PRELIMINARY DOSE ASSESSMENT FOR EMERGENCY RESPONSE EXERCISE AT DISASTER CITY USING UNSEALED RADIOACTIVE CONTAMINATION

A Thesis

by

LAINY DROMGOOLE COCHRAN

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Chair of Committee,	Craig M. Marianno
Committee Members,	John W. Poston, Sr.
	Robin R. Murphy
Head of Department,	Yassin A. Hassan

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ABSTRACT

The Department of Nuclear Engineering at Texas A&M University currently supports emergency response exercises at Disaster City, a mock community used for emergency response training that features full-scale, collapsible structures designed to simulate various levels of disaster and wreckage. Several times a year, sealed radioactive sources are used at Disaster City to create radiation fields in which emergency responders can become more familiar with dose rates and how to use their radiation detection equipment. This research seeks to enhance emergency response exercises by using unsealed radioactive sources to simulate a more realistic response environment following an incident involving the dispersion of radioactive material.

Limited exercises are performed worldwide using unsealed radioactive sources, and most of that information is not published. This research compiles that information and presents the process for selection of a short-lived radionuclide for use at Disaster City. Historically-used radionuclides were considered, as well as other short-lived radionuclides commonly utilized or capable of being produced at Texas A&M. A preliminary dose assessment for the exercise was performed based on conservative calculation methods used in assessments for unsealed contamination exercises performed at other sites. The assessment was broken into four parts: activation, distribution, exercise participation, and post-exercise monitoring. The computer code MicroShield was used to determine external exposure from the source during and after distribution. Internal exposure via inhalation and ingestion was estimated by assuming fractional

intakes of activity and converting to dose using allowable limits on intake and dose conversion factors.

The selection process identified seven radionuclides that could be used in an unsealed contamination exercise at Disaster City. Pharmaceuticals ^{99m}Tc and ^{18}F are suitable and available for purchase from nearby vendors. In addition, the Texas A&M Nuclear Science Center TRIGA reactor could be used to produce ^{24}Na, ^{56}Mn, ^{64}Cu, ^{82}Br, and ^{140}La via thermal neutron activation. It was determined from the dose assessment that a radionuclide-dependent range of 1-40 mCi can be used to achieve detectable dose rates during the exercise without exceeding assumed administrative dose limits. Tc-99m results in the lowest dose and is recommended from a radiological safety standpoint. However, the choice of which radionuclide and what activity to use for an exercise should be made based on budget and the logistics of the actual exercise.

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NOMENCLATURE

10 CFR 20	Title 10 Part 20 of the Code of Federal Regulations
ALARA	as low as reasonably achievable
ALI	allowable limit on intake
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CVM	College of Veterinary Medicine and Biomedical Sciences
DCF	dose conversion factor
DDE	deep-dose equivalent
DRDC	Defence Research and Development Canada
EHSD	Environmental Health and Safety Department
HMIS	Hazardous Materials Identification System
INL	Idaho National Laboratory
NATO	North Atlantic Treaty Organization
NCRP	National Council on Radiation Protection and Measurements
NNSS	Nevada National Security Site
NSC	Nuclear Science Center
PNNL	Pacific Northwest National Laboratory
PPE	personal protective equipment
PRex	Particle Release Experiment
SDE	shallow-dose equivalent

SRNL	Savannah River National Laboratory
TEDE	total effective dose equivalent
TODE	total organ dose equivalent
TRACER	Testing Radiation and Contamination in Emergency Response

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CHAPTER I

INTRODUCTION

I.A. Motivation

It is important for emergency responders to be able to respond to incidents involving dispersed radioactive material, including nuclear power plant accidents, transportation accidents, and terrorist attacks using radiological or nuclear devices. The National Council on Radiation Protection and Measurements (NCRP) defines emergency responders as "individuals who in the early stages of an incident are responsible for the protection and preservation of life, property, evidence and the environment" (NCRP 2005). Many of these responders include local officials, law enforcement, and other support personnel who do not regularly interact with radiation and radioactive material. Specialized programs and facilities have been created for training emergency responders on how to respond in radiological environments. NCRP Commentary No. 19 states that the overall objectives for training emergency responders in a nuclear or radiological scenario include (1) enhancing their ability to take appropriate measures to protect themselves and the public, and (2) increasing their confidence about effectively managing an emergency involving radiation or radioactive materials (NCRP 2005).

Faculty and students in the Department of Nuclear Engineering at Texas A&M University currently support exercises at Disaster City, a mock community used for emergency response training that features full-scale, collapsible structures designed to simulate various levels of disaster and wreckage (TEEX Disaster City 2005). Several times a year, sealed radioactive sources are used at Disaster City to create radiation fields in which emergency responders can become more familiar with dose rates and how to use their radiation detection equipment. This research seeks to enhance emergency response exercises by using unsealed radioactive sources, which simulate a more realistic response environment following an incident involving the dispersion of radioactive material.

I.B. Literature Review

Limited field exercises are performed worldwide using unsealed radioactive sources (Rothbacher et al. 2015), and most of that information is not published. A literature review yielded information on a variety of exercise types, including dispersion tests, emergency response field exercises, and laboratory-scale exercises. The following sections summarize the available information.

I.B.1. Dispersion Tests

In 2012, Defence Research and Development Canada (DRDC) conducted field trials where explosives were used to disperse 1 Ci of ^{140}La to simulate a radiological dispersal device (Green et al. 2016). The goal of the trials was to obtain measurements that could be used to characterize plumes and deposition patterns. La-140 was selected because of its short 40-h half-life, easily detected gamma and beta emissions, stable daughter (^{140}Ce), and availability via neutron activation of ^{139}La at a nearby research reactor. The chemical form was powdered lanthanum oxide (La_2O_3) which when explosively dispersed would create the desired range of particle sizes.

Following DRDC's example, Pacific Northwest National Laboratory (PNNL) performed the Particle Release Experiment (PRex) at the Nevada National Security Site (NNSS) in 2013 (Keillor et al. 2016). In this experiment, 1 Ci of ^{140}La in La_2O_3 form was released to simulate small-scale venting from an underground nuclear test. The purpose of the test was to obtain ground contamination measurements using different sampling and survey techniques. PNNL also considered using ^{198}Au-coated aluminosilicate microspheres for PRex, but was unable to successfully produce them in time for the exercise. Au-198 was considered because of its 2.69-d half-life and ability to be produced via neutron activation.

Detonation field tests similar to those performed by DRDC and PNNL have occurred in the Czech Republic using ^{99m}Tc in a 0.9% sodium chloride solution (Prouza et al. 2010; Rulik et al. 2013). Tc-99m was selected for its radioactive characteristics. It is a readily available gamma-ray emitter that is commonly used in diagnostic nuclear medicine due to its short 6-h half-life. Ten tests were conducted from 2007 to 2010 for the purposes of informing dispersion models. The activities in these tests ranged from 0.75 to 2.02 MBq. The technetium was diluted in 1.5 L of potassium permanganate aqueous solution in eight of the tests. Twenty dispersion tests using 6 to 8 Ci of ^{99m}Tc were also conducted in Israel from 2010 to 2014 (Sharon et al. 2014).

I.B.2. Field Exercises

NNSS also used ^{99m}Tc for its Testing Radiation and Contamination in Emergency Response (TRACER) program (Gwin 2012). In the TRACER exercises, the technetium was dissolved in water and sprayed on target areas at the T-1 site to create a realistic response environment for the exercise participants. The pre-exercise dose assessment performed by NNSS has been acquired, but was not available publically.

Savannah River National Laboratory (SRNL) conducted field exercises using 10 mCi of ^{18}F, a common radiopharmaceutical used in positron emission tomography in fluorodeoxyglucose (FDG) form (Brown et al. 2010). F-18 decays to stable ^{18}O via positron emission and has a 110-min half-life. The short half-life and stable daughter were characteristics desired by SRNL. Similar to the NNSS exercises using technetium, SRNL diluted the ^{18}F in water for distribution. SRNL considered using ^{99m}Tc but decided against it because it decays to ^{99}Tc, which has a long half-life and could create a soil-to-groundwater contamination issue (Randy Brown, SRNL, personal communication, October 2015). It is speculated that this was less of an issue for NNSS because the T-1 site is already contaminated due to its history as a nuclear weapon test site. Information about the SRNL exercise was not publically available.

Idaho National Laboratory (INL) has conducted radiological response training exercises using KBr, which was irradiated in the INL Neutron Radiography TRIGA reactor (INL 2010). The available environmental assessment document for the exercise does not discuss why KBr was selected for the exercise other than that the activated KBr is short-lived. The environmental assessment was performed under the assumption that a variety of distribution methods would potentially be used, including spreading the radionuclide as a powder or spraying it as a water solution, as well as dispersing it using compressed air and explosives. Details about the INL exercise and dose assessment were not publically available.

Aqueous ^{140}La was used to contaminate a metal drum and a square patch of grassy ground in a North Atlantic Treaty Organization (NATO) exercise in 2003 (Haslip et al. 2004). The goal of the exercise was to validate NATO protocols for radiological sampling and surveying. The details of this exercise have not been located.

I.B.3. Laboratory Exercises

Earlier research regarding mobile radiological laboratory exercises using spiked samples was also found in the review (Martincic 2000; Inn et al. 2006; Lortie et al. 2012). In these exercises, environmental samples were spiked with common long-lived fission products (e.g., ^{60}Co, ^{90}Sr, ^{137}Cs, ^{241}Am). A publication about a nuclear power plant exercise at Browns Ferry in 1985 also described the use of spiked environmental samples, but using ^{131}I and objects contaminated with ^{99m}Tc (McNees 1986). Exercises performed by DRDC in 2005 and 2006 similarly used ^{99m}Tc in a crime scene simulation (Larsson et al. 2006). Seibersdorf Laboratories in Austria provides a decontamination training course that uses unsealed radioactive sources (Stolar 2012). In this course, participants decontaminate dummies, cars, and other materials.

The difference between these laboratory exercises and the previously described dispersion tests and emergency response field exercises is that the radioactivity is contained to the samples or objects in the case of the mobile laboratory and nuclear power plant exercises. Loose surface contamination is desired for the Disaster City exercise.

I.B.4. Other Exercises

The literature review yielded a few more unique results. For example, unsealed contamination has been simulated using fluorescent powder (Heaton 1992). The powder mimics radioactivity in that it is often not visible but detectable using ultraviolet light. The issue with this approach is that this non-radioactive simulant will not properly stimulate an ionizing radiation detector, and a primary objective of the Disaster City exercise is for the participants to become familiar with using their detection equipment in radiological emergencies.

Another unique training course provided by Hotzone-Solutions was found in the review. The Hotzone-Solutions “nuclear emergency training” course is administered in the Chernobyl exclusion zone (Rothbacher et al. 2015). This course uses the radioactivity remaining from the 1986 Chernobyl accident as a source for training on using detection equipment in a contaminated environment.

I.C. Objectives

The overall objective of this research is to investigate the use of short-lived radionuclides that could be used for unsealed contamination emergency response exercises at Disaster City. The research is divided into two phases. The first phase is the determination of radionuclides that are short-lived, can be easily produced at or acquired by Texas A&M, and that generate radiation fields that can be detected by the instruments carried by responders. The second phase is a dose assessment to ensure that doses received by exercise controllers and participants are kept as low as reasonably achievable (ALARA). The results of this assessment will be used to determine which radionuclides and activities are appropriate for the Disaster City exercise.

CHAPTER II

METHODOLOGY

II.A. Radionuclide Selection

In an effort to limit the scope of this research, the distribution method was limited to dissolving the radioactive compound in water and spraying it onto surfaces within the Disaster City complex. This approach is intended to be a controlled way of distributing the unsealed radioactivity.

Several criteria were used to determine whether or not each radionuclide is suitable for an exercise at Disaster City. The selected radionuclide should (i) decay to a stable nuclide, (ii) possess a half-life appropriate for the duration of the exercise, (iii) emit detectable gamma-ray radiation, (iv) be soluble in water, (v) not cause harm to humans and the environment due to its physical or radioactive properties, (vi) be readily available, and (vii) be cost effective. It is best to use short-lived radionuclides for exercises using unsealed radioactivity in order to reduce or even eliminate the need for decontamination. It is also important to use a radionuclide that has a short half-life so that the contaminated area at Disaster City is not inaccessible for an extended period of time.

Radionuclides that have been used in past unsealed contamination exercises were evaluated against the selection criteria. Short-lived radionuclides that are commonly utilized or capable of being produced at Texas A&M were also evaluated. The College of Veterinary Medicine and Biomedical Sciences (CVM) at Texas A&M uses two radiopharmaceuticals for diagnostic procedures – ^{18}F and ^{99m}Tc. Radiopharmaceuticals possess several properties that make them attractive for use in the Disaster City exercise. These radionuclides are not harmful to humans (in properly administered doses) and have short effective half-lives such that they do not remain in the body very long.

The Texas A&M Nuclear Science Center (NSC) operates a 1 MW TRIGA research reactor, located a five-minute drive from Disaster City. The TRIGA reactor can potentially be used to produce short-lived radionuclides for use at Disaster City via thermal neutron activation. If this production method is used, the radionuclide must be in a chemical form such that other elements in the compound are either not activated or if activated, are very short-lived relative to the desired target. Additionally, it is important for the compound to be as chemically pure as possible so that no long-lived impurities are produced as a result of the activation. It is possible that multiple short-lived radionuclides could be activated in a water-soluble compound, but this assessment is limited to compounds in which a single element is radioactive and the rest of the compound is stable.

II.B. Dose Assessment

The Disaster City exercise will take place in the daytime with little-to-no wind and no precipitation. The dose assessment begins at the point that the radioactive source is received at Disaster City. Once on-site, the source will be placed into a container full of water, dissolved, and sprayed onto the desired surface. The exercise participants will not be permitted into the contaminated area until the source has been allowed to settle onto the surface. This limits exposure via inhalation to resuspension of the surface contamination. Access to the area surrounding the contaminated area will be limited to exercise controllers and participants.

The radionuclides that met the selection criteria were included in the dose assessment to determine which radionuclides result in a justifiable dose for the needed dose rates for detection. The dose assessment for the NNSS exercise (Gwin 2012) was used as the basis for the Disaster City dose assessment. The calculations were performed under the conservative assumption that no personal protective equipment (PPE) or shielding will be used.

II.B.1. External Exposure

The radionuclide was treated as a point source until dissolved in water on-site at Disaster City. The method for determining the dose rate, $\dot{X}$, in rem h^{-1} from a point source is defined in Eq. 1

$$\dot{X} = \frac{\Gamma A}{r^2} \tag{1}$$

where Γ is the specific gamma-ray constant for the radionuclide (R m^2 h^{-1} Ci^{-1}), A is the source activity (Ci), and r is the radial distance from the source (m). The dose rates at 1 cm and 30 cm were calculated to determine extremity and whole body exposures, respectively. The quality factor for gamma radiation is 1, so it was assumed that 1 R is equal to 1 rem. The source was assumed to be handled for a maximum of one minute while emptied into the sprayer.

The one-gallon sprayer that will be used to distribute the radioactive solution was assumed to be a cylinder. Therefore, external exposure calculations for the spraying portion of the exercise used a cylindrical volume for the source geometry. The height of the sprayer was assumed to be 20.32 cm (8.0 in) and 7.62 cm (3.0 in). The dose rates at 1 cm and 30 cm above the cylinder were calculated. The duration of the distribution was assumed to take a maximum of 30 min.

The cylindrical source was modeled using MicroShield, which was used in the NNSS assessment. MicroShield is a program that is used to design radiological shields and containers and assess radiation exposure to people and materials (MicroShield 2015). The point kernel method is used to calculate radiation exposure for the majority of the 16 source geometries available in MicroShield (infinite plane and infinite slab excluded). The point kernel method breaks down a distributed source into small surface or volume elements and treats each element as a point source. The source strength divided by $4\pi r^2$ is attenuated using the appropriate attenuation coefficients and buildup factors for each point. The contribution from each point at the location of interest is then summed to obtain a point kernel solution.

A study by NNSS determined that a one-gallon sprayer can cover a 308.8 m^2 area (Gwin 2012). This was the area assumed to be contaminated in the Disaster City assessment. For comparison, the area of Rubble Pile 1 at Disaster City is estimated to be 1083 m^2, so approximately 29% of the pile would be contaminated if used for this exercise. The solution was treated as a surface source once distributed. This surface source was also modeled in MicroShield as an infinite plane source with uniformly distributed activity. Integral solutions are used in MicroShield for the infinite plane geometry instead of the point kernel method. Because of this, buildup is calculated using the Taylor approximation for the infinite geometries. The dose rates at 1 cm, 30 cm, and 100 cm above the contaminated surface were calculated. The exercise participants were assumed to operate in the contaminated area for three hours.

II.B.2. Internal Exposure

Internal exposure via inhalation and ingestion at all stages of the exercise was estimated by assuming fractional intakes of activity and converting to dose. The whole body committed effective dose equivalent (CEDE), H_E, in rem was estimated using Eq. 2

$$H_E = \frac{5 * I}{ALI} \tag{2}$$

where I is the intake in μCi, ALI is the stochastic allowable limit on intake in μCi, as provided in Title 10 Part 20 of the Code of Federal Regulations (10 CFR 20), "Standards for Protection Against Radiation," (U.S. NRC 1992) and 5 is the CEDE in rem from annual intake of 1 ALI.

The committed dose equivalent (CDE), H_T, in rem to the maximum exposed organ was estimated using Eq. 3

$$H_T = I * DCF_T * 3.7x10^6 \tag{3}$$

where I is the intake in μCi, DCF_T is the organ-specific dose conversion factor for the maximum exposed organ in Sv Bq^{-1} as provided in Environmental Protection Agency Federal Guidance Report No. 11 (Eckerman et al. 1988), and $3.7x10^6$ is the conversion factor to convert from Sv Bq^{-1} to rem μCi^{-1}. Following the NNSS approach, the worker that distributes the radionuclide was assumed to inhale and ingest 1% of the activity in the sprayer. A 1 μCi intake was assumed for the exercise participants.

The contaminated area was assumed to be posted until the surface contamination levels decay below 1000 dpm/100 cm^2 ($4.51x10^{-6}$ μCi cm^{-2}), as recommended for beta-gamma emitters in NUREG-1556 Volume 7 Appendix Q (Fuller et al. 1999). The length of time needed to reach this contamination level was determined using the exponential decay equation (Eq. (4))

$$A = A_0 e^{-\lambda t} \tag{4}$$

where A is the desired contamination limit, 1000 dpm/100 cm^2, A_0 is the initial surface contamination level (μCi cm^{-2}), λ is the decay constant for the radionuclide (h^{-1}) and t is the time in hours to reach A. After the area is released, it will be monitored as long as agreed by Disaster City personnel and the Texas A&M Environmental Health and Safety Department (EHSD).

II.B.3. Accidents

Accident scenarios were also analyzed. The CEDE and CDE resulting from the negligent ingestion and inhalation of the entire source was estimated using Eq. 2 and

Eq. 3. The CEDE from a contaminated wound was estimated using Eq. 2, where the intake is treated as an ingestion of 100 cm^2 of the surface contamination level.

The external dose equivalent received by a member of the public that entered the contaminated area was estimated by multiplying the external dose rates obtained using MicroShield by a conservatively-assumed duration of 12 h. The CEDE for a member of the public cannot be estimated using Eq. 2 because the ALI corresponds to the 5 rem occupational limit for radiation workers. Instead, the CEDE for the public, $H_{E,pub}$, is estimated using Eq. 5

$$H_{E,pub} = I * DCF_E * 3.7x10^6 \tag{5}$$

where I is the intake in μCi and DCF_E is the effective dose conversion factor in Sv Bq^{-1} as provided in Environmental Protection Agency Federal Guidance Report No. 11 (Eckerman et al. 1988). An intake of 1 μCi was assumed, following the assumptions used for the exercise participants.

Skin or PPE contamination is possible during the spraying process and during the exercise. Following the NNSS approach, the computer code VARSKIN was used to evaluate skin exposure from a 1 μCi drop of the radioactive solution. VARSKIN can be used to calculate the absorbed dose from beta-particle irradiation via numerical integration of the Berger point kernel. It can also be used to calculate the absorbed dose for gamma radiation using point kernel integration (Hamby et al. 2011).

The assumption to assign a single drop of solution an activity of 1 µCi was based on the activity concentration of the solution in the spraying container. An activity of 50 mCi dissolved in one gallon of water yields an activity concentration of 13.2 µCi mL^{-1}. A drop was assumed to have a volume of 0.05 mL. Therefore, the activity of a single drop would be 0.66 µCi. This was rounded up to 1 µCi for simplification and conservativism.

The drop was assumed to cover an area of 10 cm^2 and remain on the area for a maximum of 15 min. The contamination was modeled in VARKSIN as a disk geometry with a skin-averaging area of 10 cm^2. 10 CFR 20 states that the shallow-dose equivalent (SDE) to the skin must be averaged over the 10 cm^2 of skin that receives the highest exposure. The SDE to the skin at a tissue depth of 0.007 cm (7 mg cm^{-2}) and deep-dose equivalent (DDE) at a depth of 1 cm (1000 mg cm^{-2}) were evaluated for a 1 µCi drop of each radionuclide on bare skin, skin covered by a plastic lab coat, and skin covered by two surgeon gloves. The lab coat was assumed to be 0.2 mm thick and have a density of 0.36 g cm^{-3}. A single surgeon glove was assumed to be 0.05 mm thick and have a density of 0.9 g cm^{-3}. These values were selected based on the suggested values provided in the VARSKIN manual (Hamby et al. 2011).

A spill scenario in which 1 mCi of the radioactive solution was dropped onto a 100 cm^2 area was also analyzed. This area is less than 0.01% of the total contaminated area assumed, 308.8 m^2. The 1 mCi activity was selected following the NNSS assessment approach. Equation 4 was used to determine the time needed for the spilled material to decay below the 1000 dpm/100 cm^2 surface contamination limit.

II.B.4. Administrative Dose Limits

The federal occupational dose limits are defined in 10 CFR 20.1201. The annual total effective dose equivalent (TEDE) limit for occupational workers is 5 rem. The TEDE is defined as the sum of the DDE due to whole body external exposure and the CEDE. The annual total organ dose equivalent (TODE) limit for occupational workers is 50 rem. The TODE is defined as the sum of the DDE and the CDE to the maximum-exposed individual organ or tissue.

The doses estimated for the worker that dissolves and distributes the source were compared with the administrative dose limits set by EHSD. The EHSD administrative dose limits are 10% of the occupational annual limits, or 500 mrem TEDE and 5 rem TODE. The exercise participants could potentially also be held to occupational administrative limits. However in an effort to be even more conservative, the doses estimated for the exercise participants were compared with 10% of the EHSD administrative dose limits, or 50 mrem TEDE and 500 mrem TODE.

Skin exposure from a drop of the radioactive solution on the skin was compared to the EHSD administrative dose limit for the SDE, which is 10% of the 50 rem occupational annual limit, or 5 rem. The estimated dose to a member of the public that enters the contaminated area was compared with the federal annual dose limit for individual members of the public, which is 100 mrem TEDE.

CHAPTER III

RESULTS AND DISCUSSION

III.A. Radionuclide Comparison

The seven radionuclides identified for potential use at Disaster City are located in Table 1. Their radiological properties are included in Appendix A. All of the historically-used radionuclides (^{140}La, ^{99m}Tc, and ^{18}F) were deemed suitable for the Disaster City exercise. Tc-99m is available as sodium pertechnetate ($NaTcO_4$) from a local vendor. F-18 is available as FDG from a vendor in Houston. Both chemical forms are soluble in water. Both radionuclides have short half-lives and emit low-energy gamma-rays. Tc-99m does not meet the requirement of decaying to a stable nuclide, but the decay product, ^{99}Tc, is very long-lived and was kept in the assessment.

Several radionuclides were identified for potential production using the NSC TRIGA reactor. Na-23 and ^{55}Mn are both 100% naturally monoisotopic and can be activated to short-lived ^{24}Na and ^{56}Mn, respectively. Both beta-decay to stable nuclides and emit detectable gamma-rays. Short irradiation times most likely do not allow for successive neutron captures but if they occurred, they would produce heavier sodium and manganese isotopes that have very short half-lives (several seconds to several minutes) and primarily beta-decay to stable nuclides. La-140 can also be produced via neutron activation. Lanthanum exists naturally as nearly 100% ^{139}La. The small percent that is not ^{139}La is ^{138}La, which if irradiated with neutrons would become the desired ^{139}La.

Table 1. Comparison of the radionuclides deemed suitable for use in an unsealed contamination exercise at Disaster City. The asterisk indicates a radionuclide that is not 100% naturally abundant.

Criteria	Tc-99m	F-18	La-140	Na-24	Mn-56	Cu-64*	Br-82*
Decay mode	IT, β-	EC+β+	β-	β-	β-	EC+β+, β-	β-
Stable daughter	Tc-99 (not stable)	O-18	Ce-140	Mg-24	Fe-56	Ni-64 and Zn-64	Kr-82
$T_{1/2}$ (h)	6.0	1.8	40	15	2.6	13	35
Compound soluble in water with no safety hazards	$NaTcO_4$	FDG	$La_2(SO_4)_3$	$NaHCO_3$ $NaC_2H_3O_2$ $\text{-}3H_2O$	$MnSO_4$ $\text{-}H_2O$	HMIS health rating ≥ 2	$MgBr_2$
Solubility (g/100 g H_2O)	Soluble	Soluble	2.33 (20°C)	10.3 (25°C) 50.4 (25°C)	63.7 (25°C)	--	102 (25°C)

Other naturally-stable elements that become short-lived radionuclides when activated include ^{63}Cu, which becomes ^{64}Cu (12.7-h half-life) and ^{81}Br, which becomes ^{82}Br (35.3-h half-life). The issue with these nuclides is that they are not 100% naturally abundant. The natural abundances of ^{63}Cu and ^{81}Br are 69.15% and 49.31%, respectively. The other natural isotope of copper is ^{65}Cu (30.85% natural abundance), which can be activated to become ^{66}Cu. Cu-66 has a short 5.10-min half-life but has a neutron capture cross-section of 140 b and can be activated to become ^{67}Cu (2.58-d half-life). The other natural isotope of bromine is ^{79}Br (50.69% natural abundance) which can be activated to become ^{80}Br. Br-80 has a 17.7-min half-life and a metastable state with a 4.42-h half-life. Both are shorter lived than the desired ^{82}Br, and ^{80}Br beta-decays to stable ^{80}Kr (92% branching ratio) or decays by electron capture to stable ^{80}Se. In addition, neutron capture by ^{82}Br produces ^{83}Br, which has a 2.40-h half-life. The more complicated neutron activation and resulting decay chains for producing ^{64}Cu and ^{82}Br makes them less desirable for use in the Disaster City exercise.

Suitable water-soluble compounds were identified for the radionuclides that could be produced via neutron activation. To avoid hazards due to chemical properties, only compounds with the following Hazardous Materials Identification System (HMIS) numeric hazard ratings were considered: health – 1 or 0, flammability – 0, physical hazard – 0, and personal protection – 0. The identified compounds are listed in Table 1. These compounds contain elements that will not be activated (Texas A&M Nuclear Science Center, personal correspondence, February 2016), including hydrogen, carbon, oxygen, sulfur, and magnesium. Appendix B contains the thermal neutron capture cross sections for these elements and the elements intended for activation. All water-soluble compounds of copper that met the criteria for neutron activation had HMIS health hazard ratings of two or greater. Copper was included in the dose assessment in case the chemical hazard restrictions for the actual exercise are less stringent.

III.B. Dose Calculations

A dose assessment tool was developed that allows the user to select from a library of radionuclides to perform dose estimates for an exercise at Disaster City. The library of radionuclide properties (e.g., half-life, radiation emissions and energies) is automatically input. The gamma-ray constants, ALIs, and DCFs are located in Appendix C. Other user inputs include exercise duration, activity, and the size of the contaminated area. The tool performs the previously detailed calculations to determine the estimated doses associated with each part of the exercise, including dose to the worker that dissolves and sprays the source and dose to the exercise participants.

The doses resulting from several activities ranging from 1 mCi to 100 mCi were examined for each of the seven identified radionuclides. These doses were compared to the respective administrative dose limit set for each individual. The worker TEDE is the most restrictive dose of the entire exercise. Fig. 1 displays the worker TEDE as a function of activity up to 100 mCi. Fig. 2 displays the same information as Fig. 1 but with an abscissa limited to 20 mCi because the 500 mrem TEDE administrative dose limit was exceeded for several of the radionuclides above 20 mCi. Fig. 2 shows that activities below 20 mCi are needed for ^{140}La, ^{82}Br, and ^{24}Na. Greater activities can be used for ^{56}Mn, ^{64}Cu, ^{18}F, and ^{99m}Tc. At 100 mCi, the estimated dose is still well below the administrative dose limit for ^{99m}Tc and ^{18}F.

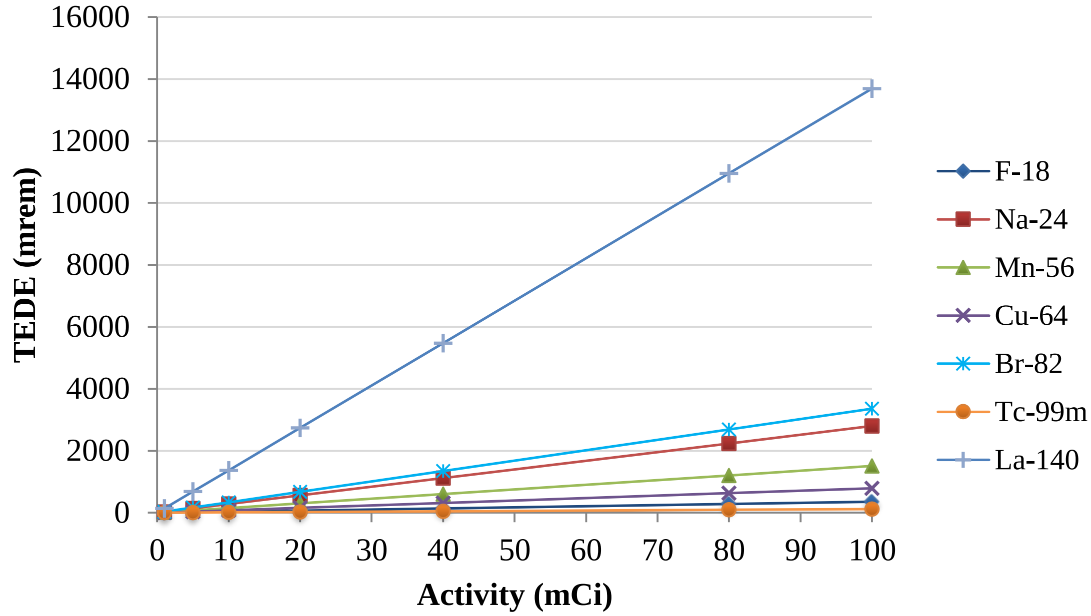

Fig. 1. Worker TEDE as a function of activity up to 100 mCi.

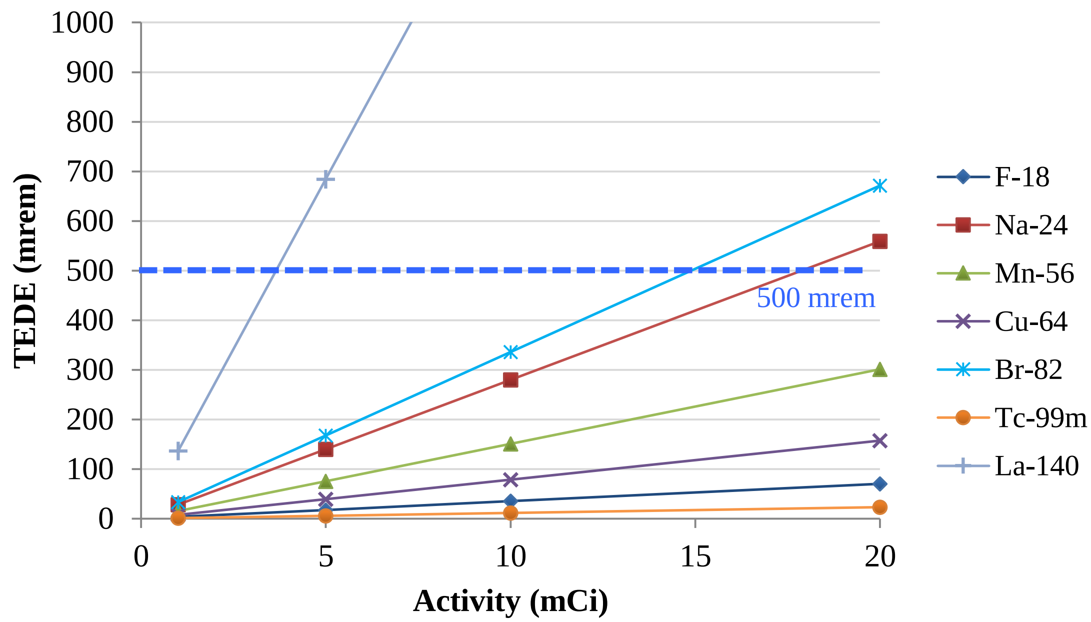

Fig. 2. Worker TEDE as a function of activity up to 20 mCi.

Fig. 3 displays the worker TODE as a function of activity. The 5 rem TODE administrative dose limit for the worker is exceeded for ^{140}La, ^{82}Br, and ^{24}Na at 80 mCi. At 40 mCi, the estimated TODEs are below the administrative dose limit for all radionuclides.

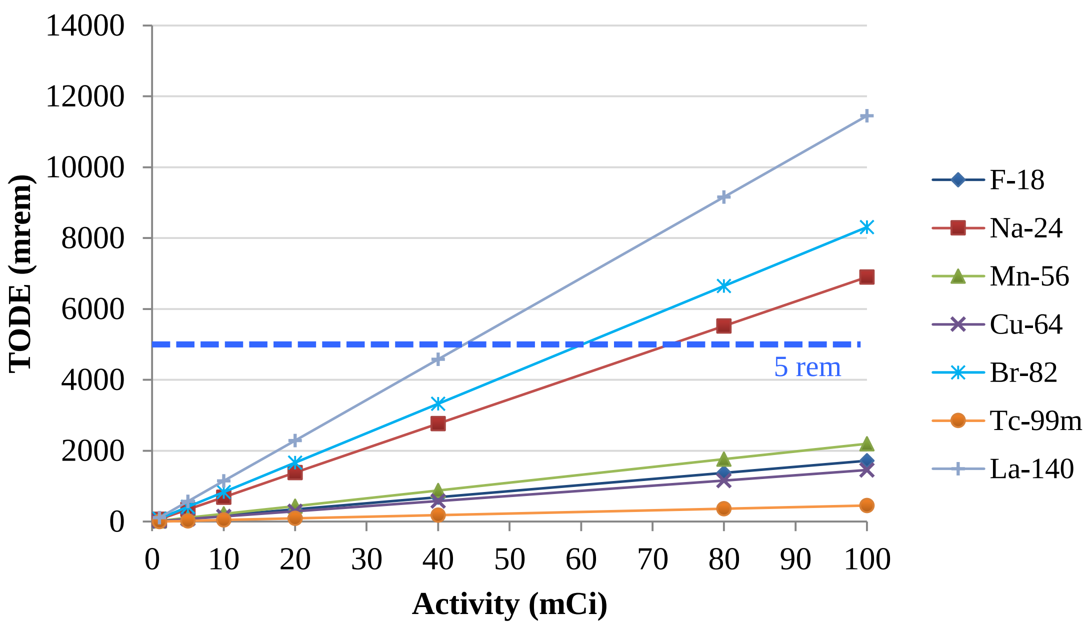

Fig. 3. Worker TODE as a function of activity.

La-140 results in the greatest TEDE and TODE to the worker among the studied radionuclides for any activity. The ranking of radionuclides in terms of dose received does not change as activity increases. Internal exposure (CEDE and CDE) contributes the most to the TEDE (shown in Fig. 4) and TODE (shown in Fig. 5), and external exposure (DDE) contributes less to the total as activity increases for each radionuclide. This is due to the 1% activity intake assumption. For the worker CEDE, ingestion contributes more to dose than inhalation because the effective DCF is larger for ingestion than inhalation. This is the case for all of the studied radionuclides. For the worker CDE, inhalation contributes more to dose than ingestion for most of the radionuclides because the limiting organ DCF for inhalation is larger than ingestion, except for ^{18}F and ^{99m}Tc. Appendix C includes the limiting organs for ingestion and inhalation for each radionuclide.

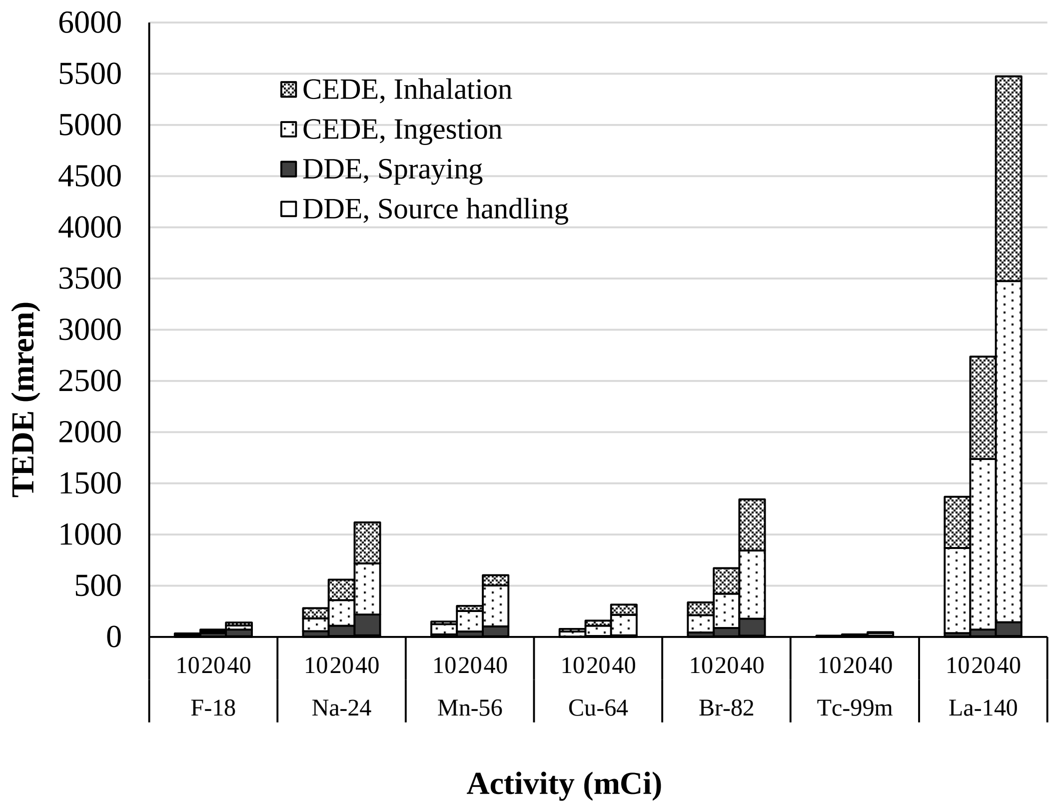

Fig. 4. The external and internal contributions to the worker TEDE are shown for 10, 20, and 40 mCi of each radionuclide.

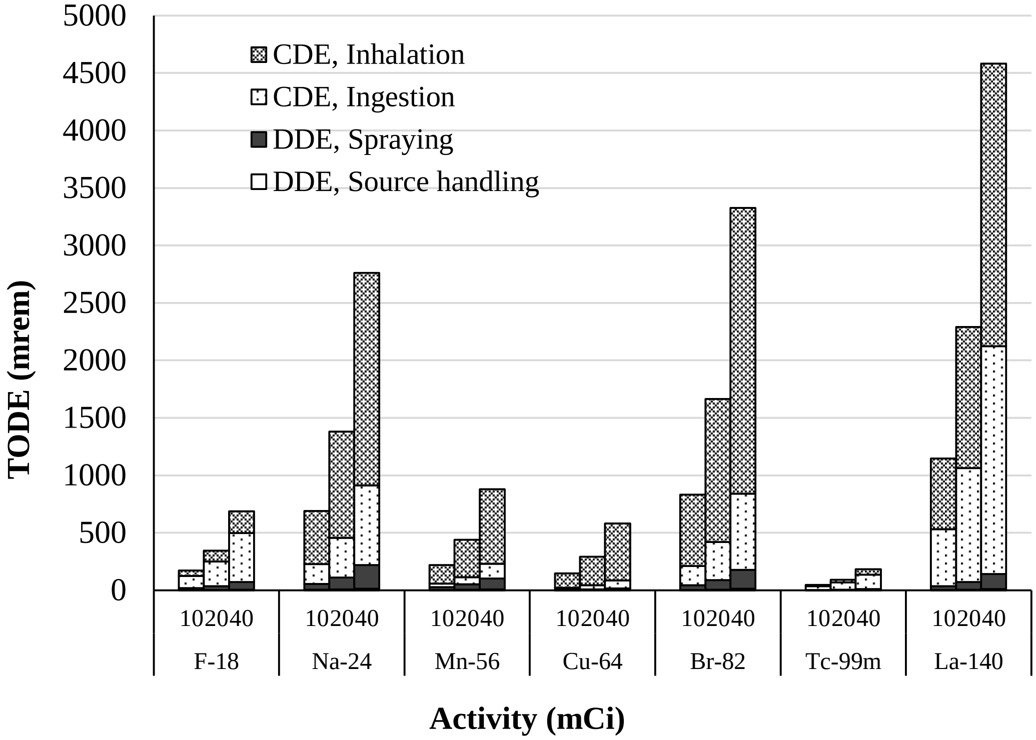

Fig. 5. The external and internal contributions to the worker TODE are shown for 10, 20, and 40 mCi of each radionuclide.

Fig. 6 displays the exercise participant TEDE as a function of activity. The estimated TEDE for an exercise participant is one to three orders of magnitude lower than the TEDE for the worker (depending on the activity). The 50 mrem TEDE administrative dose limit for the exercise participants allows for higher activities than the worker. Fig. 7 displays the exercise participant TODE as a function of activity. For all of the activities studied, the 500 mrem TODE administrative limit for the exercise participant is not exceeded.

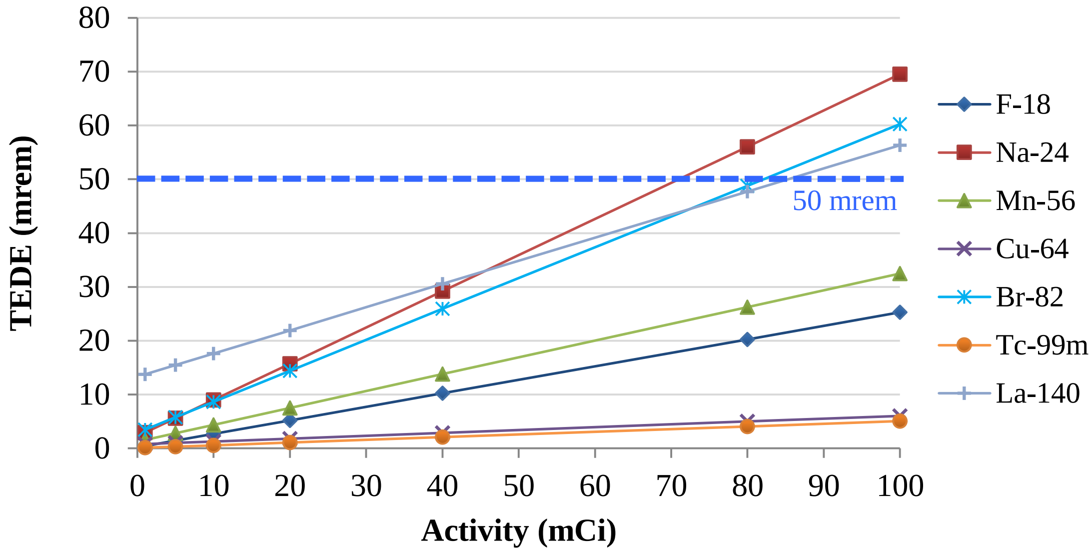

Fig. 6. Exercise participant TEDE as a function of activity.

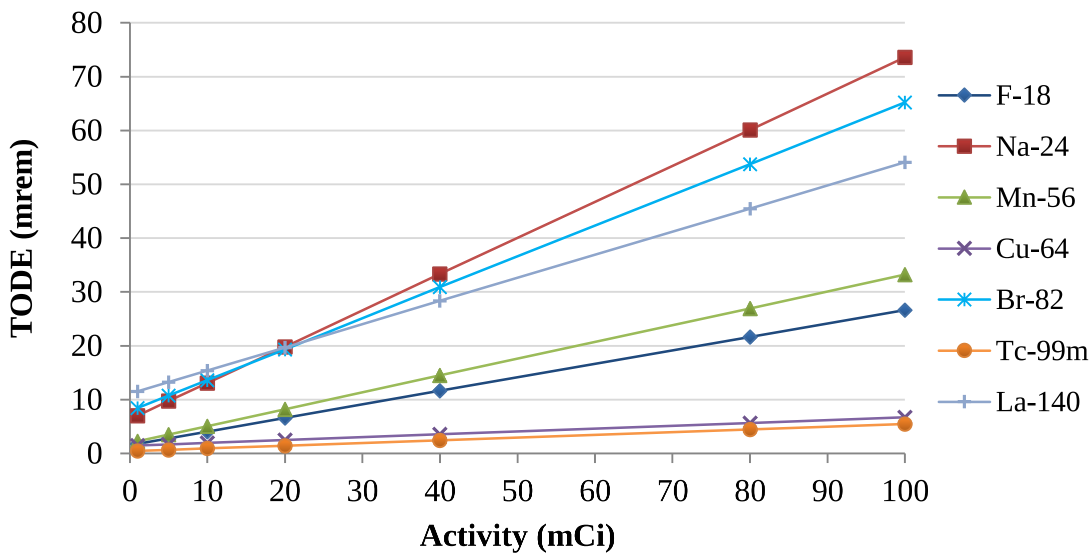

Fig. 7. Exercise participant TODE as a function of activity.

Unlike the worker TEDE and TODEs, the radionuclide that results in the greatest dose to the exercise participant changes depending on activity. Below 20 mCi, ^{140}La results in the greatest dose. As activity increases, ^{24}Na and ^{82}Br overtake ^{140}La and contribute more to total dose. This is because internal exposure remains the same no matter what amount of activity is used due to the flat 1 μCi intake assumption for the exercise participants. Therefore, external exposure contributes more to total dose as activity increases. This is seen in Fig. 8 for the TEDE and Fig. 9 for the TODE.

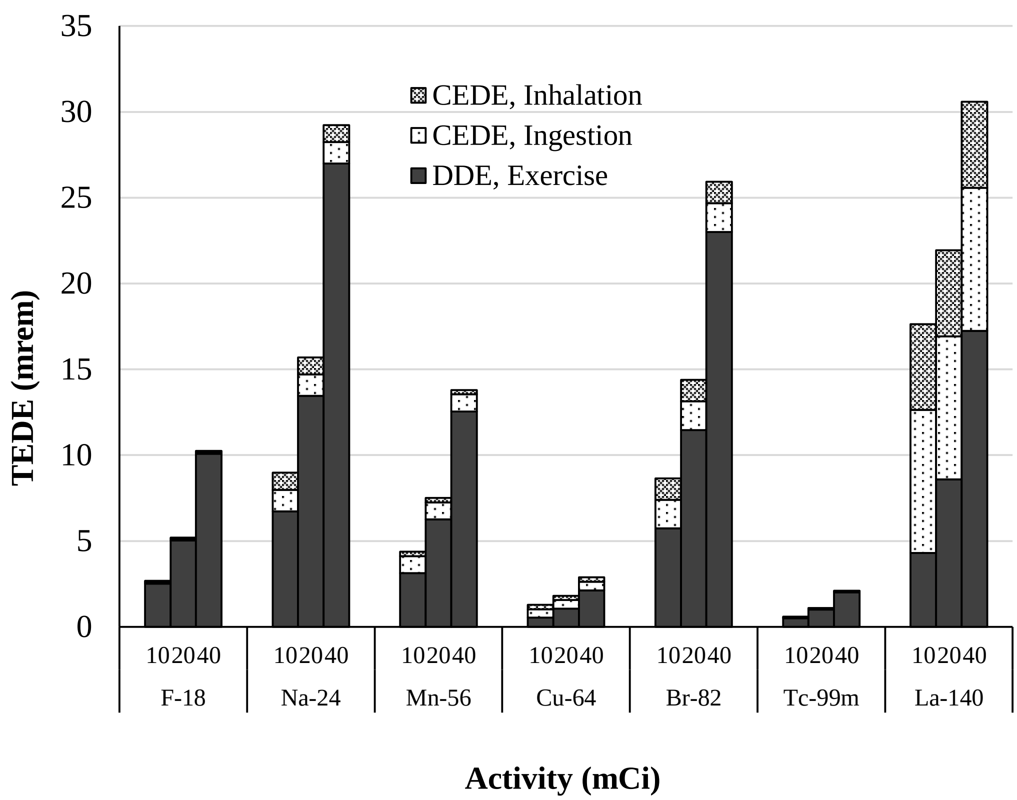

Fig. 8. The external and internal contributions to the exercise participant TEDE are shown for 10, 20, and 40 mCi of each radionuclide.

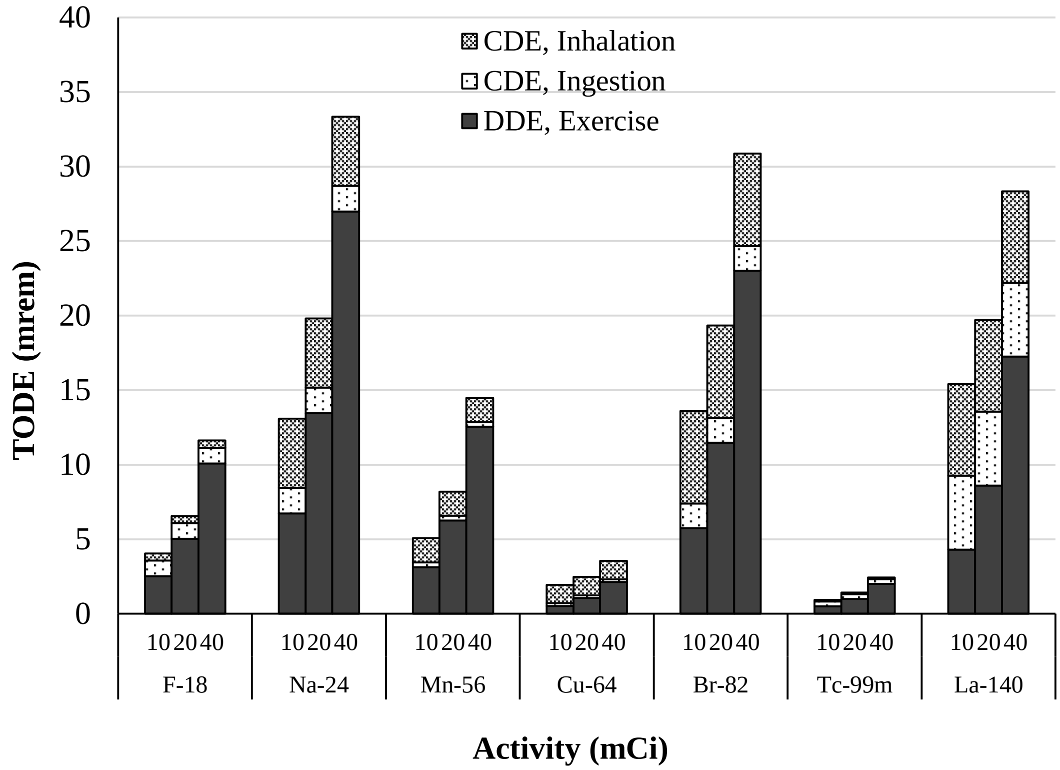

Fig. 9. The external and internal contributions to the exercise participant TODE are shown for 10, 20, and 40 mCi of each radionuclide.

Conversely, as activity decreases, the internal exposure from the assumed 1 µCi intake starts contributing the most to the total dose. Theoretically it is expected that zero dose be received if no activity is used. In Fig. 6 and Fig. 7., the TEDE and TODE for the exercise participant do not appear to approach zero as activity approaches zero. Fig. 6 and Fig. 7 only show data down to 1 mCi. In an effort to further explore this unexpected behavior, Fig. 10 displays exercise participant TEDE as a function of activities smaller than 1 mCi.

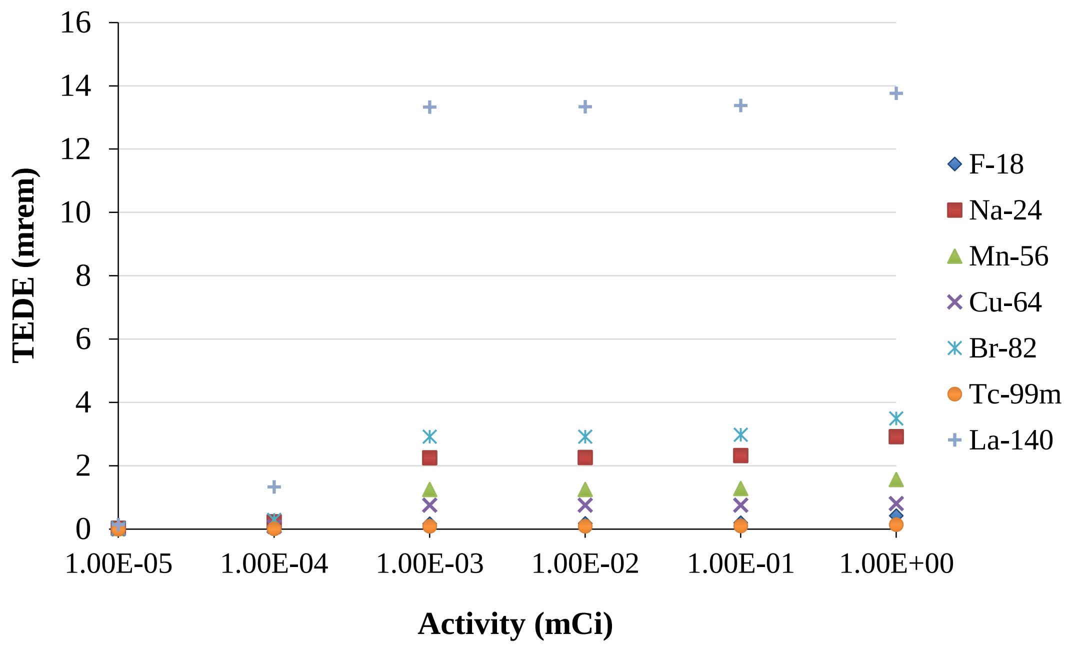

Fig. 10. Exercise participant TEDE as a function of activities less than 1 mCi.

As activity approaches 1 µCi, the assumption that the exercise participant inhales or ingests 1 µCi due to resuspension no matter the total activity used for the exercise breaks down (i.e., the exercise participant is inhaling or ingesting the entire source contents). If less than 1 µCi is used for the exercise, it is unrealistic to assume that the exercise participant takes in 1 µCi. Instead it should be assumed that the total activity used is inhaled or ingested. The TEDE then approaches zero, as shown in Fig. 10. The same behavior is expected for the exercise participant TODE.

III.C. Activity Determination

The overall analysis of TEDE and TODE for the worker and exercise participant yields that the worker TEDE restricts activity the most. Table 2 displays the maximum activities that should be used for each radionuclide and the resulting dose rate 100 cm from the contaminated surface during the exercise. Greater activities could be used for ^{18}F and ^{99m}Tc. An activity of 20 mCi was selected for these radionuclides as this is a typical activity used pharmaceutically (Brown et al. 2010). The dose rate during the 3-h exercise exceeds 1 mrem h^{-1} for several of the nuclides in Table 2. Activity could be decreased further for these radionuclides while maintaining a detectable dose rate compared to the background dose rate at Disaster City, which is about 10 µR h^{-1} based on past exercises.

Table 2. The maximum activity for the exercise was determined for the identified radionuclides. The assumed contamination area was 308.8 m^2. The asterisk indicates that a greater activity could be used without exceeding administrative dose limits.

Radionuclide	Maximum activity (mCi)	Maximum dose rate at 100 cm during exercise (mrem h^{-1})
F-18	20*	1.68
Na-24	10	2.24
Mn-56	20	2.09
Cu-64	40	0.71
Br-82	10	1.91
Tc-99m	20*	0.33
La-140	1	0.14

La-140, ^{24}Na, and ^{82}Br require more limiting activities because these radionuclides also emit gamma-rays that are higher in energy relative to the other radionuclides. La-140 causes more significant dose to the worker than ^{24}Na and ^{82}Br because of internal exposure. The absorption fraction, f_1, for ^{140}La is very low (0.001) relative to the other radionuclides (0.1-1). This value is the fraction of a stable element that reaches the body fluid after ingestion (ICRP 1979). Because f_1 is so low for ^{140}La, ingested lanthanum spends more time in the body and causes more dose. The inhalation DCFs for ^{140}La are also greater than for the other radionuclides. Tc-99m results in the lowest doses among all of the radionuclides due to its low-energy gamma-ray and small DCFs relative to the other studied radionuclides.

Using the activities listed in Table 2, the maximum duration of an exercise was determined for each radionuclide, along with the time it would take for the source to decay to a contamination level of 1000 dpm/100 cm^2. The minimum detectable dose rate for an exercise was assumed to be twice background at Disaster City, or 20 uR h^{-1}. This time-related information is captured in Table 3.

Table 3. The maximum exercise duration and time required to decay below 1000 dpm/100 cm^2 are listed for the assumed activities.

Radionuclide	**Maximum Activity (mCi)**	**Maximum exercise duration**		**Time required before release of contaminated area**	
		(h)	**(d)**	**(h)**	**(d)**
F-18	20*	11.7	0.49	19.2	0.80
Na-24	10	102	4.25	142	5.93
Mn-56	20	17.3	0.72	27.1	1.13
Cu-64	40	65.4	2.73	146	6.08
Br-82	10	232	9.67	335	14.0
Tc-99m	20*	24.3	1.01	63.0	2.63
La-140	1	113	4.71	248	10.4

The varied half-life among the radionuclides provides a way to modify the exercise. For example, multiple applications of a very short-lived radionuclide such as ^{18}F could occur in one day, or a single application of a longer-lived radionuclide could be detectable for several days. For a single application of the activities listed in Table 2, the contaminated area would be unusable at a minimum of less than a day (^{18}F) and a maximum of two weeks (^{82}Br).

Although ^{99m}Tc has a short effective half-life, it decays to stable ^{99}Tc (211,100-y half-life) which would be present in the environment for potentially thousands of years. The resulting ^{99}Tc surface activity from a single application of ^{99m}Tc would not constitute a radiological hazard or even be detectable and would not build up to the 1000 dpm/100 cm^2 limit even with hundreds of applications (Gwin 2012).

III.D. Accident Scenario Doses

Accident scenarios were examined for the activity levels restricted for worker exposure, as displayed in Table 2. The estimated doses for the contaminated wound and public exposure scenarios are listed in Table 4. For a contaminated wound, the administrative dose limits are not exceeded for any of the studied activities and radionuclides. The dose to a member of the public that enters the contaminated area was compared to the 100 mrem annual public dose limit. This dose limit is not exceeded at the activity levels restricted for worker exposure. Minimal dose to a member of the public assumed to be in the contaminated area implies that negligible dose should be received to a member of the public at the perimeter of the Disaster City site.

Table 4. Estimated doses for a contaminated wound and for a member of the public entering the contaminated area.

Radionuclide	Maximum Activity (mCi)	Contaminated Wound CEDE (mrem)	Member of Public	
			TEDE (mrem)	TODE (mrem)
F-18	20*	0.07	20.3	21.6
Na-24	10	0.41	29.5	33.3
Mn-56	20	0.65	26.4	27.0
Cu-64	40	0.65	9.24	9.93
Br-82	10	0.54	26.2	30.8
Tc-99m	20*	0.04	4.10	4.44
La-140	1	0.27	13.6	12.8

The accident scenario that results in the greatest dose is ingestion or inhalation of the entire source. Administrative dose limits are exceeded for all of the studied activities in this scenario. Table 5 displays the CEDE and CDE resulting from intakes of the activities restricted for worker exposure, as defined in Table 2.

Table 5. Estimated doses for worker ingestion or inhalation of the entire source.

Radionuclide	Maximum Activity (mCi)	Ingestion CEDE (rem)	Ingestion CDE (rem)	Inhalation CEDE (rem)	Inhalation CDE (rem)
F-18	20*	2.00	21.2	1.43	9.55
Na-24	10	12.5	17.3	10.0	46.3
Mn-56	20	20.0	6.31	5.00	32.6
Cu-64	40	20.0	7.07	10.0	49.6
Br-82	10	16.7	16.6	12.5	62.2
Tc-99m	20*	1.25	6.26	0.50	2.27
La-140	1	8.33	4.96	5.00	6.14

The computer code VARSKIN was used to determine the SDE and DDE resulting from a 1 μCi drop of solution on 10 cm^2 of skin with a 15-min exposure time. This exposure is compared to an annual SDE administrative limit of 5 rem. Fig. 11 displays the SDE results and Fig. 12 displays the DDE results. For a 1 μCi drop on the skin, the administrative SDE limit is not exceeded. The DDE is less than 2 mrad and would contribute marginally to the TODE. For the SDE, beta exposure contributes most to dose. Conversely, the DDE is driven by primarily gamma-ray exposure. This is as expected. Adding a lab coat or two surgeon gloves reduces the SDE by a factor of 1.16 to 1.54 depending on the radionuclide. The additional PPE has little effect on the DDE.

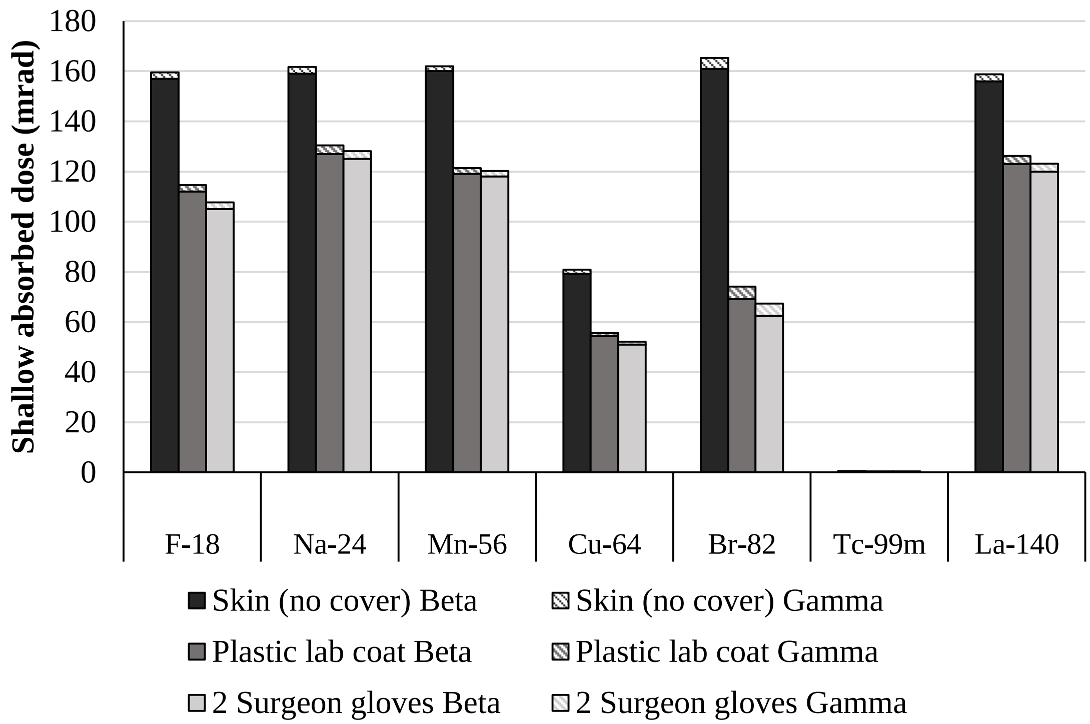

Fig. 11. The SDE is shown for a 1 μCi drop of solution on bare skin, and skin covered by a plastic lab coat or two surgeon gloves.

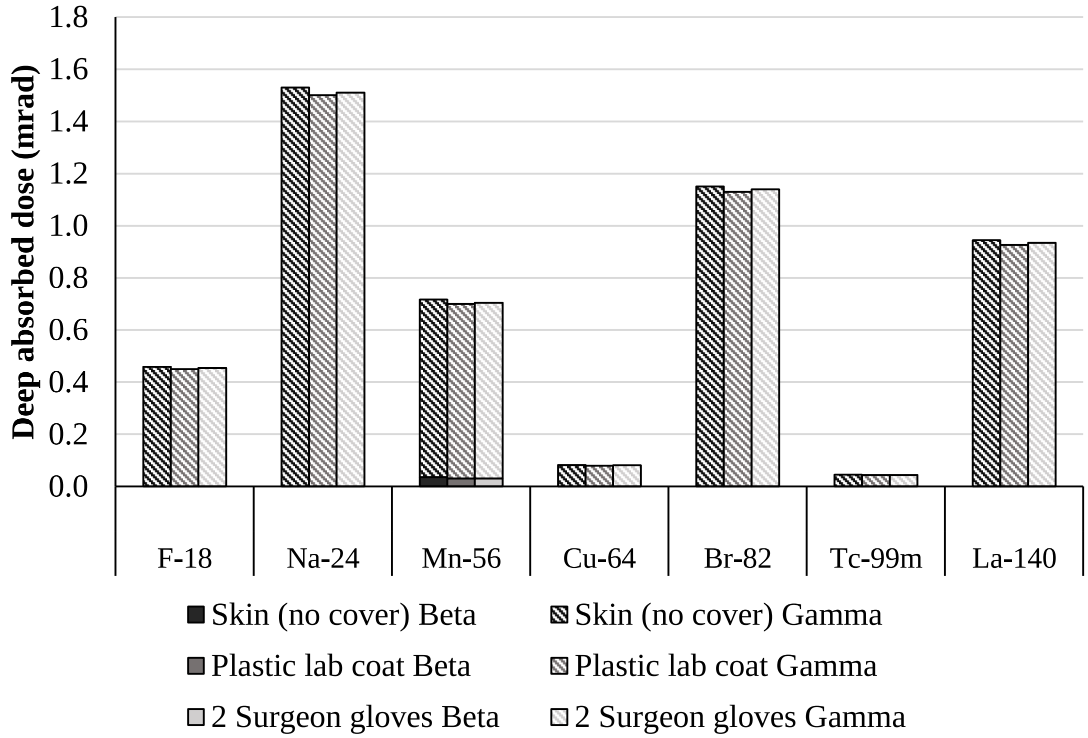

Fig. 12. The DDE is shown for a 1 µCi drop of solution on bare skin, and skin covered by a plastic lab coat or two surgeon gloves.

The dose resulting from skin exposure increases linearly with increasing activity. As much as about 30 µCi of ^{18}F, ^{24}Na, ^{56}Mn, and ^{140}La could be dropped onto a 10 cm^2 area of bare skin without exceeding the 5 rem administrative limit. Cu-64 emits fairly low-energy beta particles at low yields, so as much as about 60 µCi could be tolerated on bare skin without exceeding the administrative SDE limit. The SDE from a 1 µCi drop of ^{99m}Tc is very small because ^{99m}Tc doesn't emit beta particles. Tc-99m contributes less than 50 µrad to DDE. This is because the 140 keV gamma ray emitted by ^{99m}Tc is very low energy relative to the other studied radionuclides. The skin can tolerate a drop of nearly 10 mCi of ^{99m}Tc without exceeding the administrative SDE limit.

Table 6 displays the time needed for a 1 mCi spill of the radioactive solution over a small area to decay to the recommended 1000 dpm/100 cm^2 limit. A spill of this size requires the contaminated surface at Disaster City to be allowed to decay two to three times longer than the planned surface contamination levels, depending on the radionuclide.

Table 6. The time required for a 1 mCi spill over 100 cm^2 to decay below 1000 dpm/100 cm^2.

Radionuclide	**Time required to decay below 1000 dpm/100 cm^2**	
	(h)	**(d)**
F-18	38.6	1.61
Na-24	316	13.7
Mn-56	54.4	2.27
Cu-64	268	11.2
Br-82	744	31.0
Tc-99m	127	5.28
La-140	849	35.4

III.E. Exercise Recommendations

Dose estimates and results are known for the exercise performed by NNSS. For the actual exercise, the doses received were lower than the administrative limits, as well as the preliminary estimates used for justification of radiological safety. For the NNSS exercise using 20 mCi of ^{99m}Tc, the largest estimated dose was to the individual handling the source for one minute prior to dilution. An extremity dose of 400 mrem at 1 cm and a whole body dose of 0.5 mrem at 30 cm were estimated. In the actual exercise, the largest dose received was 20 mrem to the hand of the worker that dissolved the source. Because the actual doses received in the NNSS exercise were lower than the estimated doses, following those calculation methods implies that estimates for the Disaster City exercise are also conservative.

The results of this assessment are also conservative because the activities were not decay-corrected over the course of the exercise. If decay was accounted for, dose rates would decrease over time, which would in turn decrease exposure. In addition, no PPE was assumed for this assessment. Use of respiratory protection would restrict internal exposure even further and greatly diminish the TEDE and TODE for the worker, for which internal exposure contributes the most to the total dose. A negative pressure half-mask would decrease internal exposure by a factor of 10, and a full negative pressure facepiece would decrease internal exposure by a factor of 100 (U.S. NRC 1992). All participants are at risk of being externally contaminated. For this reason, coveralls are recommended. From a training standpoint, there is value in testing the exercise participants' ability to choose the proper PPE for the scenario, though ultimately the exercise controller should instruct the participants what to wear for safety during the exercise.

The greatest uncertainties for this assessment are the amount of material inhaled and ingested by the worker during spraying, and the external exposure to the exercise participants from the contaminated surface. A 1% intake was assumed for the worker in order to follow the conservative assumptions used by NNSS. Actual evaporation of the source and the size of the aerosols resulting from spraying is not known. Further investigation of worker intake during spraying is recommended because this contributes most to worker exposure. It is also recommended that standard resuspension factors and breathing rates be applied to determine internal exposure to the exercise participant due to resuspension. The conservative 1 μCi intake assumption used by NNSS becomes unrealistic as the total activity used for the exercise approaches 1 μCi.

The dose rate from the contaminated surface was modeled using an infinite plane source. For the infinite plane and slab geometries in the MicroShield program, only energy bins that are consistent with ANSI/ANS standard indices are allowed. User-defined photon grouping is not allowed by the software. User-defined grouping and even the auto-grouping feature typically more realistically represent the photon-energy spectrum from a given radionuclide. Care was taken to verify that the source term using the ANSI/ANS standard indices adequately characterized the actual source spectrum. For all of the radionuclides used in the assessment, the contribution of each photon energy to the total source strength was balanced with the under- or overestimation of the photon energy itself, as prescribed by the standard indices.

Actual distribution of the source onto an irregular surface such as one of the Disaster City rubble piles is not known and should be investigated further, as radioactivity could collect in certain areas of the pile and create larger dose rates. It could be valuable from a training standpoint to spray the source onto part of a rubble pile, which is not an even, flat surface. The sloped surfaces of the pile could contribute additional “shine” to participants.

Potential modifications to the exercise could be applied to tailor the exercise to a specific scenario or desired radiation dose limit. For instance, a sealed source could be added to the exercise to create an additional radiation field in which the exercise participants could operate. Additionally, the size of the contaminated area could be decreased, which would in turn concentrate the surface contamination level and create higher dose rates without increasing the total amount of activity used. The contamination could also purposely be distributed unevenly to simulate the dose rate contours that would be expected in a real incident.

According to the assessment, the worker exposure limits the activity for the exercise. Exposure to the worker that dissolves and sprays the source could be reduced by using two people – one person to dissolve the source and one person to distribute the source. For the assessment it was assumed that one person performs both tasks because this is more conservative. Entirely eliminating the need for a worker to spray the source on the desired surface would vastly decrease total dose for this exercise. This could be done by using a robot or configuring a mechanical rig that pumps the source over the surface without a person present. However, these approaches have their own cons (e.g., mechanical failure) that would need to be analyzed if used.

Radionuclide cost is also a point of consideration. The cost of activation at the NSC is $580 plus $100 for a transportation shield (Texas A&M Nuclear Science Center, personal correspondence, November 2015). The price for the compounds listed in Table 1 ranges from $1 to $35 per 10 g of material. Limited information was obtained regarding the purchase of radiopharmaceuticals. A vendor in College Station currently provides ^{99m}Tc to the Texas A&M CVM for horse scans. The vendor is contracted to supply ^{99m}Tc for a $95 weekly charge in which CVM performs five to six scans

(NuTech, Inc., personal communication, February 2016). F-18 can be purchased from a vendor in Houston. A quote for a different project involving ^{18}F at Texas A&M estimated $150 to $175 per dose ranging between 1 and 15 mCi, plus another $250 in delivery charges. A potential roadblock to using radiopharmaceuticals for the exercise at Disaster City is that radiopharmaceutical purchases are only authorized through a licensed nuclear pharmacy. The cyclotron at Texas A&M recently obtained licenses to produce isotopes for medical use, including ^{18}F, but cost information is unknown at this time (Texas A&M EHSD, personal communication, February 2016).

The logistics of an actual exercise could have a significant impact on the choice of radionuclide and needed activity. For example, it is possible that if the source is activated at the NSC, it will have to be transferred from the NSC radioactive material license to the Texas A&M EHSD radioactive material license. Regulatory contamination and radiation surveys would need to be performed and documented as part of this license transfer. In this scenario, it is recommended that the surveys and documentation be performed at Disaster City if possible in order to avoid transporting the source to EHSD prior to Disaster City, which could be an issue due to the short half-life.

The International Commission on Radiological Protection proposes an increased cancer risk of 17 percent per Sv, or 0.17 percent per rem, based on effects seen at high doses (ICRP 2007). If the 50 mrem or 500 mrem administrative dose limits were reached as a result of this exercise, this equates to a 0.0085% and 0.085% increased cancer risk, respectively. In comparison, the lifetime risk of developing a cancer among the U.S. population is 42.05% for men and 37.58% for women (ACS 2016). The increased cancer risk from potential exposure during this exercise is very small compared to this background cancer incidence. The benefit of enhanced emergency response training using unsealed contamination outweighs the potential health detriments due to radiation exposure during the exercise.

CHAPTER IV

CONCLUSIONS

The dose assessment methodology presented in this research can be used for any radionuclide and source activity so long as the radiological properties of the radionuclide are properly accounted for. The conservative results of the dose assessment indicate that an unsealed contamination exercise using the identified radionuclides (^{18}F, ^{24}Na, ^{56}Mn, ^{64}Cu, ^{82}Br, ^{99m}Tc, and ^{140}La) at Disaster City is safe from a radiological standpoint, and that ^{99m}Tc results in the lowest dose. The choice of which radionuclide and what activity to use should be made based on budget and the logistics of the actual exercise, including exercise duration and desired dose rates.

The results show that workers and exercise participants can receive measurable doses and as a result should be working under a worker permit and with proper badges and PPE. The most exposed individual for the hypothesized exercise is the worker that dissolves and distributes the source. The worker dose is manageable and can be limited by employing ALARA techniques. Only in the accident scenario where the entire source contents are ingested or inhaled are the assumed administrative dose limits exceeded. For an actual exercise, it is recommended that administrative dose limits be established for that specific exercise, taking into account the actual activity used and durations of exposures.

The results of this research provide a basis for the decision to proceed with planning an unsealed radioactive contamination exercise at Disaster City. This exercise will be a valuable addition to the few exercises currently performed worldwide using unsealed sources. Publishing the process of radionuclide selection is also useful for the radiological/nuclear incident response community because limited information is publically available.

IV.A. Future Work

Another graduate student at Texas A&M is currently performing a detailed characterization of background radiation and radioactivity levels at Disaster City such that if this exercise takes place, a contaminated area at Disaster City can be returned to known background levels. A next step towards executing this exercise at Disaster City would be to select and acquire a radionuclide, distribute it onto a surface and obtain exposure measurements to compare to the assessment.

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APPENDIX A

The radiological properties of the seven radionuclides identified for potential use at Disaster City are listed.

Radionuclide	F-18	Na-24	Mn-56	Cu-64	Br-82	Tc-99m	La-140
Decay method	β+ to O-18 (stable) Electron capture	β- to Mg-24 (stable)	β- to Fe-56 (stable)	EC+β+ to Ni-64 (61%), β- to Zn-64 (39%)	β- to Kr-82 (stable)	Primarily IT to Tc-99 (211100 y half-life to Ru-99 stable)	β- to Ce-140 (stable)
Half-life (h)	1.83	15.0	2.58	12.7	35.3	6.01	40.3
Specific activity ($Ci\ g^{-1}$)	$9.5x10^{7}$	$8.7x10^{6}$	$2.2x10^{7}$	$3.9x10^{6}$	$1.1x10^{6}$	$5.3x10^{6}$	$5.6x10^{5}$
Gamma-ray energies (keV)	511 (193%) annihilation photons	2754 (99.9%) 1369 (99.9%) 3866 (0.074%)	847 (98.8%) 1811 (26.9%) 2113 (14.2%)	511 (35.2%) annihilation photons 1346 (0.475%)	777 (83.4%) 554 (71.1%) 619 (43.5%)	140 (89%)	1596 (95.4%) 487 (45.5%) 816 (23.3%) 329 (20.3%)
Beta endpoint energies (keV)	633.5 (96.7%)	1391 (99.9%)	2848 (56.3%) 1038 (27.9%) 736 (14.6%)	1673 (17.4%) 578 (39%)	444 (98.5%) 265 (1.3%)	--	1365 (44%) 1680 (19.2%) 1244 (10.9%) 2164 (4.8%)

APPENDIX B

The thermal neutron capture cross sections for the naturally abundant isotopes of the elements that compose the compounds under consideration for neutron activation are listed. The target isotopes intended for activation are in bold font.

Nuclide	**Natural Abundance (%)**	**Thermal Neutron Capture Cross Section (b)**
H-1	99.96	0.3326
H-2	0.01	0.0005
C-12	98.93	0.0035
C-13	1.07	0.0014
O-16	99.76	0.0002
O-17	0.04	0.0005
O-18	0.20	0.0002
Na-23	**100.0**	**0.517**
Mg-24	78.99	0.054
Mg-25	10.00	0.199
Mg-26	11.01	0.038
Mn-55	**100.0**	**13.36**
S-32	94.99	0.518
S-33	0.75	0.454
S-34	4.25	0.256
S-36	0.01	0.236
Cu-63	**69.2**	**4.5**
Cu-65	30.8	2.17
Br-79	50.69	7.88
Br-81	**49.31**	**0.235**
La-138	0.09	57.2
La-139	**99.91**	**9.04**

APPENDIX C

The listed dose parameters were used for the dose assessment.

Reference	ORNL/ RSIC-45/R1	10 CFR 20		FGR 11			10 CFR 20		FGR 11		
Nuclide	**Specific Gamma Constant (R m^2 h^{-1} Ci^{-1})**	**f**	**Ingestion ALI (µCi)**	**Effective Ingestion DCF (Sv/Bq)**	**Ingestion DCF for limiting organ (Sv/Bq)**	**Limiting organ**	**Class**	**Inhalation ALI (µCi)**	**Effective Inhalation DCF (Sv Bq^{-1})**	**Inhalation DCF for limiting organ (Sv Bq^{-1})**	**Limiting organ**
F-18	6.849E-01	1	5.00E+04	3.31E-11	2.87E-10	ST wall	D	7.00E+04	2.26E-11	1.29E-10	Lung
Na-24	1.928E+00	1	4.00E+03	3.84E-10	4.68E-10	B Surface	D	5.00E+03	3.27E-10	1.25E-09	Lung
Mn-56	9.169E-01	0.1	5.00E+03	2.64E-10	8.53E-11	Gonad	D	2.00E+04	1.02E-10	4.40E-10	Lung
Cu-64	1.300E-01	0.5	1.00E+04	1.26E-10	4.78E-11	Gonad	W	2.00E+04	6.93E-11	3.35E-10	Lung
Br-82	1.612E+00	1	3.00E+03	4.62E-10	4.48E-10	Gonad	W	4.00E+03	4.13E-10	1.68E-09	Lung
Tc-99m	1.227E-01	0.80	8.00E+04	1.68E-11	8.46E-11	Thyroid	W	2.00E+05	7.21E-12	3.07E-11	Lung
La-140	1.267E+00	0.001	6.00E+02	2.28E-09	1.34E-09	Gonad	D	1.00E+03	9.33E-10	1.66E-09	Lung