Radiation Worker injected and mixed 200 MBq Fludeoxyglucose ^{18}F (FDG) with 470 ml H_2O in a commercial weed sprayer. The solution was distributed evenly over a 3 m x 3 m region in 5 min. After 45 min of evaporation, the Exercise Participant entered the area for a total of 22 min. Actual whole body (WB) doses from optically stimulated luminescence (OSL) were 10 ± 2 µSv for both the Radiation Worker and Exercise Participant. WB digital personal dosimeter readings were 4.3 ± 0.4 µSv and 3.3 ± 1.0 µSv for the Radiation Worker and Exercise Participant, respectively. Actual extremity doses to Radiation Worker's finger dosimeters were < 100 μSv (minimum detectable limit), and to exercise participant's leg OSL was < 10 μSv.

Preliminary dose assessment method was conservative for the Radiation Worker and conservatively accurate for the Exercise Participant. The predicted Radiation Worker doses were 90 μSv to the whole body (WB) and 744 μSv to the hand, both $\gg 2\sigma$ above the actual exposures. The Exercise Participant's estimated doses were 7 μSv to the WB and 15 μSv to the knee area, which were in the same order of magnitude as the actual.

After the method was adjusted to the exercise parameters, predicted doses for the Radiation Worker doses were 2.8 ± 0.8 μSv to the WB and 21.8 ± 7.5 μSv to the hand. The Exercise Participant's estimated doses were 5.2 ± 0.5 μSv to the WB and 13.4 ± 1.2 μSv to the knee area. Estimated whole body doses were in the same order of magnitude as the actual doses for both the Radiation Worker and the Exercise Participant. Comparing estimated extremity dose to the actual value was difficult, due to exposures having been below detectable limits, however, there were no obvious inconsistencies.

Further experimental data would provide stronger evidence on the validity of the dose assessment method. Suggested modifications to the procedure included ensuring that the dispersed area used to calculate the areal density in the preliminary dose assessment is the same or smaller than the actual dispersion area. More numerous sample sizes to facilitate robust statistical analysis can be achieved by performing repeated studies with multiple Radiation Workers and Exercise Participants. To overcome the challenge of analyzing doses below dosimeter detection limits, more sensitive dosimeters could be worn by personnel, such as using more digital dosimeters. A more practical option for the extremity doses would be to request the analysis and reporting of lower doses on OSLs and TLDs. Lastly, the full validation of the assessment tool would include testing the model for internal dose, skin exposure, and accidental scenarios.

Overall, the dose assessment method has shown so far to be accurate and a conservative tool to predict doses during designed exercises using unsealed sources. These findings served as a stepping stone in the goal of creating practical dispersion training exercises, so that emergency workers will be better equipped to respond effectively, efficiently, and safely during future radiation incidents.

REFERENCES

[1] L. Cochran, "Masters of Science Thesis: Preliminary Dose Assessment for Emergency Respone Exercise at Disaster City Using Unsealed Radioactive Contamination," Texas A&M University, College Station, 2016.

[2] U.S. Nuclear Regulatory Commission, "ALARA," 6 July 2018. [Online]. Available: https://www.nrc.gov/reading-rm/basic-ref/glossary/alara.html. [Accessed 10 March 2019].

[3] National Council on Radiation Protection and Measurements (NCRP), "An Overview of NCRP Commentary No. 19," [Online]. Available: https://ncrponline.org/wp-content/themes/ncrp/PDFs/Commentary-No-19-Overview.pdf. [Accessed 10 Mar 2019].

[4] State of Michigan, "Simple Tabletop Exercise, Physical Attack - Witness Account and Notification by Law Enforcement Scenario, Scenario #5, Facilitator's Guide," [Online]. Available: https://www.michigan.gov/documents/deq/deq-wb-wws-SSc5-0_272213_7.pdf. [Accessed 10 Mar 2019].

[5] L. D. Cochran, "Radionuclide Selection for Emergency Response Exercise at Disaster City® Using Unsealed Radioactive Contamination," *Health Physics,* vol. 114, no. 1, p. 7–12, 2018.

[6] B. RL, "Assessment of F-18 use as a live radiological agent in field exercises; SRNL-L2300-2010-00117," Savannah River National Laboratory, Aiken, SC, 2010.

[7] Idaho National Laboratory, "Idaho National Laboratory Radiological Response Training Range Environmental Assessment; DOE/EA-1776," Idaho National Laboratory, Idaho Falls, ID, 2010.

[8] Texas A&M Engineering Extension Service, "Disaster City®," 2015. [Online]. Available: https://teex.org/Pages/about-us/disaster-city.aspx. [Accessed 10 March 2019].

[9] S. L. Simont, "Uncertainty Analysis and Model Validation For a Retrospective Assessment Of Thyroid Dose Resulting From Atomic Weapons' Test Fallout," in *Reliability of Radioactive Transfer Models*, Brussels and Luxembourg, Springer, Dordrecht, 1988, pp. 54-65.

[10] V. Golikov, "Model validation for external doses due to environmental contaminations by the Chernobyl accident," *Health Physics,* vol. 77, no. 6, pp. 654-661, 1999.

[11] Chernobyl 131 I Release Working Group of EMRAS Theme 1 , "The Chernobyl I-131 Release: Model Validation and Assessment of the Countermeasure Effectiveness," IAEA Environmental Modelling for RAdiation Safety (EMRAS) Programme, 2007.

[12] K. Abe, "Estimation of85Kr dispersion from the spent nuclear fuel reprocessing plant in Rokkasho, Japan, using an atmospheric dispersion model," *Radiation Protection Dosimetry,* vol. 167, no. 1-3, pp. 331-335, 2015.

[13] E. Kim, "Estimation of Early Internal Doses to Fukushima Residents after the Nuclear Disaster Based on the Atmospheric Dispersion Simulation," *Radiation Protection Dosimetry,* vol. 171, no. 3, pp. 398-404, 2016.

[14] K. Andersson, "Requirements for estimation of doses from contaminants dispersed," *Journal of Environmental Radioactivity,* vol. 100, no. 2009, pp. 1005-1011, 2009.

[15] Argonne National Laboratory, "RESRAD-RDD," [Online]. Available: http://resrad.evs.anl.gov/codes/resrad-rdd/. [Accessed 10 March 2019].

[16] Sandia National Laboratories, "Turbo FRMAC," [Online]. Available: https://nirp.sandia.gov/Software/TurboFRMAC/TurboFRMAC.aspx. [Accessed 11 March 2019].

[17] Sandia National Laboratory, "SHARC (Specialized Hazard Assessment Response Capability)," [Online]. Available: https://nirp.sandia.gov/Software/SHARC/Default.aspx#Overview. [Accessed 11 March 2019].

[18] United States Environmental Protection Agency, "Tools for Calculating Radiation Dose and Risk," [Online]. Available: https://www.epa.gov/radiation/tools-calculating-radiation-dose-and-risk#dcal. [Accessed 12 March 2019].

[19] Texas A&M Environmental Health and Safety, "Radiological Safety," [Online]. Available: https://ehs.tamu.edu/programs/radiological-safety/. [Accessed 10 March 2019].

[20] Texas Department of the State, "Laws and Rules - Radiation Control Program," [Online]. Available: https://www.dshs.texas.gov/radiation/laws-rules.aspx. [Accessed 10 Mar 2019].

[21] Nuclear Regulatory Commission, "Subpart C—Occupational Dose Limits, § 20.1201 Occupational dose limits for adults.," [Online]. Available: https://www.nrc.gov/reading-rm/doc-collections/cfr/part020/part020-1201.html. [Accessed 10 March 2019].

[22] Nuclear Regulatory Commission, "Subpart D—Radiation Dose Limits for Individual Members of the Public, § 20.1301 Dose limits for individual members of the public.," 24 August 2018. [Online]. Available: https://www.nrc.gov/reading-rm/doc-collections/cfr/part020/part020-1301.html. [Accessed 10 March 2019].

[23] North Shore/LIJ Research Institute, "New Drug Application NDA 21-870: FDG F 18 Injection," [Online]. Available: https://www.accessdata.fda.gov/drugsatfda_docs/label/2005/021870lbl.pdf.

[24] Task Group for the MIRD Committee, "MIRD Dose Estimate Report No. 19: Radiation Absorbed Dose Estimates from 18F-FDG," *J Nucl Med,* vol. 43, no. http://jnm.snmjournals.org/content/43/2/210.long, p. 210–214, 2002.

[25] RxList, "Fludeoxyglucose," 3 March 2017. [Online]. Available: https://www.rxlist.com/fludeoxyglucose-drug.htm#dosage. [Accessed 10 March 2019].

[26] K. Cook, "Masters of Science Thesis: Analyzing a Canine in a Radioacitve Contaminated Working Environment," College Station, 2018.

[27] Oakridge National Laboratory, "ORNL/RSIC-45/R1, Specific Gamma-Ray Dose Constants for Nuclides Important to Dosimetry and Radiological Assessment," U.S. Nuclear Regulatory Commission, Oak Riedge, TN, 1982.

[28] G. J., "Technical justification for the TRACER program at the Nevada National Security Site," National Security Technologies, LLC, 2012.

[29] Landauer, "Luxel®+ Specifications," 2018. [Online]. Available: https://www.landauer.com/sites/default/files/product-specification-file/Luxel%2B_1.pdf. [Accessed 14 March 2019].

[30] Landauer, "Luxel®+ Dosimeter for X, Gamma, Beta, and Neutron Radiation," 2006. [Online]. Available: https://www.landauer.com.au/Content/Specsheets/LuxelSpecSheet.pdf. [Accessed 14 March 2019].

[31] Landauer, "Saturn® TLD Ring," 2017. [Online]. Available: https://www.landauer.com/sites/default/files/product-specification-file/Saturn_Ring_2.pdf. [Accessed 14 March 2019].

[32] Ludlum Measurement, Inc., "Model 25-1 Personal Radiation Monitor," 2019. [Online]. Available: https://ludlums.com/products/all-products/product/model-25-1. [Accessed 14 March 2019].

[33] Mirion Technologies (MGPI), Inc., "ULTRARADIAC-PLUS™," [Online]. Available: https://www.mirion.com/products/ultraradiac-plus-personal-radiation-monitor. [Accessed 14 March 2019].

[34] FLUKE, "451P Radiation Detector - Pressurized," [Online]. Available: https://www.flukebiomedical.com/products/radiation-measurement/radiation-safety/451p-radiation-detector-pressurized. [Accessed 14 March 2019].

[35] Pacific Northwest National Laboratory, "Compendium of Material Composition Data for Radiation Transport Modeling Rev 1," 4 March 2011. [Online]. Available: https://www.pnnl.gov/main/publications/external/technical_reports/pnnl-15870rev1.pdf. [Accessed 14 March 2019].

[36] U.S. Nuclear Regulatory Commission, "Doses in Our Daily Lives," 02 October 2017. [Online]. Available: https://www.nrc.gov/about-nrc/radiation/around-us/doses-daily-lives.html. [Accessed 11 May 2019].

[37] Los Alamos National Laboratory, "A General Monte Carlo N-Particle (MCNP) Transport Code," [Online]. Available: https://mcnp.lanl.gov/mcnp_how_to_get_to_mcnp.shtml. [Accessed 10 March 2019].

[38] "Geant4 Simulation Kit Overview," Geant4 Working Groups, [Online]. Available: https://geant4.web.cern.ch/. [Accessed 12 Mar 2019].

[39] C. Travis, "Evaluation of Remediation Worker Risk at Radioactively Contaminated Waste Sites," *Journal of Hazardous Materials,* vol. 35, pp. 387-401, 1993.

[40] L. A., "Dynamic Dose Modeling/Soil Segregation: A Method For Reducing Uncertainty and Increasing Efficiency During Radiological Decommissioning," in *WM'05 Conference*, Tucson, AZ, February 2-March 3 2005.

[41] M. Dennis, "Dose Estimates in a Loss of Lead Shielding Truck Accident," Sandia Report SAND2009-5107, August 2009.

[42] E. Waller, "Overview of Hazard Assessment and Emergency Planning Software of Use To RN First Responders," *Health Phys.*, vol. 97, no. 2, p. 145–156, 2009.

[43] K. Sanders, "Radiological Decontamination in the Urban Environment Utilizing an Irreversible Wash-Aid Recovery System," Air Force Institute of Technology AFIT-ENV-MS-18-M-233, March 2018.

[44] Nuclear Regulatory Commission, "NUREG-1575, Rev. 1: Multi-Agency Radiation Survey And Site Investigation Manual (MARSSIM)," Washington, 2000.

[45] J. McLaughlin, The Natural Radiation Environment VII, Volume 7, 1st Edition, Elsevier Science, 2005.

[46] SHB, Inc., "Microshield and MCNP5 Modeling of Natural Uranium Ore Mined Using Ablation Technology," Environmental Restoration Group, Inc., Albuquerque, NM, 2016.

[47] Grove Software, "MicroShield Specifications," [Online]. Available: http://radiationsoftware.com/microshield/ . [Accessed 10 March 2019].

[48] Grove Software, "MicroShield® User's Manual Version 11," Grove Software, 2017.

[49] J. E. Turner, Atoms, Radiation, and Radiation Protection, 3rd Edition, Weinheim: WILEY-VCH Verlag GmbH & Co., 2007.

[50] ICRP, "ICRP Publication 30: (Part 1) Limits for Intakes of Radionuclides by Workers," *Annuals of the ICRP*, vol. 2, no. 3-4, pp. 1-116, 1979.

[51] ICRP, "ICRP Publication 60: 1990 Recommendations of the International Commission on Radiological Protection," *Annuals of the ICRP*, vol. 21, no. 1-3, pp. 1-201, 1991.

[52] Nucleonica GmbH, "9 F 18," European Atomic Energy Community, [Online]. Available: https://nucleonica.com/Application/ReducedDecaySchemes/F18_TXT.htm. [Accessed 12 March 2019].

[53] NNDC, Brookhaven National Laboratory, "ENSDF Decay Data in the MIRD (Medical Internal Radiation Dose) Format for 18F," November 1996. [Online]. Available: https://www.nndc.bnl.gov/useroutput/18f_mird.html. [Accessed 12 March 2019].

[54] BNC, "BNC Model 940 SAM Eagle Instruction Manual," [Online]. Available: http://en.kanggaote.com/static/upload/products/20170608144553373_446374_853.pdf. [Accessed 12 March 2019].

[55] IEEE, "ANSI N42.34-2015 American National Standard Performance Criteria for Handheld Instruments for the Detection and Identification of Radionuclides," 2015. [Online]. Available: https://ieeexplore.ieee.org/stamp/stamp.jsp?arnumber=7551091. [Accessed 12 March 2019].

[56] Nucleonica GmbH, "55 Cs 137," European Atomic Energy Community, [Online]. Available: https://www.nucleonica.com/Application/ReducedDecaySchemes/Cs137_TXT.htm. [Accessed 12 March 2010].

[57] M. Isaksson, "Environmental Dosimetry – Measurements and Calculations," 19 October 2011. [Online]. Available: https://www.intechopen.com/books/radioisotopes-

applications-in-physical-sciences/environmental-dosimetry-measurements-and-calculations. [Accessed 16 March 2019].

[58] J. K. Shultis, Radiation Shielding, American Nuclear Society, 2000.

[59] Keisan Online Calculator, "Exponential integral En(x) Calculator," CASIO COMPUTER CO., LTD., 2019. [Online]. Available: https://keisan.casio.com/exec/system/1180573425. [Accessed 16 Mar 2019].

[60] Sun Microsystems, Inc., "Appendix D What Every Computer Scientist Should Know About Floating-Point Arithmetic," [Online]. Available: https://docs.oracle.com/cd/E19957-01/806-3568/ncg_goldberg.html. [Accessed 04 May 2019].

[61] Y. Rojavin, "Civilian nuclear incidents: An overview of historical, medical, and scientific aspects," *J Emerg Trauma Shock,* p. 260–272, 2011.

[62] Radiation Emergency Medical Management, "Radiological Dispersal Devices (RDDs)," 15 March 2019. [Online]. Available: https://www.remm.nlm.gov/rdd.htm#otherdispersalmethods.

[63] International Atomic Energy Agency, Vienna, "The Radiological Accident in Goiania," IAEA, Australia, 1988.

[64] W. R. Johnston, "Nuclear Terrorism Incidents," Johnston's Archive, 28 September 2003. [Online]. Available: http://www.johnstonsarchive.net/nuclear/wrjp1855.html. [Accessed 10 March 2019].

[65] R. K. Mullen, "Nuclear Violence," in *Preventing Nuclear Terrorism: The Report and Papers of the International Task Force on Prevention of Nuclear Terrorism,* Lexington, MA, Lexington Books, 1987, pp. 231-247.

[66] M. A. Pomper, "Nuclear terrorism – Threat or not?," *AIP Conference Proceedings,* vol. 1898, no. 1, 2017.

[67] Nuclear Regulatory Commission, "Backgrounder on Dirty Bombs," May 2018. [Online]. Available: https://www.nrc.gov/reading-rm/doc-collections/fact-sheets/fs-dirty-bombs.html. [Accessed 10 March 2019].

[68] R. Johnston, "Statistical summary of radiological accidents and other incidents causing radiation casualties," Johnston Archive, 20 Novermber 2011. [Online]. Available: http://www.johnstonsarchive.net/nuclear/radevents/radeventdata.html. [Accessed 10 March 2019].

[69] World Nuclear Association, "Fukushima: Radiation Exposure," February 2016. [Online]. Available: http://www.world-nuclear.org/information-library/safety-and-security/safety-of-plants/appendices/fukushima-radiation-exposure.aspx. [Accessed 10 March 2019].

[70] NIST, "Polyethylene X-Ray Mass Attenuation Constant," [Online]. Available: https://physics.nist.gov/PhysRefData/XrayMassCoef/ComTab/polyethylene.html. [Accessed 15 March 2019].

[71] J. K. Shultis, "Chapter 11 Radiation Shielding and Radiological Protection," in *Handbook of Nuclear Engineering*, Springer Science+Business Media LLC, 2010, pp. 1313 - 1448.

[72] U.S. Environmental Protection Agency, "Federal Guidance Report (FGR) 12: External Exposure to Radionuclides in Air, Water, and Soil," Oak Ridge National Laboratory, Oak Ridge, TN, 1993.

[73] Kansas State Univeristy, "Radiation Detector & Dosimeter Calibration Room," [Online]. Available: http://smartlab.mne.ksu.edu/rooms/Room%20Listings/Room%20002%20Ward%20-%20Radiation%20Detector%20and%20Dosimeter%20Calibration%20Room/sview. [Accessed 18 June 2019].

APPENDIX A

MICROSHIELD AS SIMULATION SOFTWARE OF CHOICE

NNSS's TRACER and Cochran's dose assessment method required the use of simulation software to estimate external exposure, and MicroShield® was chosen for this purpose. Other well-established and flexible modeling software such as Monte Carlo N-Particle (MCNP) and GEANT4 could be utilized to accurately simulate the exposure during these exercises. However, MCNP's being an export-controlled code and the requirement of having an expert user for both programs made them less practical for the current project [37] [38].

Instead, MicroShield® was the preferred software for several reasons. First, federal agencies and industries have relied on this software since 1993 to perform dose and shielding assessments; including for contaminated waste sites, decommissioning, truck accidents, and emergency planning [39] [40] [41] [42] [43] [44]. Although experimental validation studies have not been directly performed for RDD, it has been shown to agree with numerous other codes including MCNP, in settings that include uranium ores and sealed sources [45] [46].

MicroShield® regularly updated its program through periodic software revisions, according to International Commission on Radiological Protection (ICRP) dose conversion factors [47]. The current revision was MicroShield® v12. This software utilized a deterministic (point kernel) method to quickly calculate dosages [48]. It had a

Graphic User Interphase (GUI) to help easily simulate various scenarios and allowed customization of shielding and radiation sources.

APPENDIX B

PERSONAL RADIATION MONITOR VERIFICATION DETAIL

As discussed in "2.3.1 Measurements", to verify the accuracy of the readings the personal digital dosimeters were checked using a calibration standard after the experiment. Using a panoramic irradiator and a known source (Calibration chart shown in Figure 6.1), the digital personal dosimeters were tested near the same exposures as the readings acquired during the dispersion. An example procedure is as follows.

The Exercise Participant's UltraRadiac™-Plus alarm dosimeter had a reading of "0.350 mR" during the 22 May 2019 dispersion. On 24 May 2019, the detector was placed on the carousel, 60 cm from a known 4,440 MBq (120 mCi) Cs-137 source. With the corrected dose rate of 1.763 mR/min and exposure time of 0.23 s, the actual dose was 0.405 mR. The detector read 0.440, 0.411, 0.424 mR in three separate readings. Taking the average reading of 0.425 mR, the correction factor (CF) was estimated to be

$$CF = \frac{Actual\ Dose}{Detector\ Reading}$$

$$= \frac{0.405\ mR}{0.425\ mR}$$

$$= 0.95$$

Then, the corrected dose from the dispersion event was found by applying the correction factor to the average reading.

$$Corrected\ Reading = CF * Detector\ Reading$$

$$= 0.95 * 0.350\ mR$$

$$= 0.33\ mR$$

Finally, the Exercise Participant had two digital dosimeters. The corrected reading from the more accurate meter (CF closer to 1) was used, i.e., readings from the UltraRadiac™-Plus was reported rather than his Ludlum M25-1, which had a CF of 0.72.

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Calibration Chart

ALL DATA IN mR/min.
Calibration Date 1/02/14

Distance in Cm.	Unattenuated	X-9.20 Attenuator
10	73.8	8.02
20	19.2	2.09
30	7.75	0.842
40	4.35	0.473
50	2.8	0.304
60	1.95	0.212

CUSTOMER: Texas A & M
SOURCE: 120 mCi.Cs-137
MODEL 142-10, S.N. 6085

APPROVED BY

DATE

Figure 6.1 Calibration Chart of Known Source Used for Checking Digital Dosimeters.

APPENDIX C

DOSIMETER RESULTS

Below is a summary of dosimetry results and reports from the analytical lab. Note, the reported dosages were in mrem. The following personnel were not exposed to the contamination and received negligible doses: Manager, Veterinarian, Counter, and Administrator. Urine bioassays were obtained for the Radiation Worker and Exercise Participant, and results were comparable to pre-exercise bioassays.

Personnel	OSL Badges	OSL Results (Deep Dose, µSv)	TLD Rings	TLD Rings (Shallow, µSv)	Digital Alarm (µSv)	Bioassay	Results
Receiver	1 whole body (Spare 1, VV)	**10**	1 hand (2499 257SV)	**370**		None	
Radiation Worker	1 whole body (M1, WB, Spare 2)	**10**	1 hand (2499 247SX)	**< 100**	**4.3**	Urine	Negligible
	1 trunk (M2, T, waist, Spare 3)	**20**	1 hand(2499 248SW)	**< 100**			
	1 leg (M3, L, knee, Spare 4)	**10**					
Exercise Participant	1 whole body (C1, WB, Spare 5)	**10**	1 hand (2499 253SZ)	**< 100**	**3.3**	Urine	**Negligible**
	1 leg (C3, L, knee, Spare 7)	**< 10**	1 hand (2499 256SW)	**< 100**			
Manager	1 whole body (C2, T, Spare 6)	**< 10**	None			None	
Veterinarian	1 whole body (M4, WB, Spare 8)	**< 10**	L hand (2499 254SY)	**No results**		None	
			R hand (2499 251SI)	**No results**			
Counters	1 whole body (J1, WB, Spare 9)	**< 10**	1 hand (2499 245SZ)	**< 100**		None	
Technician	1 whole body (T1, WB, Spare 13)	**10**	1 hand (2499 250S2)	**< 100**			
Administrator	1 whole body (G1, WB, Spare 14)	**< 10**	1 hand (2499 249SV)	**< 100**			
			1 hand (2499 255SX)	**< 100**			

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Radiation Dosimetry Report

Account: 716483 Subaccount: 1450588 Series: A92

**No NVLAP accreditation is available from NVLAP for thermal neutron or X type dosimeters. When exposure results are reported for thermal neutrons or X type dosimeters, this report contains data that are not covered by the NVLAP accreditation.

Participant Number	Name / ID Number / Birth Date	Dosimeter	Use	Rad. Type	Rad. Quality	Period Shown Below DDE	Period Shown Below LDE	Period Shown Below SDE	Quarter to Date DDE	Quarter to Date LDE	Quarter to Date SDE	Year to Date DDE	Year to Date LDE	Year to Date SDE	Lifetime to Date DDE	Lifetime to Date LDE	Lifetime to Date SDE	Inception Date	Serial Number
For Monitoring Period:						2018-04-01 to 2018-06-30			QUARTER 2			2018			LIFETIME				
	Historical Customer Avg Control Dose	Pa				11	11	11											
03511	SPARE 1	Pa	CHEST	*P		4	4	4	4	4	4	4	4	4	4	4	4	2018/04	7177426H
03512	SPARE 2	Pa	CHEST	*P		4	4	3	4	4	3	4	4	3	4	4	3	2018/04	7177427H
03513	SPARE 3 Sprayer	Pa	CHEST	*P		5	5	4	5	5	4	5	5	4	5	5	4	2018/04	7177428H
03514	SPARE 4	Pa	CHEST	*P		4	4	4	4	4	4	4	4	4	4	4	4	2018/04	7177429H
03515	SPARE 5	Pa	CHEST	*P		4	4	3	4	4	3	4	4	3	4	4	3	2018/04	7177430H
03516	SPARE 6 Handler	Pa	CHEST	*P		3	3	3	3	3	3	3	3	3	3	3	3	2018/04	7177431H
03517	SPARE 7	Pa	CHEST	*P		3	3	3	3	3	3	3	3	3	3	3	3	2018/04	7177432H
03518	SPARE 8	Pa	CHEST	*P		3	3	3	3	3	3	3	3	3	3	3	3	2018/04	7177433H
03519	SPARE 9	Pa	CHEST	*P		3	3	3	3	3	3	3	3	3	3	3	3	2018/04	7177434H
03520	SPARE 10	Pa	CHEST	*P		3	3	3	3	3	3	3	3	3	3	3	3	2018/04	7177435H
03521	SPARE 11	Pa	CHEST	*P		3	4	4	3	4	4	3	4	4	3	4	4	2018/04	7177436H
03522	SPARE 12	Pa	CHEST	*P		4	4	4	4	4	4	4	4	4	4	4	4	2018/04	7177437H
03523	SPARE 13	Pa	CHEST	*P		4	4	4	4	4	4	4	4	4	4	4	4	2018/04	7177438H
03524	SPARE 14	Pa	CHEST	*P		3	3	3	3	3	3	3	3	3	3	3	3	2018/04	7177439H
03525	SPARE 15	Pa	CHEST	*P		6	6	5	6	6	5	6	6	5	6	6	5	2018/04	7177440H

Dose Equivalent (mrem) for Periods Shown Below. DDE-Deep Dose Equivalent LDE-Lens Dose Equivalent SDE-Shallow Dose Equivalent

* - Customer average background dose used for control subtraction

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Radiation Dosimetry Report

Account: 1311 Subaccount: 1435548 Series: S24

**No NVLAP accreditation is available from NVLAP for thermal neutron or X type dosimeters. When exposure results are reported for thermal neutrons or X type dosimeters, this report contains data that are not covered by the NVLAP accreditation.

Participant Number	Name		Dosimeter	Use	Rad. Type	Rad. Quality	Dose Equivalent (mrem) for Periods Shown Below DDE-Deep Dose Equivalent LDE-Lens Dose Equivalent SDE-Shallow Dose Equivalent												Inception Date	Serial Number
							Period Shown Below			Quarter to Date			Year to Date			Lifetime to Date				
	ID Number	Birth Date					DDE	LDE	SDE	DDE	LDE	SDE	DDE	LDE	SDE	DDE	LDE	SDE		
For Monitoring Period:							2018-04-01 to 2018-06-30			QUARTER 2			2018			LIFETIME				
	Historical Customer Avg Control Dose		S						15											
00175			S	RFINGR	*				M			M			M			M	2015/01	2499245S
00177			S	RFINGR	*				M			M			M			M	2015/01	2499247S
00178			S	RFINGR	*				M			M			M			M	2015/01	2499248S
00180			S	RFINGR	*				M			M			M			M	2015/01	2499249S
00181			S	RFINGR	*				M			M			M			M	2015/01	2499250S
00193			S	RFINGR	*				M			M			M			M	2015/01	2499253S
00195			S	RFINGR	*				M			M			M			M	2015/01	2499255S
00196			S	RFINGR	*				M			M			M			M	2015/01	2499256S
00197			S	RFINGR	*				37			37			37			37	2015/01	2499257S

* - Customer average background dose used for control subtraction

Radiation Dosimetry Report

Annual Radiation Exposure Limits (mrem) :

Whole body, blood forming organs, gonads	5,000
Lens of Eye	15,000
Extremities and Skin	50,000
Fetal (Gestation period)	500
General Public	100

Based on the US NRC Regulations, Title 10, Part 20, Code of Federal Regulations and adopted by many states. Certain state and other regulatory agencies may adhere to different limits.

Control Dosimeter: A control dosimeter is included with each shipment of dosimeters for monitoring radiation exposure received during transit. At the customer's facility, store the control in a radiation free area during the wear period.

Minimal Dose Equivalent Reported: Dose equivalents below the minimum measurable quantity for the current monitoring period are recorded as "M." The minimal reporting levels vary by the dosimeter type and radiation quality. "SL" is an elective option for the minimal dose equivalent reported where exposures less than 10 mrem report as "SL" (excludes fetal dosimeters), and/or exposures at or more than 10 mrem begin reporting at 10 mrem and report in increments of 10 mrem.

Dosimeter Type	M (DDE,LDE,SDE)	M (SDE Only)	SL
Luxel+©	1	-	10
InLight©	5	-	10
Whole Body Beta	-	10	10
U Ring	-	30	-
Neutrak© Neutron Fast	20	-	-
Neutrak© Neutron Thermal/Fast	10	-	-
Saturn Ring	-	10	10

Special Calculations: Special dose calculations can be applied to radiation workers who wear lead aprons.
EDE 1 - two dosimeters: one worn at the waist level under lead apron and one worn at the collar level outside lead apron. 1.5 (Waist DDE) + 0.04 (Collar DDE) = Assigned Deep Dose Equivalent.
EDE 2 - one dosimeter: one worn at the collar level outside lead apron. 0.3 (Collar DDE) = Assigned Deep Dose Equivalent.
EDE 122 - one dosimeter: one worn at the collar level outside lead apron. Collar DDE / 5.6 = Assigned Deep Dose Equivalent.
Calc3 - Lens of Eye dosimeter. 0.5 (Lens of Eye LDE) = Assigned Lens of Eye Dose Equivalent.
Lens.175 - Lens of Eye dosimeter. 0.175 (Lens of Eye LDE) = Assigned Lens of Eye Dose Equivalent.
EDE1-NTC EDE1 without Thyroid Collar assigned deep dose equivalent =
0.06 × (collar dose - waist dose) + waist dose
EDE1-TC EDE1 with Thyroid Collar assigned deep dose equivalent =
0.02 × (collar dose - waist dose) + waist dose
The "ASSIGNED" line follows all of the original whole body dosimeter doses with the EDE 1 or EDE 2 calculation results or Landauer's standard Dose Assessment Protocol (deep and shallow whole body dose from the highest reading whole body dosimeter, lens dose from dosimeter closest to the eye).

Ring Dosimeter Reading: Ring dosimeter readings report as a shallow dose.

Fetal Dosimeter: A declared pregnant worker will possess a fetal exposure on an extra page of the report based upon the whole body dosimeter worn closest to the fetus. The fetal dose is reported for the current wear period, plus the estimated dose from conception to declaration (if provided by customer), and the total dose from declaration to present.

Use	Description	Use	Description
AREA	Area Monitor	OEXTRM	Other Extremity
CHEST	Chest	OWHBDY	Other Whole Body
CNTRL	Control	RANKLE	Right Ankle
COLLAR	Collar	RFINGR	Right Hand Ring
EYE	Eye	RUARM	Right Upper Arm
FETAL	Fetal	RULEG	Right Upper Leg
LANKLE	Left Ankle	RWRIST	Right Wrist
LFINGR	Left Hand Ring	SPCPUR	Special Purpose
LUARM	Left Upper Arm	UPBACK	Upper Back
LULEG	Left Upper Leg	WAIST	Waist
LWBACK	Lower Back	WHBODY	Whole Body
LWRIST	Left Wrist		

Code	Radiation Quality Description (Type and/or Energy)
B	beta
BH	beta high energy, e.g. Strontium, Phosphorus
BL	beta low energy e.g. Thallium, Krypton
BS	Strontium beta
BT	Thallium beta
BU	Uranium beta
BN	beta, neutron mixture
NF	neutron fast
NT	neutron thermal
P	photon (x or gamma ray)
PB	photon, beta mixture
PBN	photon, beta, neutron mixture
PH	photon high energy greater than 200 keV
PL	photon low energy less than 40 keV
PM	photon medium energy 40 keV to 200 keV
PN	photon, neutron mixture

First Line Explanation
Participant Number: Unique number assigned by Landauer.
Name: Participant to whom the dosimeter is assigned.
Dosimeter: Badge type according to radiation monitoring needs.

Dosimeter	Code	Type of Radiation Monitored				
		Photons			Neutrons	
		X	Gamma	Beta	Fast	Fast/ Thermal
InLight Basic	B1P	Yes	Yes	Yes		
InLight Basic	B4P	Yes	Yes	Yes		
InLight Basic	B2C	Yes	Yes	Yes		
InLight Basic	B4C	Yes	Yes	Yes		
InLight LDR	L2P	Yes	Yes	Yes		
InLight LDR	L2J	Yes	Yes	Yes		
InLight LDR	L2T	Yes	Yes	Yes		Yes
InLight LDR	L2D	Yes	Yes	Yes	Yes	Yes
InLight LDR	L4P	Yes	Yes	Yes		
Luxel+	Pa	Yes	Yes	Yes		
Luxel+	Ja	Yes	Yes	Yes	Yes	
Luxel+	Ta	Yes	Yes	Yes		Yes
Luxel+ Escort	Pa	Yes	Yes			
Neutrak	N				Yes	
Neutrak	E					Yes
Ring, Single TLD	U or S	Yes	Yes	Yes		

Deep, Eye and Shallow Dose Equivalents: Deep dose equivalent (DDE) applies to external whole body exposure at a tissue depth of 1 cm (1000 mg/cm^2).
Eye dose equivalent (LDE) applies to external exposure of the lens at a tissue depth of 0.3 cm (300 mg/cm^2).
Shallow dose equivalent (SDE) applies to the external exposure of the skin or extremity at a tissue depth of 0.007 cm (7 mg/cm^2) averaged over an area 1 cm^2.

Deep, eye and shallow dose equivalents report for the time frame indicated by "For Monitoring Period." These doses represent the dose received only for the account/subaccount specified. Individual radiation component results and combined totals report in separate lines.

Quarterly accumulated results reflect total dose received within a calendar 3-months time frame and the customer defined start day. (Note: Quarterly accumulated columns are eliminated for bimonthly service or display "Not applicable.") Year to date accumulation totals dose received from the beginning of the current year to report date. Lifetime accumulation totals all dose received from inception date of dosimeter service to report date, and could include earlier dose history if supplied by customer. Reported quarterly, annual and lifetime dose accumulations represent the doses totaling from all account/subaccount dosimeters to be reported at the customer level.

Inception Date: The date Landauer began keeping dosimeter records for a given dosimeter for a badging participant on the current customer.

Serial Number: Dosimeter serial number.

Second Line Explanation
Participant's personal information consisting of ID number and birth date. This information can be suppressed on "Duplicate and Original Reports" for privacy and/or posting needs.

Notes: Text messages explaining any abnormalities or comments. The notes with message appears on a separate line below all dosimeter exposure information.

U.S. Patents
6,316,702; 6,127,685; 5,892,234

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APPENDIX D

RADIATION UNITS

Radiation Unit Conversion

For the investigation, the conversions from British to International System of Units (SI) were the following

$$1\, rad = 0.01\, Gy$$

$$1\, rem = 0.01\, Sv$$

And, the following were assumed for calculating absorbed dose in soft tissues from photon exposure (strictly speaking, the relationship should be closer to 0.95 [49]).

$$1\, roentgen \cong 1\, rad$$

Dose Equivalence Calculations

For dose equivalence, the Quality factor (Q) or Radiation Weighting Factor (w_r) for photon was one, per International Commission on Radiological Protection (ICRP) 30 and ICRP 60 [50] [51] [49, p. 363]. Therefore,

$$H = QD$$

$$1\, rem \cong 1\, rad$$

$$1\, Sv \cong 1\, Gy$$

Where

$H = Dose\ equivalent\ [\,rem\ or\ Sv]$
$Q = Quality\ Factor$
$D = Absorbed\ Dose\ [rad\ or\ Gy]$

APPENDIX E

FLUORINE-18 NUCLEAR DATA AND DETECTABILITY

Fluorine-18 is a radioisotope which decays to a stable oxygen-18 daughter with a half-life of about 110 min, and primarily by β^+ (positron) emission (Branching ratio ~ 0.97) with some probability of electron capture (Branching ratio ~ 0.03). Each positron would annihilate to two 511 keV photons. The simplified decay scheme and Evaluated Nuclear Structure Data Files (ENSDF) Decay Data in the MIRD Format are shown in Figure 6.2 and Figure 6.3, respectively. For the purpose of the current project, F-18 was assumed to decay purely by positron and instantaneously annihilate to two photons. Unless otherwise stated, analysis was based on that the source emitted two 511 keV photons per decay, with the overall angular distribution being isotropic.

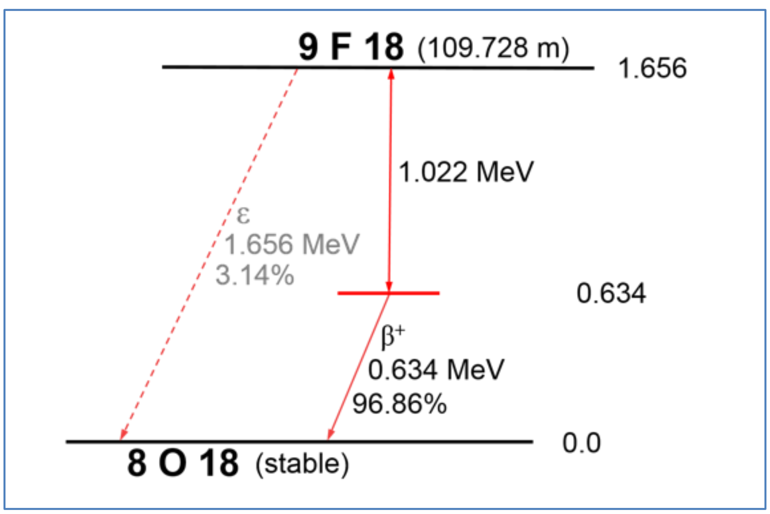

Figure 6.2 F-18 Decay Scheme of F-18 [52].

Radiations	y(i) $(\text{Bq-s})^{-1}$	E(i) (MeV)	y(i)×E(i)
β+ 1	9.67×10^{-01}	2.498×10^{-01}*	2.42×10^{-01}
$\gamma^{\pm}$	1.93	5.110×10^{-01}	9.89×10^{-01}
K X-ray	1.80×10^{-04}	5.249×10^{-04}*	9.42×10^{-08}
K X-ray	4.74×10^{-12}	5.000×10^{-04}*	2.37×10^{-15}
Auger-K	3.07×10^{-02}	5.200×10^{-04}*	1.60×10^{-05}
Listed X, γ, and $\gamma^{\pm}$ Radiations			9.89×10^{-01}
Listed β, ce, and Auger Radiations			2.42×10^{-01}
Listed Radiations			1.23

* Average Energy (MeV).

Figure 6.3 ENSDF Decay Data in the MIRD for F-18 [53].

Fluorine-18 shares nuclear characteristics with cesium-137, a radioisotope listed for its potential use in a RDD attack [26]. This also made F-18 a favorable candidate in radiation training by acting as a surrogate RDD isotope, so to speak. Both F-18 and Cs-137 emit characteristic mid-range energy photons (551 keV and 661.7 keV, respectively), which can be located and identified with Radioisotope Identification Devices (RIIDS), such as the BNC 940 SAM Eagle (Table 6.1). Not all RIIDs will contain F-18 in their nuclide library, however; since it is not listed as a required isotope in the ANSI N42.34-2015 “American National Standard Performance Criteria for Handheld Instruments for the Detection and Identification of Radionuclides” shown in Table 6.2. Decay scheme of Cs-137 is shown in Figure 6.4.

Table 6.1 BNC 940 SAM Eagle Nuclide Library (Trigger List) [54].

F18	In111	Ir192	Np237
K40	I123	Tl201	U238
Co57	I125	Ra226	Pu239
Co60	I131	Th232	Am241
Ga67	Ba133	U233	
Tc99m	Cs137	U235	

Table 6.2 ANSI 42.34 Required Radionuclide Identification Library for RIIDS [55].

^{241}Am	^{137}Cs	^{40}K	^{232}Th	DU
^{133}Ba	^{67}Ga	^{99m}Tc	^{235}U	HEU
^{57}Co	^{131}I	^{201}Tl	^{238}U	WGPu
^{60}Co	^{192}Ir	^{226}Ra	^{239}Pu	RGPu

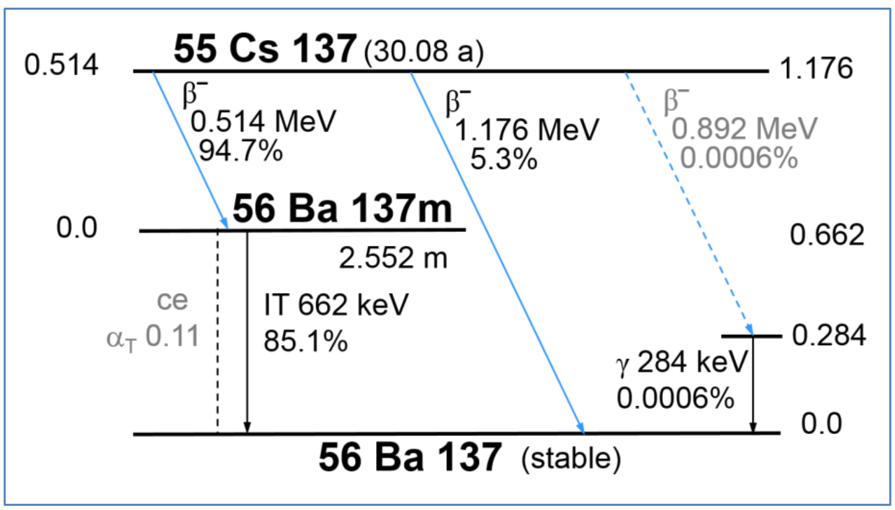

Figure 6.4 Cs-137 Decay Scheme [56].

APPENDIX F

PRE-EXPERIMENTAL PREPARATION

Prior to the experiment, equipment and the area had to be prepared. The intended contamination area was prepped to simulate a debris area (Figure 6.5). The 3 m x 3 m contamination area was pre-taped and objects such as tires, buckets, crates were placed to create a "rubble pile", then a garden duster (Figure 6.6) was utilized to create a layer of corn starch to simulate a dusty environment. A weed sprayer was prefilled with 470 ml (2 cups) of tap water. Decontamination line was also setup (Figure 6.7). A survey team with a mechanically cooled High Purity Geranium (HPGe) detector was stationed > 50 m from the room entrance (Figure 6.8). Filters from the Radiation Worker and Exercise Participant were surveyed with the goal of estimating internal dose from the activity levels. HVAC was set at negative pressure and four air exchanges per hour to limit the release of radiation to the environment.

Figure 6.5 Photo of Area to be Contaminated Shown with Debris.

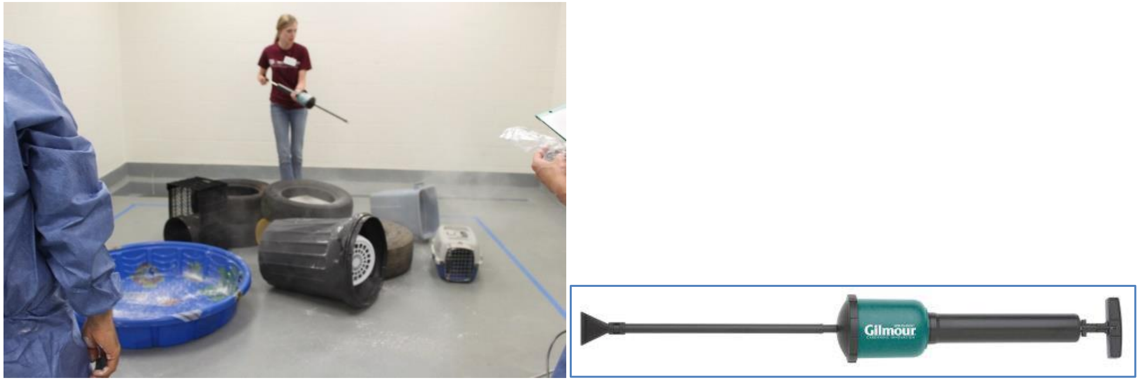

Figure 6.6 Garden Duster Used to Dispense Corn Starch.


Figure 6.7 Decontamination Line during Experiment.

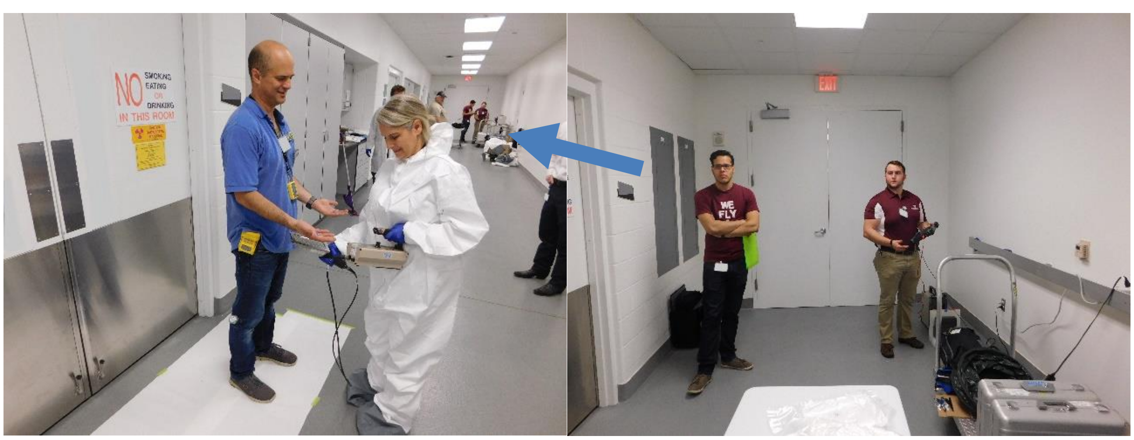

Figure 6.8 Survey Team Located at End of Hall during Experiment.

APPENDIX G

OTHER CONSIDERATIONS (FINITE VERSUS INFINITE PLANE)

One question during the investigation was “When would an infinite plane provide a reasonable estimate of the exposure?” This could be predicted by plotting the fluence ratio between an infinite source versus finite disk sources of various radii, a technique described by Isaksson [57].

Uncollided fluence at a point in a homogeneous attenuating medium from an infinite isotropic plane source can be estimated by the following equation [58, p. 166] [57] and Figure 6.9.

$$\dot{\phi}_p^{\infty} = \frac{S_A}{2} E_1(\mu_a h) \qquad \text{Equation 2}$$

Where

$\dot{\phi}_p^{\infty} = Fluence\ at\ a\ point\ p\ from\ an\ infinite\ plane\ \left[\frac{\gamma}{m^2 \cdot s}\right]$

$S_A = Gamma\ flux\ density\ at\ source\ \left[\frac{\gamma}{m^2 \cdot s}\right]$

$E_1(x) = Exponential\ Integral = \int_1^{\infty} \frac{e^{-xt}}{t} dt$

$\mu_a = Linear\ Attenuation\ in\ Air\ [cm^{-1}]$

$h = Height\ of\ point\ p\ from\ the\ plane$ [cm]

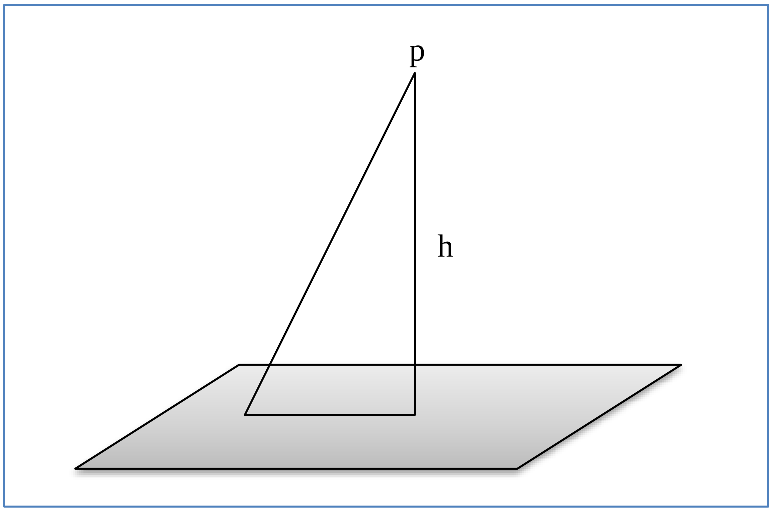

Figure 6.9 Infinite Plane Source Diagram for Theoretical Calculation.

Calculation was done with a height 100 cm, density of air was $\rho_a = 1.205 \times 10^{-3}$ g/cm^3 and μ_a was the "total minus coherent coefficient" μ in Shultis [58]. For 511 keV $\mu_a = 10.38 \times 10^{-5}\ cm^{-1}\ (interpolated)$. In this case, $\mu_a h = 10.38 \times 10^{-3}$ and the Exponential Integral was $E_1(\mu_a h) = 4.0006$ [59]. For the uncollided fluence at point p in a homogeneous attenuating medium from finite isotropic disk source shown in Figure 6.10, the fluence could be estimated with the following equation [58, p. 189] [57].

$$\dot{\phi}_p^N = \frac{S_A}{2}\left[E_1(\mu_a h) - E_1\left(\frac{\mu_a h}{\cos\alpha}\right)\right] \qquad \text{Equation 3}$$

Where

$\dot{\phi}_p^N = Fluence\ at\ a\ point\ p\ from\ a\ disk\ source\ \left[\frac{\gamma}{m^2 \cdot s}\right]$

$S_A = Gamma\ flux\ density\ at\ source\ \left[\frac{\gamma}{m^2 \cdot s}\right]$

$E_1(x) = Exponential\ Integral = \int_1^\infty \frac{e^{-xt}}{t} dt$

$\mu_a = Linear\ Attenuation\ in\ Air\ [cm^{-1}]$

$h = Height\ of\ point\ p\ from\ the\ plane$ [cm]

$\alpha = Angle\ as\ shown\ in$ Figure 6.10 $[radians]$

$r = Radius\ of\ the\ disk\ source\ [cm]$

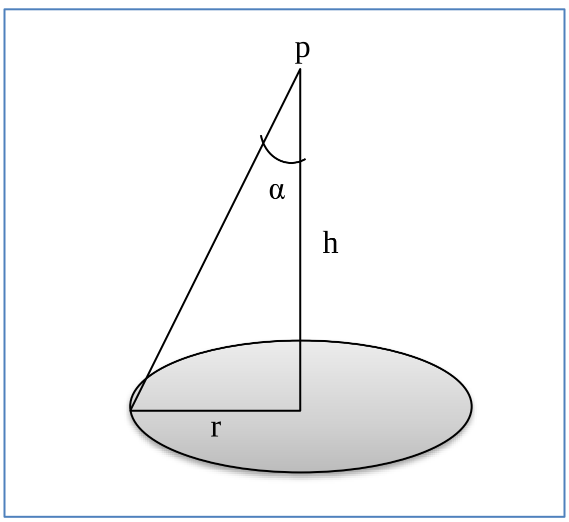

Figure 6.10 Finite Disk Source Diagram for Theoretical Calculation.

Then, for the same flux density S_A, which correlates with the areal density of contaminated surface, the fluence ratio is

$$\frac{\dot{\phi}_p^N}{\dot{\phi}_p^\infty} = 1 - \frac{E_1\left(\frac{\mu_a h}{\cos\alpha}\right)}{E_1(\mu_a h)} \qquad \text{Equation 4}$$

A plot of the variation in fluence ratio with varying angle α is shown in Figure 6.11. The ratio asymptotically approached 1 near $r = 22{,}900\ cm$. This correlated with a disk area of $1.65 \times 10^9\ cm^2$. The dispersion areas of preliminary and actual experimental contamination were $3.09\ \times 10^6$ cm^2 and 9.29×10^4 cm^2, respectively. In fact, for the experimental contamination area, Equation 4 predicted that modeling with an infinite plane would over-estimate the exposure by about 5X. This was consistent with MicroShield®, where the ratio was 5.9, as shown in Table 6.3.

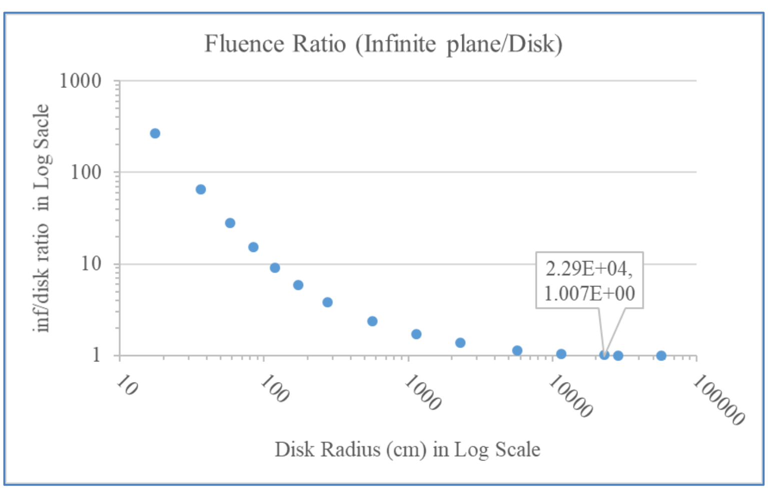

Figure 6.11 Plot of Fluence Ratio versus Radius of Disk Source.

Table 6.3 Comparing Dose Rates by MicroShield® Modeling with Finite versus Infinite Plane (μSv/hr).

Detector Distance	**1 cm**	**30 cm**	**100 cm (Whole Body)**
Finite Plane	106.2	36.7	14.3
Infinite Plane	179.7	108.5	83.5
Ratio	1.7	3.0	**5.9**

Some unexpected results were observed when finite plane dose rates were compared with infinite dose rates using MicroShield® for various planes sizes and constant areal density. Dose rate from finite plane modeling with the disk area of $1.65 \times 10^9\ cm^2$ was expected to approach the infinite plane using the analytical method (Equation 4). The exposure rate of the finite plane using MicroShield® however was 102.2 μSv/hr, greater than for the infinite plane (83.5 μSv/hr). This was surprising, for estimation from a finite plane was not expected to exceed that of the infinite plane, for the same areal density. This discrepancy was due to the differing algorithm MicroShield® uses for finite versus infinite planes. Per MicroShield® manual for infinite plane, the dose rate is solved analytically, but the finite plane uses the point-kernel method [48].

To gain a more better understanding of this discrepancy and when the predictions of finite planes approached that of an infinite plane in MicroShield®, a plot of dose rates versus square-plane areas was generated--for finite and infinite planes as shown in Figure 6.12. For a constant areal density equal to the calculated value from the experiment ($2.158 \times 10^3\ Bq/cm^2$), as the area increased the dose at 100 cm using finite plane model increased toward the prediction using the infinite plane method. At around $1.03 \times 10^8\ cm^2$, the finite plane dose rate matched the infinite plane. However, thereafter, the results were counterintuitive, where the finite plane calculations surpassed the infinite plane prediction, then actually dropped after $1.65 \times 10^9\ cm^2$. For a constant areal density the decrease in dose rate with increased area was an incorrect prediction of reality. One possible reason for these unexpected results was that when the point-kernel method was used for areas $> 1.65 \times 10^9\ cm^2$, the contribution from each kernel may be too low to be stored in the floating-point numbers [60]. This led to the summing of zeros from each kernel.

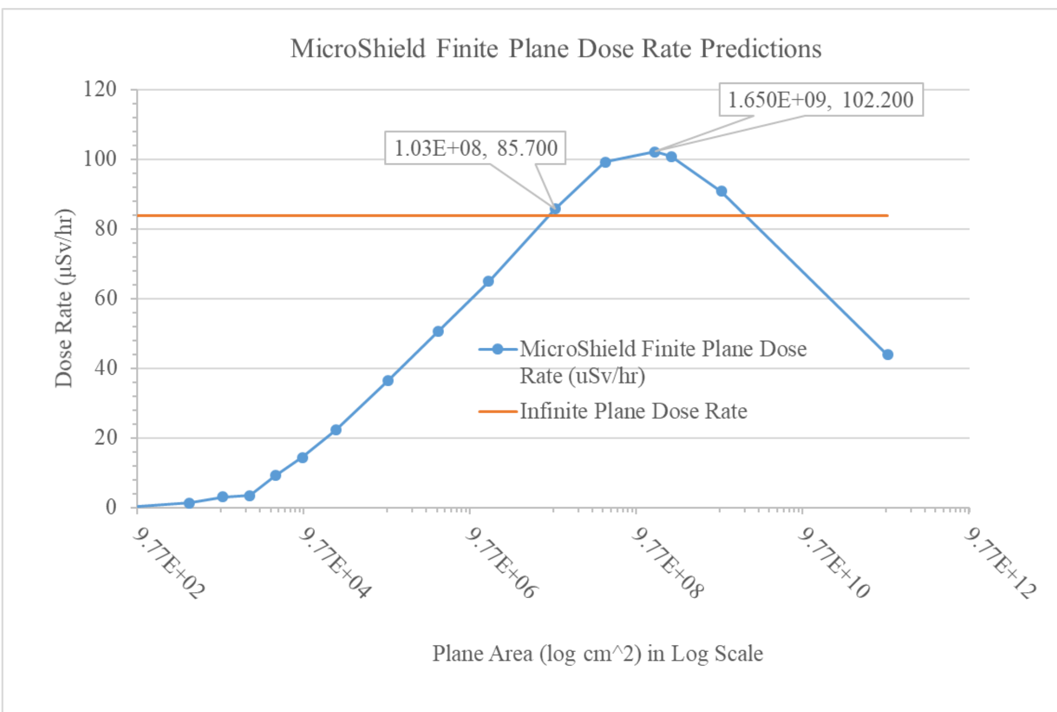

Figure 6.12 Plot of MicroShield® Finite Plane versus Infinite Plane Dose Rates.

In Summary, for the areal density of interest, MicroShield®'s finite plane predictions approached that of its infinite plane calculations when the square plane area was near $1.03 \times 10^8\ cm^2$. This correlated with the length of the square of around 10,000 cm (100 m). That is, for plane sizes less than that area, the infinite plane method might have been overly conservative, with the expected magnitude correlating with the fluence ratio plotted in Figure 6.11, above. Furthermore, for a square plane of area $> 1.65 \times 10^9\ cm^2$, or with the length of > 20,000 cm (200 m), the finite plane method using MicroShield® may be inaccurate. Nonetheless, the validity of these predictions would require more experimental data. For this study, though, MicroShield® prediction appeared to provide accurate representation of the exposure for the 304.8 cm x 304.8 cm plane F-18 was actually dispersed.

APPENDIX H

DOSIMETRY UNCERTAINTIES AND MINIMUM DETETION LIMITS

Figure 6.13, Figure 6.14, and Figure 6.15 were responses to inquiries on uncertainties of the dosimeters and lower detection limit that can be requested on the analysis of OSLs and TLDs. These may be useful for planning for future experiments to facilitate readings at doses lower than the LOD in the current project.

GENERAL QUESTIONS / luxel saturn Request # (49092

2 messages

LDR_Landauer_Customer_Service_Records <CUSTTECH@landauer.com> Wed, Mar 20, 2019 at 12:16 PM

To: "chen0378@tamu.edu" <chen0378@tamu.edu>

Resolution: 03/20/2019	The accuracy of the Luxel + dosimeter is +/- 15% at the 95% confidence interval for photons about 20 keV and beta particles above 200 keV. The precision of the Luxel + dosimeter is +/- 2 mrem The accuracy of the Saturn ring dosimeter is +/- 20%

Figure 6.13 Uncertainties of OSL and TLD per Landauer Customer Service.

On Fri, May 10, 2019 at 8:12 AM <custserv@landauerinc.com> wrote:

Hello,

Thank you for contacting LANDAUER Client Services.

The minimum dose we can detected is 0.001 mRem but we will report anything less than 1 mRem as M for minimal. There are no additional fees pertaining to analysis it is included in the dosimeter pricing. To obtain a quote please contact our new accounts department at 800-300-0735.

Thank you,
LANDAUER Client Services
Maxine

Figure 6.14 Response From Landauer on Requesting Lower Detection Limits on OSL and TLD.

On Wed, May 8, 2019 at 12:34 PM Phillips, Derek L <dphillips@tamu.edu> wrote:

Sir,

I had mentioned to Dr. Marianno that Landauer has produced a report for me in which the measurements were reported to three decimal places for an extra charge. At the time I also inquired about the +- 2mrem precision for the OSL badges, but they did not respond to that question. Are we looking at asking for additional information for the past dosimetry badges or at future measurements?

Derek

Derek Phillips | Associate Health Physicist
Environmental Health and Safety | Texas A&M University
4472 TAMU | College Station, TX 77843-4472

ph: 979.845.5868 | fax: 979.862.7804 | dphillips@tamu.edu

Figure 6.15 Response from Radiation Safety on Requesting Lower Detection Limits on OSL and TLD.

APPENDIX I

MCNP MODELING

Instead of calculating exposure rates with MicroShield®, other modeling software could be utilized. Table 6.4 and Table 6.5 are the results when MCNP was used for the final (refined) dose assessment. The MCNP codes and visedX visualizations are shown below. Notably, the dose prediction ranges between MicroShield® and MCNP using point detectors overlapped, therefore were statically equal. The predictions using point detectors were also in the same order of magnitude as the whole body dosimetry results. On the other hand, MCNP estimates using Phantom with Moving Arms and Legs (PIMAL) were lower than MicroShield®. For the whole body, PIMAL estimates compared with MicroShield® was an order magnitude lower for the Radiation Worker and ~ 50% less for the Exercise Participant. Compared with actual dosimetry, PIMAL estimates was an order magnitude lower for the Radiation Worker and in the same order of magnitude for the Exercise Participant. Therefore, MCNP using point detectors is a reasonable alternative for modeling exposure rates in future assessments, due to its consistency with MicroShield® and dosimetry in this investigation.

Table 6.4 Radiation Worker Results Compared with MicroShield® and MCNP Predictions.

	Whole Body (µSv)	**Extremity (µSv)**
Actual Dosimetry	10 ± 2 (OSL)	< 100 (TLD)
Digital Dosimeter	4.3 ± 0.4	-
MicroShield®	2.8 ± 0.8	21.8 ± 7.5
MCNP Point Detector	2.5 ± 0.7	21.0 ± 7.3
MCNP PIMAL Phantom (Male)	0.5 ± 0.3	N/A

Table 6.5 Exercise Participant Results Compared with MicroShield® and MCNP Predictions.

	Whole Body (µSv)	**Extremity (µSv)**
Actual Dosimetry	10 ± 2 (OSL)	< 10 (OSL)
Digital Dosimeter	3.3 ± 1.0	-
MicroShield®	5.2 ± 0.5	13.4 ± 1.2
MCNP Point Detector	6.0 ± 0.5	15.3 ± 1.4
MCNP PIMAL Phantom (Male)	2.7 ± 0.2	N/A

MCNP with Point Detectors

MCNP Code and VisedX of Radiation Worker Exposure During Spraying (Cylinder Source)

```
Dose Rates above cylinder of F-18
C Cell Cards
10 100  -1          -10     20  -30        imp:p=1        $Source in water
20 300  -0.001205   -10     30  -40        imp:p=1        $Space above source
30 200  -0.944     (10:   -20:  40)
                   (-50     60  -70)       imp:p=1        $HDPE
40   0             (50:   -60:  70) -100   imp:p=1        $Outside source and space above;
where detectors are
50   0              100                    imp:p=0        $Outside world

C Surface Cards
C Source, origin is mid-height
10 CZ       7           $Cylinder, source outer wall, shield inner wall
20 PZ      -1.5         $Bottom of Source
30 PZ       1.5         $Top of Source
C Space above source
40 PZ      31.5         $Top of space
C Shielding
50 CZ       7.5         $Shield outer wall
60 PZ      -2.0         $Bottom of shield
70 PZ      33.5         $Top of shield
C Outside World
100 SO      150

C Data Cards
Mode p
SDEF Cell=10 POS=0 0 0 RAD=D2 AXS=0 0 1 EXT=D3 ERG=0.511
C Radius of sources
SI2     0   7           $From r=0 to max radius
SP2    -21  1           $Uniform distribution
C Height of sources
SI3     1.5             $Extend both ways at this
SP3     -21 0           $Uniform distribution
C Materials
M100    1001    0.666657        $H20; Hydrogen
        8016    0.333343        $Oxygen
M200    1001    0.666662        $HDPE
        6000    0.333338
```

```
M300    6000    0.000150    $Air (Dry, Near Sea Level)
        7000    0.784431    $Density -0.001205
        8000    0.210748
       18000    0.004671
F5:p    0   0   38.5  0          $Point detector closer one
        0   0   88.5  0          $Point detector farther one
FM5     4.01006E+8           $Gamma emissions/sec; 2 gammas per decay
C Convert flux to dose rate
DE5  log 0.01    0.015   0.02    0.03    0.04    0.05    0.06
         0.08    0.1     0.15    0.2     0.3     0.4     0.5
         0.6     0.8     1.0     1.5     2.0     3.0     4.0
         5.0     6.0     8.0     10.0
DF5  log 2.78e-6 1.11e-6 5.88e-7 2.56e-7 1.56e-7 1.20e-7 1.11e-7 $ICRP-21, 2013
MCNP6 manual Table 11-2
         1.20e-7 1.47e-7 2.38e-7 3.45e-7 5.56e-7 7.69e-7 9.09e-7
         1.14e-6 1.47e-6 1.79e-6 2.44e-6 3.03e-6 4.00e-6 4.76e-6
         5.56e-6 6.25e-6 7.69e-6 9.09e-6
fc5     **********Dose to cube in rem/hr*************
NPS 100000
```

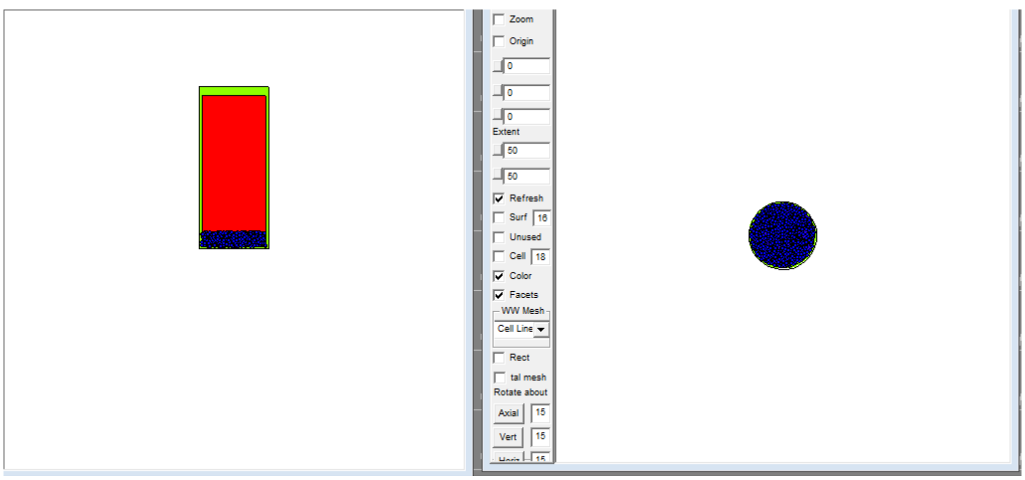

Figure 6.16 VisedX Visualization of MCNP Using Point Detectors for Cylinder Source during Spraying by Radiation Worker.

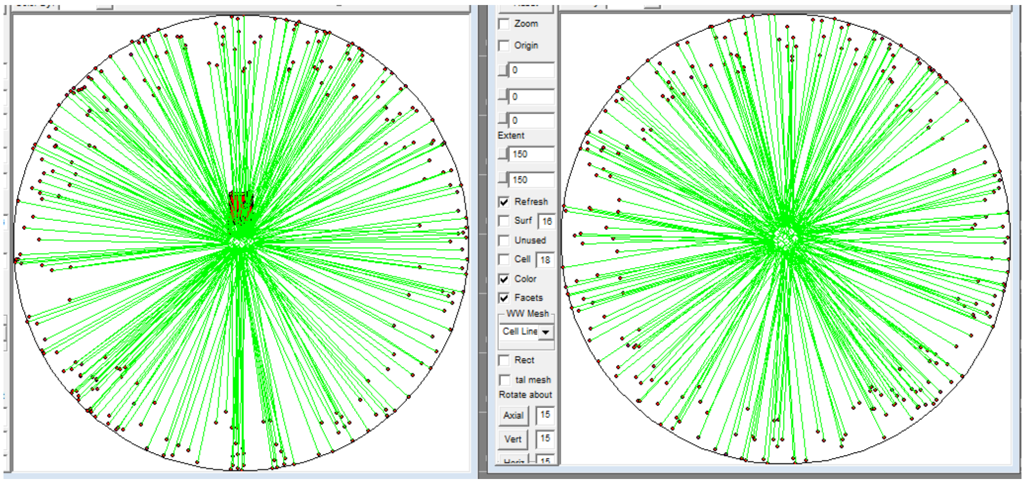

Figure 6.17 VisedX Visualization of MCNP Using Point Detectors for Cylinder Source during Spraying Showing Isotropic Distribution.

MCNP Code and VisedX Exercise Participant Exposure (Plane Source)

```
C Cell Cards
10 1001     -2.3       -10   20  -30        imp:p=1        $Base
20   0               (10: -20:  30) -100    imp:p=1           $Detector region, 1/4 mfp in air is
2401 cm
30   0                100                imp:p=0           $Outside world

C Surface Cards
C Infinite Plane
C ****************   Change with different Plane Size ************
10 CZ      300         $Concrete Base
20 PZ      -25
30 PZ       25         $Plane at origin, where source emitting from
C Outside World
C ****************   Change with different Plane Size ************
100 SO      500

C Data Cards
Mode p
SDEF POS= 0 0 0 X=D1    Y=D2    Z=25.0001   ERG=0.511
C Source just above surface, so VisEd shows tracks
C ****************   Change with different Plane Size ************
SI1 H  -152.4  152.4       $X-spand
SP1 D   0   1
SI2 H  -152.4  152.4       $Y-spand
SP2 D   0   1
C Materials
M1001   1000    0.305330    $Concrete, Ordinary (NIST)
        6000    0.002880    $Density -2.3 g/cc
        8000    0.500407
       11000    0.009212
       12000    0.000725
       13000    0.010298
       14000    0.151042
       19000    0.003578
       20000    0.014924
       26000    0.001605
M1002   6000    0.000150    $Air (Dry, Near Sea Level)
        7000    0.784431    $Density -0.001205 g/cc
        8000    0.210748
       18000    0.004671
M1003   1000    0.630454    $Tissue, Soft (ICRP)
        6000    0.117588    $Density -1.0
```

```
         7000    0.010804
         8000    0.239601
        11000    0.000299
        12000    0.000033
        15000    0.000261
        16000    0.000377
        17000    0.000230
        19000    0.000310
        20000    0.000035
        26000    0.000005
        30000    0.000003
C Detector
F5:p    0   0    26.0  0          $Point detector closest
         0   0    55.0  0         $Point detector mid dist
         0   0   125.0  0         $Point detector farthest
C
C ****************   Change with different Plane Size ************
FM5     4.01E+08          $Gamma emissions/sec; 2 gammas per decay
C Convert flux to dose rate
DE5  log 0.01    0.015   0.02    0.03    0.04    0.05    0.06
        0.08    0.1     0.15    0.2     0.3     0.4     0.5
        0.6     0.8     1.0     1.5     2.0     3.0     4.0
        5.0     6.0     8.0     10.0
DF5  log 2.78e-6 1.11e-6 5.88e-7 2.56e-7 1.56e-7 1.20e-7 1.11e-7 $ICRP-21, 2013
        1.20e-7 1.47e-7 2.38e-7 3.45e-7 5.56e-7 7.69e-7 9.09e-7
        1.14e-6 1.47e-6 1.79e-6 2.44e-6 3.03e-6 4.00e-6 4.76e-6
        5.56e-6 6.25e-6 7.69e-6 9.09e-6
fc5    **********Dose to cube in rem/hr*************
NPS 1E6
```

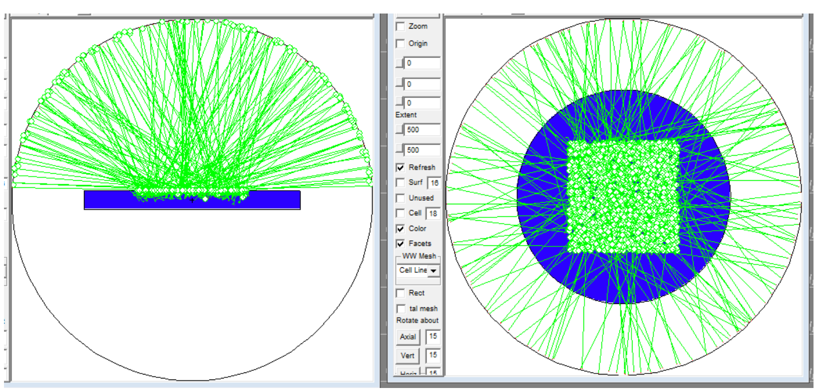

Figure 6.18 VisedX Visualization of MCNP Using Point Detectors for Exposure of Exercise Participant to Isotropic Plane Source.

MCNP with PIMAL

<u>MCNP Code and VisedX of Radiation Worker Exposure during Spraying (Cylinder Source)</u>

(Due to the length of the PIMAL code it was omitted; but is available upon request)

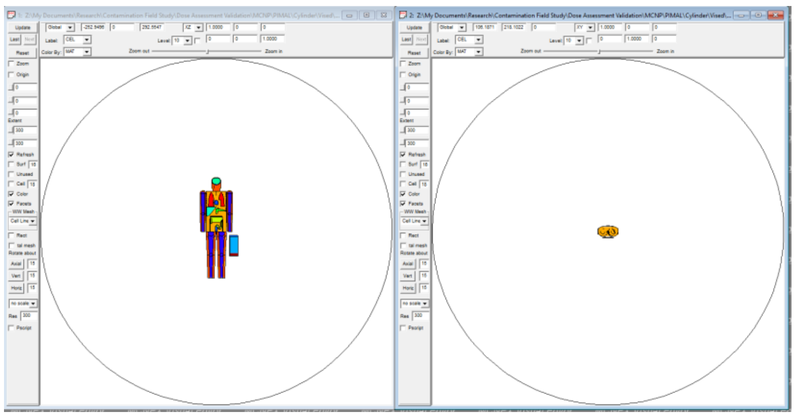

Figure 6.19 VisedX Visualization of MCNP Using PIMAL for Cylinder Source during Spraying by Radiation Worker.

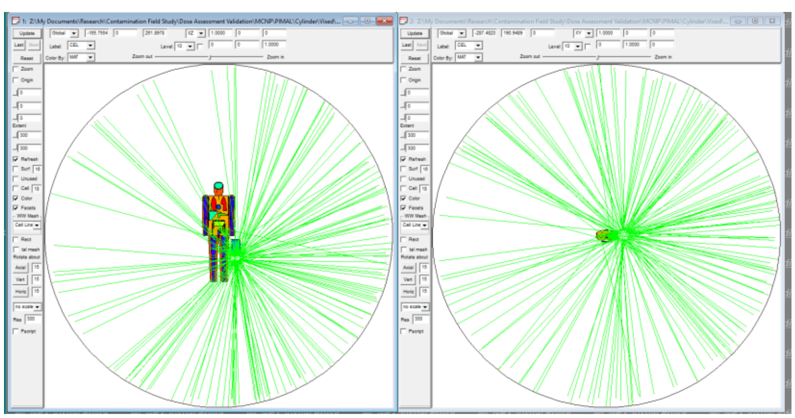

Figure 6.20 VisedX Visualization of MCNP Using PIMAL for Cylinder Source during Spraying Showing Isotropic Distribution.

MCNP Code and VisedX Exercise Participant Exposure (Plane Source)

(Due to the length of the PIMAL code it was omitted; but is available upon request)

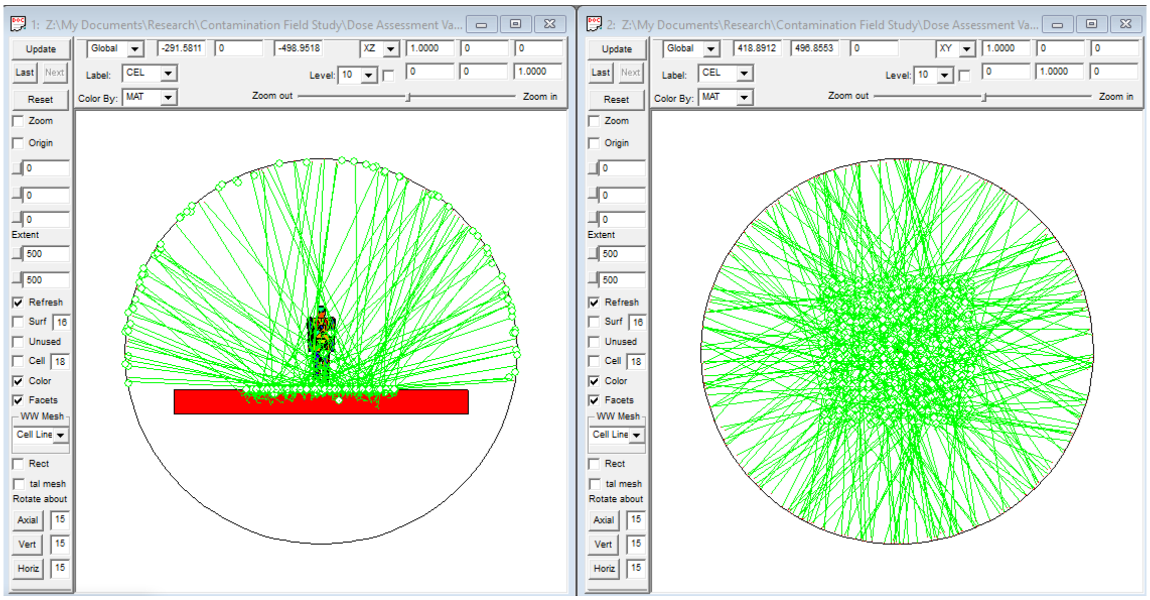

Figure 6.21 VisedX Visualization of MCNP Using PIMAL for Exposure of Exercise Participant to Isotropic Plane Source.