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A Methodology to Quantify the Release of Spent Nuclear Fuel from Dry Casks during Security-Related Scenarios

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Abstract

Assessing the risk to the public and the environment from a release of radioactive material produced by accidental or purposeful forces/environments is an important aspect of the regulatory process in many facets of the nuclear industry. In particular, the transport and storage of radioactive materials is of particular concern to the public, especially with regard to potential sabotage acts that might be undertaken by terror groups to cause injuries, panic, and/or economic consequences to a nation. For many such postulated attacks, no breach in the robust cask or storage module containment is expected to occur. However, there exists evidence that some hypothetical attack modes can penetrate and cause a release of radioactive material. This report is intended as an unclassified overview of the methodology for release estimation as well as a guide to useful resource data from unclassified sources and relevant analysis methods for the estimation process.

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ABBREVIATIONS/DEFINITIONS

AED	Aerodynamic equivalent diameter
BCL	Battelle Columbus Laboratories
BWR	Boiling water reactor
CRUD	Chalk River unidentified deposit
DOE	Department of Energy
EF	Enrichment factor
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (Translation: Society for Plant and Reactor Safety)
GSD	Geometric standard deviation
GWd/MTHM	Gigawatt-days per metric ton of heavy metal
HED	High energy device
INEL	Idaho National Engineering Laboratory
IRSN	Institut de Radioprotection et de Sûreté Nucléaire (Translation: Radioprotection and Nuclear Safety Institute)
MMD	Mass median diameter
PWR	Pressurized water reactor
SFR	Spent fuel ratio
SNF	Spent nuclear fuel
SNL	Sandia National Laboratories

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1 INTRODUCTION

1.1 Background

Assessing the risk to the public and the environment from a release of radioactive material produced by accidental or purposeful forces/environments is an important aspect of the regulatory process in many facets of the nuclear industry. In particular, the transport and storage of radioactive materials is of particular concern to the public, especially with regard to potential sabotage acts that might be undertaken by terror groups to cause injuries, panic, and/or economic consequences to a nation. For many such postulated attacks, no breach in the robust cask or storage module containment is expected to occur. However, there exists evidence that some hypothetical attack modes can penetrate and cause a release of radioactive material.

1.2 Objective

While a detailed analysis of a security scenario that causes release is classified, the evaluation process may be described in a generic manner that identifies the principal mathematics and assumptions behind the determination of a release fraction. That framework together with a compendium of experimental data, analyses, and analytical tools provide a useful understanding of potential consequences of many scenarios of concern. This report is intended as an overview of the methodology for release estimation as well as a guide to useful resource data from unclassified sources and relevant analysis methods for the estimation process.

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2 DETERMATION OF RELEASE FRACTION

This chapter presents the general methodology and terminology associated with the determination of release fractions from radiological casks subjected to malevolent events. Previous models proposed by the Department of Energy (DOE) and the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) (Translation: Radioprotection and Nuclear Safety Institute) are presented along with the current model being used to determine source terms for security-related scenarios.

2.1 Previous Source Term Models

The DOE, in its Handbook of Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities, provides a general formulation for estimation of the source term for released airborne materials as shown below.

$$\text{Source Term} = \text{MAR} \cdot \text{DR} \cdot \text{ARF} \cdot \text{RF} \cdot \text{LPF}$$

where

MAR = Material-at-Risk (Curies or grams)

DR = Damage Ratio

ARF = Airborne Release Fraction

RF = Respirable Fraction of the ARF, and

LPF = Leakpath Factor

The first four terms define the amount of respirable material created within the container where the material was located. Among these,

- DR relates to the damaged volume created by an attack as compared to the total volume of material at risk.
- ARF relates to how the energy imparted by the attack mode is converted to particles small enough to be airborne within the containment volume and, therefore, available for release should there be an escape path to the environment.
- RF is a measure of what fraction of ARF is in a size range that would make it respirable and thus contribute to long term radiological impacts if there is a release to the environment.
- LPF is intended to capture the effect of voiding a fraction of the container volume carrying particles to the environment, deposition within the container, and deposition that occurs along the path to release to the environment. The last two of which are particle size dependent.

Most of the content of the "Handbook" is devoted to providing values for ARF and RF for materials and configurations relevant to DOE's production facilities. This handbook can be an important source of data for the calculations of interest here.

In a recent report by Loiseau, et al. from the IRSN, the same basic formulation is adopted, except for the LPF factor.

$$\text{Source Term} = \text{MAR} \cdot \text{FDAM} \cdot \text{FSUS} \cdot \text{FRES} \cdot \text{FEXP} \cdot \text{FRED}$$

where

MAR = Material At Risk

FDAM = Damaged Fraction

FSUS = Fraction of material in suspension or in aerosol form

FRES = Respirable Fraction of FSUS

Their formulation replaces the LPF in the formulation of the "Handbook" with the last two factors in the multiplicative chain.

- FEXP = The fraction of the source that is expelled from the containment volume as a result of pressurization effects

- FRED = Reduction Factor that includes deposition processes and chemical reactions within the containment volume and deposition along the exit path way.

The two formulations differ only in the fact that the Loiseau formulation gives more specificity in the LPF than in the DOE handbook. This formulation also acknowledges the contribution of a pressurized gas within the container as a vehicle to force aerosols to the external environment.

By inspection it is clear that the "source term" is measured in units such as Curies or grams of a specific material or mixture. But frequently, the source term is non-dimensionalized using MAR, which might be the entire content of the volume or a discrete definable sub-part of the total content. A relevant example is a spent fuel cask or storage module in which the MAR might be the Cs-137 content of all the contained assemblies. The source term would then be expressed in Cs-137 curies. In that case the non-dimensional source term in the sense usually used would be $\text{ST} = \text{DR} \cdot \text{ARF} \cdot \text{RF} \cdot \text{LPF}$, where these factors are all related to Cs-137 release. This type of formulation has the advantage of allowing application to various initial loadings of the material of interest as long as the physical and chemical properties of the materials of concern are similar.

2.2 Current Source Term Model

The current source term model is described next. This model builds on the concepts of the DOE and IRSN models, but also incorporates a limited set of data from full-scale testing (Sandoval 1983 and Lange 1994). The representation of the model presented in this report has been simplified in order to demonstrate the logic underlying its derivation. Concepts such as gap fines in the outer rim of spent fuel and spallation of Chalk River Unidentified Deposits (CRUD) are ignored. However, the doses derived from the source terms defined by this model are unlikely to be significantly different from the full model as these enhanced treatments do not affect the release fractions of the transuranic radioisotopes, which dominate the dose. The transuranic species are treated the same in both models. Also, the reader is cautioned when analyzing casks containing boiling water reactor (BWR) fuel assemblies. All of the available large scale test data this is incorporated into this model was collected using pressurized water reactor (PWR) fuel assemblies. Previous treatments of BWR casks have considered the PWR equivalent cask when using this model.

Like the DOE model, the source term (ST) is proportional to the amount of material that is damaged during the causal event (m_{Dam}). This term is expressed in the DOE and IRSN models as the product of $MAR \cdot DR$ and $MAR \cdot FDAM$, respectively. However, this mass is defined as only the mass of affected fuel in the first cell for spent fuel applications. This limitation of the damaged mass is due to observations from full-scale testing conducted by Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) (Translation: Society for Plant and Reactor Safety) (Lange, 1994). The amount of this material that is aerosolized in the respirable range is defined by two separate source terms, ST_{Prompt} and $ST_{Delayed}$. The respirable range is defined as particles smaller than 10 μm aerodynamic equivalent diameter (AED).

The prompt source term (ST_{Prompt}) defines the amount of respirable particles ejected immediately from the cask as a result of the action of the high energy device (HED). This quantity is determined by multiplying the damaged mass by the release fraction determined from the full-scale transportation cask test performed by Sandoval at Sandia National Laboratories (RF_{SNL}) (Sandoval, 1983) and by the spent fuel ratio (SFR). The spent fuel ratio is defined as the amount of respirable aerosol generated by an assault on spent fuel divided the amount of respirable aerosol created by the exact same action on a surrogate material. Luna gives these values as RF_{SNL} = 7.6E-4 and SFR = 3 (Luna, 1999). Finally, the enrichment factor (EF) describes the enhanced release of volatile radioisotopes as a result of the HED interaction. Luna's interpretation of the SFR data from Battelle Columbus Laboratories (BCL) (Schmidt, 1982) and Idaho National Engineering Laboratory (INEL) (Alvarez, 1982) gives EF = 5 (Luna, 1999). Discussions about these and other parameters in the model are presented in more detail in the following sections.

The delayed source term ($ST_{Delayed}$) gives the amount of respirable particles that are transported by the gas inside the cask and the fuel itself. This term assumes the aerosol is well mixed with the internal gas. The release fraction (RF_{HED}) used to define this source term comes from examination of the work of Jardine, Molecke, and Alvarez in which they measured the amount of respirable aerosol that is generated when fracturing various brittle materials as a function of impact energy. Two deposition factors are applied to further modify this quantity. The first, $f_{Dep,\,Cask} = 0.7$, describes the amount of aerosols that deposit within the cask. The second fraction, $f_{Dep,\,Esc} = 0.4$, gives the amount of aerosols that deposit in the leak path during blowdown of the cask (Sprung, 2000). Finally, the last term in the equation defines the fraction of the respirable aerosol that will be swept from the cask by the internal gas. Note that this quantity approaches a value of 1 as the total volume of gas in the cask at STP becomes significantly larger than the cask free volume. To be clear, the cask free volume is defined as the volume of the empty cask interior minus the volume occupied by the fuel assemblies and support structures.

$ST_{Prompt} = \left(m_{Dam} \text{ or } A_{Dam}\right) \cdot RF_{SNL} \cdot SFR \cdot \left(EF\right)$, Note: Mass or activity are acceptable to define the source term

$$ST_{Delayed} = \left(m_{Dam} \text{ or } A_{Dam}\right) \cdot SFR \cdot \left(EF\right) \cdot \left(RF_{HED} - RF_{SNL}\right) \cdot \left(1 - f_{Dep,\, Cask}\right) \cdot \left(1 - f_{Dep,\, Esc}\right) \cdot \left[1 - \frac{V_{Free}}{V_{Free} + V_{Cask} + V_{Rods}}\right]$$

$ST_{Tot} = ST_{Prompt} + ST_{Delayed}$

where

m_{Dam} = mass of damaged fuel in first cell that is aerosolized in the respirable range

A_{Dam} = activity of damaged fuel in first cell that is aerosolized in the respirable range

SFR = spent fuel ratio = $\dfrac{RF_{Spent\ Fuel}}{RF_{Surrogate}}$, Aerosols AED < 10 μm

EF = enrichment, or enhancement, factor for volatile radioisotopes (e.g. Cs-137)

ST_{Tot} = total amount of respirable aerosols released from the container

ST_{Prompt} = respirable fraction of aerosols that are promptly released

$ST_{Delayed}$ = respirable fraction of aerosols that swept out of the container by gas blowdown

RF_{SNL} = respirable fraction of aerosols measured in the SNL full-scale test

RF_{HED} = respirable fraction of aerosols generated by the HED

$f_{Dep,\, Cask}$ = fraction of respirable aerosol deposited within the cask

$f_{Dep,\, Esc}$ = fraction of respirable aerosol deposited while escaping

V_{Free} = cask free volume (Physical free volume of gas defined within the cask)

V_{He} = "excess" volume of helium backfill gas at STP $\left(V_{He} = V_{Free} \cdot \left[\left(\frac{T_o}{T_i}\right) \cdot \left(\frac{P_i}{P_o}\right) - 1\right]\right)$

STP = standard temperature and pressure, T_o and P_o

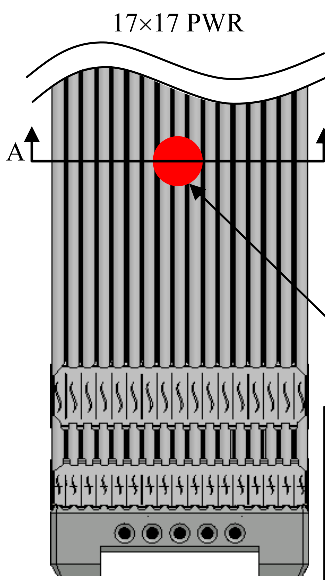
T_i = initial cask gas temperature

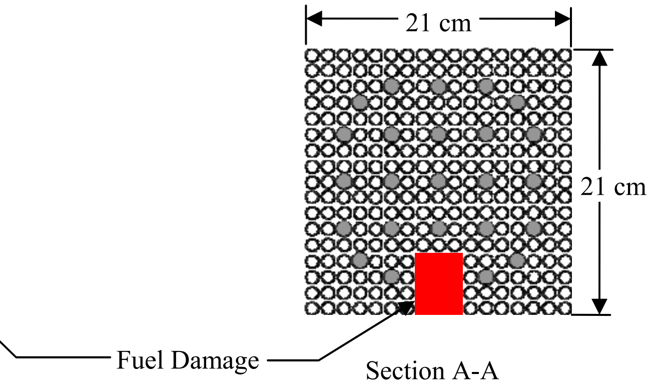
P_i = initial cask gas pressure

V_{Rods} = volume of plenum and decay product gases from damaged fuel pins at STP

2.2.1 Damage Mass (m_{Dam})

The action of an HED penetrating into radioactive material causes some characteristic volume of damage. This volume then translates to a mass of damaged material. For the example of spent nuclear fuel, the simplest way to calculate the damaged mass from the volume of damage is to assume the fuel exists in a homogenous state. Essentially, the mass or Curie content of the entire assembly for any particular radionuclide is assumed to exist in equal measure throughout the volume of the assembly where fuel exists, i.e., the assembly footprint times the length of fuel. Figure 2.1 shows the application of this concept to a typical 17×17 PWR fuel assembly. Note that the fuel pins are shown discreetly in the figure, but are assumed to be spread homogenously throughout the volume of the fuel assembly in the expression for m_{Dam}. In the example given in the figure inset, the mass of uranium from a typical, fresh fuel assembly is determined for a hole with a diameter of 3 cm and a depth of 4 cm. This principal can also be used for damage volumes other than cylinders as well as for determining Curies of release rather than mass.





$$\text{Inventory}_{\text{Dam}} = \frac{\text{Damage volume}}{(\text{Outer dimension})^2 \cdot \text{Fuel Length}} \cdot \text{Inventory}_{\text{Fuel}}\,(\text{mass or activity})$$

e.g., Assume a penetration creates a hole 3 cm in diameter with a depth of 4 cm

$$m_{\text{Dam}} = \frac{0.25 \cdot \pi \cdot (0.03\ \text{m})^2 \cdot 0.04\ \text{m}}{(0.21\ \text{m})^2 \cdot 3.7\ \text{m}} \cdot 460\ \text{kg U} = 8.0\text{E-}2\ \text{kg U}$$

Figure 2.1 Schematic showing hypothetical fuel damage in PWR 17×17 fuel assembly.

2.2.2 Sandia Release Fraction (RF_{SNL})

A test series was conducted in the early 1980's comprised of three tests, two quarter-scale and one full-scale (Sandoval, 1983). The full-scale test involved an obsolete, single assembly spent fuel cask that was attacked using a relatively large HED. This HED device fully penetrated one wall and the assembly but did not fully penetrate the second wall. The cask was loaded with a single, unpressurized 15×15 PWR fuel assembly using surrogate DUO_2 pellets in Zircaloy cladding. The cask was placed inside a tank that was sealed after the HED detonation to facilitate collection of aerosols produced by the attack. The amount of respirable aerosol collected amounted to 3 grams, which was about 7.6E-4 of the mass of DUO_2 within the hole in the spent fuel assembly produced by the action of the HED.

The quarter-scale test mimicked the dimensions of the full scale test with one exception: the fuel assembly was composed of 25 fuel rods that were shorter versions of those used for the full scale test. Attempts were made to scale the HED, but the resulting damage to the scaled cask and assembly was slightly greater than in the full scale test in that both walls of the cask were penetrated. The amount of respirable material released to the containment tank was approximately six times that for the full scale test relative to the mass of DUO_2 within the hole produced by the HED. This has been taken to mean that a through hole minimizes the amount of material trapped within the cask when the fast moving gas and particle cloud flows right through the cask.

2.2.3 Spent Fuel Ratio (SFR)

Spent fuel is highly fractured and is affected by neutron bombardment as well as growth and decay of fission products over time. Although spent fuel is mostly UO_2, the DUO_2 aerosol data from SNL, INEL, and BCL experiments must be scaled in order to scale to the results expected from spent fuel. Note that the DUO_2 surrogate material in these tests was constructed of unbroken pellets, further necessitating the use of a correcting factor. A factor referred to as the Spent Fuel Ratio (SFR) was developed in order to link results for spent fuel and DUO_2 surrogates. Values for SFR that were considered by Sandoval for use in his source term report (Sandoval, 1982) ranged from about 0.4 to 6.

Using data from INEL (Alvarez, 1982), BCL (Schmidt, 1982), and Kraftwerk Union (Ruhmann, 1985) estimates of SFR were developed (Luna, 1999 and Luna, 2004). These reviews of previously available data suggest that the value SFR is approximately 3 but had a possible range of 1 to 12. Luna postulates that a SFR smaller than unity is implausible given the highly fractured condition of spent fuel compared to the relatively solid DUO_2 surrogate pellets.

2.2.4 Enrichment Factor (EF)

Certain radioactive fission products, primarily Cs-134 and Cs-137, may volatilize at temperatures readily achieved by interaction with a HED. The use of an enrichment factor attempts to account for this phenomenon. Enrichment factors are reported in both the SFR reports by BCL (Schmidt, 1982) and INEL (Alvarez, 1982). These factors appear to be size dependent and range from EF ≈ 1 to 11 for particles AED > 4 μm and < 0.5 μm, respectively. An intermediate value of EF = 5 was adopted for the Yucca Mountain evaluation and is accepted for this report as well (Luna, 1999).

2.2.5 Release Fraction from HED (RF_{HED})

The amount and size distribution of the particles produced by impacts delivered to brittle materials have been the subject of several experimental programs related to nuclear fuel and nuclear waste disposal forms. A primary study was performed at Argonne National Laboratory (Jardine, 1982). For these experiments, a single cylindrical sample was placed between two hardened steel plates inside a sealed sampling chamber. Each specimen received a dynamic impact by dropping a mass from a predetermined height. The resulting particulate was diameter-classified to produce a size distribution expressed in log-normal terms with a mass median diameter (MMD) and a geometric standard deviation (GSD). Over a wide variety of brittle materials (glasses, ceramics, rock, cements) with a factor of 2 to 3 in material density, the fraction of the total pellet mass in particles smaller than 10 μm AED was directly related to the energy density (J/cc).

Another study of particular interest is the work conducted at Sandia National Laboratories to re-examine the spent fuel ratio (Molecke, 2006). Although no tests were conducted using actual spent fuel, a large number of tests were performed using surrogates, CeO_2 and DUO_2, to measure the mass fraction of respirable particles produced from interaction with HED's. These results represent the largest database most relevant to the energy densities of interest. Finally, a small number of results are available from the original SFR determination efforts (Alvarez, 1982).

All these data are plotted in Figure 2.2. A power-law fit was applied to the data (black line). A large amount of scatter is evident about the curve fit. An upper and lower confidence interval were defined by adding and subtracting, respectively, two times the standard error of the line to the curve fit constant. The hashed red portion in the upper right of the plot shows the region of interest for HED assault scenarios. Previous analysis of this data by Luna indicated that 5% of the affected, or damaged, mass would be in the respirable range. As shown on this plot, this estimate is still valid, but any value of about 0.7 to 13% is also possible. Given that no observed respirable percentage in this dataset exceeded 5%, the estimate of 5% is maintained for the remainder of this report.

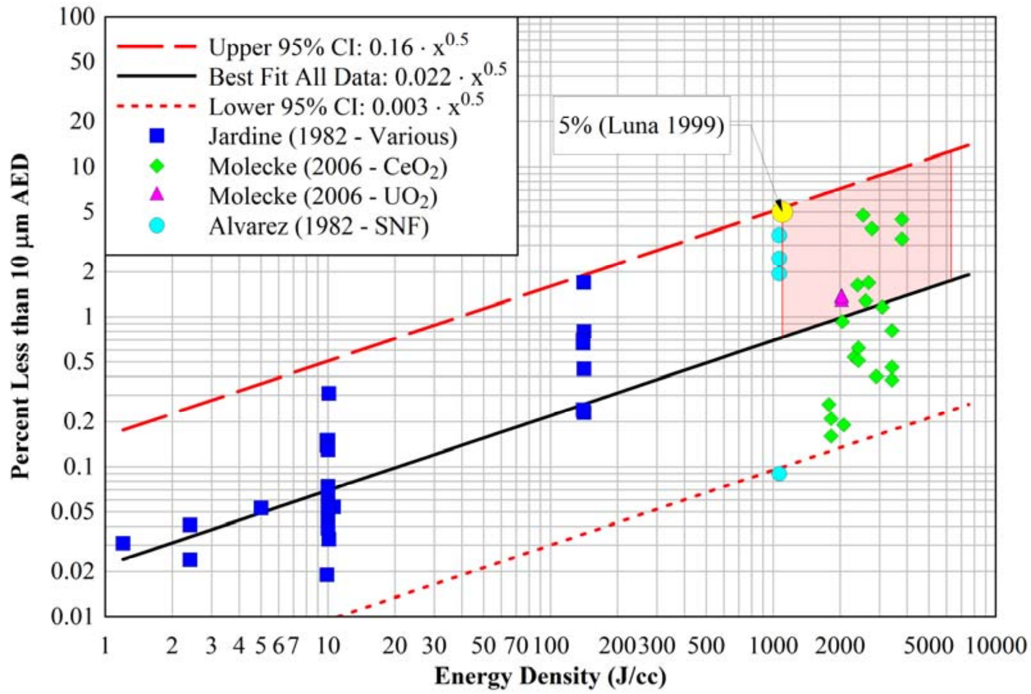


Figure 2.2 Mass of aerosols with AED < 10 µm normalized by affected mass as a function of energy density.

2.2.6 Deposition Fraction in the Cask ($f_{Dep,\ Cask}$)

In order to complete the model empirically, an estimate of the amount of respirable aerosol deposited in the cask was needed. Because this is an empirical model the fraction deposited was varied between 0 and 1 to determine the deposition fraction that optimized the prediction of the GRS full-scale, SNL full-scale, and SNL quarter-scale results.

For a 5% respirable generation (RF_{HED}) from the initial energy deposition, the cask deposition fraction needed to be approximately 0.7, i.e., about 30% of the respirable aerosol originally created was available for release through the entrance hole in the cask. With greater or smaller respirable aerosol creation, the deposition fraction would have to be correspondingly larger or smaller in order to best match the five experiments that served as the database for determining $f_{Dep,\ Cask}$.

2.2.7 Deposition Fraction while Escaping ($f_{Dep,\ Esc}$)

The fraction of respirable aerosols that deposit while escaping from the cask is based on estimates from Figure 7.11 in NUREG/CR-6672 (Sprung, 2000). This estimation is based on the amount of time the aerosols for various radioactive species are resident in the cask. This residence time is in turn proportional to the size of the leak path. The deposition fraction asymptotically approaches constant values for large breaches in the cask. The value chosen for the current model treatment is equivalent to the average deposition fraction for all species with the exception of noble gases, or $f_{Dep,\ Esc}$ = 0.4. The deposition fraction for the different species varies between 0.35 to 0.5.

2.2.8 Aerosol Size Distributions

When inputting the source terms into dose consequence/dispersion models, some codes allow the user to define aerosol sizes. Figure 2.3 shows the cumulative mass fraction as a function of AED for the BCL spent fuel data (Schmidt, 1982) and an aerosol distribution described by a MMD = 1.5 μm and GSD = 2.5. This choice of size parameters represents the highest possible percentage of the smallest aerosols based on the data and is therefore a slightly conservative choice for aerosol transport. An aerosol with MMD = 1.5 μm and GSD = 2.5 is recommended for dispersion modeling of the respirable source terms provided by this model.

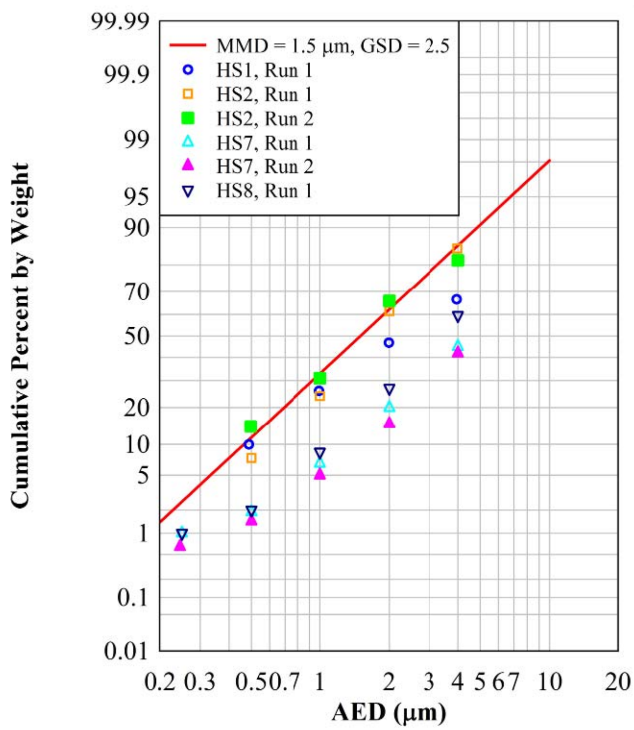


Figure 2.3 Aerodynamic particle size distributions from spent fuel acted on by a high energy device.

Data reproduced from Schmidt (1982) Figure 5-4.

2.3 Model Comparisons with GRS Results

The results obtained in the GRS test series (Lange, 1994) are compared to the current model predictions for these test conditions are shown in Table 2.1. The model predicts the measured respirable aerosol to within an error of 36%. The model also tends to overestimate the source term. The fact that the model reasonably predicts the GRS results is no guarantee that it is a reliable model for general use. However, incorporation of the Sandoval data and a reasonable phenomenological model provides a somewhat stronger level of confidence in its use.

Table 2.1 Comparison of the GRS test results to the current model predictions.

Test Identifier	Test (g)	Model (g)	Error (%)
GRS 1	1.05	1.02	2.9
GRS 2	0.962	1.31	-36
GRS 3	0.375	0.422	-13

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3 SUMMARY

The heuristic model developed to determine the source term from penetrations to dry casks resulting in damage to the spent fuel has been documented in this report. This model captures the current, best understanding of the complex physics involved with high energy devices (HED) acting on spent nuclear fuel. The model also makes use of the limited database available from large-scale testing conducted with DUO_2 surrogates. Some simplifications are invoked for presentation purposes in this report but are not expected to significantly impact the dose calculated from the model given here and the full model.

Table 3.1 summarizes the model parameters. A range of possible values is also defined for each parameter where available from the original source materials. As evident in the table, many of these variables have ranges of an order of magnitude or greater. However, the best estimate values shown in the table have been shown capable of reproducing the releases from the GRS full-scale tests within 36%.

Table 3.1 Summary of model parameters.

Parameter	Best Estimate	Possible Range	Source
$RF_{\mathbf{SNL}}$	7.6E-4	Undefined	Sandoval 1983, Luna 1999
$RF_{\mathbf{HED}}$	0.05	0.007 – 0.13	Alvarez 1982, Jardine 1982, Molecke 2006
SFR	3	0.4 – 12	Alvarez 1982, Schmidt 1982, Sandoval 1983, Ruhmann, 1985, Luna 1999, Luna 2004
EF	5	1 – 11	Alvarez 1982, Schmidt 1982
$f_{\mathbf{Dep,\ Cask}}$	0.7	Undefined	Sandoval 1982, Lange 1994
$f_{\mathbf{Dep,\ Esc}}$	0.4	0.35 – 0.5	Sanders 2000

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APPENDIX A EXAMPLE CALCULATION

Problem Statement: A cask with 24 PWR assemblies that have an effective burnup of 45GWd/MTHM and are 10 years from offload is attacked resulting in a penetration with a diameter of 3 cm and a depth of 4 cm. Determine the source terms for Am-241 and Cs-137.

Given: The cask is filled with helium at P_i = 5.1 bar (5 atm) and T_i = 600 K. The free volume within the cask is 6 m^3. Each fuel rod contains 7.5E-4 m^3 of gas at STP. Assume the gas in the cask at the end of the depressurization is at STP, P_o = 1.0 bar (1 atm) and T_o = 298.15 K.

The rod-to-rod pitch is approximately 1.26 cm.

The inventory of a single PWR fuel assembly 10 years from offload with a final burnup of 45 GWd/MTHM is given in Table A.1.

Table A.1 Inventory of the most important radioisotopes for dose consequences in a single PWR fuel assembly at 10 years from offload and a burnup of 45 GWd/MTHM.

Radionuclide	PWR Inventory (Ci)
AM-241	1130
CE-144	75
CM-244	2653
CO-60	2326
CS-134	4353
CS-137	51140
EU-154	3209
KR-85	2938
PU-238	2625
PU-239	128
PU-240	128
PU-241	51220
RU-106	315
SR-90	35170
Y-90	35180

Americium-241:

$$\frac{A_{Dam}}{A_{Fuel}} = \frac{V_{Dam}}{V_{Fuel}} = \frac{0.25 \cdot \pi \cdot (0.03\ \text{m})^2 \cdot 0.04\ \text{m}}{24 \cdot (0.21\ \text{m})^2 \cdot 3.7\ \text{m}} = 7.2\text{E-6}, \text{ where A is activity in Curies}$$

$$\frac{ST_{Prompt}}{A_{Fuel}} = \frac{A_{Dam}}{A_{Fuel}} \cdot RF_{SNL} \cdot SFR \cdot (EF) = (7.2\text{E-6}) \cdot (7.6\text{E-4}) \cdot (3) \cdot (1) = 1.6\text{E-8}, \text{ where EF} = 1$$

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$$\frac{ST_{Delayed}}{A_{Fuel}} = \frac{A_{Dam}}{A_{Fuel}} \cdot SFR \cdot (EF) \cdot (RF_{HED} - RF_{SNL}) \cdot (1 - f_{Dep,\,Cask}) \cdot (1 - f_{Dep,\,Esc}) \cdot \left[1 - \frac{V_{Free}}{V_{Free} + V_{He} + V_{Rods}}\right]$$

$$V_{He} = V_{Free} \cdot \left[\left(\frac{T_o}{T_i}\right) \cdot \left(\frac{P_i}{P_o}\right) - 1\right] = 6\text{ m}^3 \cdot \left[\left(\frac{298\text{ K}}{600\text{ K}}\right) \cdot \left(\frac{5.07\text{ bar}}{1.01\text{ bar}}\right) - 1\right] = 9\text{ m}^3$$

$$V_{Rods} = \frac{(\text{Diameter} \cdot \text{Depth})_{Dam}}{\text{pitch}^2} \cdot V_{Single\ Rod} = \frac{(0.03\text{ m} \cdot 0.04\text{ m})}{(0.0126\text{ m})^2} \cdot (7.5\text{E-4 m}^3) = 5.7\text{E-3 m}^3$$

$$\left[1 - \frac{V_{Free}}{V_{Free} + V_{He} + V_{Rods}}\right] = \left[1 - \frac{6\text{ m}^3}{6\text{ m}^3 + 9\text{ m}^3 + 5.7\text{E-3 m}^3}\right] = 0.6$$

$$\frac{ST_{Delayed}}{A_{Fuel}} = (7.2\text{E-6}) \cdot (3) \cdot (1) \cdot (0.05 - 7.6\text{E-4}) \cdot (1 - 0.7) \cdot (1 - 0.4) \cdot [0.6] = 1.1\text{E-7}$$

$$\frac{ST_{Tot}}{A_{Fuel}} = \frac{ST_{Prompt} + ST_{Delayed}}{A_{Fuel}} = (1.6\text{E-8}) + (2.7\text{E-7}) = 1.3\text{E-7}$$

$$ST_{Tot} = \frac{ST_{Prompt} + ST_{Delayed}}{A_{Fuel}} \cdot A_{Fuel} = (1.3\text{E-7}) \cdot (24\text{ Assemblies}) \cdot \left(\frac{1130\text{ Ci}}{\text{Assembly}}\right) = \boxed{3.5\text{E-3 Ci}}$$

Cesium-137:

$$\frac{A_{Dam}}{A_{Fuel}} = 7.2\text{E-6}, \ (\text{see Americium-241 calculation for details})$$

$$\frac{ST_{Prompt}}{A_{Fuel}} = \frac{A_{Dam}}{A_{Fuel}} \cdot RF_{SNL} \cdot SFR \cdot (EF) = (7.2\text{E-6}) \cdot (7.6\text{E-4}) \cdot (3) \cdot (5) = 8.2\text{E-8, where EF} = 5$$

$$\frac{ST_{Delayed}}{A_{Fuel}} = \frac{A_{Dam}}{A_{Fuel}} \cdot SFR \cdot (EF) \cdot (RF_{HED} - RF_{SNL}) \cdot (1 - f_{Dep,\,Cask}) \cdot (1 - f_{Dep,\,Esc}) \cdot \left[1 - \frac{V_{Free}}{V_{Free} + V_{He} + V_{Rods}}\right]$$

$$V_{He} = 9\text{ m}^3 \ (\text{see Americium-241 calculation for details})$$

$$V_{Rods} = 5.7\text{E-3 m}^3 \ (\text{see Americium-241 calculation for details})$$

$$\left[1 - \frac{V_{Free}}{V_{Free} + V_{He} + V_{Rods}}\right] = 0.6 \ (\text{see Americium-241 calculation for details})$$

$$\frac{ST_{Delayed}}{A_{Fuel}} = (7.2\text{E-6}) \cdot (3) \cdot (5) \cdot (0.05 - 7.6\text{E-4}) \cdot (1 - 0.7) \cdot (1 - 0.4) \cdot [0.6] = 5.7\text{E-7}$$

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$$\frac{ST_{Tot}}{A_{Fuel}} = \frac{ST_{Prompt} + ST_{Delayed}}{A_{Fuel}} = (8.2E\text{-}8) + (5.7E\text{-}7) = 6.6E\text{-}7$$

$$ST_{Tot} = \frac{ST_{Prompt} + ST_{Delayed}}{A_{Fuel}} \cdot A_{Fuel} = (6.6E\text{-}7) \cdot (24 \text{ Assemblies}) \cdot \left(\frac{51140 \text{ Ci}}{\text{Assembly}}\right) = \boxed{0.81 \text{ Ci}}$$

Observations: For casks pressurized beyond $P_i > 2$ bar, the delayed release dominates the total source term. Also, the gas released from the rods is negligible compared to the helium from the cask for internal cask pressures greater than $P_i > 2$ bar.

The enrichment factor of EF = 5 is primarily for isotopes of cesium but may also be applied to ruthenium-106. An enrichment factor of EF = 1 is appropriate for all other species in Table A.1.

If the contribution to the dose from krypton-85 is required, the analyst should use the ratio of the gas volume from damaged rods to the volume of gas from a single rod scaled to the entire inventory of Kr-85 as shown below. Here, the volume of gas from damaged rods is more proportional to the mid-plane area of the damage volume normalized by the square of the pitch.

$$ST_{KR85} = \frac{(\text{Diameter} \cdot \text{Depth})_{Dam}}{\text{pitch}^2} \cdot \frac{A_{Fuel}}{N_{Rods}}$$

$$ST_{KR85} = \frac{(0.03 \text{ m} \cdot 0.04 \text{ m})}{(0.0126 \text{ m})^2} \cdot \left(\frac{2938 \text{ Ci}}{264}\right) = \boxed{84 \text{ Ci,}} \text{ for the previous example problem}$$

Enhanced spallation and release of cobalt-60 contained in the CRUD is ignored in this modeling treatment but does not significantly affect the final dose.

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