SAND2016-12871C

EXAMPLE OF INTEGRATION OF SAFETY, SECURITY, AND SAFEGUARD USING DYNAMIC PROBABILISTIC RISK ASSESSMENT UNDER A SYSTEM-THEORETIC FRAMEWORK

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Transportation of spent nuclear fuel (SNF) is expected to increase in the future, as the nuclear fuel infrastructure continues to expand and fuel takeback programs increase in popularity. Analysis of potential risks and threats to SNF shipments is currently performed separately for safety and security. However, as SNF transportation increases, the plausible threats beyond individual categories and the interactions between them become more apparent.

A new approach is being developed to integrate safety, security, and safeguards (3S) under a system-theoretic framework and a probabilistic risk framework. At the first stage, a simplified scenario will be implemented using a dynamic probabilistic risk assessment (DPRA) method. This scenario considers a rail derailment followed by an attack. The consequences of derailment are calculated with RADTRAN, a transportation risk analysis code. The attack scenarios are analyzed with STAGE, a combat simulation model. The consequences of the attack are then calculated with RADTRAN. Note that both accident and attack result in SNF cask damage and a potential release of some fraction of the SNF inventory into the environment.

The major purpose of this analysis was to develop the input data for DPRA. Generic PWR and BWR transportation casks were considered. These data were then used to demonstrate the consequences of hypothetical accidents in which the radioactive materials were released into the environment. The SNF inventory is one of the most important inputs into the analysis. Several pressurized water reactor (PWR) and boiling water reactor (BWR) fuel burnups and discharge times were considered for this proof-of-concept. The inventory was calculated using ORIGEN (point depletion and decay computer code, Oak Ridge National Laboratory) for 3 characteristic burnup values (40, 50, and 60 GWD/MTU) and 4 fuel ages (5, 10, 25 and 50 years after discharge).

The major consequences unique to the transportation of SNF for both accident and attack are the results of the dispersion of radionuclides in the environment. The dynamic atmospheric dispersion model in RADTRAN was used to calculate these consequences. The examples of maximum exposed individual (MEI) dose, early mortality and early morbidity, and soil contamination are discussed to demonstrate the importance of different factors.

At the next stage, the RADTRAN outputs will be converted into a form compatible with the STAGE analysis. As a result, identification of additional risks related to the interaction between characteristics becomes a more straightforward task. In order to present the results of RADTRAN analysis in a framework compatible with the results of the STAGE analysis, the results will be grouped into three categories:

- *Immediate negative harms*
- *Future benefits that cannot be realized*
- *Additional increases in future risk*

By describing results within generically applicable categories, the results of safety analysis are able to be placed in context with the risk arising from security events.

I. INTRODUCTION

Transportation of spent nuclear fuel (SNF) is expected to increase in the future, as the nuclear fuel infrastructure continues to expand and fuel takeback programs increase in popularity. Analysis of potential risks and threats to SNF shipments is currently performed separately for safety and security. However, as SNF transportation increases, the plausible threats beyond individual categories and the interactions between them become more apparent.

A new approach is being developed to integrate safety, security, and safeguards (3S) under a system-theoretic framework and a probabilistic risk framework. The conceptual design of these frameworks are described in Ref. 1. At the first stage, a simplified scenario will be implemented using a dynamic probabilistic risk assessment (DPRA) method. This scenario considers a rail derailment followed by an attack. The consequences of derailment are calculated with RADTRAN (Ref. 2), a transportation risk analysis code. The attack scenarios are analyzed with STAGE (Ref. 3), a combat simulation model. The consequences of the attack are then calculated with RADTRAN. The diagram in Fig. 1 shows the connections between the dynamic event tree, RADTRAN, and STAGE.

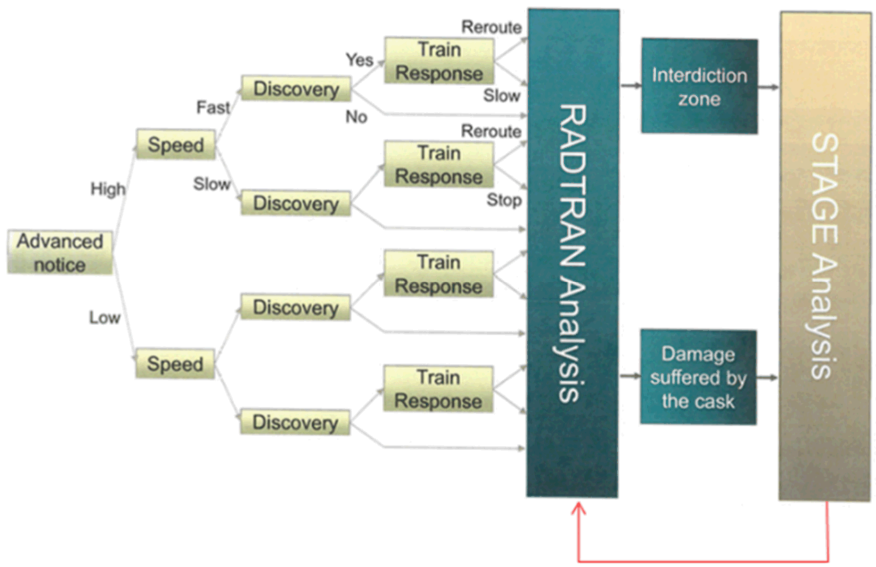

Fig. 1. Diagram of Integration of 3S Using Dynamic Probabilistic Risk Assessment.

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I. ANALYTICAL APPROACH

II.A. Dynamic Probabilistic Risk Assessment (DPRA)

Dynamic probabilistic risk assessment (DPRA) methodologies are a 'bottom up' approach to systematically describe a system entering different failure states based on individual component failures. In the integrated 3S analysis, the order of events within a scenario is not fixed and should not be pre-specified, due to the inherent uncertainties of the scenario (e.g., safety event before a security event or vice versa). In addition, there are many input files that need to be prepared for a variety of different models, which can lead to errors from improper preparation. Both of these concerns challenge the event-tree/fault-tree methodology of conventional probabilistic risk assessment (PRA).

By constructing dynamic event trees (DETs) integrated 3S scenarios can be assessed organically. By automating the creation and assessment of scenarios, transcription errors can be eliminated and the system analysis space can accommodate an unknown sequence of events. In addition, the transition between safety, security and safeguards analysis can be handled at a level of individual runs, which allows for a seamless transition between each analysis. The output from a safety code can be fed into a security code, which has its output fed into a safeguards code. Additionally, the order of implementation of each 3S code may be shuffled in order to explore potential differences that may occur.

II.B. RADTRAN

RADTRAN© (Copyright: Sandia National Laboratories 2006) is the national standard for transportation risk assessment computer codes. The international version, INTERTRAN (Ref. 4), is based on RADTRAN. RADTRAN combines user-determined meteorological, packaging, demographic, transportation, and material data with health physics data to calculate the expected radiological consequences and accident risk of transporting radioactive material.

RADTRAN was initially developed for the "Environmental Impact Statement (EIS) for the Transportation of Radioactive Materials by Air and Other Means" (NUREG-0170) in 1977 (Ref. 5). RADTRAN 6.10, from 2014 is the version used for this effort.

II.C. Scenario Toolkit and Generation Environment (STAGE)

Sandia has used the commercial-off-the-shelf (COTS) Presagis International computer combat model Scenario Toolkit and Generation Environment (STAGE) (Ref. 3) to develop a novel vulnerability analysis tool to aid in the design and evaluation of nuclear security applications. The STAGE software interface consists of five editors:

- The **database editor**, allows the user to define all of the various computational sensors, weapon effects, combat speeds, load outs, armaments, and other variables that serve as the data foundation for the STAGE software.
- The **mission editor** is the logic based behavioral model consisting of an 'if/then' structure that is flexible to model such choices as navigation/locomotion, detection/sensing, and weapon deployment/operation to more complex behaviors such as communication, mission switching/adaptation, and defeat of physical protection barriers
- The **script editor**, serves as the behavior editor that allows the user to define the prioritization logic (e.g., human/vehicle differentiation, weapon target preference, target selection, and ammunition use) that allows the entities in the simulation to automatically react to the environment.
- The scenario editor organizes the information from the previous three editors into the simulation environment; includes 2D/3D graphical and numerical data displays.
- The run time editor conducts that actual simulation.

STAGE allows the user to focus more on the complex behaviors of the scenario and less on plotting the exact course of entities; allows entities within STAGE to dynamically plan paths, recognize and avoid obstacles or harsh terrain, and stay on defined pathways such as roads or sidewalks (Ref. 3). The ability to react with intelligent behaviors to dynamic simulation environment changes at the entity level exemplifies the overall flexibility that STAGE has in modeling higher fidelity security analysis for nuclear applications (Ref. 6).

II.D. Analysis of Dynamics Accident Progression Trees (ADAPT)

The DPRA method will use the Sandia National Laboratories (SNL) software, Analysis of Dynamic Accident Progression Trees (ADAPT) (Ref. 7). The ADAPT software was developed by the Ohio State University as part of a SNL laboratory directed research and development project to generate dynamic event trees (DETs). System simulators such as RADTRAN or STAGE can be linked with ADAPT to determine possible scenarios based on the branching and stopping rules provided by the user. ADAPT can keep track of scenario likelihoods and graphically display the DETs, as well as all simulator output as a function of time.

ADAPT has a distributed computing architecture: a simulator driver, centralized server, a database storage area, and a graphical user interface (GUI) based client side software. Figure 2 provides a schematic overview of the ADAPT computational infrastructure. The ADAPT framework is an open architecture that will allow easy replacement of the component modules, and algorithms used in those components (Ref. 8). The ADAPT system components assume that for a single event tree, a single simulator is used to follow the transient.

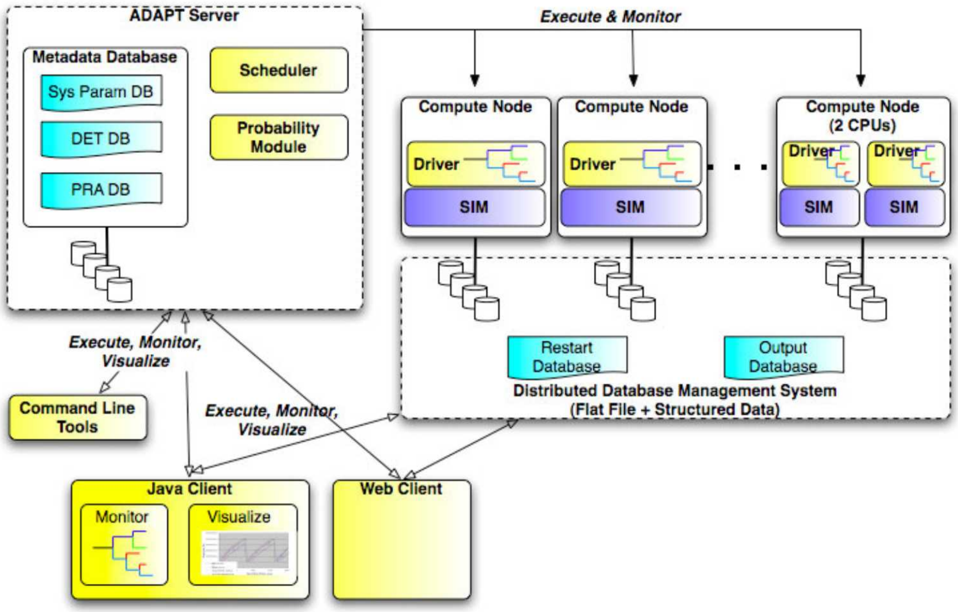

Fig. 2: ADAPT System Architecture (Ref. 8)

With the use of simulator agnostic Driver, ADAPT currently provides means to process the output of a single simulator and edit/modify (i.e., the Apply Edit Rules in Figure 3) input files for dynamic branching. The Driver for the simulator (i.e., RADTRAN and STAGE, for this work) will read input, conduct check-pointing, allow users to define stopping conditions, and utilize output to detect stopping conditions. The Driver also allows for a user-defined edit-rules file to be created. Figure 3 provides an illustration of the Driver's workflow interaction with a plant simulator.

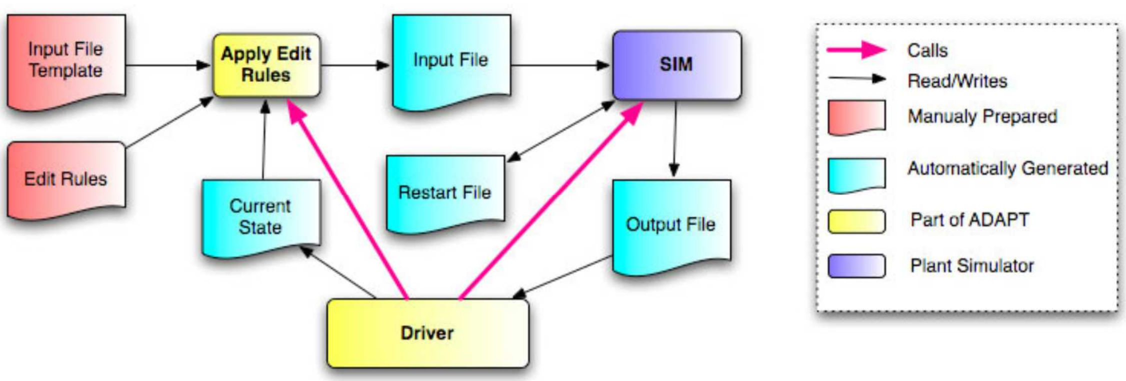

Fig. 3: ADAPT Driver (Ref. 8)

ADAPT maintains a distributed database. ADAPT has the capability to access the DETs, the input and output files of the simulations, and the metadata. ADAPT also maintains basic statistics within the metadata. ADAPT allows the use of XML schemas to describe the metadata schema and create a generic framework which will allow for the design and deployment of multiple schemas for the multiple simulator drivers (Ref. 8). The ADAPT database system can efficiently process analysis queries such as plotting system variables which may require accessing multiple output files stored over multiple nodes. The ADAPT database is also capable of pruning event trees.

In the ADAPT software, a pluggable scheduling interface was designed and implemented with three basic scheduling techniques: (1) random scheduling, (2) first-come first-served scheduling, and (3) greedy staging minimization (Ref. 8). When a compute node becomes idle, random and first-come first-served scheduling techniques either pick a random task from the task queue, or pick the very first one in the queue respectively. When a compute node becomes idle, the greedy staging minimization algorithm first scans the task queue for a task whose parent had been executed on the same compute node; if such task exists it is picked and executed on that node. Otherwise the first task in the queue is executed in that node.

III. INPUT DATA

This analysis considered two generic SNF transportation casks – a PWR cask with a 24 fuel assembly capacity and a BWR cask with a 52 fuel assembly capacity. The design of these generic casks is loosely based on the AREVA TN24 dual-purpose storage and transportation casks (Ref. 9). The TN24 casks have been used worldwide. Over 300 of these casks were delivered and 400 have been ordered worldwide (Ref. 9). Note that unlike in US, in the other countries the SNF is (will be) transported in the bare fuel transportation casks (e.g. uncanistered fuel). These casks are designed as the rail cask, but they also are (can be) transported by the heavy haul truck.

The most important inputs into the analysis as discussed below are

- radionuclide inventory, which determines what is in the release
- release fractions, which determine how much is released
- dispersion parameters, which determine the plume size, and air and soil concentrations
- exposure parameters (breathing rate, evacuation time, population density)

III.A. Radionuclide Inventory

Several pressurized water reactor (PWR) and boiling water reactor (BWR) fuel burnups and discharge times were considered for this proof-of-concept. The inventory was calculated using ORIGEN (point depletion and decay computer code, Oak Ridge National Laboratory) (Ref. 10) for 3 characteristic burnup values (40, 50, and 60 GWD/MTU) and 4 fuel ages (5, 10, 25 and 50 years after discharge). Note that NUREG-2125 (Ref. 11), the most recent analysis of the spent fuel transportation risk assessment, considered PWR fuel with discharge time 14 years and burnup 45 GWD/MTU. However, it is likely that the SNF transported by the other countries will have a wide range of different burnups and ages.

A large number of radionuclides are present in SNF (e.g., 204 radionuclides are found in 5-year-old, 60 GWD/MTU fuel). The approach used was to include all the radionuclides considered in NUREG-2125 and any additional radionuclides (if present) that contribute to >90% of the human health effects (e.g., 69 SOARCA radionuclides, Ref. 12). An example of calculated inventories is provided in Table I for 60 GWD/MTU PWR and BWR fuel assemblies. The additional radionuclides represent 22-26% of the total activity. However, the normalized activity of these radionuclides is less than 0.02%. The normalized activities were calculated by dividing the actual activities by the corresponding A2 (radiotoxicity) values (Ref. 13) and expressing these obtained values as percent of total normalized activity.

TABLE I. Example of Calculated Inventories for 60 GWD/MTU PWR and BWR Assemblies.

Isotope	**Assembly Activity (Tera Becquerel)**							
	PWR				**BWR**			
	5yr	**10yr**	**25yr**	**50yr**	**5yr**	**10yr**	**25yr**	**50yr**
Radionuclides considered in NUREG-2125								
am241	19.22	31.58	54.187	67.246	8.95	14.41	24.384	30.136
am242	0.11001	0.10734	0.099715	0.088191	0.05	0.05	0.044664	0.039502
am242m	0.11052	0.10783	0.10017	0.088596	0.05	0.05	0.04487	0.039684
am243	1.0333	1.0328	1.0313	1.0289	0.39	0.39	0.39416	0.39323
ce144	176.08	2.0771	3.41E-06	7.79E-16	55.95	0.66	1.08E-06	2.47E-16
cm243	0.61188	0.54323	0.38011	0.20964	0.25	0.22	0.15259	0.084153
cm244	226.81	187.33	105.55	40.566	66.49	54.92	30.942	11.892
co 60	18.353	9.5135	1.3252	0.049599	15.75	8.16	1.137	0.042557
cs134	1142.2	213.51	1.3945	0.000318	350.87	65.59	0.42837	9.78E-05
cs137	2837.7	2529.1	1790.4	1006.8	1019.70	908.76	643.33	361.75
eu154	133.06	88.956	26.578	3.5491	49.91	33.37	9.9697	1.3313
eu155	60.312	29.103	3.27	0.085552	23.13	11.16	1.2539	0.032805
kr 85	200.64	145.4	55.34	11.062	69.13	50.10	19.069	3.8116
pu238	137.87	132.53	117.74	96.653	46.18	44.40	39.439	32.379
pu239	3.9103	3.91	3.9089	3.9071	1.96	1.96	1.9616	1.9605
pu240	8.3927	8.4972	8.7094	8.8655	4.52	4.55	4.6076	4.648
pu241	1767.3	1386.9	670.32	199.55	780.81	612.75	296.15	88.157

pu242	0.084109	0.084108	0.084107	0.084105	0.03	0.03	0.027813	0.027813
ru106	341.4	11.365	0.000419	1.71E-11	113.32	3.77	0.000139	5.69E-12
sb125	61.621	17.558	0.40615	0.000763	20.75	5.91	0.13674	0.000257
sr 90	1942.4	1722.3	1200.5	657.86	685.32	607.64	423.55	232.1
te125m	15.089	4.2994	0.099455	0.000187	5.08	1.45	0.033484	6.29E-05
u234	0.018084	0.019991	0.02528	0.032815	0.01	0.01	0.009883	0.012407
y 90	1942.9	1722.7	1200.8	658.03	685.49	607.79	423.66	232.16
Additional Radionuclides								
ba137m	2687.30	2.40E+03	1695.5	953.39	965.60	861	609.23	342.57
cm242	0.66	8.92E-02	0.082464	0.072934	0.29	0.04	0.036937	0.032668
np239	1.03	1.03	1.0313	1.0289	0.39	0.3.951	0.39416	0.39323
pr144	176.09	2.08	3.41E-06	7.79E-16	55.96	0.660	1.08E-06	2.47E-16
pr144m	1.68	0.0198	3.25E-08	7.43E-18	0.53	6.30E-03	1.03E-08	2.36E-18
rh106	341.40	11.4	0.000419	1.71E-11	113.32	3.77	0.000139	5.69E-12
te127	0.00	6.82E-09	5.17E-24	0	0.00	2.80E-09	2.12E-24	0
te127m	0.00	6.96E-09	5.27E-24	0	0.00	2.86E-09	2.17E-24	0
TOTAL	**14245.39**	**10658.11**	**6938.86**	**3710.25**	**5140.18**	**3903.55**	**2530.39**	**1343.99**
% additional	22.52	22.61	24.45	25.73	22.10	22.17	24.09	25.52

Figure 4 shows the total activity of generic PWR and BWR casks as a function of age and burnup. Also shown in this figure is the total activity of the cask considered in NUREG-2125. Note that the NUREG-2125 cask had 26 assemblies while the generic PWR cask has 24 assemblies. The total activity spans over a large range with PWR cask activity being about 1.3 higher than BWR cask activity.

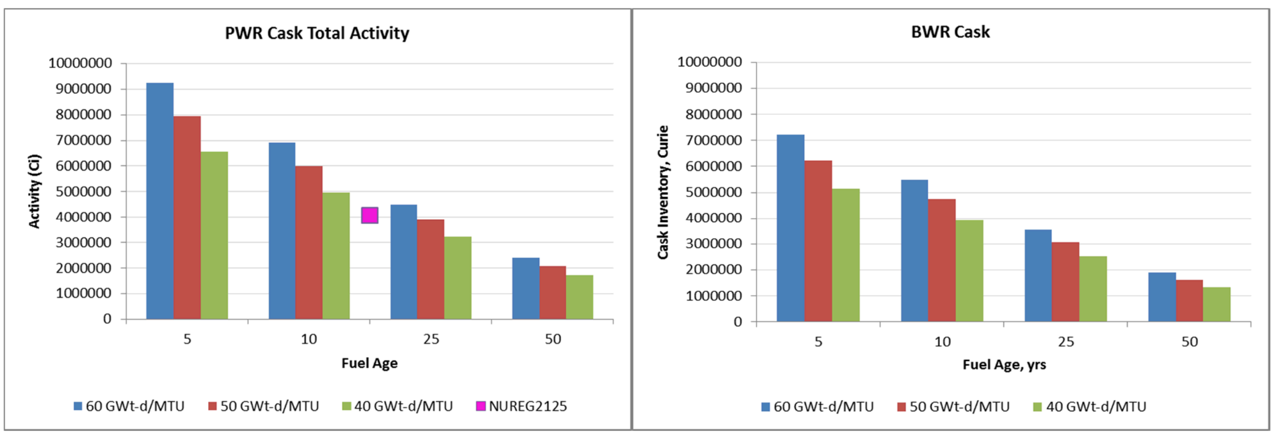

Fig. 4. Generic Transportation Cask Inventory.

Figure 5 shows the normalized activities of 60 GWD/MTU PWR and BWR fuel assemblies. In both cases (PWR and BWR), the major contributors with regard to radiotoxicity are Pu-238, Am-241, and Cm-244 (76%-89% of radiotoxicity). Cs-137, Pu-241, Pu-240, Pu-241, Sr-90, and Y-90 are smaller contributors (11%-22%) and their contribution decreases with time. The total contribution from the remaining radionuclides is 0.75% to 2% and it decreases with time as well.

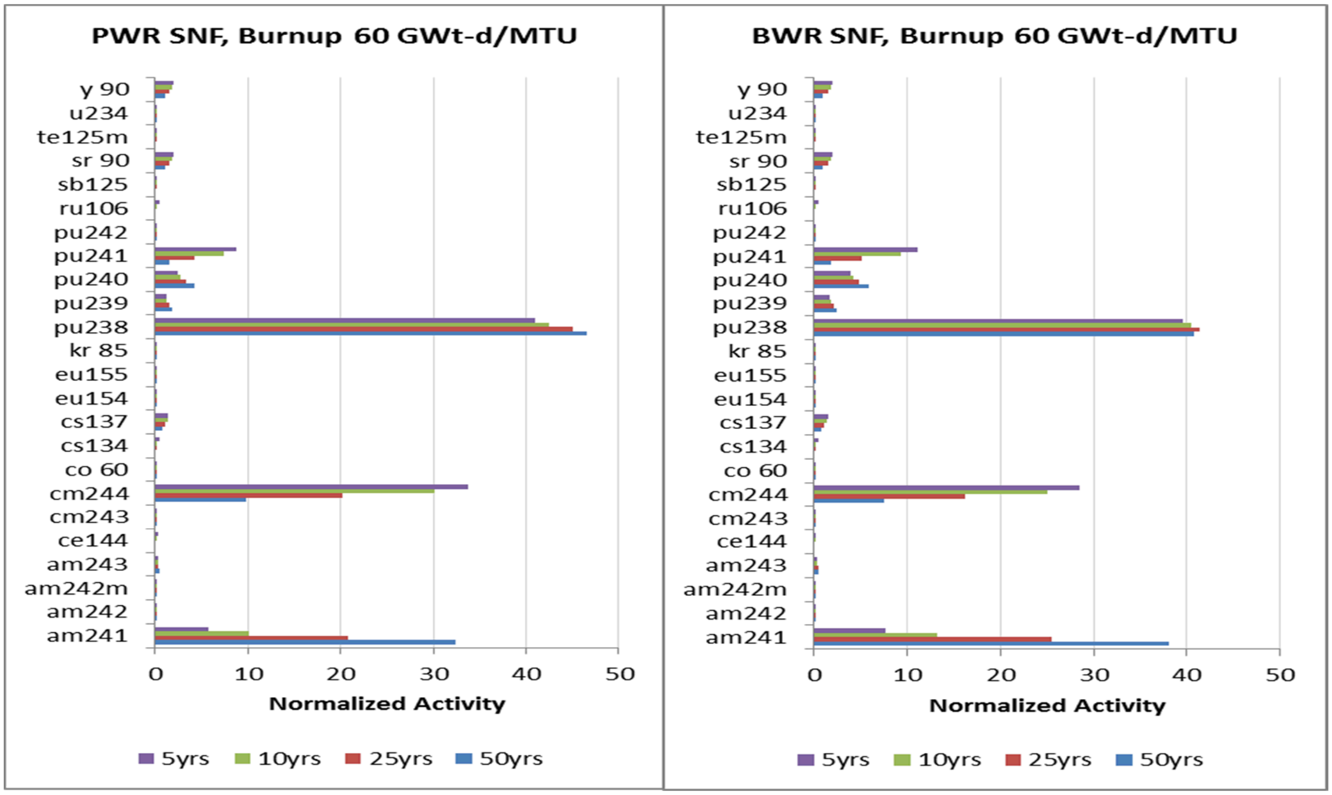

Fig. 5. Normalized Inventory of PWR and BWR Assemblies.

Figure 6 provides an example of how the activity of the major contributors changes with time. The activities of Cm-244, Cs-137, Pu-238, Pu-241, Sr-90, and Y-90 decrease with time. The activities of Pu-239 and Pu-240 do not change (long decay times). The activity of Am-241 increases with time (in-growth). The same tendencies were observed for the other burnups and BWR fuel.

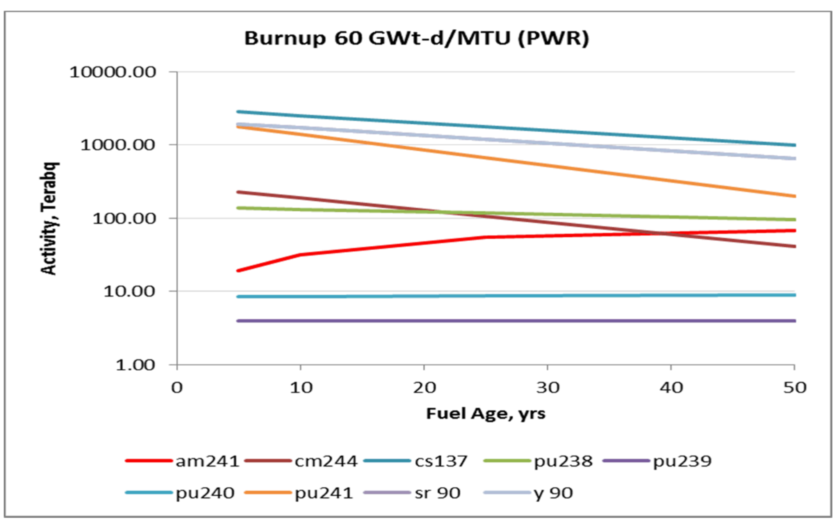

Fig. 6. PWR Assembly Radionuclide Activity as a Function of Time.

The Am-241 and Pu-238 activities per PWR assembly are shown in Figures 7 and 8 as a function of age and burnup. Am-241 activity increases with age and burnup. Pu-238 activity decreases with age and increases with burnup. In summary, the major difference between the SNF of different age and burnup is in the activities of Am-241, Cm-244, Pu-238, Sr-90, Y-90, and Cs-137.

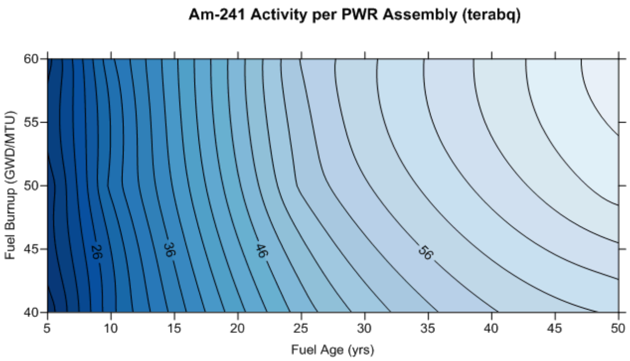

Fig. 7. Am-241 Activity as a Function of PWR Assembly Age and Burnup.

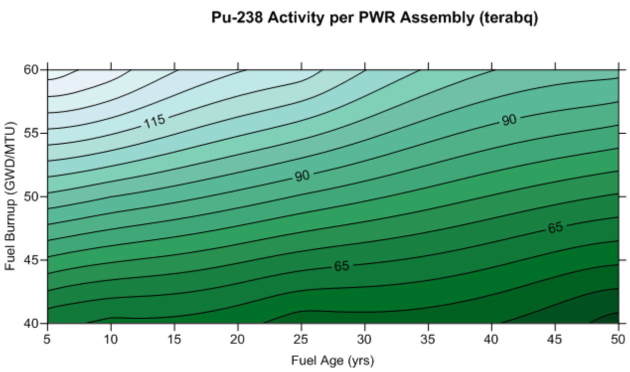

Fig. 8. Pu-238 Activity as a Function of PWR Assembly Age and Burnup.

III.B Release Fractions

Both accident and attack may result in damage to the transportation cask and fuel. The fission product gases and particles from the damaged rods may be released into the cask and ultimately into the environment. The aerolized material (InventoryxRelease FractionxAerolized Fraction) is the source of external exposure. The respirable material (Aerolized MatterxRespirable Fraction) is the source of exposure via inhalation including inhalation of resuspended material. The release fractions are defined for the physical/chemical group in RADTRAN. For SNF, these groups are classified as: gas, CRUD, particle, and volatile.

The release fraction defined for the accidents involving a transportation cask with uncanistered PWR fuel in NUREG-2125are shown in Table II in parentheses. It was hypothesized that if the cask and SNF are damaged in an attack, the release fractions of particles and volatiles from rods to cask would be 50 times higher than in accident scenarios considered in NUREG-2125. It was assumed that the release fractions of CRUD and gas would be the same as in NUREG-2125 accidents. It was further assumed that 100% of gases released in the cask would be released into the environment. The same values as NUREG-2125 were assumed for the aerosol and respirable fractions for chemical/physical forms. Note that a probabilistic description of the security system's effectiveness against expected attacks is calculated by STAGE. In RADTRAN calculations, the probability was set equal to 1.0 to calculate dose, and not dose risk.

TABLE II. Release Fractions Considered in the Analysis.

Group	**Release Fraction**		**Total Release Fraction**	**Aerosol Fraction**	**Respirable Fraction**	**Total Respirable**
	Rods to Cask	**Cask to Environment**				
Gas	0.12	1 **(0.8)**	0.12 (0.096)	1	1	0.12 (0.096)
CRUD	1 (1)	0.001 (0.001)	0.001 (0.0101)	1 (1)	0.05 (0.05)	$5.0*10^{-5}$ ($5*10^{-5}$)
Particle	$1.68*10^{-4}$ ($4.8*10^{-6}$)	1 (0.7)	$1.68*10^{-4}$ ($3.36*10^{-6}$)	1 (1)	0.05 (0.05)	$8.4*10^{-6}$ ($1.68*10^{-7}$)
Volatile	$7.50*10^{-4}$ ($3.0*10^{-5}$)	1 (0.5)	$7.5*10^{-4}$ ($1.5*10^{-5}$)	1 (1)	0.05 (0.05)	$3.75*10^{-5}$ ($7.5*10^{-7}$)
NOTE: The values considered in NUREG-2125 are shown in parenthesis						

III.C. Dispersion and Exposure Parameters

The dispersion of released material is calculated with the dynamic atmospheric dispersion model, which is one of the options available in RADTRAN. The dispersion model parameters include

- release height,
- source height,
- source width,
- heat flux,
- wind speed,
- thermal lofting temperature,
- atmospheric mixing height,

- atmospheric stability class,
- rainfall, and
- deposition velocities

The source height and width are defined by the dimensions of the transportation cask. The release height is defined by the position of the cask (on railcar or on the ground) and its dimensions. The release height was assumed to be 2.0 m (cask on the railcar). The other parameters related to the weather and the deposition velocity will come from the DPRA method.

This analysis assumed the same weather parameters as in two cases considered in NUREG-2512. The first case is for a very stable meteorology with Pasquill stability F and wind speed 0.5 m/s. The second case is for a neutral weather conditions with the Pasquill stability D and wind speed 4.7 m/s. Additionally, the second case was considered with and without precipitation. The deposition velocity was assumed to be 0.0 for gas and 0.01for CRUD, volatiles, and particles; as was done in NUREG-2125.

The exposure parameters are evacuation time and breathing rate. The evacuation time will also come from the DPRA method. In this analysis the RADTRAN default value is equal to 24 hrs (the one used in NUREG-2125 as well). The RADTRAN default breathing rate was specified. A different breathing rate can be specified via the DPRA method if a subpart of the exposed population group is very different from the average.

III.D. Security Scenario Input Parameters

STAGE is based on modeling the transportation scenario in terms of the physical parameters of the transportation vehicle itself—including size, weight and average/abnormal speed. In addition, the physical dimensions of the SNF cask is necessary to provide as realistic a simulation as possible within the STAGE environment. STAGE also models the reliability of various physical protection system (PPS) components, like sensors and locks, placed on the SNF cask and transportation vehicle itself. More specifically, these PPS, in terms of key parameters such as size of the sensor field, probability of detection, probability of alarm communication and delay time provided. Lastly, in its traditional role as a 'force-on-force' simulator, the differing capabilities, skills and resources of both the adversary and response forces are included— including response time, weapons usage proficiency, destructive power of various weapons and proficiency in securing the scene of a security event. Per the mission and script editors, STAGE also models fairly complex and coordinated adversary attacks, including diversion, multi-prong and swarm attacks. This flexibility also aids in modeling the varying capabilities of different types (e.g., local guards to the national police force) of response to an adversary attack. (Ref. 3, Ref. 6)

For this analysis, ADAPT will use the output from the RADTRAN analysis to influence several key input variables for STAGE—to include the response time of the onboard security force, initiation of secondary (and tertiary) waves of response from local law enforcement, ability for response force members to execute related tasks (e.g., securing the site) and reliability metrics of security technologies.

IV. CONSEQUENCE ANALYSIS

The consequences were calculated for 12 BWR and 12 PWR scenarios (i.e., different combinations of fuel age and burnup) (stability F, wind speed 0.5 m/s). A few additional scenarios (e.g., different release fractions and weather conditions) were considered for the PWR fuel. The consequences were characterized using the following parameters:

- A dose to the maximally exposed individual (MEI) during evacuation
- Number of early fatalities
- Total activity of the soil within the contaminant plume at the time of release

Note that the additional parameters will be added to the list during the next stage of this analysis (i.e., combined RADTRAN-STAGE analysis with the DRPA method).

Fig. 9 shows the dose to MEI as a function of age for a PWR fuel with different burnup assuming high release fraction (Table II). In addition, a MEI was calculated for the same release fractions as in NUREG-2125 for 5-year-old 60 GWD/MTU fuel assemblies (release fractions shown in parentheses in Table II). This MEI dose is about 50 times smaller than the corresponding dose with the high release fractions of particles and volatiles. This demonstrates that the results in Figure 9 can be easily scaled up or down to represent the release fractions values of interest. The NUREG-2125 MEI dose is shown in Fig. 9 for comparison. The dose is smaller because the fuel is older and burnup is lower.

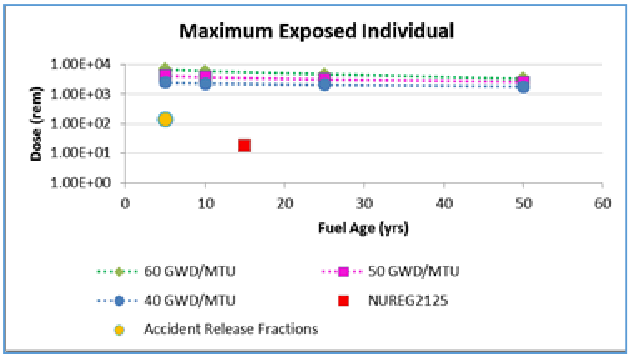

Fig.9. MEI during Evacuation Due to Release from a Damaged PWR SNF.

Figure 10 shows the MEI dose as a function of PWR fuel burnup and age. There is a steep increase in dose for a young high burnup fuel.

Fig. 11 shows the soil activity within the plume at the time of release as a function of age for a PWR fuel with different burnups assuming high release fraction (Table II). In addition, the soil activity was calculated for the same release fractions as in NUREG-2125 for a 5-year-old 60 GWD/MTU fuel assemblies (release fractions shown in parentheses in Table II). This soil activity is 41 times smaller than the corresponding soil activity with the high release fractions of particles and volatiles. The scaling in this case is non-linear and the results can't be scaled up or down in the same way as for MEI dose. The value corresponding to the NUREG-2125 case is shown in Fig. 11 for comparison. The soil activity is smaller because the fuel is older and burnup is lower.

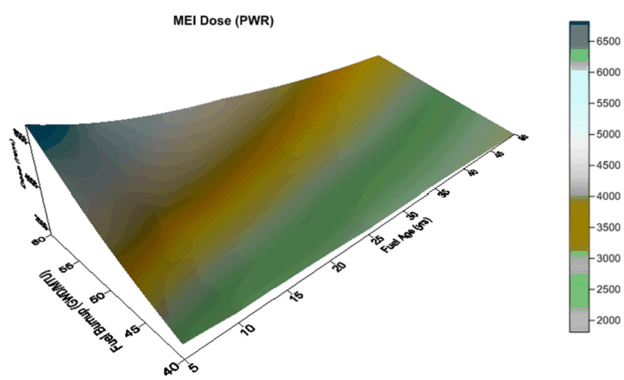

Fig 10. MEI Dose as a Function of Fuel Age and Burnup (Release from PWR SNF).

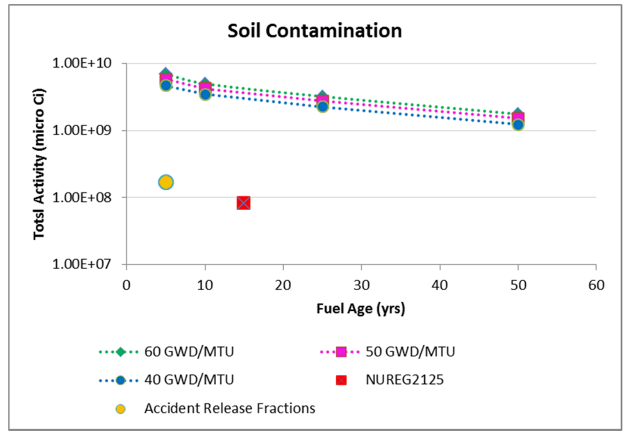

Fig. 11. Total Soil Activity within the Plume at the Time of Release from a Damaged PWR SNF.

Figure 12 shows the total soil activity as a function of PWR fuel burnup and age. There is a steep increase in dose for recently discharged high burnup fuel. The soil activity is more a function of the fuel burnup than the fuel age. This can be seen as a steep increase in the activity for the high burnup fuel.

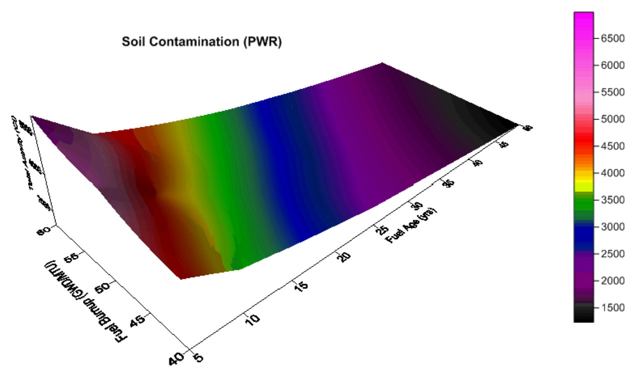

Fig. 12. Soil Activity as a Function of Fuel Age and Burnup (Release from PWR SNF).

As it was noted in NUREG-2125, there are no acute health effects due to the releases related to the accident. The number of early fatalities and morbidity is zero for all the considered releases. However, there are acute health effects which exist for the release fractions considered in this analysis. Figure 13 shows the number of early fatalities as a function of age for a PWR fuel with different burnups. Note that the number of fatalities is calculated for the population density of 1 person per km2. The number of early fatalities exhibits a similar behavior as the doses. There is a steep increase in the number of early fatalities for a recently discharged high burnup fuel.

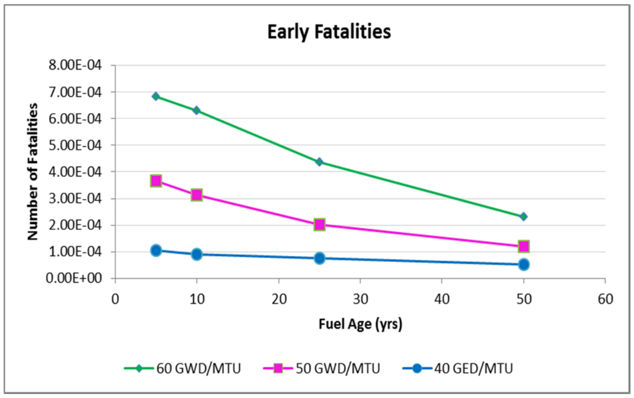

Fig. 13. Number of Early Fatalities Due to Release from a Damaged PWR SNF.

Figure 14 compares the MEI doses for PWR and BWR fuel. The tendency is very similar for both types of fuel. The doses related to PWR fuel are 1.22 to 1.34 times higher than the doses related to BWR fuel, which is proportional to the difference in the total activities of the PWR and BWR casks.

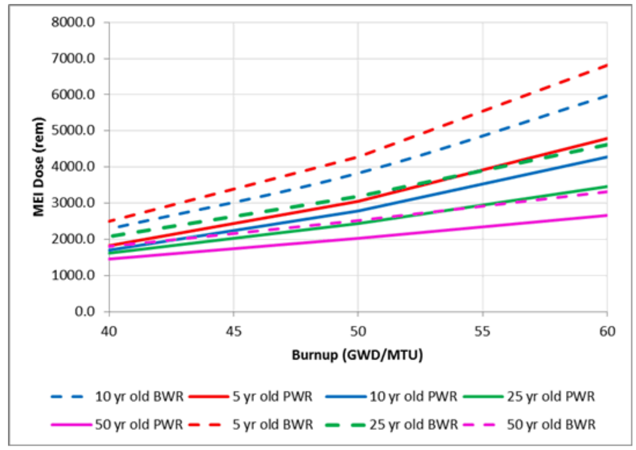

Fig. 14. MEI Dose from Damaged PWR and BWR SNF.

Figure 15 shows the percent contribution by each radionuclide to MEI dose from 5-year-old 60 GWD/MTU burnup and 50-year-old 40 GWD/MTU burnup PWR fuel for the following weather conditions:

- stability class F with wind 0.5 m/s and no precipitation
- stability class D with wind 4.7 m/s and no precipitation
- stability class D with wind 4.7 m/s and moderate precipitation

-

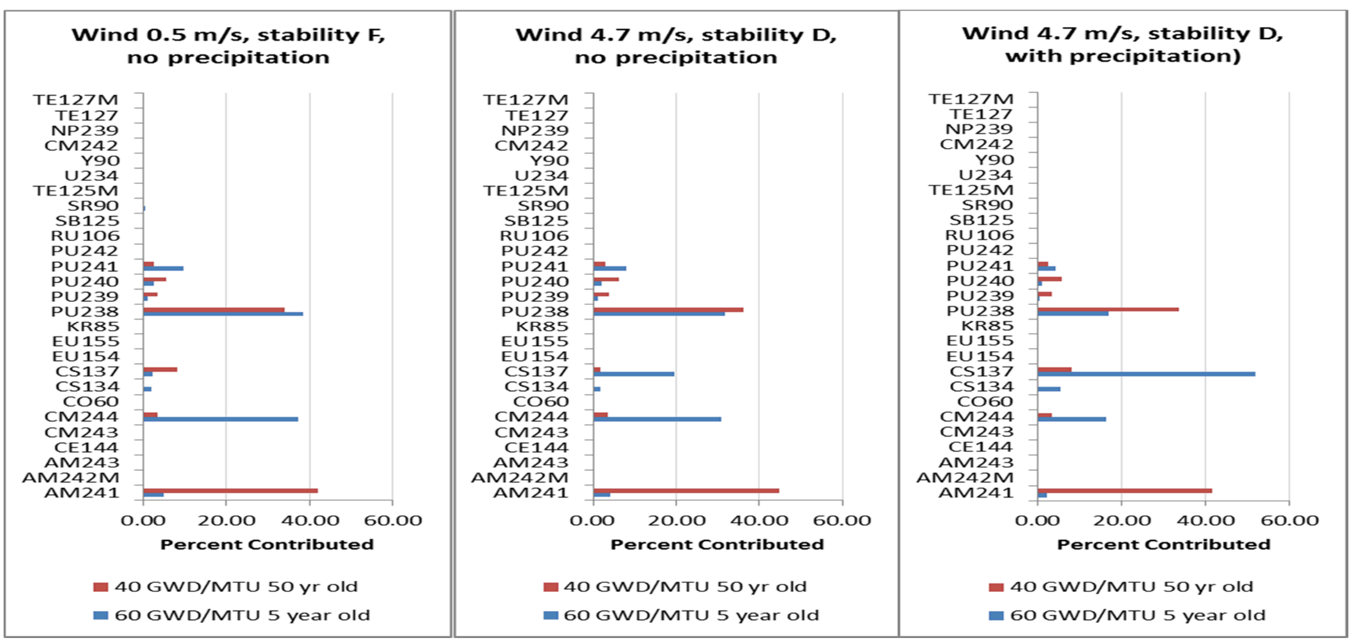

Fig. 15. Radionuclide Contributions to the MEI Dose from Damaged PWR SNF for Different Weather Conditions.

The major difference for the recently discharged high burnup fuel is from the Cs-137 contribution to the total dose, which is significantly large for the stability D class with the higher wind than for the stability class F with low wind. As a result, the plume is larger, MEI doses are smaller, and the soil activity is higher in the second case (stability class F weather). The precipitation (plume washout) increases the Cs-137 (volatile group) contribution and further amplifies these impacts.

The differences are small for the older low burnup fuel. However, similar impacts were observed, but with much smaller amplitudes. Note that the same weather conditions were considered in NUREG-2125 and the conclusion was made that the weather has low impact. This difference to NUREG-2125 is because the considered fuel was older and had a lower burnup.

Figure 16 compares the contributions of the different exposure pathways to the total MEI dose for the different scenarios.

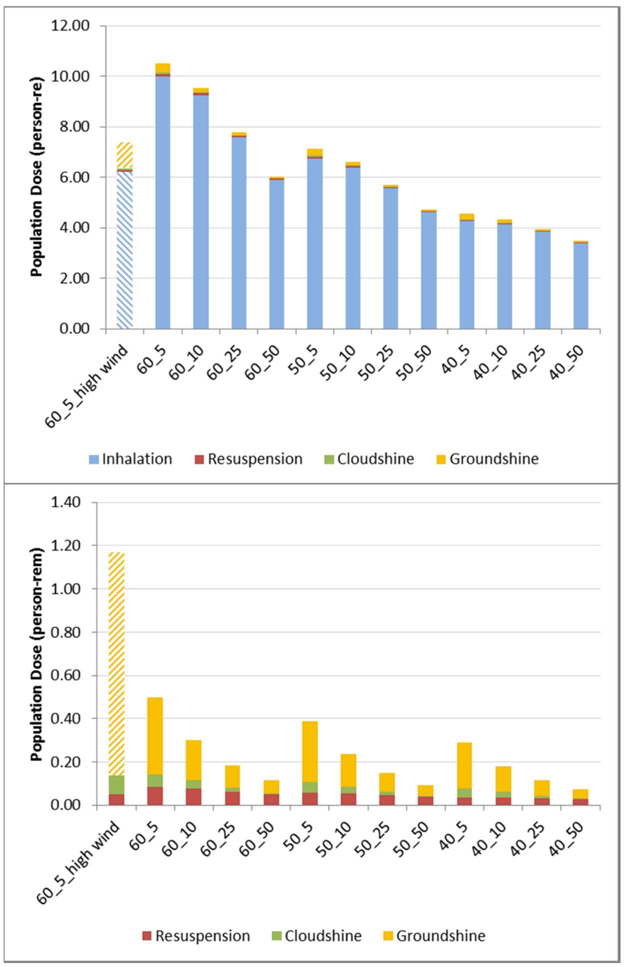

Fig. 16. Exposure Pathway Contributions to the MEI Dose in Different PWR Fuel Scenarios.

The major exposure pathway in all scenarios is the inhalation pathway. There is a smaller contribution from the groundshine (external) pathway. The external exposure pathway (ground-shine) contribution increases with higher wind and precipitation. The contributions of cloudshine and inhalation from resuspension pathway are significantly smaller.

Because STAGE 1 uses dynamic path planning (which enables entities in the simulation to navigate within the environment to achieve their objectives without extensive or prescriptive preplanning), ADAPT is used to link the consequences outlined above to that ability to combat a (near) immediate adversary attack on the derailed train transporting SNF. Traditional STAGE-based security analysis (Ref. 3) evaluates expected security system component performance (e.g., a sensor's probability of detection or the time required for a response force member to mobilize) against an expected set of adversary capabilities. This approach, however, does not explicitly consider a security-scenario playing out in a hazardous environment—let alone an environment that dynamically changes from 'normal' to 'hazardous.' In this scenario, once the train derails, these expected security system component performance values are no longer valid in the newly created 'hazardous' (e.g., radioactive) environment surround the derailed train. More specifically, the higher the MEI and/or soil contamination, the slower the on-board security force is able to engage with an adversary. Similarly, as the MEI and/or soil contamination increase, the ability for the response force to adequately engage the adversary (e.g., weapons usage proficiency) and the performance reliability of security technologies decrease. Here, the RADTRAN-calculated consequence values are translated via ADAPT into degradation factors for STAGE simulation inputs. Interestingly, in contrast, the train derailment provided an advanced alert to local law enforcement agents—resulting in a faster deployment of secondary and tertiary waves of response to derailment site (before they were notified of the adversary action). An increased initial response time, plus a degraded response force capability, results in an increased ability for an adversary to either steal SNF or conduct a sabotage operation to exacerbate the radiological release. Ultimately, ADAPT provides a framework by which the interdependencies between traditional safety (e.g., RADTRAN) and security (e.g., SNL uses of STAGE) analysis result in increased (e.g., radiological release degrading ability to stop an adversary attack) consequences related to SNF transportation.

V. CONCLUSIONS

By using ADAPT to drive both the RADTRAN and STAGE calculations, it is possible to generate thousands of simulation runs to better characterize and capture the uncertainties associated with this scenario. Each individual RADTRAN run can then be translated into STAGE in order to determine the effects that a specific

[1] Due to unforeseen software complications, the STAGE analysis was incomplete at the time of writing this article. In that vein, the proceeding discussion is based on the logic used to initiate the analysis. The authors apologize for a lack of formal results to support this discussion.

evolution of an accident sequence would have on the resultant security scenario without needing to cluster results beforehand. Follow-on analysis will also incorporate using different branching points within the DET (e.g., different procedures related to real-world safety and security expectations) to evaluate effects on scenario-related consequences.

This analytical approach, then, helps explicitly identify interdependencies (and their cumulative consequence-related effects) between traditional safety and security analysis. For example, the MEI dose in conjunction with the soil contamination inform the size of the region around the cask that will need to be cordoned off, while release from the plume will affect the evacuation efforts. If the release is large enough, the efforts of security personnel to prevent access to the cask could be conflicted by the contamination of the area. Further, as part of the transition between RADTRAN and STAGE, several parameters of the RADTRAN analysis need to be translated into their security effects (summarized in the preceding section). More specifically, because in the proposed scenario potential radioactive releases from damage to the cask may create a hazardous environment near the cask which could be potentially difficult for security personnel to operate within for extended periods of time.

In addition, this analytical approach helps identify new mechanisms by which to identify potentially-conflicting and/or beneficial responsibilities. For example, applying the MEI safety metric to STAGE analysis determines the degree of immediate lifesaving response that is required after the derailment. If the on-board response personnel are cross-trained in emergency medicine, there may be a conflict (which can be modeled in using the 'if/then' logic to prioritize tasks within STAGE) between performing lifesaving and combatting adversary tasks during a security incident. On the other hand, there are also gains from local law enforcement being alerted to the safety event, and arriving more rapidly than expected by the security analysis. The nature of these potential interdependencies indicates that the form of cross-training given to onboard personnel is important—namely that the benefits of ensuring that security forces are also trained as first responders in the event of a safety accident may result in miscommunications and unclear priorities that hamper security efforts. Future work related to this project includes: a more in-depth mapping of RADTRAN outputs to STAGE inputs, mapping STAGE outputs to RADTRAN inputs, evaluating the related recursivity and expanding this analysis to include safeguards modeling using PR-CALC.[2]

This paper summarizes one example of how a time dependent, dynamic control theoretic complex system model can provide both new approaches to mitigate increasingly complex risk and methodologies to assess, manage, mitigate and eliminate the complex risks of SNF transportation.

VI. REFERENCES

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