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Radiological Source Terms Resulting From Sabotage to Transportation Casks

Final Report

Prepared by N. E. Miller, A. W. Fentiman, M. R. Kuhlman,
H. N. Ebersole, B. D. Trott, J. E. Orban

Battelle's Columbus Division

Prepared for
U.S. Nuclear Regulatory
Commission

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Prepared by
N. E. Miller, A. W. Fentiman, M. R. Kuhlman,
H. N. Ebersole, B. D. Trott, J. E. Orban

Battelle's Columbus Division
505 King Avenue
Columbus, OH 43201-2693

Prepared for
Division of Regulatory Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
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ABSTRACT

The Nuclear Regulatory Commission (NRC) promulgated a rule, 10 CFR 73.37, which established requirements for safeguarding shipments of spent fuel to reduce the risk from acts of sabotage of highly radioactive materials. After the rule became effective, experimental programs conducted by Battelle for the NRC and by Sandia for the DOE showed the consequences of an attack using explosives on a shipment of PWR spent fuel were significantly less than had been indicated by earlier analytical studies. As a result, NRC is considering modifying the safeguards requirements.

In support of NRC's efforts to modify the rule, Battelle has conducted additional experimental studies to evaluate the consequences of attacks on shipments of high-temperature gas-cooled reactor (HTGR) spent fuel, non-power reactor (NPR) spent fuel, and vitrified high-level waste (HLW). Model casks containing surrogates of the spent fuels or high-level waste were penetrated by the jet from a precision shaped charge. Air samples collected after each test were used to estimate the quantities of respirable material released after the cask was penetrated. Results of the tests were scaled by specially developed scaling factors to estimate the releases that may occur from attacks on full-sized shipments of the materials. It was concluded that the sabotage of shipments of HTGR spent fuel, NPR spent fuel, or vitrified HLW should have no greater consequences than those predicted for shipments of PWR spent fuel.

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FINAL REPORT
on
RADIOLOGICAL SOURCE TERMS RESULTING FROM
SABOTAGE TO TRANSPORTATION CASKS
to
U.S. Nuclear Regulatory Commission

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) promulgated a rule, 10 CFR 73.37, which establishes the requirements for safeguarding shipments of spent fuel. A major objective of this rule is to reduce the radiological risk from sabotage of shipments of spent fuel. After the rule became effective, an experimental program conducted for the NRC by Battelle^{(1)*} with real spent fuel in model shipping casks indicated that approximately 9 g of uranium and associated fission products might be released as airborne material from an attack on a truck shipment of pressurized water reactor (PWR) spent fuel. An independent study by Sandia⁽²⁾ with surrogate fuel in a full-size cask predicted a similar quantity of fuel released. Both programs indicated that the consequences of an attack using explosives on a shipment of spent PWR fuel were significantly less than had been indicated by earlier analytical studies. As a result, the NRC is considering modifying the safeguards requirements.

In support of NRC's efforts to modify the rule, Battelle also estimated the radiological source term that might result from attacks using explosives on shipments of other types of highly radioactive wastes. The results of that study⁽³⁾ indicated that potential releases from attacks on shipments of cladding hulls, activated power reactor components, ion-exchange resins from reactor accident cleanup, and "intermediate-level" wastes fixed in concrete would have negligible health effects to the maximally exposed

* References are listed at the end of this report.

individual and to the general public. The study concluded that slight, but measurable, health effects could occur from attacks on shipments of high-level wastes solidified in glass or in Synroc.

The safeguards requirements apply to any type of spent-fuel shipments from NRC-licensed facilities. The most common type of spent-fuel shipments to storage or disposal facilities expected in the future is from commercial light-water power reactors (LWR). The data from the PWR fuel studies⁽¹⁾ can be applied to most LWR fuel shipments. Other shipments will be made with spent fuel from a high-temperature gas-cooled reactor (HTGR) and from non-power reactors (NPR). Also, within 20 years there are expected to be shipments of high-level waste (HLW) to a Federal repository. Thus, the program reported here was planned to obtain experimental data on the potential releases of radionuclides from explosive attacks on shipments of HTGR spent fuel, NPR spent fuel, and vitrified HLW. Together with the information from previous programs, these data will provide a technical basis for confirming the adequacy of regulatory positions.

The program was divided into three tasks, one each for HTGR spent fuel, NPR spent fuel, and vitrified HLW. The technical approach used in this program was to measure the release of respirable material resulting from a shaped-charge attack on a scale-model shipping cask containing surrogate spent fuel or high-level waste. The data from these tests were extrapolated, using specially developed scaling factors, to estimate the release from a full-sized shipping cask. Radionuclide source terms were calculated for attacks on each type of shipment using the experimentally measured releases and typical spent fuel or HLW compositions. These source terms were compared with a similarly developed source term for attacks on an LWR fuel shipment to compare the potential consequences of attacks on shipments of these highly radioactive materials.

Section 2 of this report contains a description of the experimental apparatus and sampling methods. Sections 3, 4, and 5 contain characterizations of the spent fuel or HLW being simulated, descriptions of the surrogate materials and model casks, and results of each series of tests. A summary and the conclusions drawn from the results of the experiments are given in Sections 6 and 7, respectively. The methods used for scaling the experimental data to explosive attacks on full-sized shipping casks are described in Appendix A. A brief description of the method for developing the statistical model used for

treating the experimental data is given in Appendix B. The raw data and the methods used to analyze the samples for thorium or uranium are given in Appendix B. A brief description of the methods used to analyze the samples for thorium or uranium is given in Appendix C.

2.0 TECHNICAL APPROACH

All tests in the three tasks were conducted with the same apparatus and similar procedures in an explosion testing facility at Battelle's West Jefferson Site. The tests were similar to those done earlier with PWR spent fuel⁽¹⁾. Figures 1 and 2 show the experimental setup. The scale-model cask loaded with surrogate fuel was clamped to a fixed stand near the center of a 670-liter steel test chamber. The shaped charge was suspended outside of the chamber and aligned with a laser so that the jet would pass through an entrance hole in the chamber, through the model cask, out through an exit hole in the chamber, and into a massive steel stop block. Prior to entering the chamber the jet penetrated a 2-inch-thick steel conditioning block. The pressure pulse in the chamber was measured during each shot. The precision-shaped charges used in the tests are described in Appendix A.

Samples of the airborne material released during each test were collected inside the test chamber. Three types of sampling devices were used: (1) cascade impactors, (2) Millipore[®] filters, and (3) deposition plates.

Air was drawn from the chamber through two parallel cascade impactors during the period from 0.5 to 5 minutes after the charge was detonated. The cascade impactor consists of six stages stacked in a column. The largest particles ($>16\text{ }\mu\text{m}$)* are deposited in the first stage. Particles with a smaller diameter (between 8 and $16\text{ }\mu\text{m}$) are deposited in the next stage, and so forth. Particles $<0.5\text{ }\mu\text{m}$ are collected in a backup filter. The flow rate of the air through each sampler was measured prior to each test by inserting an in-line flow meter upstream from the sampler. Data from analyses of deposits in successive stages of the impactors were used to determine the particle size distribution of the aerosol released from the model cask to the chamber at the time of sample collection. The impactor data reveals the fraction of the total aerosol that is respirable and gives a measure of the respirable mass.

Air was also drawn from the chamber through Millipore[®] filters during three successive time periods. A separate filter was used for each interval.

* All particle sizes given in this report are in terms of equivalent aerodynamic diameters, i.e., the diameter of a particle of unit density that has the same settling rate as the particles of interest.

FIGURE 1. EXPERIMENTAL ASSEMBLY WITH SHAPED CHARGE, MODEL CASK, AND ALIGNMENT LASER

FIGURE 2. EXPERIMENTAL TEST CHAMBER WITH AIR SAMPLERS

Analyses of deposits on the filters were used to determine the aerosol concentration in the chamber during those time intervals. The airborne concentration was calculated from the mass of heavy metal in the sample, the flow rate of air through the sampler, and the duration of the sample period. The concentration was corrected for dilution of the air in the test chamber by air drawn into the chamber during the sampler operation and during previous sample collections. By extrapolating these data to the time of detonation, the concentration of particulates in the air immediately after the attack could be estimated.

Four deposition plates were placed on the floor of the test chamber. These plates collected debris from the cask that immediately fell to the floor of the test chamber as well as smaller particles that settled out of the air within an hour after the attack. The measured deposition on these plates was used to estimate the total release from the cask during the test. These data are used to qualitatively assess the validity of the tests and are not used to determine the release of respirable size material.

After each test a health-physics technician took smears from surfaces around the test area to check for contamination. The results from counting these smears showed that the deposition on the inner surface of the test chamber was low in comparison with that on the floor of the chamber. This indicates that no significant deposition of airborne material occurred on the inner wall of the test chamber.

Uranium or thorium was used as a tracer element to estimate the release of radionuclides that would occur with spent fuel or high-level waste materials. In all of the experiments and analyses, it was assumed that the release of heavy metal (uranium or thorium) in the surrogate spent fuel or high-level waste would be proportional to the radionuclides which would be released from real spent fuel or high-level waste. This assumption was shown to be valid in prior studies⁽¹⁾.

For the HTGR tests the tracer element in the surrogate spent fuel was thorium. The samples collected during the tests were analyzed for thorium by digesting each sample in a HNO_3/HF solution, evaporating to dryness on a planchet, and alpha counting the residue with a gas proportional counter. These were compared with standards prepared from thorium nitrate solutions

which also accounted for self-shielding depending on the mass of residue on the planchet. The surrogate NPR spent fuel and high-level waste glass used uranium as the tracer element. The particulate samples from these tests were digested in hydrochloric acid, added to a $\text{NaF}/\text{Na}_2\text{CO}_3$ flux and fused into pellets prior to being analyzed for uranium by a fluorometric technique.

The mass of tracer element in the respirable material released in each test was determined from the analyses of the samples collected in the Millipore® filters and the cascade impactors. An average release of respirable material was calculated from the data from all tests for each type of spent fuel or high-level waste.

The scaling of the experimental data from the model tests to the results of a sabotage attack on a full-scale shipping cask is based on the assumption that the respirable mass of radioactive material released from the cask is proportional to the momentum transferred from the weapon to the cask and contents. The momentum transferred was computed by the method described in Appendix A for the scale-model casks used in the experiments. The data was treated by a statistical model as follows: Let

R_i = predicted respirable mass (m.g. thorium, for example)
released in an attack on the i -th scale model

and

M_i = momentum transferred to model i .

For each type of cask (or fuel) we have

$$R_i = C M_i \quad ,$$

or

$$\ln(R_i/M_i) = \ln(C) \quad .$$

The quantity R_i represents the predicted mass of respirable material released from the model casks. However, the mass actually released on a single direct hit is a random quantity that must be characterized by a statistical distribution. From the experimental data estimates of the respirable masses

released from the model casks were obtained. For the j -th cask of model size i the estimate is given by

$$R_{ij} = V \cdot \hat{D}_{ij} \hat{P}_{ij}$$

where:

V = volume of the test chamber,

$\hat{D}_{ij}$ = estimated initial concentration of airborne material
in the test chamber,

and

$\hat{P}_{ij}$ = estimated fraction of the total airborne mass that
is respirable.

The concentration, $\hat{D}_{ij}$, is obtained by extrapolation of the filter data and the fraction $\hat{P}_{ij}$ is computed from the cascade impactor data.

These estimates contain two sources of variability: the shot-to-shot variability, due primarily to the differences in casks and charges used; and the variability due to errors in measurements. Based on the statistical assumptions, presented in Appendix B, we have the following model for estimating the proportionality constant C :

$$Y_{ij} = \ln(\hat{R}_{ij}/M_i) = \ln C + a_{ij} + e_{ij} \quad ,$$

where

Y_{ij} = the observed response from shot j of scale model i ,

a_{ij} = the random component due to shot-to-shot differences

and

e_{ij} = the random component associated with the measurement
error of the instruments.

The random error terms are assumed to be normally distributed.

To estimate the release from a full-scale attack the constant C is estimated by

$$\hat{C} = \exp(\bar{Y})$$

where $\bar{Y}$ is the average of the observed responses.

The average respirable mass for a full-scale attack is estimated by

$$\hat{R}_{fs} = \hat{C} M_{fs} ,$$

where M_{fs} is the calculated momentum transferred to a full-scale target.

The median respirable mass was selected as a quantity of interest because it represents the value that would be exceeded in exactly 50 percent of the attacks. According to the model developed in Appendix B, the distribution of releases follows a log-normal distribution. Thus the median is somewhat smaller than the arithmetic mean. The median of a log-normal distribution is usually estimated by the geometric mean.

For each set of experiments (HTGR, NPR, and HLW) 95 percent confidence intervals for the median respirable mass released in a full-scale direct hit were computed. The confidence intervals characterize the experimental uncertainty associated with our estimate of the median full-scale release.

An estimate of the shot-to-shot variability of the actual release in full-scale attacks was obtained in each experiment. This estimate was used to compute an "upper bound" on the respirable mass that would be realized in a single incident. This upper bound is an estimate of the respirable release amount that would be exceeded in approximately 2.5 percent of the direct hits on a full-scale target. Due to the relatively small number of tests that were performed, a conservative bound, having a high level of statistical confidence, was determined not to be practical. For this reason, the value given represents our best estimate of the highest level of respirable material that would be released in a "direct hit" attack on a full-scale shipment.

Estimates and confidence intervals for the proportionality constants (C) and the median respirable mass released in a full-scale attack ($\hat{R}_{fs}$) are presented, along with the upper bound on the amount released in a single attack, in the Test Results section (3.3, 4.3, and 5.3) describing the three sets of experiments (HTGR, NPR, and HLW). The development of the statistical model is presented, along with the data, in Appendix B.

A theoretical radionuclide content was established for each material from published data for HTGR spent fuel and for high-level waste, and from calculated data for a generic NPR spent fuel. The concentrations of the radionuclides were related to the concentrations of tracer element in the surrogate materials. Then, from the median mass of respirable tracer element calculated to be released in an attack on a full-scale shipment, the associated quantity of radionuclides that theoretically would be released was determined. By this method a radionuclide source term was developed for an attack on each type of shipment.

3.0 EXPERIMENTS WITH SURROGATE HTGR SPENT FUEL

This section presents information on HTGR spent fuel, describes the surrogate spent fuel used in our tests, and discusses the results of the tests.

3.1 HTGR Spent Fuel

The Fort Saint Vrain (FSV) Reactor is the only commercial high-temperature gas-cooled reactor (HTGR) which has shipped and is expected to ship spent HTGR fuel in this country in the near future. A FSV fuel element is a prismatic hexagonal block of graphite 14.2 in. across flats and 31.2 in. high with 0.5-in. diameter fuel holes and 0.63 in. diameter coolant holes drilled longitudinally in the block. The fuel holes hold rods containing pyrolytic-carbon-coated fuel particles in a graphite matrix. The 210 fuel holes and 102 coolant holes are located in a triangular array on a 0.74-in. pitch with approximately one coolant hole between two fuel holes. The coolant holes are open ended while the fuel holes are closed at both ends. The FSV HTGR uses a mixture of fissile and fertile material in the reactor core. The fissile material is highly enriched uranium carbide; the fertile material is thorium carbide.

3.2 Surrogate HTGR Spent Fuel and Model Cask

For our studies the staff of Fort Saint Vrain furnished us with rods containing only coated thorium carbide particles to simulate the fuel. (This surrogate fuel was attractive because it had mechanical properties almost identical to those of the real fuel but contained no fissile material which would require stringent accountability procedures.) These rods were inserted in holes in graphite blocks to simulate spent HTGR fuel. The graphite blocks, obtained from the same source and of the same grade as the FSV graphite, were 4 in. by 5 in. in cross section and 2, 4, or 8 in. thick. The three different thicknesses were designed to provide information on the attenuation of the jet

by graphite to assist in scaling the results to full-size fuel elements. Coolant and fuel holes in the model fuel element had the same size and location as in the FSV HTGR fuel elements. The location of holes for a 4 in. thick block is illustrated in the drawing in Figure 3. "Fuel" rods were placed only in the last two rows of fuel holes so similar quantities of thorium carbide would be affected in tests using all three thicknesses of graphite blocks. Coolant holes were drilled in each row but, except for the two rows of fuel, the other fuel holes were left undrilled since the solid graphite and graphite-matrix fuel rods were expected to respond similarly to the jet from the shaped charge.

Each graphite block was inserted into a 6-in. square steel tube with a 1/4-in. thick wall which simulated the containment vessel of the cask. Components of the surrogate fuel and cask are shown in the photograph in Figure 4. The vessel was closed at each end with 1/4-in. thick end plates welded to the tubing. Shielding blocks of 1.75-in. thick steel were tack welded to two opposite sides of the containment vessel.

3.3 Test Results

Ten tests were conducted with HTGR fuel. Initially, six tests were conducted with simulated HTGR cask models, two each with 2, 4, and 8-in. thick fuel blocks. In each of these tests, either experimental difficulties prevented the collection of a complete set of samples or the final sample analyses caused the data to be suspect. As a result, four additional tests were run with 4-in. thick fuel blocks. In each of the latter four tests, one or two sampling devices were incapacitated during the test by shrapnel puncturing an air line or shorting an electrical power line. However, the loss of some samplers is expected in this highly hostile environment and is one reason multiple samplers are employed.

The physical damage to the casks is illustrated by the photographs of the jet entrance and exit holes shown in Figures 5 and 6, respectively. It can be seen that the entrance and exit holes are approximately the same size. The splattering of metal on the entrance face, and the spalling of metal around the exit hole are typical occurrences in all of the tests. After penetrating the 2-in. thick steel conditioning block, 4-in. of steel in the

Note: The arrow indicates the location and direction of the path of the jet

FIGURE 3. LOCATION OF HOLES IN A FOUR-INCH-LONG SURROGATE HTGR FUEL BLOCK

FIGURE 4. SURROGATE EIGHT-INCH HTGR FUEL BLOCK AND CASK COMPONENTS

Fuel block was for a preliminary test that did not contain surrogate fuel pins.

FIGURE 5. JET ENTRANCE HOLE IN MODEL HTGR CASK

FIGURE 6. JET EXIT HOLE IN MODEL HTGR CASK

cask walls, and the graphite block, the jet still penetrated an average of 3 in. into the steel stop block.

Because of the malfunction of some sampling devices, as discussed above, only 16 of a possible 30 filter samples were collected during the 10 tests. Insufficient filter data was obtained for Tests 3 and 10 to calculate aerosol release. The value of one sample, 9A, was too low to be credible and thus it and sample 9C were not used. The data from the remaining 14 filter samples are shown in Figure 7 where they are plotted as the log of airborne concentration of thorium ($\mu\text{g/liter}$) versus the average time (min.) after the shot in which the sample was collected. The results of Tests 5 and 6 with the 8-in. thick surrogate fuel blocks show lower values than the other tests indicating the jet may have been attenuated more by penetrating the additional graphite. However, because of the overall scatter in the data, it is not known if this is a real effect. The data from Tests 1 and 2 with the 2-in. thick blocks are similar to the data from the 4-in. thick blocks.

The log of the airborne concentration is expected to vary linearly with settling time after the shot. By extrapolating the data from each test to zero time using the restricted least squares method, the airborne concentration of thorium immediately after the shot, D_{ij} was estimated. From the initial thorium concentration in the 670-liter-test chamber the total mass of airborne thorium was calculated. The data from the cascade impactors are plotted in Figure 8 as particle diameter versus cumulative percent by weight on log-probability graph paper. From the data for each test, the fraction of the airborne material which was respirable, P_{ij} , ($\leq 4 \mu\text{m}$ in diameter)* was calculated. The product of the values of the estimated initial concentration of thorium, D_{ij} , the volume of the test chamber, V , and the fraction of respirable particles, P_{ij} , gives the estimated mass of respirable thorium particles, R_{ij} , released in each test.

* Particles with an aerodynamic diameter of $\leq 3.5 \mu\text{m}$ are generally considered to be respirable. The cascade impactors used in these experiments provide information on the mass of material with diameter $\leq 4 \mu\text{m}$. We have, therefore, assumed that particles with diameter $\leq 4 \mu\text{m}$ are respirable. This is a conservative but realistic assumption.

FIGURE 7. FILTER DATA FOR HTGR TESTS

FIGURE 8. CASCADE IMPACTOR DATA FOR HTGR TESTS

As stated earlier, the scaling of the measured release from the model experiments to the release from a full-scale shipping cask is based on the assumption that the release of aerosol is proportional to the transfer of momentum from the weapon to the cask and contents. In Appendix B, it is shown that the estimated proportionality constant, C , for the HTGR tests is calculated from the amount of respirable release from the seven successful tests as

$$C = 12.44 \times 10^{-3} \text{ mg sec/mm}^4$$

The 95 percent confidence interval is $(4.3, 35.8) \times 10^{-3}$. In Appendix A, the momentum transfer calculated for the full scale cask, FSV-1,* is

$$M_{fS} = 8.16 \times 10^4 \text{ mm}^4/\mu\text{sec}$$

Thus the estimated median release from a full scale event is

$$R_{fS} = C M_{fS}$$

$$R_{fS} = 12.44 \times 10^{-3} \times 8.16 \times 10^4 = 1015 \text{ mg}$$

of respirable thorium. Also as shown in Appendix B, the upperbound is estimated to be 9765 mg of respirable thorium.

Based upon previous tests with LWR fuel⁽¹⁾, it was assumed that fission products and transuranic elements would be released from HTGR spent fuel in proportion to the release of the original heavy metal in the surrogate fuel, in this case, thorium. Table 1 gives the concentrations of selected radionuclides** which the HTGR spent fuel would theoretically contain. The concentrations are expressed as Curies per gram of thorium in the surrogate fuel. The fission product and transuranic content of spent HTGR fuel has been characterized by Hamilton⁽⁴⁾ utilizing calculations with the GARGOYLE code. The quantity of each radionuclide was obtained from the GARGOYLE data for once-through fuel with 180 days of decay. The fuel was assumed to be in the reactor for four years with a burnup typical of HTGR fuel. This quantity, expressed in Curies per fuel element, was divided by the number of fuel rods in an

* Shipping cask for HTGR spent fuel, Certificate of Compliance No. 6346, owned by the General Atomic Company.

** Mattsen⁽⁵⁾ has identified the radionuclides that dominate the health effects from the release of respirable material from spent fuel. Only these radionuclides, shown in Tables 1 through 7, are considered in the source terms in this study.

TABLE 1. SPECIFIC ACTIVITY OF SIGNIFICANT RADIONUCLIDES IN
HTGR AND LWR SPENT FUELS WITH 180 DAYS
OF DECAY

Nuclide(a)	HTGR Fuel Ci/g of Th(b)	LWR Fuel Ci/g of U(c)
Sr-90	1.8 E-1(d)	7.0 E-2
Ru-106	6.7 E-2	3.6 E-1
Cs-134	2.8 E-1	1.8 E-1
Cs-137	1.8 E-1	9.8 E-2
Ce-144	9.7 E-1	6.4 E-1
Pu-238	7.0 E-3	2.2 E-3
Pu-239	5.5 E-6	3.0 E-4
Pu-240	9.5 E-6	4.7 E-4
Pu-241	3.9 E-3	1.2 E-1
Pu-242	1.5 E-7	1.7 E-6
Am-241	5.5 E-6	2.1 E-4
Am-243	2.3 E-6	1.5 E-5
Cm-242	1.1 E-3	1.8 E-2
Cm-244	4.3 E-4	1.4 E-3

- (a) Radionuclides with significant health effects from spent fuel release⁽⁵⁾.
- (b) Calculated from GARGOYLE data⁽⁴⁾. The units are chosen on the assumption the initial heavy metal is thorium to correspond with the surrogate fuel.
- (c) Calculated from data in Reference (6).
- (d) E-1 = $\times 10^{-1}$.

element and by the density of thorium in a surrogate fuel rod to give Curies/g of thorium in the spent fuel. Then for a given mass of thorium determined to have been released in a test, it is assumed that the associated Curies of radionuclides theoretically would be released also. Table 1 also contains the calculated concentrations of selected radionuclides in LWR spent fuel for comparison. The burnup of the LWR fuel is 29,300 MWD/T which is an average of the typical end-of-life burnups for BWR and PWR fuel.

Table 2 presents the amounts of each of the most hazardous radionuclides that would be released with the 1.015 g of thorium. These amounts were calculated by multiplying the concentrations in Table 1 by 1.015 g, or the median mass of respirable thorium estimated to be released. The source term for the release of 9 g of uranium from LWR spent fuel from a similar attack is also given in Table 1 for comparison.

For the attack on a shipment of HTGR spent fuel, the 1.015 g of thorium predicted to be released as respirable-size particles is equivalent to approximately 1.2 cm³ of fuel*. The 9 g of uranium predicted to be released from an attack on a LWR fuel shipment⁽¹⁾ is approximately 1 cm³. However, the relatively lower concentration of radionuclides in HTGR spent fuel results in a relatively smaller quantity of radionuclides released than from LWR fuel. All the fission products are released in smaller quantities (<30 percent) than those from LWR fuel. Of the transuranic nuclides, only plutonium-238 is released in a significant quantity and that is 36 percent of that from the LWR fuel.

* The volume of HTGR fuel refers only to the volume of fuel rods and does not include graphite from the fuel block.

TABLE 2. ESTIMATED SOURCE TERMS OF SELECTED RADIONUCLIDES
RESULTING FROM AN ATTACK WITH A SHAPED CHARGE ON
SHIPMENTS OF HTGR AND LWR SPENT FUELS

Nuclide(a)	Curies(b)	
	HTGR	LWR(c)
Sr-90	1.8 E-1(d)	6.3 E-1
Ru-106	6.8 E-2	3.2 E 0
Cs-134	2.8 E-1	1.6 E 0
Cs-137	1.8 E-1	8.8 E-1
Ce-144	9.8 E-1	5.8 E 0
Pu-238	7.1 E-3	2.0 E-2
Pu-239	5.6 E-6	2.7 E-3
Pu-240	9.6 E-6	4.2 E-3
Pu-241	4.0 E-3	1.1 E 0
Pu-242	1.5 E-7	1.5 E-5
Am-241	5.6 E-6	1.9 E-3
Am-243	2.3 E-6	1.4 E-4
Cm-242	1.1 E-3	1.6 E-1
Cm-244	4.4 E-4	1.3 E-2

(a) Radionuclides dominating the health effects(5).

(b) All spent fuels assumed to be shipped 180 days after discharge from the reactor.

(c) From estimated release in prior experimental study(1).

(d) E-1 = $\times 10^{-1}$.

4.0 EXPERIMENTS WITH SURROGATE NON-POWER REACTOR SPENT FUEL

In this section are discussions of the different types of non-power reactors considered in this study, descriptions of the surrogate spent fuel selected for the experiments, and the results of the tests.

4.1 NPR Spent Fuel

Four types of non-power reactors were examined to select one representative type of NPR spent fuel to use in our study to evaluate the relative risk associated with a sabotage attack on a shipment of NPR spent fuel. The types examined were AGN, TRIGA, pool-type, and tank-type reactors licensed for operation by the NRC. To estimate the magnitude of the potential release of radionuclides from shipments of each of the four types of spent fuel, we considered reactor power level, percent of time at power, fuel burnup, and shipping configuration. We chose fuel characteristic of pool-type and tank-type reactors for our experiments. The reasons for this choice are given below.

The AGN reactors (Models 201 and 211) are very low power training reactors. There are presently eight of these reactors listed as being in operation with power levels from 0.1 to 15 watts⁽⁶⁾. The core of the AGN reactor is about 25 cm in diameter and composed of polyethylene discs containing a dispersion of 20 percent enriched UO_2 particles. The owners of these reactors report operating about 100 hours per year. Upon inspection it can be seen that the total radioactivity in this fuel is very low, and a shipment of AGN spent fuel would be similar to a small shipment of low-level waste and not comparable to a shipment of other types of spent fuel.

There are 20 TRIGA reactors listed⁽⁶⁾ as being in operation, but we have considered only those 11 reactors operating at 1 MW or greater. The TRIGAs use individual fuel rods 1.47 in. in diameter with $UZrH_x$ fuel in a stainless steel cladding. The core of 70 to 100 rods is operated in an open pool of water which provides moderation, cooling, and shielding. The maximum power level is 1.5 MW, and the two reactors operating at this level report less than 400 hours per year of operation. This is typical of the low

operating times for all TRIGAs with 900 hours the maximum annual usage reported. There is one exception; a reactor (designated here as the P-T reactor) was converted from a Pulstar design to allow the use of TRIGA-type fuel in 4-rod assemblies. The P-T reactor reports operating at 1 MW for nominally 2500 hours per year.

There has been little TRIGA fuel shipped to date. Most of that which has been shipped was in the BMI-1* cask in a basket for individual rods. The P-T reactor operators have also used BMI-1 for their shipments, but because of their 4-rod assemblies, they have used the standard basket for MTR-type fuel assemblies. Because of the difference in the configuration of the baskets, a jet from a shaped charge could disrupt more fuel in the P-T fuel shipments than in other TRIGA fuel shipments. Our conclusion is that the shipments of P-T spent fuel offer a potentially higher risk from sabotage than other TRIGA fuel because the reactor's higher operating time yields a higher transuranic content and requires more frequent refueling, and because of the unique shipping configuration. However, the P-T spent fuel shipments presents less of a hazard than the pool- or tank-type spent fuel discussed below.

Finally, we considered the pool- and tank-type reactors with an operating power of 1 MW or greater. Of the eleven non-TRIGA, NRC-licensed reactors operating at 1 MW or greater, nine use plate-type metal fuel elements and two use Zircaloy-clad UO_2 rods <0.5 in. in diameter. The latter two operate at 1 or 2 MW. Seven of the eleven reactors are pool-type and four are tank-type. The pool-type reactors operate at 1 to 5 MW; the tank-type operate at 5 to 10 MW.

Both types of reactors have their core in a deep, open pool of light water which provides shielding during operation and refueling. The tank-type reactor uses a sealed enclosure (a tank) to separate the core from the surrounding pool. The plate-type fuel assemblies from both pool- and tank-type reactors are very similar. The most unique is a tank-type reactor with assemblies pie-shaped in cross section with highly curved plates. Six of these assemblies make up a full annular core. The others use MTR-type assemblies with flat or slightly curved plates in assemblies with rectangular cross sections. The compositions of the plates in all of these fuel

* A shipping cask for MTR-type spent fuel, Certificate of Compliance No. 5957, owned by Battelle's Columbus Division.

assemblies are similar. The fuel in most plates is highly enriched uranium as a uranium-aluminum alloy or $U-Al_x$ particles suspended in an aluminum matrix. To increase the uranium loading for the new low enriched fuels, the fuel is in the form of U_3O_8 particles in aluminum. This change in composition is not expected to cause a significant change in the response of the fuel to an explosive attack relative to that of the older-type fuel. However, most fuel shipments in the next few years will be with $U-Al_x$ fuel.

The pool- and tank-type reactors, which operate at higher power with a higher percent of operating time, present the highest risk from sabotage of spent fuel shipments of all the NRC-licensed, non-power reactors because of their greater radionuclide content. Since spent fuel from pool- and tank-type reactors is similar, we believe the risk from sabotage of shipments from any of them can be estimated by the data from experiments conducted with spent fuel from any one of these reactors. All of these spent fuels can be shipped in similar configurations in similar shipping casks. As a result, we selected a generic 5 MW reactor as a model for our experiments.

4.2 Surrogate NPR Spent Fuel and Model Cask

As a part of their program for developing new low-enriched fuel for non-power reactors, DOE has fabricated a number of different types of miniature fuel plates for irradiation testing. DOE supplied us with 17 of these miniature plates for use in this experimental program. These plates were 4.5 in. long, 2 in. wide, and 0.05 in. thick. Each plate was of similar construction to NPR fuel in use today with $U-Al_x$ particles in an aluminum matrix clad with aluminum.

The NPR spent fuel assembly was represented by four miniature fuel plates held in specially fabricated aluminum side plates to form a miniature fuel assembly as shown in Figure 9. The surrogate fuel assembly was inserted into a containment vessel of 3 in. by 4 in. steel tubing with a 0.25-in. thick wall. The assembly was centered vertically in the tube with wooden blocks. The vessel was closed with 0.25-in. thick steel end plates. Shield blocks of 1.75-in. thick steel were tack welded to two opposite sides of the containment vessel. A surrogate NPR fuel assembly in a model cask is shown in Figure 10.

FIGURE 9. SURROGATE NPR FUEL ASSEMBLY

FIGURE 10. MODEL CASK WITH SURROGATE FUEL ASSEMBLY

4.3 Test Results

Four tests were run with the model NPR shipping casks. Typical external damage to the cask is shown in the photographs of the entrance and exit holes in Figures 11 and 12. The photograph in Figure 13 shows some of the internal damage to the containment vessel and fuel element.

It was necessary to recover the fuel plates from the damaged models for accountability purposes. None of the fuel assemblies could be removed intact from the models because distortion of the container walls caused binding of the side plates. In one model it was possible to force one side plate from the assembly allowing the other side plate and fuel plates to be removed as a unit from the container. This assembly and the model cask are shown in Figure 14. Note that the hole in the fuel plates is much larger than the hole in the cask. It appears that a large fraction of the fuel plates has been removed by the attack. However, when the weight of the four damaged fuel plates was compared with the weights of four undamaged fuel plates, only about 5 weight percent of the plates was removed by the attack. Not all of that material was lost from the cask, as some fragments of fuel plate and container wall remained in the cask. Much of the large hole was a result of distortion of the plates rather than removal of material.

All samples collected in the first two tests contained much smaller amounts of uranium than expected and the results are not credible. One explanation is that fans used in the test chamber to promote mixing of the air and representative sampling caused rapid dilution of the air in the chamber. The fans were not used in subsequent testing.

The filter data from Tests 3 and 4 are shown plotted in Figure 15. Extrapolation of the data from each test to time zero gives an estimate of the concentration of airborne material, D_{ij} , immediately after the attack for that test. Data from the cascade impactors for Tests 3 and 4 are shown plotted in Figure 16. The fraction of airborne uranium particles, P_{ij} , which were respirable ($<4 \mu\text{m}$ in diameter) were calculated from these data for each test. Then the estimated total mass of respirable uranium, R_{ij} , released in each test was calculated by

$$R_{ij} = D_{ij} P_{ij} V$$

where V is the volume of the test chamber.

FIGURE 11. JET ENTRANCE HOLE IN MODEL NPR CASK

FIGURE 12. JET EXIT HOLE IN MODEL NPR CASK

FIGURE 13. INTERNAL DAMAGE TO NPR CASK AND CONTENTS

FIGURE 14. JET EXIT HOLES IN NPR MODEL CASK AND SURROGATE FUEL ASSEMBLY

FIGURE 15. PLOT OF FILTER DATA FROM NPR TESTS

FIGURE 16. PLOT OF CASCADE IMPACTOR DATA FROM NPR TESTS

As shown in Appendix B, the proportionality constant, C , for the scaling equation was calculated to be

$$C = 0.748 \text{ mg } \mu\text{sec}/\text{mm}^4$$

with a 95 percent confidence interval of (0.132, 4.24).

The momentum transfer to a full-scale cask as calculated in Appendix A is

$$M_{fS} = 6106 \text{ mm}^4/\mu\text{sec}.$$

From this the estimated median release from a full-scale event is

$$R_{fS} = C M_{fS} = 0.748 \times 6106 = 4567 \text{ mg}$$

of respirable uranium with a 95 percent confidence interval of (806, 2570) mg. The upper bound, shown in Appendix B is 47.6 g of uranium.

The radionuclide content of the generic spent NPR fuel was calculated using the ORIGEN code. The burnup of the fuel is assumed to be typical of end-of-life NPR fuel. The input data included neutron energy and cross sections typical of a PWR. These data are not identical to those for the highly enriched fuel in a NPR reactor. However, we do not believe the significantly large additional effort to adjust the cross sections and energy spectrum is warranted for this application. Table 3 shows the specific activities from the ORIGEN calculations for those radionuclides which dominate the health effects⁽⁵⁾ from the release of respirable material from NPR spent fuel. For comparison, Table 3 also contains similar data for LWR spent fuel. Based upon the data in Table 3, and the calculated mass of released respirable material, R_{fS} , the estimated quantities of significant radionuclides that would be released in a full-scale event are given in Table 4. Similar data from a release from a shipment of LWR fuel are also given for comparison.

It can be seen in Table 3 that the concentration of most of the fission products (expressed as Curies/g of uranium) is higher in the NPR fuel than in LWR fuel. This may be surprising when one thinks of an NPR as a small, low-power reactor. Actually, the power density (watts or fissions per cm^3 of fuel material) for our generic NPR and a typical PWR are very similar. However, the NPR uses fuel highly enriched in uranium-235 which results in it having much less total uranium per unit volume than the PWR which uses low-enriched fuel. Thus, when the concentration of fission products, which may be similar on a volumetric basis, are related to uranium content, the NPR fuel

TABLE 3. SPECIFIC ACTIVITY OF SIGNIFICANT RADIONUCLIDES IN
NPR AND LWR SPENT FUELS WITH 180 DAYS OF DECAY

Nuclide(a)	Generic NPR Fuel Ci/g of U(b)	LWR Fuel Ci/g of U(c)
Sr-90	2.3 E-1(d)	7.0 E-2
Ru-106	2.2 E-1	3.6 E-1
Cs-134	2.3 E-1	1.8 E-1
Cs-137	2.3 E-1	9.8 E-2
Ce-144	3.4 E 0	6.4 E-1
Pu-238	1.2 E-3	2.2 E-3
Pu-239	5.1 E-6	3.0 E-4
Pu-240	3.5 E-6	4.7 E-4
Pu-241	3.7 E-4	1.2 E-1
Pu-242	1.5 E-9	1.7 E-6
Am-241	4.3 E-7	2.1 E-4
Am-243	6.6 E-9	1.5 E-5
Cm-242	1.0 E-5	1.8 E-2
Cm-244	2.6 E-7	1.4 E-3

(a) Radionuclides with significant health effects from 5-year-old spent fuel release(5).

(b) From ORIGEN calculations by Battelle.

(c) Calculated from data in Reference (6).

(d) E-1 = $\times 10^{-1}$.

has the higher concentration. The concentration of transuranic radionuclides is lower in the NPR fuel than the LWR fuel because of the shorter irradiation time and because of the lower uranium-238 content from which many of them occur.

The estimated mass of respirable uranium released by the attack on the NPR spent fuel shipment is approximately half of that previously estimated⁽¹⁾ to be released by an attack on a shipment of LWR spent fuel. As shown in Table 4, the predicted quantities of three of the fission products released in the NPR attack exceed those in an LWR attack. The greatest relative release is for cerium-144 which has a significant health effect. However, the relative quantities of the transuranics, which dominate the health effects from LWR fuel, are less in the NPR release.

TABLE 4. ESTIMATED SOURCE TERMS OF SELECTED RADIONUCLIDES
RESULTING FROM AN ATTACK WITH A SHAPED CHARGE ON
SHIPMENTS OF NPR AND LWR SPENT FUELS

Nuclide(a)	Curies(b)	
	NPR	LWR(c)
Sr-90	1.1 E 0	6.3 E-1(d)
Ru-106	1.0 E 0	3.2 E 0
Cs-134	1.1 E 0	1.6 E 0
Cs-137	1.1 E 0	8.8 E-1
Ce-144	1.6 E+1	5.8 E 0
Pu-238	5.5 E-3	2.0 E-2
Pu-239	2.3 E-5	2.7 E-3
Pu-240	1.6 E-5	4.2 E-3
Pu-241	1.7 E-3	1.1 E 0
Pu-242	6.9 E-9	1.5 E-5
Am-241	2.0 E-6	1.9 E-3
Am-243	3.0 E-8	1.4 E-4
Cm-242	4.6 E-5	1.6 E-1
Cm-244	1.2 E-6	1.3 E-2

(a) Radionuclides dominating the health effects⁽⁵⁾.

(b) All spent fuels assumed to be shipped 180 days after discharge from the reactor.

(c) From estimated release in prior experimental study⁽²⁾.

(d) E-1 = $\times 10^{-1}$.

5.0 EXPERIMENTS WITH SURROGATE HIGH LEVEL WASTE

In this section are discussions of some types of vitrified high-level waste, a description of the surrogate high-level waste glass used in the tests, and the results of the tests.

5.1 Description of High-Level Waste Glass

The DOE has designated borosilicate glass as the reference waste form for high-level waste (HLW) for geologic disposal. As a result, Synroc, which had been included in a previous study⁽³⁾, was not considered in this experimental program. At the Savannah River Plant, DOE is proceeding with plans to build the Defense Waste Processing Facility for retrieving, treating, and solidifying the high-level waste stored in tanks at the site. The HLW glass from the facility will be stored onsite until it can be shipped to a federal repository for disposal. Shipments of defense HLW may begin near the end of the century. This may be the first high-level waste glass to be transported in this country. Plans are being formulated for solidification and disposal of other defense high-level waste glass at DOE's Hanford and Idaho sites.

DOE is also proceeding with plans to solidify the commercial high-level waste stored at West Valley, New York. This waste glass may also be available for shipment about the end of the century. With no reprocessing of commercial spent fuel expected in this country in the near future, no other HLW shipments are anticipated for several decades.

Commercial HLW is generally characterized as the waste stream from the reprocessing of relatively high burnup (33,000 MWD/T) commercial power reactor fuel. The waste is assumed to have the equivalent of ten years of decay after discharge from the reactor. Most of the uranium and plutonium will have been removed during reprocessing, but the waste will contain all of the other transuranic nuclides and most of the fission products. The commercial HLW at West Valley is not typical of the above-described HLW as it is produced from lower burnup fuel, is more dilute, and will have been out of the reactor three decades or more prior to shipping. Defense HLW also results from lower burnup fuel, is diluted with inert materials, and will have

considerably more than ten years of decay when shipped. Defense waste can be expected to be less concentrated in radionuclides than reference commercial high-level waste. For this study, we have taken a worst-case approach and selected commercial HLW glass as the reference source term.

5.2 Surrogate Canisters of HLW Glass and Model Cask

A well-documented glass formulation, PNL 76-68, was used to simulate the vitrified waste in our surrogate canisters of HLW glass. Four weight percent UO_2 was added to this glass to be the tracer element for our samples.

Six miniature canisters of surrogate HLW glass were prepared. The canisters were fabricated from 7-in. lengths of stainless steel tubing with 1/8-in. thick walls. Four canisters were 1.5 in. in diameter and two were 3 in. in diameter. The glass was cast in the canisters to a height of 6 inches. The ends of the canisters were closed with welded end plates 1/8-in. thick. A photo of a 1.5-in. diameter canister prior to closure is shown in Figure 17.

Each canister was inserted into a square steel tube with 1/8-in. thick walls which formed the containment vessel for the model cask. The ends of the cask were closed with welded end plates. Shield blocks of 1-3/4-in. thick steel were welded to two opposite sides of the containment vessel. The components for a 3-in. canister and cask are shown in Figure 18.

5.3 Test Results

Six tests were conducted with canisters of surrogate HLW glass. The first four used 1.5-in. diameter canisters of glass and the last two used 3-in. diameter canisters. The entrance and exit holes in a model cask for a 3-in. diameter canister are shown in Figures 19 and 20, respectively. The shaped charge was aimed closer to the lower side of the cask to hit near the center of the glass column. It can be seen that the entrance and exit holes made by the jet are nearly the same diameter. Inspection of the holes through the models showed that all were plugged at the exit from the canister.

FIGURE 17. 1.5-INCH SURROGATE CANISTER OF HLW

FIGURE 18. COMPONENTS OF MODEL CASK FOR 3-INCH SURROGATE CANISTER OF HLW

FIGURE 19. JET ENTRANCE HOLE IN MODEL HLW CASK

FIGURE 20. JET EXIT HOLE IN MODEL HLW CASK

Plugging occurs when a part of the molten-metal jet solidifies and does not completely penetrate the target.

On the third test the jet struck the HLW canister off center and much of the jet missed the canister so the data were not used. The data from the filter samples are shown plotted in Figure 21. These values for the airborne uranium concentration in the test chamber after the shots are consistent with each other except for two data points. In the first and second tests with 1.5-in. diameter canisters, the mass of uranium was unexplainably high in the samples collected from the second filter approximately 15 minutes after the shot. The filter data from each test were extrapolated to time zero to obtain the initial concentration of airborne uranium, D_{ij} , by the restricted least squares method as described in Appendix B.

Since the jet would disrupt nominally twice the volume of HLW glass while penetrating a 3-in. diameter canister than a 1.5-in. diameter canister, one would expect higher concentrations of airborne uranium from tests with the larger canisters. However, the filter data do not support this expectation. This effect may result from the plugging of the exit hole.

The cascade impactor data are plotted in Figure 22. The fraction of the total mass of airborne uranium, P_{ij} , which was respirable ($<4 \mu\text{m}$ diameter) was determined for each test from the impactor data. The estimated total mass of respirable uranium, R_{ij} , released in each test was then calculated as the product of initial concentration, D_{ij} , the fraction of respirable uranium, P_{ij} , and the volume of the test chamber, V , i.e., $R_{ij} = D_{ij} P_{ij} V$.

The scaling method was applied to the data in a different manner than the other two series of tests because it had been observed that the glass reacted in a different manner to the jet than did the graphite or the thin aluminum plates. Momentum transfer values were determined for each of the two different-sized models. Next an estimated proportionality constant was calculated for each model size using the experimental data from that size model and the appropriate momentum transfer value. The scaling law was used to determine the momentum transfer value for the full-scale event and estimated release values were calculated with each value for the proportionality constant. The two estimated releases were similar but the greater value, associated with the data from the smaller model, was used for the source term

FIGURE 21. FILTER DATA FROM HLW TESTS

FIGURE 22. CASCADE IMPACTOR DATA FROM HLW TESTS

calculations. The estimated median release of respirable uranium was 0.035 g with a 95 percent confidence interval of (0.023 to 0.052) g. The estimated upper bound was 0.085 g of uranium. The released mass of 0.035 g of uranium corresponds to a release of 0.33 cm³ of HLW glass. The application of the scaling to these data is described in Appendix A and the statistical treatment of the data is given in Appendix B.

The concentrations of significant radionuclides in the commercial HLW glass are given in Table 5. These values are based on a published radionuclide content of LWR fuel⁽⁷⁾ which is assumed to be reprocessed six months after discharge to remove 99 percent of the plutonium and uranium. The values were then calculated for 10 years of decay after discharge. The values in units of Ci/cm³ assume 280 kg of vitrified waste per metric ton of uranium reprocessed, and a glass density of 3 g/cm³.

The associated radionuclides which would theoretically be released with the estimated release of 0.33 cm³ of glass are given in Table 6.

TABLE 5. SPECIFIC ACTIVITY OF SIGNIFICANT RADIONUCLIDES IN
REFERENCE HIGH-LEVEL WASTE GLASS AND LWR
SPENT FUEL

Nuclide	Curies per cm ³ (a)	
	HLW Glass, 10 Years Old	LWR Fuel, 1/2 Year Old
Sr-90	5.7 E-1(b)	6.1 E-1
Ru-106	5.1 E-3	3.1 E 0
Cs-134	7.5 E-2	1.6 E 0
Cs-137	8.1 E-1	8.6 E-1
Ce-144	1.4 E-3	5.6 E 0
Pu-238	2.0 E-4	1.9 E-2
Pu-239	3.0 E-5	2.7 E-3
Pu-240	4.8 E-5	4.1 E-3
Pu-241	7.5 E-3	1.0 E 0
Pu-242	1.7 E-7	1.5 E-5
Am-241	2.1 E-3	1.8 E-3
Am-243	1.5 E-4	1.3 E-4
Cm-242	7.2 E-8	1.6 E-1
Cm-244	9.6 E-3	1.2 E-2

(a) Calculated from ORIGEN data(6) assuming 280 kg of glass per metric ton of fuel reprocessed and with densities of 3.0 and 10.4 g/cc for glass and LWR fuel respectively.

(b) E-1 = $\times 10^{-1}$.

TABLE 6. ESTIMATED SOURCE TERMS OF RADIONUCLIDES RESULTING FROM AN ATTACK WITH A SHAPED CHARGE OF SHIPMENTS OF HIGH-LEVEL WASTE GLASS AND LWR SPENT FUEL

Radionuclide(a)	Curies	
	HLW Glass(b)	LWR Spent Fuel(c)
Sr-90	1.9 E-1(d)	6.3 E-1
Ru-106	1.7 E-3	3.2 E 0
Cs-134	2.5 E-2	1.6 E 0
Cs-137	2.7 E-1	8.8 E-1
Ce-144	4.6 E-4	5.8 E 0
Pu-238	6.6 E-5	2.0 E-2
Pu-239	9.9 E-6	2.7 E-3
Pu-240	1.6 E-5	4.2 E-3
Pu-241	2.5 E-3	1.1 E 0
Pu-242	5.6 E-8	1.5 E-5
Am-241	6.9 E-4	1.9 E-3
Am-243	4.9 E-5	1.4 E-4
Cm-242	2.4 E-8	1.6 E-1
Cm-244	3.2 E-3	1.3 E-2

(a) Radionuclides dominating the health effects(5).

(b) Assumes decay equivalent to 10 years after reactor discharge.

(c) Assumes 180 days after reactor discharge.

(d) E-1 = $\times 10^{-1}$.

6.0 ANALYSIS OF RESULTS

A summary of the estimated source terms of radioactive material released from attacks on shipments of HTGR, NPR, and LWR spent fuel and vitrified HLW is shown in Table 7. To assist in comparing the source terms, the table also contains the relative contributions of the more significant nuclides in LWR fuel to the health effects (latent cancer fatalities). Because some radionuclides decay more quickly than others, a radionuclide's relative contribution to health effects changes with time. Table 7 shows the relative contributions of the most hazardous radionuclides in fuel that has been cooled 120 days and fuel that has been cooled five years. Note that the same nuclides contribute to health effects regardless of how long the fuel has been cooled. Only their relative contributions change. Therefore, these nuclides are the ones whose concentrations we calculated in our source terms regardless of the cooling time of the fuel being considered.

In Figure 23 the quantities of the radionuclides in the source terms which contribute significantly to the health effects are plotted for the HTGR, NPR, and HLW releases relative to that of the LWR release. This graph shows that the quantities of these radionuclides predicted to be released by attacks on shipments of HTGR spent fuel and vitrified HLW are less in all cases than that from LWR fuel. The release of some fission products from attacks on shipments of NPR spent fuel exceed those of LWR fuel; however, the quantities of released transuranic nuclides are all less. When the relative increases and decreases in the quantities of radionuclides are considered, we believe that the latent cancer fatalities resulting from an attack on a shipment of NPR spent fuel will be approximately the same as that predicted for the LWR fuel shipment⁽²⁾.

TABLE 7. SUMMARY OF SOURCE TERMS AND RADIOLOGICAL EFFECTS FOR ESTIMATED RELEASES FROM ATTACKS ON SHIPMENTS OF HTGR, NPR, AND LWR SPENT FUELS AND VITRIFIED HLW

Nuclide	Source Terms (Curies)				Percent Contribution (>1%) to Latent Cancer Fatalities ^(a)	
	HTGR(b)	NPR(b)	HLW(c)	LWR(b)	120-Day Decay	5-Year Decay
Sr-90	1.8 E-1 ^(d)	1.1 E 0	1.9 E-1	6.3 E-1	-	-
Ru-106	6.8 E-2	1.0 E 0	1.7 E-3	3.2 E 0	12.3	-
Cs-134	2.8 E-1	1.1 E 0	2.5 E-2	1.6 E 0	-	-
Cs-137	1.8 E-1	1.1 E 0	2.7 E-1	8.8 E-1	-	-
Ce-144	9.8 E-1	1.6 E+1	4.6 E-4	5.8 E 0	18.5	-
Pu-238	7.1 E-3	5.5 E-3	6.6 E-5	2.0 E-2	24.5	45.7
Pu-239	5.6 E-6	2.3 E-5	9.9 E-6	2.7 E-3	3.0	5.6
Pu-240	9.6 E-6	1.6 E-5	1.6 E-5	4.2 E-3	3.7	6.9
Pu-241	4.0 E-3	1.7 E-3	2.5 E-3	1.1 E 0	1.3	1.9
Pu-242	1.5 E-7	6.9 E-9	5.6 E-8	1.5 E-5	-	-
Am-241	5.6 E-6	2.0 E-6	6.9 E-4	1.9 E-3	1.9	19.4
Am-243	2.3 E-6	3.0 E-8	4.9 E-5	1.4 E-4	-	-
Cm-242	1.1 E-3	4.6 E-5	2.4 E-8	1.6 E-1	20.7	-
Cm-244	4.4 E-4	1.2 E-6	3.2 E-3	1.3 E-2	11.5	17.7

(a) From Tables 11 and 12 of Mattsen's Memo⁽⁵⁾.

(b) Spent fuel 180 days after reactor discharge.

(c) Vitrified HLW with equivalent of 10 years' decay after discharge.

(d) E-1 = $\times 10^{-1}$.

FIGURE 23. QUANTITIES OF SIGNIFICANT RADIONUCLIDES IN THE SOURCE TERMS FOR THE RELEASES (BY ATTACKS ON SHIPMENTS OF HTGR AND NPR SPENT FUEL AND HLW) NORMALIZED TO THOSE OF LWR SPENT FUEL

7.0 CONCLUSION

When the experimental results from model tests with surrogate HTGR and NPR spent fuel and vitrified HLW are scaled to full-size shipments of these materials, the release of radioactive materials by a sabotage attack with a shaped charge are predicted to be less than those that occur from an attack of LWR spent fuel. A similar attack on a shipment of NPR spent fuel is predicted to result in higher releases of some fission products and lower releases of all transuranic radionuclides than those from an attack on shipment of LWR spent fuel. Thus, the sabotage of shipments of HTGR and NPR spent fuels and vitrified HLW should have no greater consequences than those predicted⁽²⁾ for shipments of LWR spent fuel.

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APPENDIX A

DEVELOPMENT AND APPLICATION OF THE SCALING LAW

In these series of tests we fired a small precision-shaped charge against sub-scale targets and measured the amount of respirable radioactive material released. These data were then scaled to estimate the respirable release generated when the full-scale shaped charge penetrates the full-scale target. The method used to derive this estimate is described in this appendix.

Charge Description and Functioning

The modern shaped charge consists of an explosive body with a metal lined cavity, typically conical, always symmetric about a centerline. The charge is initiated on the end opposite the cavity, on the centerline. As the detonation wave progresses through the explosive and encounters the liner, it causes the liner to collapse onto the centerline, ejecting a thin, very high-velocity jet of liner material out the front of the cavity. This is followed by a slower moving, more massive slug of liner material. The jet exhibits a velocity gradient through its length which stretches the jet and eventually breaks it into individual, aligned segments or particles. This process is called "particulation".

When the jet strikes a target it is capable of great penetration due to the extremely high pressures generated, and length of the impacting jet. At these high pressures, physical characteristics of the target such as hardness and strength play only minor roles in resisting penetration, while the target density is the major factor.

Table A-1 lists some of the physical characteristics of the two shaped charges. As mentioned, the charge used in the laboratory tests is a small precision device. The full-scale charge is an Army Engineer demolition charge developed in the 1940s primarily for penetration of concrete structures such as paved highways. It is not a high precision device.

TABLE A-1. SHAPED CHARGE PHYSICAL CHARACTERISTICS

	Precision	Full-Scale
<u>Liner</u>		
Cone Diameter, mm	65	234
Cone Angle, Degrees	44	60
Material	Copper	Iron
<u>Explosive</u>		
Fabrication Method	Pressed	Cast
Composition	HMX Based	Composition B
Mass, kg	0.45	13.61

Scaling Development

In scaling the release of material from the precision laboratory shaped charge to the release from the full-scale threat, it is necessary to relate the release to the measurable or modelable parameters of the jet responsible for the release.

As the jet strikes and penetrates the target, it imparts momentum to the target in both axial and radial directions. This transfer of momentum is in the form of a high-pressure wave traveling through the target medium. As the wave progresses, it crushes the medium, thereby releasing material. The amount of material that the pressure wave can crush and release before attenuating to a level below the release limit of the target medium is assumed to be proportional to the amount of momentum transferred to the medium. This is the major premise of this scaling method.

We first calculate the amount of radial and axial momentum transferred to the target from the jet. We consider penetration through a homogeneous medium at steady state, i.e., the jet velocity, V_j , penetration velocity V_p , and jet diameter d_j are assumed to be constant*. We know that the

* In actuality, the mentioned quantities are not constant but vary slowly along the length of the jet. Thus at any point on the jet this assumption is a good approximation.

stagnation pressure at the center of symmetry of the jet-target interface P_s is equal in the target and the jet material. From incompressible inviscid hydrodynamics and application of the Euler equation we get:

$$P_s = \frac{1}{2} \rho_j (V_j - V_p)^2 = \frac{1}{2} \rho_t V_p^2 \quad (1)$$

where ρ is the density and the subscripts j and t denote jet and target materials, respectively. From Equation (1) comes the expression

$$V_j = V_p (\sqrt{\rho_j/\rho_t} + 1) / (\sqrt{\rho_j/\rho_t}) \quad (2)$$

The mass flow rate of jet material into the jet impact region is

$$G_j = \left(\frac{\pi}{4}\right) d_j^2 \rho_j \Delta V \quad (3)$$

where

$$\Delta V = V_j - V_p \quad .$$

Along a streamline the Euler equation can be written, in the absence of viscosity, as

$$\frac{\rho q^2}{2} + P = \text{constant} \quad (4)$$

Before impact, the pressure P on the jet material is zero, and after the jet material has assumed a straight line in generally a rearward direction along the crater walls the pressure is again zero because the jet material is along a free surface. Thus, along all streamlines the jet exit speed, $-q$, is equal but in the opposite direction the jet entrance speed, q . Momentum is first transferred axially as the jet is turned 90 degrees into the outward radial direction, and then radially as the jet is turned rearward. The radial momentum transfer rate would be $G_j \Delta V$ if the jet turned through exactly 90 additional degrees and exited parallel to the incoming jet. This condition

appears to be approached in many jet penetrations and can serve as an approximation to the radial momentum transfer rate.

We obtain an estimate of the radial momentum transferred to the target per unit depth of penetration by dividing the radial momentum transfer rate by the penetration velocity:

$$p_r = G_j \Delta V / V_p \quad (5)$$

or from Equations (2) and (3):

$$p_r = \left(\frac{\pi}{4}\right) d_j^2 \rho_t V_p \quad (6)$$

We also need an expression for the axial momentum transferred to the target from the jet. In the coordinate system with the origin moving with the jet target interface we see that the jet material enters and exits the interface at equal speeds. Therefore, the amount of axial momentum transferred to the target per unit length of jet is equal to twice the momentum contained in that length of jet, as in a perfect elastic collision, or

$$I = 2 \left(\frac{\pi}{4}\right) \rho_j d_j^2 \Delta V \quad (7)$$

The axial momentum transfer rate is thus

$$I \Delta V = \frac{\pi}{2} \rho_j d_j^2 \Delta V^2 = \frac{\pi}{2} \rho_t d_j^2 V_p^2 \quad (8)$$

By dividing by the penetration velocity V_p , we obtain the axial momentum transferred per unit depth of penetration or

$$p_a = \frac{\pi}{2} \rho_t d_j^2 V_p \quad (9)$$

The sum of the radial and axial momentum transferred per unit depth of penetration is

$$p_T = p_r + p_a = 3\left(\frac{\pi}{4}\right) \rho_t d_j^2 V_p \quad (10)$$

Our major premise says that the amount of material released is proportional to the total momentum transferred to the target, or

$$R = K \int_{X_1}^{X_2} \left(3 \left(\frac{\pi}{4} \right) \rho_t d_j^2 V_p \right) dX \quad (11)$$

where K is the proportionality constant, X_1 and X_2 are the starting and stopping locations of the target zone responsible for the release and R is the amount of material released. Although the density of the jet material does not appear explicitly in this equation, it is implicit, as it affects the penetration velocity. Since both charges are being fired into similar targets, we can combine some of the constants in Equation (11) to yield

$$R = C \int_{X_1}^{X_2} V_p d_j^2 dX \quad (12)$$

Scaling Application

The constant C should be the same for both charges when fired into similar targets. We need only find an expression for $V_p d_j^2$ for the precision laboratory charge, integrate that over the position of the target zone along the jet path yielding the release, and divide it into the measured release to obtain a value for C.

A computer code which models the jet formation and subsequent penetration of a specific target for a given shaped charge was run for the precision laboratory shaped charge against the target type(s) used in the laboratory tests. The code, written by J. E. Backofen, formerly of Battelle, treats the jet as if it comprised a large number of segments. A number of parameters are calculated, including penetration velocity, the diameter of each jet segment at jet formation time, the amount of stretching which each undergoes during its flight, and the time and position where the segment particulates. From knowledge of the relative position of the charge and the target, we can calculate the diameter of the individual jet segments when they

impact the target zone of interest, assuming no further stretching after particulation.

From these computer runs, values or expressions for V_p and d_j over the position of the target zone along the jet path yielding the release were extracted. These were substituted into Equation (12) along with the measured value of R , allowing a value of C to be calculated for that target type.

The computer code was next run for the full-scale charge against the corresponding full-scale target. Again, values or expressions for V_p and d_j over the applicable zones were extracted. These were inserted into Equation (12) along with the corresponding calculated value of C , thus yielding a predicted full-scale release.

The following sections give a more detailed account of how the scaling method was applied to each of the three full-scale target types, including HTGR spent fuel, NPR spent fuel, and HLW glass.

HTGR Spent Fuel

As mentioned in the main body of the report, the release data from the 4-in.-thick targets and the 8-in.-thick targets were used in the scaling effort. Table A-2 lists the penetration velocity V_p and the jet diameter d_j of each jet segment responsible for penetration through the radioactive release-producing zone, as well as the starting and stopping locations of the portions of that zone that each segment penetrates, as extracted from the computer program. These values were substituted into the integral $\int_{x_1}^{x_2} V_p d_j^2 dx$, to yield the momentum transfer values given in Table A-3. The resulting values were then divided into the measured respirable release for each of the test targets, from which was calculated an estimated value for $\hat{C}$ of $12.44 \times 10^{-3} \text{ mg } \mu\text{sec}/\text{mm}^4$.

The computer code was next run for the full-scale charge against the full-scale target, which contains 14.173 in. or 360 mm of release-producing material. The resulting values of V_p and d_j through this material are shown in Table A-4.

These values were substituted into the integral $\int_{x_1}^{x_2} V_p d_j^2 dx$, yielding a value of $81570.625 \frac{\text{mm}^4}{\mu\text{sec}}$. This number was next multiplied by $\hat{C}$ to

TABLE A-2. LABORATORY TEST COMPUTER RUN DATA, HTGR SPENT FUEL

Target	Position of Release-Producing Zone, Ignoring Air Spaces, mm		Penetration Velocity, V_p , mm/ μ sec	Jet diameter, d_j , mm
	Starting	Stopping		
Shots 1 and 2	101.60	113.40	4.62	1.688
	113.40	125.19	4.56	1.688
	125.19	136.96	4.48	1.688
	136.96	148.74	4.41	1.688
	148.74	152.40	4.34	1.688
Shots 3,4,7,8,9, and 10	152.40	160.50	4.34	1.688
	160.50	172.80	4.27	1.412
	172.80	185.12	4.20	1.412
	185.12	197.46	4.13	1.412
	197.46	203.2	4.05	1.412
Shots 5 and 6	254.00	258.97	3.77	1.412
	258.97	271.28	3.69	1.412
	271.28	283.51	3.62	1.412
	283.51	295.73	3.55	1.412
	295.73	304.8	3.48	1.412

TABLE A-3. VALUES OF MOMENTUM TRANSFER, HTGR SPENT FUEL TESTS

Target	$x_1 \int_{x_1}^{x_2} v_p^2 d_j^2 dx \cdot \frac{\text{mm}^4}{\mu\text{sec}}$
Tests 1 and 2	652
Tests 3 and 4	456
Tests 5 and 6	365

TABLE A-4. FULL-SCALE COMPUTER RUN DATA, HTGR SPENT FUEL

Target	Position of Release-Producing Zone, Ignoring Air Spaces, mm		Penetration Velocity, V_p , mm/ μ sec	Jet diameter d_j , mm
	Starting	Stopping		
Full Scale	188.00	229.48	4.27	7.242
	229.48	270.48	4.21	7.242
	270.48	311.48	4.09	7.242
	311.48	354.93	3.98	7.688
	354.93	398.37	3.87	7.688
	398.37	441.82	3.75	7.688
	441.82	488.04	3.64	8.072
	488.04	534.26	3.53	8.072
	534.26	548.00	3.41	8.072

obtain the estimated median release of respirable material from the full-scale charge against the full-scale target of 1015 g. The 95 percent confidence is (0.353, 2.917). The estimated upper bound is 9.77 g. The statistical treatment of the data for the estimated release is given in Appendix B.

NPR Spent Fuel

Table A-5 lists the penetration velocity V_p and the jet diameter d_j of each jet segment responsible for penetration through the four radioactive plates, as well as the width of each plate, ΔX . Since the penetration velocities and jet diameters are constant through each plate, and since each of the plates are separated from each other, the expression $R = C \int_{x_1}^{x_2} V_p d_j^2 dX$ becomes $R = C \sum (V_{pi} d_{ji}^2 \Delta X_i)$, where the index i refers to the plate number.

The values from Table A-5 were substituted into the sum $\sum (V_{pi} d_{ji}^2 \Delta X_i)$, resulting in a value of $21.712 \frac{\text{mm}^4}{\mu\text{sec}}$.

TABLE A-5. LABORATORY TEST COMPUTER RUN DATA, NPR SPENT FUEL

Plate No., i	V_{pi} , mm/ μsec	d_{ji} , mm	ΔX_i , mm
1	3.75	1.688	0.508
2	3.75	1.688	0.508
3	3.75	1.688	0.508
4	3.75	1.688	0.508

This was then divided into the measured respirable release from Tests 3 and 4 resulting in an estimated value for the constant C of $0.748 \frac{\text{mg } \mu\text{sec}}{\text{mm}^4}$.

The computer code was next run for the full-scale charge against the full-scale target, which contained four separate fuel assemblies, each containing 16 radioactive plates each, for a total of 64 plates. The results of this run are shown in Table A-6.

TABLE A-6. FULL-SCALE COMPUTER RUN DATA, NPR SPENT FUEL

Plate No., i	V_{pi} , mm/ μ sec	d_{ji} , mm	ΔX_i , mm
1 through 16	2.97	8.0722	0.508
17 through 33	2.88	8.0722	0.508
34	2.84	8.0722	0.508
35 through 56	2.79	8.0722	0.508
57 through 64	2.70	8.4638	0.508

Again, these values were substituted into the sum $\sum (V_{pi} d_{ji}^2 \Delta X_i)$ yielding a value of $6105.59 \frac{\text{mm}^4}{\mu\text{sec}}$. This number was next multiplied by the calculated value of $\hat{C}$ to obtain an estimated median release of respirable material from the full-scale charge against the full-scale target of 4.567 gm. The 95 percent confidence interval is (0.806, 25.7) g. The estimated upper bound is 47.6 g. The statistical treatment of the data is shown in Appendix B.

HLW Glass

When the computer code was run for the two test configurations for the HLW glass scaling, greater total penetrations were predicted in both cases than were observed in the actual tests. This is due in part to the fact that glass exhibits a resistance to penetration greater than would be predicted by the relatively simple hydrodynamic considerations employed by the code. It was felt necessary, therefore, to scale the position axis in the computer simulations to match that observed in the actual tests, and thereby scaling the associated jet parameters. In other words, the total penetration predicted by the computer code should match the total penetration observed in the actual tests.

The derivation of the momentum transferred to each of the two different models are shown on pages A-13 and A-14. Using the computer generated values for V_p and Equation (2), a linear relationship between V_j and the position of the jet (X_C) in the solid portion of the computer target (i.e., ignoring air spaces) was established for both computer runs. A linear relationship was also established between d_j and the same position coordinates for both runs.

The computer position axis (X_C) was next scaled to the actual test series axes (X_A). This was done by multiplying the computer position coordinate by a factor consisting of the total penetration of the jet through solid portions of the target as observed in the actual tests divided by the total penetration of the jet through solid portions of the target as predicted by the computer code.

Using Equation (2), the two expressions for V_j were converted into equations for V_p through the release-producing glass zone.

Derivation of Momentum Transfer for Tests 1-4

$$V_j = 8.2877 - (.015967) X_C \quad (\text{mm}/\mu\text{sec})$$

$$d_j = 0.34828 + (.0063497) X_C \quad (\text{mm})$$

$$X_A = \left(\frac{\text{Total Penetration Observed}}{\text{Total Penetration Predicted}} \right) X_C = \left(\frac{255.97}{465.58} \right) X_C$$

$$X_C = 1.81889 X_A$$

$$V_j = 8.2877 - (0.029042) X_A \quad (\text{mm}/\mu\text{sec})$$

$$d_j = 0.34828 + 0.011549 X_A \quad (\text{mm})$$

$$V_p = V_j \left(\frac{\sqrt{\rho_j/\rho_t}}{1 + \sqrt{\rho_j/\rho_t}} \right) = V_j \left(\frac{\sqrt{8.94/3.0}}{1 + \sqrt{8.94/3.0}} \right) = (0.63320) V_j \quad (\text{Copper jet into glass})$$

$$V_p = 5.2478 - (0.018389) X_A \quad (\text{mm}/\mu\text{sec})$$

$$V_p d_j^2 = 0.63656 + 0.039985 X_A + 5.5202 \cdot 10^{-4} X_A^2 - 2.4527 \cdot 10^{-6} X_A^3$$

glass located from $X_A = 101.63$ to $X_A = 133.38$

$$\int_{X_A = 101.63}^{X_A = 133.38} V_p d_j^2 dX_A = 284.21 \frac{\text{mm}^4}{\mu\text{sec}}$$

$$X_A = 101.63$$

Derivation of Momentum Transfer for Tests 5-6

$$V_j = 7.3996 - 0.010609 X_C \quad (\text{mm}/\mu\text{sec})$$

$$d_j = 0.27124 + 0.0063085 X_C \quad (\text{mm})$$

$$X_A = \left(\frac{\text{Total Penetration Observed}}{\text{Total Penetration Predicted}} \right) X_C = \left(\frac{317.52}{479.87} \right) X_C$$

$$X_C = 1.51131 X_A$$

$$V_j = 7.3996 - 0.016033 X_A \quad (\text{mm}/\mu\text{sec})$$

$$d_j = .27124 + 0.0095341 X_A \quad (\text{mm})$$

$$V_p = V_j \left(\frac{\sqrt{\rho_j/\rho_t}}{1 + \sqrt{\rho_j/\rho_t}} \right) = V_j \left(\frac{\sqrt{8.94/3.0}}{1 + \sqrt{8.94/3.0}} \right) = (0.63320) V_j \quad \text{(Copper jet into glass)}$$

$$V_p = 4.6854 - (0.010152) X_A \quad (\text{mm}/\mu\text{sec})$$

$$V_p d_j^2 = (0.34471) + (0.023486) X_A + (3.7339 \cdot 10^{-4}) X_A^2 + (-9.2281 \cdot 10^{-7}) X_A^3$$

glass located from $X_A = 101.6$ to $X_A = 171.45$

$$X_A = 171.45$$

$$\int V_p d_j^2 dX_A = 570.02 \quad \frac{\text{mm}^4}{\mu\text{sec}}$$

$$X_A = 101.6$$

We now had expressions for V_p and d_j for both target types. These were combined to form expressions for $V_p d_j^2$ for both target types, and were then integrated over the respective release-producing glass zones.

The estimated proportionality constant, $\hat{C}$, for equation 12 was calculated for each model from the appropriate value for momentum transfer above. For the model used in Tests 1 to 4 $\hat{C}$ is 5.93×10^{-7} with a 95 percent confidence interval of $(3.30, 10.6) \times 10^{-7}$. For the model used in Tests 5 and 6, $\hat{C}$ is 2.61×10^{-7} , $(1.2, 5.6) \times 10^{-7}$.

The computer code was next run for the full-scale charge against the full-scale target. In a manner similar to that described above, linear relationships between V_j and d_j and the computer position coordinates (ignoring air spaces) were established.

We obtained two estimates of the total respirable release from the full-scale charge against the full-scale target, one each by comparing the full-scale charge computer run to each of the two different test models. We scaled the position axis once using the scaling factor developed for Tests 1 to 4 and once using the scaling factor developed for Tests 5 and 6. This gave us two scaled full-scale expressions for V_j and two for d_j . It should be noted that, once scaled, the computer code predicts that the full-scale charge would not totally penetrate the release-producing glass zone of the full-scale target. When using the scaling factor associated with Tests 1 to 4, the computer predicts that the jet would only penetrate 50 percent of the 23.25-in.-diameter glass zone, while using the factor associated with Tests 5 and 6 indicates penetration of 66 percent of the glass.

The V_j equations were next converted to expressions for V_p as before. Expressions for $V_p d_j^2$ were calculated and integrated over the applicable zones, as before.

The values of the integrals were next multiplied by the corresponding proportionality constants to yield two predictions of full-scale release. The higher of the two values, or 0.035 g of uranium was used in the source term calculations. The 95 percent confidence interval on this value is (0.023, 0.052). The upper bound was calculated to be 0.085 g of uranium. The statistical treatment of the data is shown in Appendix B.

Derivation of Predicted Release for Full-Scale Charge

$$V_j = 6.8951 - 0.0056212 X_C \quad (\text{mm}/\mu\text{sec})$$

$$d_j = 6.1910 + 0.0058643 X_C \quad (\text{mm})$$

For comparison to Tests 1 to 4

$$\text{using } X_A = \left(\frac{255.97}{465.58}\right) X_C, \text{ then } X_C = 1.81889 X_A$$

and

$$V_p = V_j \left(\frac{\sqrt{7.85/3.0}}{1 + \sqrt{7.85/3.0}} \right) = 0.61797 V_j \quad (\text{Steel jet into glass})$$

$$V_p = 4.2610 - 0.0063183 X_A \quad (\text{mm}/\mu\text{sec})$$

$$d_j = 6.1910 + 0.010666 X_A \quad (\text{mm})$$

$$V_p d_j^2 = 163.32 + 0.32058 X_A - 3.4973 \cdot 10^{-4} X_A^2 - 7.1877 \cdot 10^{-7} X_A^3$$

jet penetrates glass from $X_A = 174.6$ to $X_A = 470.32$

$$X_A = 470.32$$

$$\int V_p d_j^2 dX_A = 58734 \quad \frac{\text{mm}^4}{\mu\text{sec}}$$

$$X_A = 174.6$$

$$R = C \int V_p d_j^2 dX_A = (5.93 \cdot 10^{-7}) (58734) = 0.035 \text{ gm}$$

For comparison to Tests 5 and 6

$$\text{using } X_A = \left(\frac{317.52}{479.87}\right) X_C, \text{ then } X_C = 1.51131 X_A$$

and

$$V_p = 0.61797 V_j \quad (\text{steel jet into glass})$$

$$V_p = 4.2610 - 0.0052499 X_A$$

$$d_j = 6.1910 + 0.0088628 X_A$$

$$V_p d_j^2 = 163.32 + 0.26638 X_A - 2.4142 \cdot 10^{-4} X_A^2 - 4.1237 \cdot 10^{-7} X_A^3$$

jet penetrates glass from $X_A = 174.6$ to $X_A = 566.05$

$$X_A = 566.05$$

$$\int V_p d_j^2 dX_A = 77892 \quad \frac{\text{mm}^4}{\mu\text{sec}}$$

$$X_A = 174.6$$

$$R = C \int V_p d_j^2 dX_A = (2.61 \cdot 10^{-7}) (77892) = 0.020 \text{ gm.}$$

APPENDIX B

STATISTICAL MODEL AND DATA ANALYSIS

This appendix contains the development of a statistical model, and required data analysis, for predicting the respirable mass that could be released in an attack on a shipment of radioactive materials. The theoretical model is presented in Section B.1 and the measurement data along with the estimates and uncertainty limits are presented in Section B.2. Section B.3 contains a discussion of the model assumptions and limitations.

B.1 The Statistical Model

The experimental results from tests on scale model versions of HTGR, NPR and HLW shipments are used to predict the respirable mass that would be released in a full-scale attack. The estimated respirable release on the j-th shot on model size i is given by

$$\hat{R}_{ij} = V \cdot \hat{D}_{ij} \cdot \hat{P}_{ij},$$

where V is the volume of the test chamber,

$\hat{D}_{ij}$ = estimated initial concentration obtained through
extrapolation of the Millipore® filter data

and

$\hat{P}_{ij}$ = estimated fraction of the total mass released
that is respirable.

The fraction $\hat{P}_{ij}$ is computed from the cascade impactor data.

The statistical model for estimating the initial concentration is presented in Section B.1.1 and the model for estimating the fraction respirable is discussed in Section B.1.2. In Section B.1.3 the theory for estimating the proportionality constant in the scaling equation $R=CM$ is presented. Variance component estimates and the confidence interval methods are discussed in Section B.1.4.

B.1.1 A Model for Millipore® Filter Measurements

Air was drawn from the chamber through the Millipore® filters during three consecutive time periods. Analysis of the deposits on the filters were used to determine the aerosol concentrations during those time periods. The measured concentration obtained from the k -th filter on the j -th shot with cask model i is denoted by X_{ijk} . Assuming an exponential decay in the airborne concentrations the statistical model is given by

$$W_{ijk} = \alpha_{ij} + \beta_{ij} t_{ijk} + \epsilon_{ijk}, \quad k = 1, \dots, n_{ij},$$

where

$$W_{ijk} = \ln X_{ijk},$$

t_{ijk} = the midpoint of the k -th sampling time period,

n_{ij} = the number of filter measurements collected,

and

ϵ_{ijk} = the random measurement error.

The error terms are assumed to be independent and normally distributed with mean zero and variance σ_ϵ^2 . That is

$$\epsilon_{ijk} \sim N(0, \sigma_\epsilon^2).$$

For the j -th shot α_{ij} and β_{ij} are constants with $\beta_{ij} < 0$ so that α_{ij} represents the initial and maximum log-concentration of airborne material. Shot-to-shot variability is accounted for by letting

$$\alpha_{ij} = \alpha_i + g_{ij},$$

where

α_i = the average log-concentration of airborne material
among shots using model size i ,

and

g_{ij} = random effect on the log-concentration due to
shot-to-shot differences.

The random effects g_{ij} are assumed to be independent of the measurement errors and normally distributed with mean zero and variance σ_g^2 . That is

$$g_{ij} \sim N(0, \sigma_g^2).$$

Estimates of the initial log-concentration were obtained using the method of restricted maximum likelihood with $\beta_{ij} < 0$. Thus the estimate of α_{ij} is given by

$$\hat{\alpha}_{ij} = \bar{w}_{ij} - \hat{\beta}_{ij} \bar{t}_{ij},$$

where

$$\bar{w}_{ij} = \sum_k w_{ijk}/n_{ij},$$

$$\bar{t}_{ij} = \sum_k t_{ijk}/n_{ij},$$

and $\hat{\beta}_{ij}$ is the minimum of zero and the usual least squares estimate of the slope, β_{ij} , from the regression equation $w_{ijk} = \alpha_{ij} + \beta_{ij} t_{ijk}$. The restriction $\beta_{ij} < 0$ is necessary to insure that large measurement errors do not result in an unrealistically low estimate of the initial concentration, since the airborne concentration will not increase during the measurement period.

Conditioned on the j -th shot, the distribution of $\hat{\alpha}_{ij}$ is approximately normal with mean $\alpha_i + g_{ij}$ and variance $d_{ij} \sigma_e^2$, where

$$\begin{aligned} d_{ij} &= n_{ij}^{-1} + \bar{t}_{ij}^2 / \sum_k (t_{ijk} - \bar{t}_{ij})^2, \text{ if } \hat{\beta}_{ij} < 0 \\ &= n_{ij}^{-1}, \text{ if } \hat{\beta}_{ij} = 0. \end{aligned}$$

The estimate of the initial concentration of airborne material on the j -th shot is given by

$$\hat{D}_{ij} = \exp \{ \hat{\alpha}_{ij} \}.$$

Finally the estimate of the error variance σ_e^2 is the average of estimates obtained from all shots in an experiment (HTGR, NPR, or HLW). That is

$$\hat{\sigma}_e^2 = \frac{\sum_i \sum_j (n_{ij}-1) s_{ij}^2}{\sum_i \sum_j (n_{ij}-1)}$$

where S_{ij}^2 is the usual unbiased estimate of the error variance in the normal linear regression of W_{ijk} on t_{ijk} .

B.1.2 A Model for Cascade Impactor Measurements

During the first five minutes, following the detonation of the charge, air was drawn through two parallel cascade impactors consisting of six stages stacked in a column. The total mass of particles having diameters in each of seven interval ranges was determined from the analyses of the deposits in the six stages and on a backup filter.

The estimated fraction of respirable mass obtained from impactor k on shot j of model size i is denoted by $\hat{p}_{ijk}$. It was calculated from the empirical distribution using the formula $\hat{p}_{ijk} = X_{ijk}/M_{ijk}$ where X_{ijk} is the total mass of particles collected on impactor k with diameters less than $4 \mu\text{m}$ and M_{ijk} is the total mass of all particles collected on impactor k . We assume that $\hat{\rho}_{ijk} = \ln(\hat{p}_{ijk})$ has a normal distribution with mean $\rho_i + h_{ij}$ and variance σ_ρ^2 , where

ρ_i = average log-fraction of respirable mass achieved
with model size i ,

and

h_{ij} = random effect on the fraction respirable due to
shot-to-shot differences.

The random effect h_{ij} is assumed to be normally distributed with mean 0 and variance σ_h^2 . Thus σ_h^2 represents the variance of the log-fraction respirable that is due to the shot-to-shot differences. The variance term σ_ρ^2 combines the measurement error with the error due to the differences in the two cascade impactors. An estimate of σ_ρ^2 is given by

$$\hat{\sigma}_\rho^2 = \frac{\sum_i \sum_j (m_{ij}-1) T_{ij}^2}{\sum_i \sum_j (m_{ij}-1)},$$

where m_{ij} is the number of impactor measurements collected ($m_{ij} = 1$ or 2) and

$$T_{ij}^2 = (\hat{\rho}_{ij1} - \hat{\rho}_{ij2})^2/2, \text{ if } m_{ij} = 2.$$

Averaging the estimates from the two impactors,

$$\hat{\rho}_{ij} = \sum_k \hat{\rho}_{ijk} / m_{ij} ,$$

the estimated respirable fraction of the total mass released on the j-th shot is given by

$$\hat{P}_{ij} = \exp \{ \hat{\rho}_{ij} \} .$$

Since the distribution of $\hat{\rho}_{ij}$ is $N(\rho_i + h_{ij}, \sigma_\rho^2/m_{ij})$, the distribution of $\hat{P}_{ij}$ is log-normal.

The cascade impactor data was also used to compute an alternative estimate of the respirable mass released on each shot. This estimate is given by

$$\tilde{R}_{ij} = \frac{V X_{ij}}{F_{ij} T_{ij}}$$

where V is the volume of the test chamber (l),

X_{ij} = total mass (μg) collected on shot j with particle size less than $4 \mu\text{m}$,

F_{ij} = flow rate (l/min) of the cascade impactor on shot j ,

and

T_{ij} = measurement time period (min.) for shot j .

Although this estimate is considered to be biased, it is useful to compare its value with the estimate R_{ij} when attempting to identify invalid test results.

B.1.3 Estimating the Full-Scale Respirable Release

For each shot in the HTGR, NPR, or HLW experiments, the estimated respirable mass released was computed according to the formula

$$\hat{R}_{ij} = V \hat{D}_{ij} \hat{P}_{ij} .$$

It is straightforward to show, based on the model and assumptions discussed in Section B.1.1 and B.1.2, that $\hat{R}_{ij}$ may be represented by

$$\hat{R}_{ij} = R_i A_{ij} E_{ij} ,$$

or

$$\ln \hat{R}_{ij} = \ln R_i + a_{ij} + e_{ij}$$

where $a_{ij} = \ln A_{ij}$ represents the random effect of shot-to-shot differences and $e_{ij} = \ln E_{ij}$ represents the random measurement errors. These random effects are independent and normally distributed. Specifically,

$$a_{ij} \sim N(0, \sigma_g^2 + \sigma_h^2)$$

and

$$e_{ij} \sim N(0, d_{ij} \sigma_e^2 + \sigma_p^2/m_{ij}) .$$

The quantity R_i represents the median respirable release from all direct hit attacks on casks of model size i . From the scaling equation we have $R_i = CM_i$, where M_i is the corresponding momentum transferred to the target. Substituting into the previous equations we have

$$\ln (\hat{R}_{ij}/M_i) = \ln C + a_{ij} + e_{ij} .$$

For convenience, let $Y_{ij} = \ln (\hat{R}_{ij}/M_i)$ represent the observed response from the j -th shot. Finally we compute the estimate of C by

$$\hat{C} = \exp \{ \bar{Y} \} ,$$

where $\bar{Y}$ is the average of the observed responses. The estimated proportionally constant is then applied to the scaling equation to obtain

$$\hat{R}_{fS} = \hat{C} M_{fS} ,$$

where $\hat{R}_{fS}$ is the estimated median full-scale respirable release and M_{fS} is the calculated momentum transferred to a full-scale target.

B.1.4 Confidence Intervals and Variance Component Estimates

An approximate 95 percent confidence interval for C was computed in each of the three experiments. The confidence intervals denoted by (C_L, C_U) , characterize the statistical precision of the estimates C.

From the model presented in Section B.1.2 we have

$$\text{Var } \{Y_{ij}\} = \tau^2 + n_{ij}^2 ,$$

where $\tau^2 = \sigma_g^2 + \sigma_h^2$ and $n_{ij}^2 = d_{ij} \sigma_e^2 + \sigma_p^2/m_{ij}$.
Therefore

$$\text{Var } \{\bar{Y}\} = (\tau^2 + \bar{n}^2)/N,$$

where N is the number of valid tests performed and

$$\begin{aligned} \bar{n}^2 &= \sum_i \sum_j n_{ij}^2 / N = \bar{d} \sigma_e^2 + \bar{f} \sigma_p^2 , \\ \bar{f} &= \sum_i \sum_j (m_{ij})^{-1} / N , \end{aligned}$$

and

$$\bar{d} = \sum_i \sum_j d_{ij} / N.$$

The variance estimates $\hat{\sigma}_e^2$ and $\hat{\sigma}_p^2$ (from Sections B.1.1 and B.1.2, respectively) were used to obtain the estimate of the overall measurement variance component

$$\hat{n}^2 = \bar{d} \hat{\sigma}_e^2 + \bar{f} \hat{\sigma}_p^2 .$$

Finally by letting

$$s_y^2 = \sum_i \sum_j (Y_{ij} - \bar{Y})^2 / (N-1)$$

we obtain the estimate of the overall shot-to-shot variance component

$$\hat{\tau}^2 = s_y^2 - \hat{n}^2 .$$

The approximate confidence limits for C were then computed according to the formulas

$$C_L = \exp \{ \bar{Y} - 2 \hat{\sigma}_{\bar{y}} \}$$

and

$$C_U = \exp \{ \bar{Y} + 2 \hat{\sigma}_{\bar{y}} \} ,$$

where

$$\hat{\sigma}_{\bar{y}} = (\hat{\tau}^2 + \hat{\eta}^2) / N .$$

Confidence limits for the median respirable release in a full-scale attack were computed using the formulas

$$R_L = C_L M_{fs}$$

and

$$R_U = C_U M_{fs} .$$

To determine an upper bound for the respirable release in a full-scale attack the following model was used: Let R_{ij} represent the actual release achieved on the j-th attack on a cask with calculated momentum transfer of M_i . Then according to the model presented we have

$$R_{ij} = C M_i A_{ij} = R_i A_{ij} ,$$

where $a_{ij} = \ln A_{ij}$ is the random effect of shot-to-shot differences. Since a_{ij} has a $N(0, \tau^2)$ distribution, no more than 2.5 percent of the direct hit attacks on such a shipment would exceed the value $R_i \cdot \exp \{2\tau\}$. Thus the estimated upper bound on the respirable release from a single attack is given by

$$R_i^* = \hat{C} M_i \exp \{2\tau\} .$$

The application of the upper bound to the full-scale attack is based on the assumption that the amount of variation, from shot-to-shot, as a percent of the average release is constant with respect to the momentum transferred to the target.

B.2 Experimental Results

B.2.1 Basic Measurement Data

The data from the air samples collected in the three series of experiments are presented in the following tables. There are four tables for each experiment. The first table for each experiment shows the samples which were successfully collected for each test. The second table gives the mass of uranium or thorium which was collected in each size range by the cascade impactors, the time period after the shot in which the sample was collected, and the volumetric flow rate of air through the sampler. The third table gives the mass of uranium or thorium collected on the Millipore® filters, the time period after the shot in which the sample was collected, and the volumetric flow rate of air through the filter. The concentration of uranium or thorium in the chamber during each sampling period was calculated from the mass of metal on the filter and the total volumetric flow of air through the filter. The concentrations were corrected for dilution of the air in the chamber during the sampling operation. The fourth table gives the masses of uranium or thorium collected on each deposition plate during each test. The location of the deposition plate is given as the numbers on a clock dial with the jet traveling from 6 to 12.

TABLE B-1. SAMPLES COLLECTED DURING TEN HTGR TESTS

Test No.	<u>Cascade Impactors</u>		<u>Filters</u>			<u>Deposition Plates</u>			
	A	B	A	B	C	2	4	8	10
1	-	x	-	x	x	-	-	-	-
2	x	x	-	-	x	-	-	-	-
3	-	x	-	-	-	x	x	x	x
4	-	x	-	x	x	x	x	x	x
5	-	-	-	x	x	x	x	x	x
6	-	-	x	x	x	x	x	x	x
7	x	-	-	x	x	x	x	x	x
8	x	x	-	x	x	x	x	x	x
9	x	x	x	-	x	-	-	-	-
10*	-	-	-	-	-	-	-	-	-

* A power failure occurred after the shot so air samples were not collected and deposition plate samples were not analyzed.

TABLE B-2. CASCADE IMPACTOR DATA FROM HTGR TESTS

Model i	Test j	Impactor k	Time Period, min.	Flow Rate l/min.	Mass of Thorium (ug) per Size Range (um)							Percent Respirable $\hat{P}_{ijk}$
					16	8-16	4-8	2-4	1-2	0.5-1.0	0.5	
1	1	B	0.5 to 5.0	12.35	65	31	29	100	46	28	60	65.2
1	2	A	0.5 to 5.0	12.8	55	58	119	34	21	10	110	42.3
1	2	B	0.5 to 5.0	12.35	150	19	125	120	77	55	63	51.7
2	3	B	0.5 to 5.0	12.35	32	12	42	61	16	10	24	55.2
2	4	B	0.5 to 5.0	12.35	77	15	142	27	130	10	20	43.8
2	7	A	0.5 to 5.5	11.7	240	24	85	170	58	32	16	44.2
2	8	A	0.5 to 5.5	12.9	270	10	128	53	65	24	10	26.7
2	8	B	0.5 to 5.5	11.7	228	52	39	260	56	30	22	53.6
2	9	A	0.5 to 5.5	12.9	400	100	142	185	67	10	12	29.5
2	9	B	0.5 to 5.5	11.7	295	86	162	335	58	24	21	44.6

TABLE B-3. FILTER DATA FROM HTGR TESTS

Model i	Test j	Filter k	Time Period min.	Midpoint t_{ijk}	Flow Rate l/min	Thorium Concentration ug/l X_{ijk}
1	1	B	5 - 6	5.5	9.9	7.85
1	1	C	10 - 12	11.0	9.9	8.80
1	2	C	10 - 12	11.0	9.9	7.71
2	4	B	10 - 12	11.0	10.0	13.00
2	4	C	15 - 17	16.0	12.4	5.03
3	5	B	10 - 12	11.0	10.0	2.02
3	5	C	20 - 25	22.5	12.4	1.32
3	6	A	6 - 9	7.5	14.0	7.33
3	6	B	10 - 12	11.0	10.4	7.50
3	6	C	20 - 25	22.5	11.8	2.20
2	7	B	15 - 21	18.0	11.6	8.96
2	7	C	25 - 40	32.5	12.4	0.99
2	8	B	15 - 22	18.5	11.6	7.95
2	8	C	25 - 41	33.0	12.4	3.54
2	9	A	5 - 11	8.0	13.3	0.49
2	9	C	25 - 40	32.5	12.2	5.05

TABLE B-4. DEPOSITION PLATE DATA FROM HTGR TEST

Test No./Plate No.	Mass of Thorium, ug*			
	2	4	8	10
3	1,100	860	1,100	1,600
4	833	279	277	287
5	1,170	1,200	410	180
6	680	790	575	581
7	5,970	4,825	5,275	1,043
8	34,100	24,600	15,700	3,850
9	3,080	6,050	3,080	3,450

* The average area of each deposition plate was 79 cm²; the area of the chamber floor was 6295 cm².

TABLE B-5. SAMPLES COLLECTED DURING FOUR NPR TESTS

Test No.	Cascade Impactors		Filters			Deposition Plates					Extra
	A	B	A	B	C	2	4	8	10		
1	-	-	x	x	x	x	x	x	x		
2	x	x	x	x	x	x	x	x	x		
3	x	x	x	x	x	x	x	x	x	x	
4	x	x	x	x	x	x	x	x	x		

TABLE B-6. CASCADE IMPACTOR DATA FROM NPR TESTS

Model i	Test j	Impactor k	Time Period, min.	Flow Rate ℓ/min.	Mass of Uranium (ug) per Size Range (um)							Percent Respirable P _{ijk}
					16	8-16	4-8	2-4	1-2	0.5-1.0	0.5	
1	2	A	0.5 to 5.0	14.6	0.9	2.7	5.5	10.2	5.5	17.9	27.8	87.1
1	2	B	0.5 to 5.0	14.9	6.8	1.7	11.0	20.8	32.4	28.7	37.8	86.0
1	3	A	0.5 to 5.0	12.6	52.9	6.6	24.7	43.5	77.8	160	114	82.4
1	3	B	0.5 to 5.0	12.6	55.2	10.7	30.9	56.1	112	363	205	88.4
1	4	A	0.5 to 5.0	12.6	84.8	12.7	28.9	54.2	55.2	60.0	71.9	65.6
1	4	B	0.5 to 5.0	12.6	174	20.4	41.0	41.3	35.4	68.0	91.3	50.1

TABLE B-7. FILTER DATA FROM NPR TESTS

Model i	Test j	Filter k	Time Period min.	Midpoint t_{ijk}	Flow Rate l/min	Uranium Concentration ug/l X_{ijk}
1	1	A	5 - 10	7.5	14.6	0.006
1	1	B	10 - 12	11.0	14.9	0.01
1	1	C	20 - 25	22.5	14.9	3.50
1	2	A	5 - 12	8.5	14.6	0.07
1	2	B	15 - 22	18.5	14.9	0.03
1	2	C	20 - 25	22.5	14.9	0.06
1	3	A	5 - 10	7.5	14.6	51.3
1	3	B	15 - 20	17.5	14.9	26.4
1	3	C	25 - 40	32.5	14.9	15.6
1	4	A	5 - 10	7.5	14.6	15.1
1	4	B	15 - 20	17.5	14.9	10.5
1	4	C	25 - 40	32.5	14.9	7.7

TABLE B-8. DEPOSITION PLATE DATA FROM NPR TESTS

Test Number/Plate Number	Mass of Uranium, ug*				Extra
	2	4	8	10	
1	478	824	620	647	--
2	559	1893	924	1247	-
3	2922	5096	2588	3309	4042
4	2586	3860	4990	3144	--

* The average area of each deposition plate was 79 cm²; the area of the test chamber floor was 6295 cm².

TABLE B-9. SAMPLES COLLECTED DURING SIX HLW TESTS

Test No.	Cascade Impactors		Filters			Deposition Plates			
	A	B	A	B	C	2	4	8	10
1	x	x	-	x	x	x	x	x	x
2	x	x	x	x	x	x	x	x	x
3	x	x	x	x	x	x	x	-	-
4	x	x	x	x	x	x	x	x	-
5	x	x	x	x	x	x	x	x	x
6	x	-	x	x	x	x	x	x	x

TABLE B-10. CASCADE IMPACTOR DATA FOR HLW TESTS

Model i	Test j	Impactor k	Time Period, min.	Flow Rate L/min.	Mass of Uranium (ug) per Size Range (um)							Percent Respirable P _{ijk}
					16	8-16	4-8	2-4	1-2	0.5-1.0	0.5	
1	1	A	0.5 to 5.5	12.0	67.3	7.3	41.9	11.5	8.3	3.7	4.0	19.1
1	1	B	0.5 to 5.5	11.5	50.0	6.0	13.6	10.4	5.5	2.3	6.7	26.3
1	2	A	0.5 to 5.5	12.0	61.2	11.8	20.5	30.6	8.9	3.9	10.7	36.7
1	2	B	0.5 to 5.5	11.5	33.9	11.3	15.2	23.3	11.4	7.7	12.9	47.8
1	3	A	0.5 to 5.5	12.5	2.1	0.4	0.8	0.5	0	0.1	0.3	21.4
1	3	B	0.5 to 5.5	11.7	0.2	0.7	1.8	0.9	0.6	2.3	0.2	59.7
1	4	A	0.5 to 5.5	12.5	6.6	1.2	2.6	3.2	1.0	2.0	0.7	39.9
1	4	B	0.5 to 5.5	11.7	7.4	1.5	2.7	6.8	2.5	1.7	1.2	29.0
2	5	A	0.5 to 5.5	12.5	11.3	1.3	4.2	2.2	1.9	1.2	1.8	27.7
2	5	B	0.5 to 5.5	11.7	15.8	2.7	2.4	8.1	2.4	1.1	1.1	37.8
2	6	A	0.5 to 5.5	12.5	6.4	2.2	1.4	2.2	0.7	0.4	1.1	30.6

TABLE B-11. FILTER DATA FROM HLW TESTS

Model i	Test j	Filter k	Time Period min.	Midpoint t_{ijk}	Flow Rate l/min	Uranium Concentration ug/l X_{ijk}
1	1	B	15 - 20	17.5	14.0	1.680
1	1	C	25 - 40	32.5	14.1	0.186
1	2	A	5 - 10	7.5	14.0	0.874
1	2	B	15 - 20	17.5	14.0	2.16
1	2	C	24 - 39	31.5	14.0	0.240
1	3	A	5 - 10	7.5	11.4	0.209
1	3	B	15 - 21	18.0	10.8	0.096
1	3	C	25 - 40	32.5	11.9	0.074
1	4	A	5 - 11	8.0	11.3	0.569
1	4	B	15 - 21	18.0	11.6	0.310
1	4	C	25 - 40	32.5	11.8	0.196
2	5	A	5 - 11	8.0	11.3	0.521
2	5	B	15 - 20	17.5	11.4	0.403
2	5	C	25 - 40	32.5	11.9	0.260
2	6	A	5 - 11	8.0	10.7	0.509
2	6	B	15 - 21	18.0	10.9	0.280
2	6	C	25 - 40	32.5	11.4	0.148

TABLE B-12. DEPOSITION PLATE DATA FROM HLW TESTS

Test No./Plate No.	Mass of Thorium, ug*			
	2	4	8	10
1	790	1281	872	852
2	3,405	635	156	674
3	14.7	84.7	--	--
4	156	334	221	--
5	291	326	203	365
6	308	256	14.7	805

* The average area of each deposition plate was 79 cm²; the area of the chamber floor was 6295 cm².

B.2.2 Estimates and Statistical Uncertainty Limits

Selected parameters and estimates for individual shots are presented in Tables B-13, B-14, and B-15 for the HTGR, NPR, and HLW experiments, respectively. The description and formulas for the estimates were given in Section B.1; however, for convenience, descriptions of the important quantities are presented below. In each case, the subscripts i and j refer to the j -th shot with model size i .

M_i	=	Calculated momentum transferred ($\text{mm}^4/\mu\text{sec}$) to the i -th scale model target.
N_{ij}	=	Number of Millipore filter measurements successfully collected on the j -th shot.
$\hat{D}_{ij}$	=	Estimated initial and maximum concentration ($\mu\text{g/l}$) of thorium or uranium achieved on the j -th shot.
m_{ij}	=	Number of cascade impactor estimates of the fraction respirable mass achieved on the j -th shot.
$\hat{P}_{ij}$	=	Estimated fraction of the total mass of thorium or uranium that is considered respirable.
$\hat{R}_{ij}$	=	Estimated respirable mass (μg) of thorium or uranium achieved on the j -th shot.
$\tilde{R}_{ij}$	=	Alternative estimate of respirable mass determined solely from the cascade impactor data.
Dist.	=	Distance (inches) between the center of the jet entrance hole and the centerline of the glass cylinder--HLW experiments only.
VD	=	Volume (in^3) of glass disrupted, in HLW tests by calculating the volume common to intersecting cylinders assuming the jet hole is uniform .5 in. diameter. Glass diameter in shots 1, 2, 3, and 4 is 1.25 in. and 2.75 in. in shots 5 and 6.
Fans	=	1 if mixing fans were used; otherwise, Fans = 0.

TABLE B-13. ESTIMATES FOR HTGR TESTS

Model (i)	M _i	Shot j	n _{ij}	$\hat{D}_{ij}$	m _{ij}	$\hat{P}_{ij}$	$\hat{R}_{ij}$	$\tilde{R}_{ij}$	Fans	r _{ij}	$\overline{DP}_{ij}$
1	652	1	2	8.312	1	0.6520	3.6310	2.8210	1	*	*
1	652	2	1	7.710	2	0.4230	2.1850	2.9166	1	*	*
2	456	3	0	*	1	0.5520	*	1.3382	1	4	1165.0
2	456	4	2	105.109	1	0.4380	30.8465	2.2544	1	4	419.0
3	365	5	2	3.035	0	(0.4500) ⁽¹⁾	0.9150	*	1	4	740.0
3	365	6	3	16.198	0	(0.4500)	4.8838	*	1	4	656.5
2	456	7	2	138.075	1	0.4420	40.8915	3.1610	0	4	4278.2
2	456	8	2	21.843	2	0.3783	5.5363	2.8968	0	4	19562.5
2	456	9	2	1.573	2	0.3627	0.3823	3.9313	0	*	3915.0

(1) Based on shots 1, 2, 4, 7, and 8.

APPENDIX C

PROCEDURES FOR SAMPLE ANALYSES

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TABLE B-14. ESTIMATES FOR NPR TESTS

Model (i)	M_i	Shot j	n_{ij}	$\hat{D}_{ij}$	m_{ij}	$\hat{P}_{ij}$	$\hat{R}_{ij}$	$\tilde{R}_{ij}$	Fans	r_{ij}	$\overline{DP}_{ij}$
1	21.7	1	3	0.0594	0	*	*	*	1	4	642.25
1	21.7	2	3	0.0759	2	0.8655	0.0440	0.9111	1	4	1155.75
1	21.7	3	3	67.5859	2	0.8535	38.6452	6.6846	0	5	3591.40
1	21.7	4	3	17.7467	2	0.5733	6.8163	2.7964	0	4	3645.00

TABLE B-15. ESTIMATES FOR HLW TESTS

Model (i)	M _i	Shot j	n _{ij}	$\hat{D}_{ij}$	m _{ij}	$\hat{P}_{ij}$	$\hat{R}_{ij}$	$\tilde{R}_{ij}$	Fans	r _{ij}	$\overline{DP}_{ij}$	Dist.	VD
1	284	1	2	(0.673007) ⁽¹⁾	2	0.2231	0.100599	0.29	0	4	948.75	0.0	.2399
1	284	2	3	(0.994167)	2	0.4190	0.279093	0.62	0	4	1217.50	0.0	.2399
1	284	3	3	0.251327	2	0.3570	0.060100	0.02	0	2	49.70	0.5	.1244
1	284	4	3	0.747890	2	0.3396	0.170200	0.10	0	3	237.00	0.25	.2178
2	570	5	3	0.656390	2	0.3362	0.147900	0.11	0	4	296.25	0.28	.5251
2	570	6	3	0.727457	1	0.3073	0.149800	0.14	0	4	345.92	0.0	.5365

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(1) Estimates modified due to large measurement error.

r_{ij} = Number of measurements from deposition plates obtained on the j-th shot.

$\overline{DP}_{ij}$ = Average mass (μg) of thorium or uranium collected on the deposition plates.

After examining the results from the individual shots it was decided that only data from shots 1,2,4,5,6,7, and 8 in the HTGR experiments, shots 3 and 4 from the NPR experiments, and shots 1,2,4,5, and 6 in the HLW experiments would be used in the analyses. A discussion of the data not used was presented in the Test Results Sections 3.3, 4.3, and 5.3.

Since cascade impactor measurements were not available for shots 5 and 6 in the HTGR experiments, the average fraction respirable obtained from the other shots was used to complete the calculations. In the HLW experiments, separate analyses were performed on data from the two scale models due to a modification in the application of the scaling law. For shots 1 and 2 of the HLW experiments the estimated uranium concentration determined from samples collected in Millipore® filter B were unusually high compared to the corresponding estimates obtained with the other shots and also when compared to the estimates determined from samples collected in filters A and C. Instead of eliminating the filter B measurements, we chose a more conservative approach by modifying the regression equations used in calculating the initial concentrations in shots 1 and 2. The slopes of these modified equations were taken to be the average slope determined from shots 4, 5, and 6.

Table B-16 presents the estimates and 95 percent confidence limits for the median respirable release in a full-scale attack on each of the four types of shipments. Estimates of the maximum respirable release, exceeded by only 2.5 percent of the direct hit attacks; the proportionality constant; and the variance components are also presented. The percent variation due to measurement error was computed by the formula

$$\hat{v} = 100 \left(1 - \frac{\hat{\tau}}{\hat{\eta}^2 + \hat{\tau}^2} \right) .$$

In the HTGR tests, for example, 19 percent of the variation in individual test results can be attributed to errors in measurement. Thus 81 percent of the variation is due to shot-to-shot differences.

TABLE B-16. FINAL ESTIMATES

			HTGR	NPR	HLW(1)	HLW(2)
Median Respirable Release	$\hat{R}_{fs}$	(gm.)	1.015	4.567	0.035	0.020
95% Confidence Interval	(R_L, R_U)		(.353, 2.917)	(.806, 25.7)	(.023, .052)	(.013, .029)
Maximum Release	R^*_{fs}	(gm.)	9.765	47.6	0.085	0.049
Percent of Median	$100 R^*_{fs}/\hat{R}_{fs}$	%	962	1042	245	245
Momentum Transferred	M_{fs}	(mm ⁴ /usec)	81571	6106	58734	77892
Proportionality Constant	$\hat{C}$	($\frac{mg}{mm^4 usec}$)	12.44×10^{-3}	.748	5.93×10^{-7}	2.61×10^{-7}
95% Confidence Interval	(C_L, C_U)		$(4.3, 35.8) \times 10^{-3}$	(.132, 4.24)	$(3.30, 10.6) \times 10^{-7}$	$(1.2, 5.6) \times 10^{-7}$
Estimate of $\ln C$	$\bar{Y}$		-4.387	-.2905	-7.430	-8.251
Measurement Variance Component	$\hat{\sigma}_n^2$		.670	.133	.221	.261
Shot-to-Shot Variance Component	$\hat{\sigma}_t^2$		1.281	1.372	.0398	.0398
Percent Variation Due to Measurement	$\hat{v}$	%	19.0	4.5	61.0	63.6

B.3 Model Assumptions and Limitations

In order to characterize the uncertainty in the estimates presented in this report, it is necessary to make some assumptions concerning the distribution of both measurement and experimental errors. Nonparametric estimates, requiring no distributional assumptions, can be used for certain parameters; however, because of the small sample sizes, there would be unacceptably large uncertainties associated with these estimates. Furthermore, the interpretation of the variance components is not possible without distributional assumptions.

The assumptions presented in Section B.1 appear to be reasonable, but there was not sufficient data available to statistically verify them. The small amount of data also made it impractical to be conservative in calculating upper bounds for the respirable amount released in a single attack. Instead of stating with 95 percent confidence, for example, that no more than 2.5 percent of the direct hit attacks will result in a release of more than R^* , we only estimated the release that would be exceeded in 2.5 percent of the direct hit attacks.

In order to overcome these limitations, it would be necessary to increase the number of test shots performed to perhaps twenty shots per experiment. Considering the precision required, we feel this would not be practical.

APPENDIX C

PROCEDURES FOR SAMPLE ANALYSES

Generalized Procedures for Thorium Analyses

Thorium carbide deposits on filters, glass slides, and deposition plates were prepared using a burn and leach procedure and analyzed by gross alpha counting. First, samples containing thorium carbide were oxidized to thorium oxide at 900 C for 6 hours. In the case of samples collected on glass fiber filters, thermal oxidation was preceded by a concentrated HF digestion step to remove silica. Then the thorium oxide was digested in concentrated HNO_3 /0.05 M HF at 90 C until completely dissolved. The digested solution was concentrated by evaporation and then transferred to a tared aluminum counting planchet, which was heated in an open flame to eliminate water. The planchets were then stored in a desiccator until analysis.

Standards were prepared from fresh, reagent-grade $\text{Th}(\text{NO}_3)_4 \cdot 4\text{H}_2\text{O}$. Aliquots of a concentrated stock solution and appropriately diluted standards were evaporated on counting planchets, with final thorium contents ranging from 10 micrograms (the method detection limit) to 10 milligrams. A calibration curve was constructed from the standard counting data. The effect of shielding by substances other than thorium, such as graphite and iron, was assessed by counting a series of 1 milligram thorium standards, to which NaCl had been added in amounts ranging from 1 to 500 milligrams. A Shielding Correction Factor was calculated as the ratio of the count rate for an unshielded 1 milligram thorium standard divided by the count rate for each of the shielded standards.

Final quantification of an unknown thorium sample was achieved by multiplying the background-corrected alpha count rate of the unknown by the appropriate Shielding Correction Factor and then by comparison of the corrected unknown count rate with the standard calibration curve.

Generalized Procedures for Uranium Analysis
of NPR Fuel and Simulated HLW Glass

Uranium-containing deposits on filters, glass slides, and deposition plates were digested in acid and analyzed by fluorometry. Samples containing NPR fuel were digested in 50 milliliters 1:10 HCl at 90 C and evaporated to 5-10 ml final volume. In the case of samples collected on glass fiber filters, the HCl digestion was preceded by a concentrated HF digestion step to remove silica. Samples containing simulated HLW glass were prepared using a sequential HF, HOOH, H₂SO₄, and HNO₃ digestion procedure at 90 C. Digestion was complete when the supernate was clear and had evaporated to a volume of approximately 5 to 10 milliliters. Fully digested samples were transferred to volumetric flasks, diluted to the appropriate volume marker with deionized water, and stored until analysis.

In preparation for the fluorometric analyses, a 100 microliter aliquot of each digested sample was fused in 90% NaF/10% Na₂CO₃ flux on a platinum disc using a Rotary Fusion Burner fueled by propane. Each sample was analyzed in triplicate, along with 10 milligram per liter uranium standards, (prepared from fresh, reagent-grade UO₂(NO₃)₂·6H₂O), blanks, and standard additions (spiked) samples. The standard additions samples permitted an evaluation of the extent of quenching, or inhibition of fluorescence. Samples having un-acceptedly high quench factors were reanalyzed. The total uranium concentration of a digested sample was determined by comparing the fluorescence of the unknown with that of the 10 milligram per liter uranium standard. Total uranium content of each unknown in micrograms was calculated by multiplying the uranium concentration of the unknown in micrograms per milliliter by the final volume of digested sample in milliliters.

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N.E. Miller, A.W. Fentiman, M.R. Kuhlman,
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12 SUPPLEMENTARY NOTES

13 ABSTRACT (200 words or less)

The Nuclear Regulatory Commission (NRC) promulgated a rule, 10 CFR 73.37, which established requirements for safeguarding shipments of spent fuel to reduce the risk from acts of sabotage of highly radioactive materials. After the rule became effective, experimental programs conducted by Battelle for the NRC and by Sandia for the DOE showed the consequences of an attack using explosives on a shipment of PWR spent fuel were significantly less than had been indicated by earlier analytical studies. As a result, NRC is considering modifying the safeguards requirements.

In support of NRC's efforts to modify the rule, Battelle has conducted additional experimental studies to evaluate the consequences of attacks on shipments of high-temperature gas-cooled reactor (HTGR) spent fuel, non-power reactor (NPR) spent fuel, and vitrified high-level waste (HLW). Model casks containing surrogates of the spent fuels or high-level waste were penetrated by the jet from a precision shaped charge. Air samples collected after each test were used to estimate the quantities of respirable material released after the cask was penetrated. Results of the tests were scaled by specially developed scaling factors to estimate the releases that may occur from attacks on full-sized shipments of the materials. It was concluded that the sabotage of shipments of HTGR spent fuel, NPR spent fuel, or vitrified HLW should have no greater consequences than those predicted for shipments of PWR spent fuel.

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1. Sabotage
2. Transportation Casks
3. HTGR spent fuel
4. Non-power reactor spent fuel
5. Vitrified high-level waste

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APPENDIX A

DEVELOPMENT AND APPLICATION OF THE SCALING LAW

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APPENDIX B

STATISTICAL MODEL AND DATA ANALYSIS

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