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United States Nuclear Regulatory Commission Spent Fuel Transportation Risk Assessment

Draft Report for Comment

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

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Division of Spent Fuel Storage and Transport
Office of Nuclear Materials Safety and Safeguards
U.S. Nuclear Regulatory Commission
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ABSTRACT

The U. S. Nuclear Regulatory Commission (NRC) is responsible for promulgating regulations for the packaging of spent fuel (and other large quantities of radioactive material) for transport that provide for public health and safety during transport (Title 10 of the Code of Federal Regulations, Part 71). In 1977 the NRC published NUREG-0170, an assessment of the adequacy of those regulations to provide the assurance of safety. In that assessment, the measure of safety was risk of radiation doses to the public under routine and accident transport conditions, and the risk was found to be acceptable. Since that time there have been two affirmations of this conclusion for spent nuclear fuel transportation, each using improved tools and information that supported the earlier studies. This report presents the results of a fourth investigation into the safety of spent nuclear fuel transportation. The risks associated with spent nuclear fuel transportation come from both the radiation given off by the spent fuel, which is attenuated (but not eliminated), by the shielding provided by transportation casks, and the possibility of the release of some quantity of radioactive material during a severe accident. This investigation shows the risk from the radiation emitted from the casks to be a small fraction of that from naturally occurring background radiation and the risk from accidental release of radioactive material to be several orders of magnitude less than from naturally occurring background. Because there have been only minor changes to the radioactive material transportation regulations between NUREG-0170 and this risk assessment, the calculated dose due to the external radiation from the cask under routine transport conditions is similar to that found in earlier studies. The improved analysis tools and techniques, improved data availability, and reduction in the number of conservative assumptions has made the estimate of average accident collective dose risk from the release of radioactive material in this study approximately five orders of magnitude less than was estimated in NUREG-0170. The results demonstrate that the regulations of the NRC continue to provide adequate protection of public health and safety during the transportation of spent nuclear fuel.

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ACKNOWLEDGEMENTS

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Comment [MF1]: 1) Table of Contents needs further checking. See list of contents for Chapter 5 as example. There is no Sect 5.5 listed and Sub-sections for Sect 5.5 are listed under Sect 5.4.

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ACRONYMS AND ABBREVIATIONS

Comment [MF2]: Some acronyms not listed here. A thorough search needs to be done for all acronyms.

ALARA	as low as reasonably achievable
AMAD	activity median aerodynamic diameter
Bq	becquerel
BWR	boiling water [nuclear] reactor
c.g.	center of gravity
CAFE	Container Analysis Fire Environment
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
Ci	curie
CoC	Certificate of Compliance
CRUD	Chalk River Unidentified Deposits
DOE	[U.S.] Department of Energy
DOT	[U.S.] Department of Transportation
DU	depleted uranium
EIS	Environmental Impact Statement
EPA	[U.S.] Environmental Protection Agency
EQPS	equivalent plastic strain
FE	finite element
FEM	finite element method
FDR	Final Design Report
G	gravitational acceleration
GWD	gigawatt days
HAC	hypothetical accident condition
HLW	high-level [radioactive] waste
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
km	kilometers
kph	kilometers per hours
ksi	kilopounds per square inch
LOS	loss of [lead] shielding
MEI	maximally exposed individual
MJ	million Joules
MN	million Newtons
MPC	multi-purpose canister
mph	miles per hour
mrem	millirem
MTU	metric tons of uranium
MWD	megawatt-days
NP	nuclear plant
NRC	[U.S.] Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PWR	pressurized water [nuclear] reactor
rem	Roentgen equivalent man
SAR	Safety Analysis Report

Deleted: Analysis

SNF	spent nuclear fuel
Sv	sieverts
TBq	terabecquerels
TEDE	total effective dose equivalent
TI	transport index

CHEMICAL SYMBOLS

Cm	curium
Co	cobalt
Cs	cesium
I	iodine
Kr	krypton
Pb	lead
Pu	plutonium
U	uranium

EXECUTIVE SUMMARY

The U. S. Nuclear Regulatory Commission (NRC) has conducted several risk assessments and other analyses to evaluate the safety of transportation of spent power reactor nuclear fuel over the past 35 years. Regulations, shipping practices, and cask designs for transporting radioactive material have remained essentially unchanged during this time. Therefore, the *actual* per-shipment risk over this period of time would also have remained essentially the same. What *has* changed during this period is the *calculated* risks, brought about by an improvement in the ability to evaluate the responses of the casks and their spent fuel contents to accident environments. The improvements include advancements in the tools available to determine those responses and to calculate the consequences and risks that result from their response. This has resulted in a decrease in the calculated per shipment risk. The calculated consequences and risks in this study are several orders of magnitude less than those calculated in previously performed risk assessments.

Comment [MF3]: The new Executive Summary is excellent. Small suggestions:
a. P. xvi discusses release of radioactive material from non-canistered rail packages for "exceptionally severe impacts" (1st para) and "exceptionally severe accidents" (4th bullet at bottom). Then 7th bullet at bottom notes "extra-regulatory accidents (accidents involving a release of radioactive material and loss of lead shielding accidents)". If the "exceptionally severe accidents" and "exceptionally severe impacts" are indeed beyond regulatory DBA, then perhaps the "extra-regulatory accidents" wording should be used consistently along with a brief explanation of what "extra-regulatory" means.
b. P. xvi – 1st 5 words seem to be editing error.

Deleted: 40

Comment [MF4]: True for average accident collective dose risk, but not true for other conditions according to chapter 6. Suggest tightening up wording.

In this study the risk associated with the transportation of spent nuclear fuel was estimated by examining the behavior of three NRC-certified casks during routine transportation and in transportation accidents. Two of these casks are designed for transport by railroad: the first has steel gamma shielding and has an inner welded canister for the spent fuel; the other has lead gamma shielding and can transport spent fuel either within an inner welded canister (referred to in this report as canistered fuel) or without an inner canister (referred to as directly loaded fuel). The third cask has depleted uranium gamma shielding and is designed to transport directly loaded spent fuel by highway. The response of these casks is typical of that of other cask designs. The use of certified casks designs means this risk assessment includes the factors of safety that are typically included in cask designs but were not specifically considered in previous risk assessments.

The risks associated with routine shipments (incident free) and those associated with shipments where an accident occurs are calculated separately. In routine transportation, the risk and the consequence are the same. For this case, the dose to residents living along a transportation route, the dose to people sharing the highway or railway, the dose to people at stops, and the dose to transportation workers are all calculated. Regulations allow limited external radiation from the cask. The dose to members of the public from this radiation during routine transportation is a small fraction of the naturally occurring background radiation dose exposed individuals experience during the same time as the shipment.

If there is an accident during the shipment, the most likely result is that there is no damage to the cask, but the vehicle is stopped for a period of time, exposing people in the vicinity of this stop (nearby residents, emergency response workers, etc.) to the allowed external radiation from the cask. If the accident is more severe, the shielding effectiveness of the cask could be reduced. If the cask is involved in a fire, the plastic neutron shielding material could melt, resulting in a slightly elevated amount of radiation coming from the cask. If the lead shielded cask were to be involved in an exceptionally severe long-lasting fire, there could be a reduction in the effectiveness of the gamma shielding. The response of the cask to fire accidents was determined using detailed computer analyses. Even in the worst-case fires analyzed, no cask experienced a seal failure that could have led to a release of radioactive material from the spent fuel cask.

For impact accidents, the steel shielded cask with inner welded canister and the depleted uranium shielded cask have no release and no loss of shielding effectiveness even under the most severe impacts studied, which encompass all historic or even realistic accidents. The lead shielded cask experiences some loss of shielding effectiveness during severe impacts and in the case where spent fuel is transported without an inner welded canister there could be some release of radioactive material for exceptionally severe impacts.

If material were to be released, the dispersion of that material would be affected by the weather conditions at the accident location. The risk assessment uses national average weather conditions because the time and location of an accident are not known. The number of people exposed to the dispersed material is a function of the population density at the site of the accident, as determined from census data. The amount of material released, the dispersion, and the population density are combined to determine the consequence (potential affects) of a release. The estimated dose from the most severe accident scenarios evaluated in this study is less than that required to produce an immediate injury and is similar to a single dose from a cancer therapy regimen.

Risk assessments do not only Accident risk is the product of the consequence of the accident and its probability. The probability of an accident that has an effect on the cask is the product of the probability that the cask is involved in an accident and the conditional probability that the accident is severe enough to reduce the shielding or containment effectiveness of the cask. The conditional probability is based on state accident statistics for all types of heavy trucks and railcars. The accident probability is determined by multiplying these state-by-state accident rates by the distance traveled within each state. This was done for sixteen representative truck routes and sixteen representative rail routes.

Comment [MF5]: Sentence does not make sense.

The following findings are reached from this study:

- The collective dose risks from routine transportation are vanishingly small. Theses doses are about four to five orders of magnitude less than collective background radiation dose.
- The routes selected for this study adequately represent the routes for spent nuclear fuel transport, and there was relatively little variation in the risks per kilometer over these routes.
- Radioactive material would not be released in an accident if the fuel is contained in an inner welded canister inside the cask.
- Only rail casks without inner welded canisters would release radioactive material, and only then in exceptionally severe accidents.
- If there were an accident during a spent fuel shipment, there is only about one in a billion chance the accident would result in a release of radioactive material.
- If there were a release of radioactive material in a spent fuel shipment accident, the dose to the maximum exposed individual would be less than 2 Sv, about the dose given in a single radiotherapy treatment to cancer patients.
- The collective dose risks for the two types of extra-regulatory accidents (accidents involving a release of radioactive material and loss of lead shielding accidents) are negligible compared to the risk from a no-release, no-loss of shielding accident.
- The risk of loss of shielding from a fire is negligible.

- None of the fire accidents investigated in this study resulted in a release of radioactive material.

Based on these findings, this study reconfirms that radiological impacts from spent fuel transportation conducted in compliance with NRC regulations are low, in fact generally less than previous, already low, estimates. Accordingly, with respect to spent fuel transportation, the previous NRC conclusion that the regulations for transportation of radioactive material are adequate to protect the public against unreasonable risk is also reconfirmed by this study.

CHAPTER 1

INTRODUCTION

1.1 Organization of this Report

The body of the report consists of a public summary and six chapters. The chapters describe the risk analysis qualitatively. Each chapter in this study has an associated appendix that describes the analytical methods and calculations used to arrive at the results discussed in the chapters. Descriptions of programs, calculations, and codes used are in the relevant appendices.

1.1.1 Chapter 1 and Appendix I

This chapter provides an introduction to the study, a brief background discussion, a discussion of risk as applied to transportation of radioactive materials, a discussion of cask selection, and a review of the organization of the report. Appendix I contains a glossary of special terms used in this study.

1.1.2 Chapter 2 and Appendix II

Chapter 2 and Appendix II discuss RADTRAN analysis of incident-free transportation. During routine ("incident-free") transportation, spent fuel transportation casks deliver an external dose to anyone in proximity to the shipment. This chapter describes the consequence of the external dose. In most previous transportation risk studies the regulatory maximum dose rate, 0.1 mSv/hour at two meters from the cask, was assumed to be the external dose rate from every cask evaluated in the particular study. The present study uses the actual predicted external dose rate from NRC certified casks, as reported in the Safety Analysis Reports (SARs) for those casks.

1.1.3 Chapter 3 and Appendix III

Chapter 3 and Appendix III address the structural analyses used to determine the cask response to accidents and the parameters that determine loss of lead gamma shielding and releases of radioactive material. The results of detailed analyses of impacts of the casks with impact limiters onto rigid targets at speeds of 48, 97, 145, and 193 kph (30, 60, 90, and 120 mph) in end, corner, and side-on orientations are given. Also provided are results for impacts onto other surfaces or other objects. The response of the fuel assemblies carried by the casks is also discussed.

1.1.4 Chapter 4 and Appendix IV

Chapter 4 and Appendix IV address the thermal analyses used to determine the cask response to accidents and the parameters that determine loss of lead gamma shielding and potential releases of radioactive material. The results from analyses of fires that completely engulf the cask as well as ones that are off-set from the cask are given. The temperature response of the cask seals, the shielding material, and the spent fuel is provided.

1.1.5 Chapter 5 and Appendix V

Chapter 5 and Appendix V address RADTRAN analysis of transportation accidents, development of accident event trees and conditional probabilities, development of the radionuclide inventory and radioactive materials releases and dispersion of released material in

the environment. The chapter also discusses accidents in which there are no releases – the most likely accidents – in which the radioactive cargo is not affected at all, but the vehicle is held for many hours at the accident location before being permitted to continue.

1.1.6 Chapter 6 and Appendix VI

Chapter 6 summarizes the results of the analyses. Appendix VI contains a "plain language" summary of this study.

1.1.7 Bibliography

The bibliography is placed after the appendices. It contains all cited references and other bibliographic material. Citations in the text (e.g., Sprung et al., 2000, Figure 7.1) include specific page, figure, or table references where appropriate.

1.2 Historical Transportation Risk Studies and the Purpose of this Analysis

The purpose of this study is analysis of the radiological risks of transporting spent nuclear fuel in routine transportation and transportation accidents, using the latest available data and modeling techniques. This study primarily analyzes cask behavior rather than the behavior of the spent fuel being transported. It is the latest in a series of assessments of this type and analyzes the behavior of casks certified by NRC carrying fuel of known isotopic composition and burnup. The studies that preceded this one were based on conservative and generic assumptions.

This study is not intended to be a risk assessment for any particular transportation campaign, and does not include the probabilities or consequences of malevolent acts. The study does not address the acceptance of the risks associated with transportation of SNF, but can be used to inform such discussions.

The NRC certifies casks used to transport spent nuclear fuel under Title 10 of the *Code of Federal Regulations* Part 71 (10 CFR Part 71). The adequacy of these regulations was confirmed by NUREG-0170, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes* (NRC, 1977), an environmental impact statement for transportation of all types of radioactive material by road, rail, air, and water. Some of the conclusions drawn from this environmental impact statement were:

- The average radiation dose to members of the public from routine transportation of radioactive materials is a fraction of the existing background radiation dose.
- The radiological risk from accidents in transporting radioactive materials is very small compared to the non-radiological risk from accidents involving large trucks or freight trains.
- The regulations in force at the time of the environmental impact statement (1981) were "adequate to protect the public against unreasonable risk from the transport of radioactive materials" (46 FR 21629, April 13, 1981).

The risk assessment of NUREG-0170 was based on very conservative estimates of risk parameters and on models available at the time (which would be considered imprecise today). The NRC concluded that the regulations were adequate because even very conservative estimates of risk parameters did not result in unacceptable risk. NRC also recognized that the agency's

policies on radioactive materials transportation should be "subject to close and continuing review." In implementing this policy, two comprehensive contractor reports dealing with spent fuel transportation have been issued since 1977: the Modal Study (Fischer et al., 1987) and NUREG/CR-6672 (Sprung et al., 2000).[1] The Modal Study was the first intensive examination of vehicle accident statistics, and the first to categorize the frequency of severe accidents by structural and thermal response of a transportation cask. The Modal Study concluded that the frequency of accidents severe enough to produce significant cask damage was considerably less than NUREG-0170 had estimated. Although the Modal Study was not a risk analysis, since it did not consider the radiological consequence of accidents, risks less than those estimated in NUREG-0170 could be inferred.

NUREG/CR-6672 built on the Modal Study by refining the mechanical stress/thermal stress combinations and recasting them as a matrix of accident related impact speeds and fire temperatures. In addition, NUREG/CR-6672 developed expressions for the behavior of spent fuel in accidents and potential release of this material, and analyzed the potential releases. The enhanced modeling capabilities available for NUREG/CR-6672 allowed analyses of the detailed structural and thermal response of transportation casks to accidents. NUREG/CR-6672 also used results of experiments by Lorenz (1980), Sandoval, et al. (1988), and Sanders, et al. (1992) to estimate releases of radioactive material from the fuel rods to the cask interior and from the cask interior to the environment, following very severe accidents. The radionuclides available for release in the accidents studied in NUREG/CR-6672 are from relatively low burnup (30 gigawatt-days per metric ton uranium, or GWD/MTU) and relatively high burnup (60 GWD/MTU) pressurized water reactor (PWR) and boiling water reactor (BWR) fuel, although the transportability of the high burnup fuel was not considered. NUREG/CR-6672 studied the behavior of two generic truck casks and two generic rail casks—each generic cask encompassed design features of several NRC certified casks.

The risks calculated in NUREG/CR-6672 were several orders of magnitude less than the estimates of NUREG-0170, and concluded that no radioactive material would be released in more than 99.99 percent of accidents involving spent fuel shipments. These smaller risk estimates resulted from the use of refined and improved analytical and modeling techniques, exemplified by the finite element analyses of cask structure, and some experimental data which were substituted for the engineering judgments used in NUREG-0170.

In addition to the risk assessments sponsored by NRC mentioned above, there have been many other studies on the subject of spent fuel transportation. Perhaps one of the most independent, objective, authoritative, and recent analyses of spent fuel transportation (co-sponsored by NRC) is the report entitled "Going the Distance? – The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States" by the National Research Council (Committee on Transportation of Radioactive Waste, 2006). This reference is recommended to readers interested in further information on transportation package safety, transportation risk, and particularly for

[1] "Modal Study" and "NUREG/CR-6672" are the names by which these documents are referred to in the general transportation literature. The actual titles are in the bibliography of this document.

its coverage of societal topics that are beyond the scope of the technical risk assessment in the present study. One of the findings of "Going the Distance" was:

The radiological risks associated with the transportation of spent fuel and high level waste are well understood and are generally low, with the possible exception of risks from releases in extreme accidents involving very long duration, fully engulfing fires.

In part because of that finding, the NRC sponsored several studies to investigate the potential consequence from severe historical fire accidents if a spent fuel cask had been involved. Two of these studies investigated tunnel fires (Adkins, et al., 2006, Adkins, et al., 2007) and one investigated the response of a spent fuel cask to an accident below a highway overpass (Bajwa, et al., 2011). While these three studies examined environments that had actually occurred, they made assumptions about the placement of a cask within that environment that would cause the most damage to the cask, without consideration of the probability of that placement. This study also evaluates severe fire accident consequences (but not modeling any particular historical accidents), as well as their associated probabilities, in order to provide a risk perspective.

The present study analyzes the behavior of three currently certified casks carrying Westinghouse 17×17 PWR fuel assemblies with 45 gigawatt-days per metric ton of uranium (GWD/MTU) burnup, the highest burnup that any of the three casks were certified to carry as of 2008 (the time of the analyses—some of the casks have already had changes to their allowed contents). In the future these casks may be certified to carry higher burnup fuel that has been cooled for a longer time, and with a similar source term (a brief discussion on the effect of this change is provided in Section 6.3). For routine transportation, the risks are slightly larger than those estimated in NUREG/CR-6672 because although the actual external dose rates are less than the regulatory maximum used in the other studies, populations along the routes have increased significantly. For accidents, the radiological risks calculated in the current study are at least an order of magnitude less. The reduction in the estimates of risk from those in NUREG-0170 and NUREG/CR-6672 is the result of new data (such as event trees, and accident probabilities) and observations, and improved modeling techniques.

1.3 Risk

Understanding transportation risk is integral to understanding the environmental and related human health impact of radioactive materials transportation. A large amount of data exists for deaths, injuries, and damage from traffic accidents, but there are no data on health effects caused by radioactive materials transportation because no such effects have been observed. Therefore, both regulators and the public rely on estimates of risk to gauge the potential effects of radioactive materials transportation. The risk estimates consider the potential accidents and events, where they could occur, and how severe they might be. Risk estimates include estimating the likelihood and the severity of transportation accidents, as well as the calculation of exposure of workers and members of the public to ionizing radiation from routine transportation.

Risk is usually defined by answering the questions posed by the risk "triplet":

- What can happen (the scenario)?
- How likely is it (the probability)?

- What is the outcome if it happens, i.e., how bad is it (the consequence)?

A risk number (quantitative risk) is calculated by multiplying the probability and consequence for a particular scenario. The probability of a scenario is always less than or equal to one, because the maximum probability of an event is one (100%); an event with 100% probability (probability = 1) of occurrence is an event that is certain to happen. In reality, very few events are certain to happen or certain not to happen (zero probability). The probability of most events is between these two extremes. Transportation accidents involving large trucks, for example, have a very low probability. The probability of a traffic accident for all vehicles is about 0.000002 per mile (or two in 1,000,000 miles) according to the Department of Transportation Bureau of Transportation Statistics (DOT, 2007), and the probability of a particular traffic accident scenario is much smaller still, as shown in the event trees in Appendix V (Figures V-1 and V-2) of this document.

Comment [MF6]: Is this the right number? Previous versions listed "0.0001 per mile (or one in 100,000 miles)" so it was not clear if the right number was one in 10,000 or one in 100,000. However, this represents a significant change from what was listed previously (i.e. either a factor of 5 or 50).

1.3.1 Accident Data

The only data available to estimate the future probability of a scenario are how often that scenario has occurred in the past. The probability of the scenario can be considered the same as its historical frequency. In the case of transportation accidents, enough accidents must have occurred that future accidents per kilometer can be predicted with reasonable accuracy. That is, the sample must be large enough to be sampled randomly. The most applicable frequency would be the frequency of accidents involving vehicles carrying spent nuclear fuel, but there have been too few of these for a statistically valid prediction.[2] The sample size could have been increased by using international data, but regulations and practices in other countries are not consistent with those in the U.S. In any case, there have not been enough accidents worldwide involving spent fuel transportation to provide an adequate statistical data base. Even accidents involving all hazardous materials transportation do not provide a large enough data base from which to generate statistics on a state-by-state basis. The database used in this study is the frequency of highway accidents involving large semi-trailer trucks and the frequency of freight rail accidents (DOT, 2007). Freight rail accident frequency is based on accidents per railcar-mile.

1.3.2 Spent Nuclear Fuel Transportation Scenarios

Transportation risk is categorized in this study by several scenarios, the most probable of which is routine transportation of spent nuclear fuel (SNF) without incidents or accidents between the beginning and end of the trip. Routine transportation is an example of the risk triplet:

- What can happen? The scenario is routine incident-free transportation.
- How likely is it? The probability is 100% (even if the shipment is involved in an accident, it still has an incident-free segment and dose).

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[2] The Bureau of Transportation Statistics lists accidents per year for all classes of hazardous materials. The 2009 database lists 76 class 7 (radioactive materials) rail and highway incidents in the past ten years; http://www.phmsa.dot.gov/staticfiles/PHMSA/DownloadableFiles/Files/tenyr_ram.pdf. These data did not specify the type of radioactive material involved. Not all of these incidents are accidents by DOT definition.

- What if it happens? The consequence is a radiation dose less than one percent of background to individuals near the cask or along the route.

The doses and risks from routine transportation are analyzed in Chapter 2.

The accident scenarios discussed in this study are:

1. Accidents in which the spent fuel cask is not damaged or affected.
   - Minor traffic accidents ("fender benders," flat tires) resulting in minor damage to the vehicle.
   - Accidents in which damage to the vehicle is enough that it cannot move from the scene of the accident under its own power. There is no damage to the spent fuel cask that results in increased radiation in this type of accident.
   - Accidents involving a traffic death and/or injury but no damage to the spent fuel cask that results in increased radiation in this scenario.
2. Accidents in which the spent fuel cask is affected.
   - Accidents involving loss of shielding (either neutron shielding or gamma shielding) but no release of radioactive material.
   - Accidents in which there is a release of radioactive material.

In the first type of accidents, the only potential radiation dose to the public is from exposure of members of the public to external radiation emanating from the cask while the vehicle is stopped. In this study all of these accidents assume that the vehicle is stopped for ten hours. Only the second type of accident involve loss of shielding or release of radioactive material.

Traffic accident statistics (accident frequencies) are used in the analysis to calculate risks. Average traffic accident frequencies since 1996 for large semi-trailer trucks are about 1.3 accidents per million highway kilometers (which is about the same as the accident rate for all highway vehicles), and for freight rail, average frequencies since 1996 are about one accident per ten million railcar kilometers. The overall accident probability is the product of the probability that an accident will happen and the conditional probability that it will be a particular type of accident.

The consequence of an accident scenario could be a dose of ionizing radiation, either from external radiation from a stationary cask or from radioactive material released in an accident. The risk is the product of the overall accident probability and the consequence, and is called "dose risk."

Comment [MF7]: ONRL suggested inclusion of this concise description in the Summary Section, but that was not done.

1.4 Regulation of Radioactive Materials Transportation

Transportation of radioactive materials is regulated by the DOT, as part of hazardous materials transport regulations, primarily under 49 CFR Parts 173 to 178, and by the NRC under 10 CFR Part 71. The regulations of 10 CFR Part 20 are also relevant. NRC transportation regulations apply primarily to the packages being transported, and DOT regulations include labeling, occupational and vehicle standards, registration requirements, and reporting requirements, as well as packaging regulations. In general, the DOT packaging regulations apply to industrial and

Type A packaging, and the NRC regulations apply to Type A(F) fissile materials packaging and Type B packaging. Industrial and Type A non-fissile packages are designed to resist the stresses of routine transportation and are not certified to maintain their integrity in accidents, though many do. Type B packages are used to transport very hazardous quantities of radioactive materials. They are designed to maintain their integrity in severe accidents because the NRC recognizes that any transport package and vehicle may be in traffic accidents. This study addresses the transportation of spent nuclear fuel, and thus concerns itself only with Type B packaging. [For the balance of this report, the term "cask" will be used to refer to the contents plus the packaging.]

Nuclear fuel that has undergone fission in a reactor is both extremely hot and extremely radioactive when it is removed from the reactor. In order to cool the fuel thermally and to allow the highly radioactive and short-lived fission products in the fuel to decay, the fuel is discharged from the reactor into a large pool of water. The fuel usually remains in the pool as long as there is space for it. After the fuel has cooled sufficiently, it can be removed from the pool to dry surface storage at the reactor, or it could be transported to a storage site or other destination. In the U.S., there is currently very little transportation of spent commercial power reactor fuel and there are no plans to transport SNF before it has cooled for five years. The transportation casks used are rated for heat load, and this rating often determines the cooling time needed for the fuel to be transported. The required cooling time may also be driven by shielding or other considerations.

10 CFR Part 71

The NRC recognizes that vehicles carrying radioactive materials are as likely to be in accidents as any vehicles of similar size traveling on similar routes. Transportation packages for very radioactive materials like spent nuclear fuel are therefore designed to maintain their integrity in severe accidents[3]. Packages that can meet this requirement are Type B packages and include the casks considered in this analysis, the NAC-STC (NAC, 2004) and Holtec HI-STAR 100 (Holtec International, 2004) rail casks and the GA-4 (General Atomics, 1998) legal weight truck casks.

Type B packages are designed to pass the sequential series of tests described in 10 CFR 71.73, summarized below.

1. A 9-meter (30-foot) drop onto an essentially unyielding horizontal surface. "Essentially unyielding" in this context means that the target is hard enough and heavy enough that the package absorbs nearly all of the impact energy, and the target absorbs very little energy. This test condition is more severe than most transportation accidents.
2. A 1-meter (40-inch)[4] drop onto a fixed 15-cm (6-inch) diameter steel cylinder, to test the package's resistance to punctures.
3. A 800°C (1475°F) fire that fully engulfs the package for 30 minutes.

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[3] Although release of a specific quantity of each radionuclide is allowed by regulation, Type B casks are typically designed to remain leak-tight.

[4] When discussing the regulations, the conversion between SI units and English units are those in the regulations. In other locations within this report actual conversion factors are used.

4. Immersion under 0.9 meters (3 feet) of water. Casks carrying spent fuel are also required to withstand a non-sequential immersion in 200 meters (670 feet) of water for one hour.

Figure 1-1 illustrates this sequence of tests.

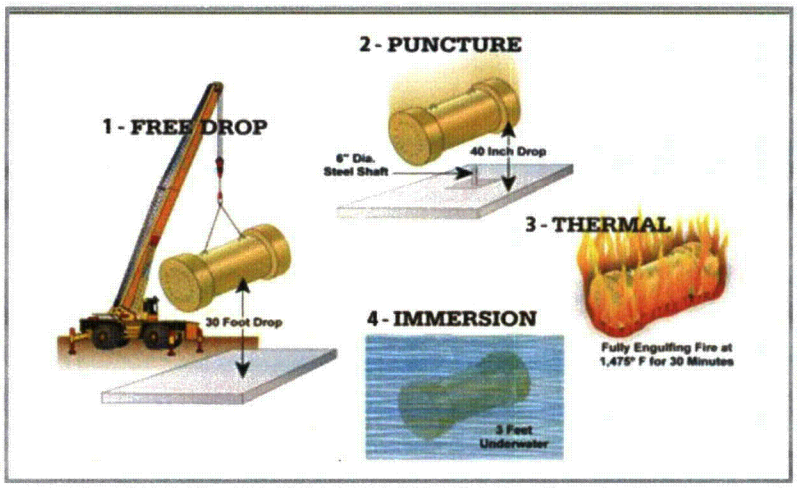


Figure 1-1. The four tests for Type B packages.

The package tests in 10 CFR Part 71 Section 73 were developed to envelope real life accidents. These tests are not intended to represent any specific transportation route, any specific historical transportation accident, or a "worst-case" accident. These tests are intended to simulate the damaging effects of a severe transportation accident in a manner that provides international acceptability, uniformity, and repeatability. These tests are the same as those used by all International Atomic Energy Agency (IAEA) member states.

The tests are performed on a package design (either physically using a full-scale prototype or sub-scale test unit, or via computational modeling), but not on every package that will be used to transport spent nuclear fuel. A package designer may create computer models to evaluate the performance of a package design and/or components of the package design, may build full-size or scale model packages for physical testing, or may incorporate references to previous satisfactory demonstrations of a similar nature. In practice, the safety analysis performed for Type B packages often incorporates a combination of physical testing, computer modeling, and engineering evaluation. The packaging Safety Analysis Report (SAR) contains the information about the package design's performance in the tests and an evaluation against the acceptance criteria in 10 CFR Part 71. The SAR is used to apply for package certification. During the certification, the NRC reviews the SAR to ensure that the package design meets all criteria specified in 10 CFR Part 71.

NRC regulations specify that release of material from the package can be no more than that allowed to be shipped in a non-accident resistant Type A package. The regulation also specifies a

maximum post-test external radiation dose rate of 0.01 Sv per hour (1 rem/hr) at one meter (40 inches) from the package surface.

10 CFR Part 20

This section of the Code of Federal Regulations prescribes the largest allowable radiation dose that a member of the public may receive from NRC-licensed facilities, exclusive of background radiation, diagnostic or therapeutic radiation, or material that has been discharged to the environment in accordance with NRC regulations. This section of the code does not apply to transport, but gives doses that can be compared to those calculated in this study. These doses are:

- 1 mSv per year (100 mrem per year) total effective dose equivalent (TEDE), including both external and committed internal dose.
- 0.02 mSv per hour (2 mrem per hour) in any unrestricted area from external sources. As shown in Table 2-12, for example, the doses from routine, incident-free transportation are considerably below these limits.
- 5 mSv per year (500 mrem per year) from a licensed facility if the licensee can show the need and expected duration of doses larger than 1 mSv per year.

Although the regulations state clearly that these dose limits do not include background, background can provide a useful comparison to other sources of radiation exposure, since it affects everyone. The average background radiation dose in the United States is 0.0036 Sv per year. This Part also regulates occupational doses to 0.05 Sv per year (5 rem per year) TEDE.

1.5 Selection of Casks

Past risk assessments for the transportation of spent fuel have used generic cask designs with features similar to real casks, but generally without all of the conservatisms that are part of real cask designs, such as assumptions on material strengths and energy absorbing capabilities of impact limiters. In this effort, the risk assessment was performed using actual cask designs with all of the features that contribute to their robustness. Because it is too costly and time consuming to examine all casks, a subset of casks was chosen for the risk assessment. Appendix I lists the various spent fuel casks that were certified by the NRC at the time the study began, gives options for the method of choosing the casks to be used, gives some of the important features of the various cask designs, and finally concludes with the casks chosen.

Table 1-1 lists the casks that were certified by the NRC as of 2006 (the date when the cask selections for this study were made) for the transportation of irradiated commercial light water power reactor fuel assemblies. Those above the heavy line are older designs that were no longer used, but still had valid certificates. Those below the heavy line were more modern and additional units of these designs could be built. The casks chosen for this study came from the later group. A brief description of each of these casks is included in Appendix I.

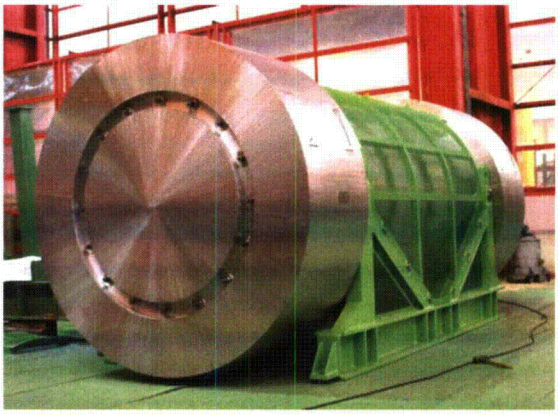
Table 1-1. NRC-certified commercial light water power reactor spent fuel casks

Cask	Package ID	Canister	Contents (Number of assemblies)	Type
IF-300	USA/9001/B( )F	No	7 PWR, 17 BWR	Rail
NLI-1/2	USA/9010/B( )F	No	1 PWR, 2 BWR	Truck
TN-8	USA/9015/B( )F	No	3 PWR	Overweight[a]
TN-9	USA/9016/B( )F	No	7 BWR	Overweight[a]
NLI-10/24	USA/9023/B( )F	No	10 PWR, 24 BWR	Rail
NAC-LWT	USA/9225/B(U)F-96	No	1 PWR, 2 BWR	Truck
GA-4	**USA/9226/B(U)F-85**	**No**	**4 PWR**	**Truck**
NAC-STC	**USA/9235/B(U)F-85**	**Both**	**26 PWR**	**Rail**
NUHOMS®-MP187	USA/9255/B(U)F-85	Yes	24 PWR	Rail
HI-STAR 100	**USA/9261/B(U)F-85**	**Yes**	**24 PWR, 68 BWR**	**Rail**
NAC-UMS	USA/9270/B(U)F-85	Yes	24 PWR, 56 BWR	Rail
TS125	USA/9276/B(U)F-85	Yes	21 PWR, 64 BWR	Rail
TN-68	USA/9293/B(U)F-85	No	68 BWR	Rail
NUHOMS®-MP197	USA/9302/B(U)F-85	Yes	61 BWR	Rail

[a]Overweight truck
The casks in bold type are the ones that were selected for this study.

The casks chosen for detailed analysis are the NAC-STC (Figure 1-2) and the HI-STAR 100 (Figure 1-3) rail casks. The GA-4 truck cask (Figure 1-4) was used to evaluate truck shipments, but detailed impact analyses of this cask were not performed because prior analyses of both truck and rail casks have shown that truck casks have significantly lower probability of release of radioactive material in impact accidents (Sprung, et al., 2000). The impact analyses from Sprung et al. were used to assess the response of the GA-4 cask. The complete Certificates of Compliance (COC) for each of these casks (as of April 12, 2010) is included in Appendix I. The NAC-STC cask was chosen because it is certified for transport of spent fuel either with or without an internal welded canister and, for transport or spent fuel without an internal canister, its certificate of compliance allows use of either elastomeric o-rings or metallic o-rings. Even though there were five casks in the group that use lead as their gamma shielding, only the NAC-STC cask of this group can transport fuel that is not contained within an inner welded canister. This ensured, as noted in the analyses of Chapters 3 to 5, that the maximum potential for radioactive material released into the environment was considered. The HI-STAR 100 rail cask was chosen because it was the only all-steel cask in the group that was certified for transport of fuel in an inner welded canister. The GA-4 truck cask was chosen because it has a larger capacity than the NAC-LWT, and therefore was more likely to be used in a large spent fuel transportation campaign. The chosen casks included all three of the most common shielding options: lead, depleted uranium (DU), and steel. Table 1-2 summarizes the chosen casks.

The choice of rail casks allowed comparison between directly loaded and canistered fuel, comparison between a Steel-Lead-Steel cask and an All-Steel cask, and comparison between elastomeric o-ring seals and metallic o-ring seals.



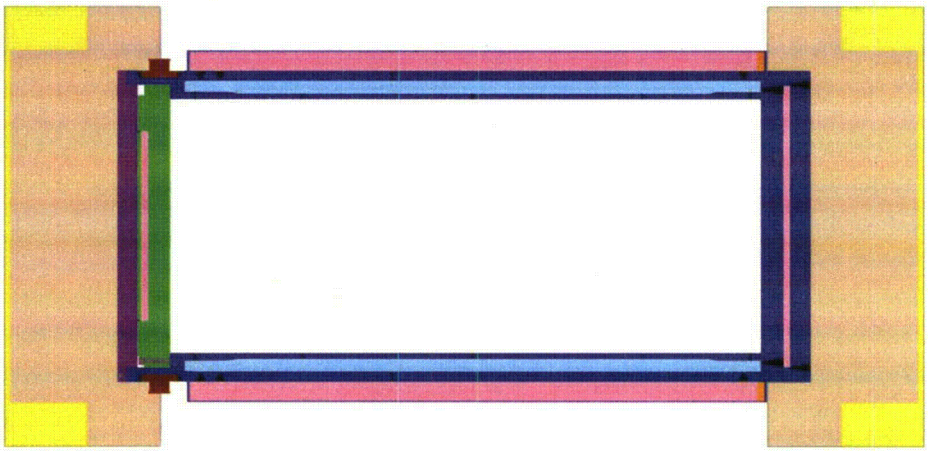


Figure 1-2. Photograph and cross-section of the NAC-STC cask (courtesy of NAC International).

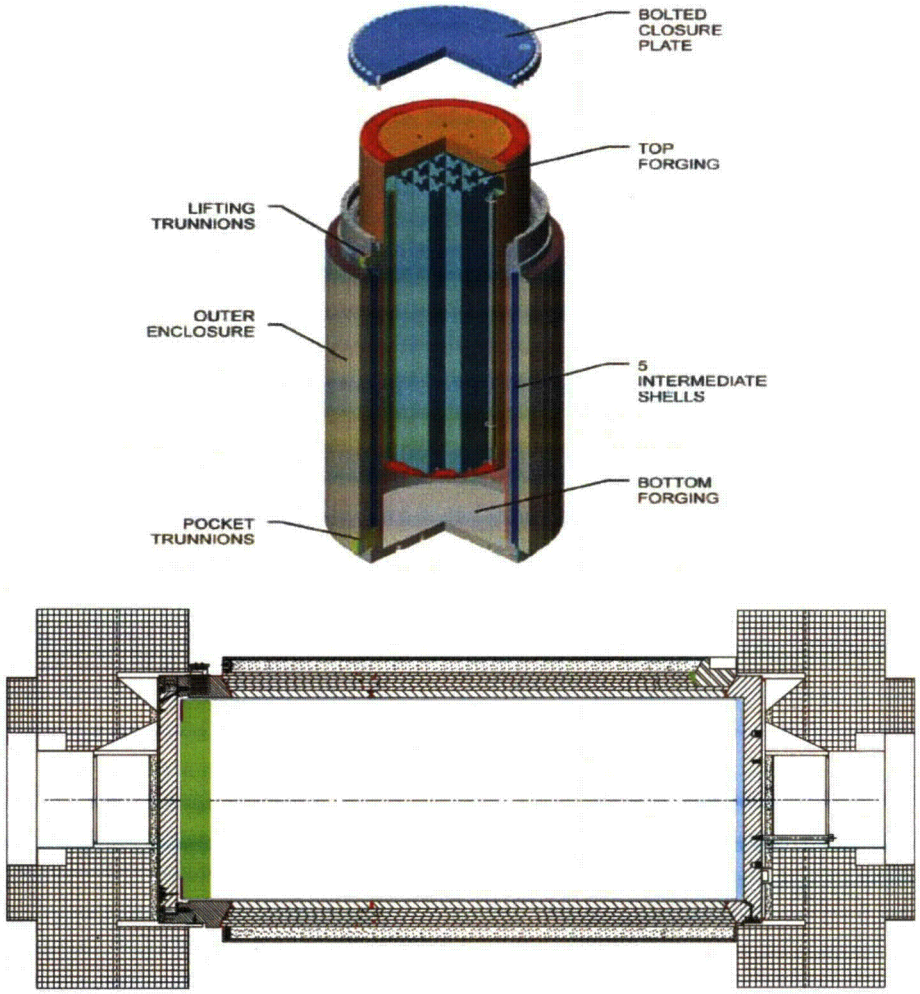


Figure 1-3. Basic layout and cross-section of the HI-STAR 100 rail transport cask (from Haire and Swaney, 2005, and Holtec International, 2004).

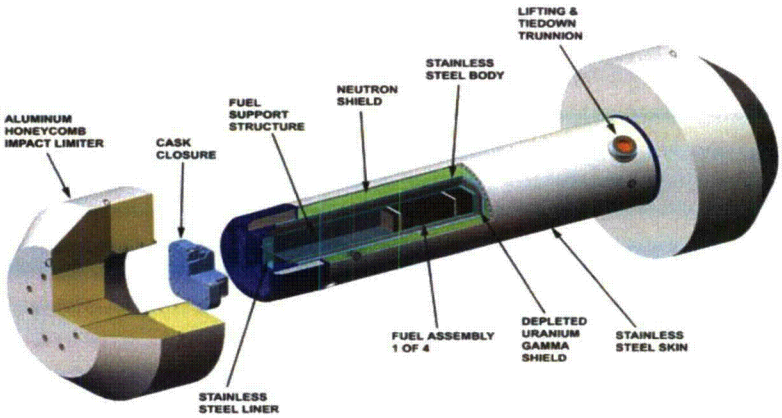


Figure 1-4. GA-4 cask (courtesy of General Atomics).

The detailed analyses of this report use the geometry and properties of these specific casks, but other similar casks are likely to respond in the same manner as these casks. Therefore, in the rest of this report the HI-STAR 100 rail cask will be referred to as Rail-Steel, the NAC-STC rail cask will be referred to as Rail-Lead, and the GA-4 truck cask will be referred to as Truck-DU.

Table 1-2. Casks chosen and reasons for selection

Cask Chosen	Type of Cask	Reason for Consideration in this Study
HI-STAR 100 Rail Cask[4]	Rail-Steel Cask	This was the only all-steel cask in the group that was certified for transport of fuel in an inner welded canister
NAC-STC Rail Cask[5]	Rail-Lead Cask	Only the NAC-STC cask of this group can transport fuel that is not contained within an inner welded canister, thus ensuring the maximum potential for radioactive material released into the environment was considered.
GA-4 Truck Cask	Truck-DU	The GA-4 truck cask was chosen because its large capacity made it more likely to be used in any large transportation campaign.

[5] The choice of rail casks allowed comparison between directly loaded and canistered fuel, comparison between a Steel-Lead-Steel cask and an All-Steel cask, and comparison between elastomeric o-ring seals and metallic o-ring seals.

CHAPTER 2

RISK ANALYSIS OF ROUTINE TRANSPORTATION

2.1 Introduction

NUREG-0170 (NRC, 1977) was the first comprehensive assessment of the environmental and health impact of transporting radioactive materials. It documented estimates of the radiological consequences and risks associated with the shipment by truck, train, plane, or barge of about 25 different radioactive materials, including power reactor spent fuel. However, little actual data on spent nuclear fuel transportation was available in 1977, and computational modeling of such transportation was, relatively speaking, in its infancy.

The RADTRAN computer code (Taylor and Daniel, 1977) is the computational tool used in this chapter to estimate risks from routine[6] transportation of spent nuclear fuel. RADTRAN was initially developed by SNL for NRC for the NUREG-0170 risk assessment. During the past several decades, the calculation method and RADTRAN code have been improved to stay current with computer technology, and supporting input data have been collected and organized. The basic RADTRAN analysis approach has not changed since the original development of the code, and the risk assessment method employed in the RADTRAN code is accepted worldwide; about 25 percent of the five hundred RADTRAN users are international.[7]

RADTRAN 6.0, integrated with the input file generator RADCAT, (Neuhauser et al., 2000;[8] Weiner et al., 2009) is the version used in this study. The incident-free module of RADTRAN, the model used for the analysis in this chapter, was validated by measurement (Steinman et al., 2002), and verification and validation of RADTRAN 6.0 are documented in Dennis, et al., 2008.

This chapter discusses the risks to the public and workers when the transportation of the casks containing spent fuel takes place without incident, and the transported casks are undamaged. Non-radiological vehicular accident risk, which is orders of magnitude larger than the radiological transportation risk, is not analyzed in this study. The risks and consequences of accidents and incidents interfering with routine transportation are discussed in Chapter 5.

This chapter includes the following:

- A brief discussion of ionizing radiation emitted during transportation.
- A description of the RADTRAN model of routine transportation.
- Radiation doses from a single routine shipment to:
  - Members of the public who live along the transportation route and near stops

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[6] The term "routine transportation" is used throughout this document to mean incident or accident-free transportation

[7] The currently registered RADTRAN users are listed on a restricted-access web site at Sandia National Laboratories.

[8] Neuhauser, et al. (2000) is the technical manual for RADTRAN 5, and is cited because the basic equations for the incident-free analyses in RADTRAN 6 are the same as those in RADTRAN 5. The technical manual for RADTRAN 6 is not yet available.

- Occupants of vehicles that share the route with the radioactive shipment
- Various groups of people at stops
- Workers

Detailed results of the RADTRAN calculations for this analysis are provided in Appendix II. All references are listed in the bibliography. A discussion of RADTRAN use and applications are provided in Weiner, et al (2009).

2.2 Radiation Emitted during Routine Transportation

The RADTRAN model for calculating radiation doses is based on the well-understood behavior of ionizing radiation. It can be absorbed by various materials, including air. Absorption of ionizing radiation depends on the energy and type of radiation and on the absorbing material.

Spent nuclear fuel, the subject of this analysis, is very radioactive, emitting ionizing radiation in the form of alpha, beta, gamma, and neutron radiation. The casks that are used to transport spent nuclear fuel have thick walls that absorb most of the emitted ionizing radiation and thereby shield the public and the workers. Figure 2-1 shows two generic cask diagrams on which the shielding is identified. This generic cask does not show the cross section of any of the three actual casks used in this study.

Alpha and beta radiation cannot penetrate the walls of the casks (both are actually absorbed well by a few millimeters of paper and plastic). The steel and lead layers of the cask wall absorb most of the gamma and neutron radiation emitted by spent fuel, although adequate neutron shielding also requires a layer of a neutron absorber like a polymer or boron compound. In certifying spent fuel casks, the NRC allows very low external dose rates due to gamma and neutron radiation. For spent uranium-based fuel, the external dose rate is typically dominated by the gamma radiation.

Absorbed radiation dose is measured in sieverts (Sv) in the Standard International system, rem or millirem in the historic English unit system (millirem is abbreviated as mrem in this document). Average U. S. background radiation from naturally occurring and some medical sources is 0.0036 Sv (360 mrem) per year (Shleien et al., 1998, Figure 1.1). Recent increased diagnostic use of ionizing radiation, as in computerized tomography, has suggested increasing the average background to 0.0062 Sv (620 mrem), and this background value is cited on the NRC website. The present study, however, uses the older value of 0.0036 Sv per year. A single dental x-ray delivers a dose of 4×10^{-5} Sv (4 mrem), and a single mammogram delivers 1.3×10^{-4} Sv (13 mrem) (Stabin, 2009). The maximum radiation dose rate from a spent fuel cask allowed by regulation is 10^{-4} Sv per hour (10 mrem/hour), measured at two meters (about six feet) from the outside of the cask (10 CFR Part 71)), or about 0.00014 Sv/hour (14 mrem per hour) at one meter from a cask four to five meters long.

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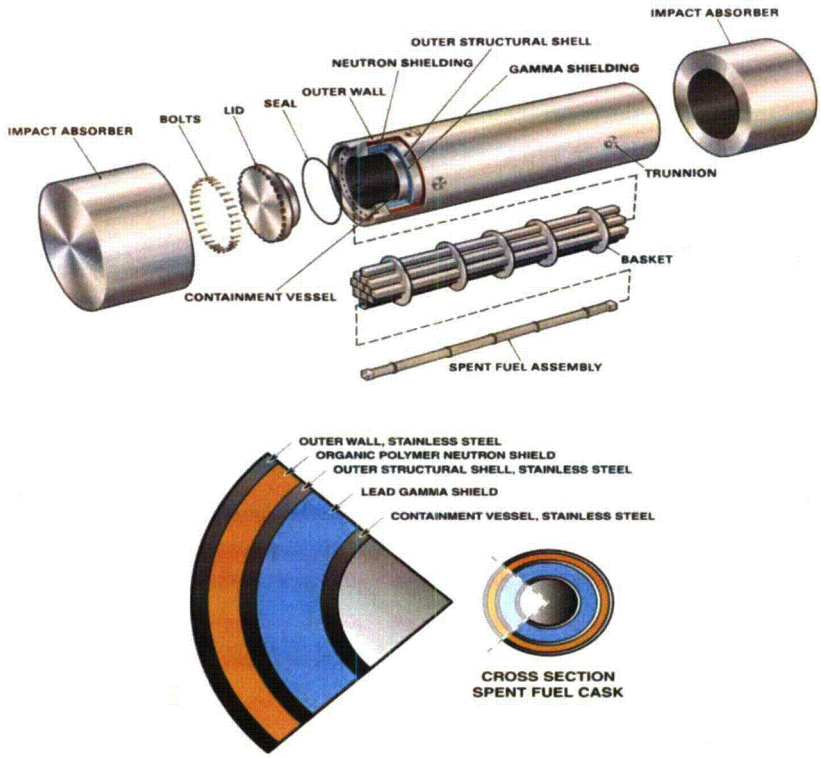

Figure 2-1. The upper figure is an exploded view of a generic spent fuel cask. The lower figure is a cross-section of the layers of the cask wall (Sandia National Laboratories archive).

The external radiation doses from the casks in this study (Figures 1-2 to 1-4), determined from values reported in the cask Safety Analysis Reports, are shown in Table 2-1 (Holtec, 2004, NAC, 2004, General Atomics, 1998).

Table 2-1. External radiation doses from the casks in this study

	Truck-DU	Rail-Lead	Rail-Steel
Transportation mode	Highway	Rail	Rail
Dose rate Sv/hr (mrem/hr) at 1 m	0.00014 (14)	0.00014(14)	0.000103 (10.3)
Gamma fraction	0.77	0.89	0.90
Neutron fraction	0.23	0.11	0.10

The calculated radiation dose to workers and members of the public from a routine shipment is based on the external dose rate at one meter from the spent fuel cask, as shown in Figure 2-2. This dose rate, when expressed in mrem per hour (or mSv per hour times 100), is numerically equal to the transport index, or TI. Doses from the external radiation from the cask depend on the external dose rate, the distance of the receptor from the cask, the exposure time, and intervening shielding.

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2.3. The RADTRAN Model of Routine, Incident-Free Transportation

2.3.1 The Basic RADTRAN Model

For analysis of routine transportation, RADTRAN models the cask as a sphere with a radiation source at its center, and assumes that the dimensions of the trailer or railcar carrying the cask are the same as the cask dimensions. The emission rate of the radiation source is based on the TI instead of on a shielding calculation. The radiation source is modeled as a virtual source at the center of the sphere shown in Figure 2-2 that produces the same TI as the cask. The diameter of this spherical model, called the "critical dimension," is the longest dimension of the actual spent fuel cask.

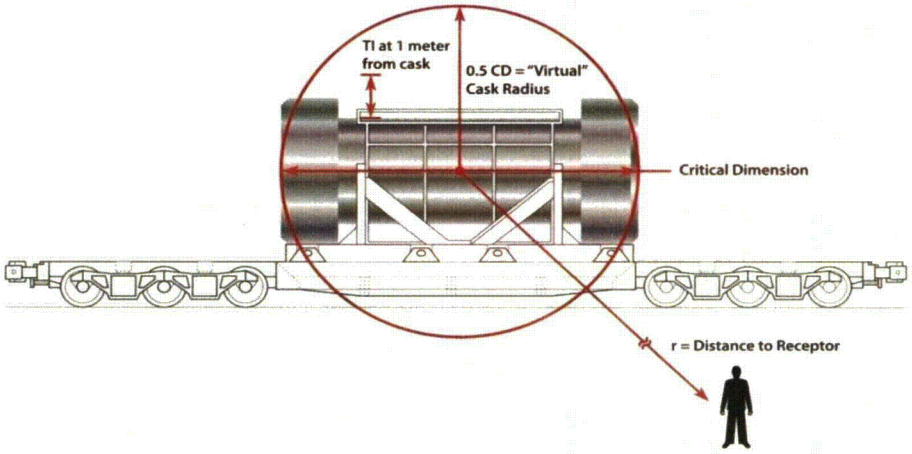

Figure 2-2. RADTRAN model of the vehicle in routine, incident-free transportation. The cask in this diagram is positioned horizontally, and the critical dimension is the cask length. (TI = transport index, CD = critical dimension, r = radius)

When the distance to the receptor (*r* in Figure 2-2) is much larger than the critical dimension, RADTRAN models the dose to the receptor as proportional to $1/r^2$. When the distance to the receptor *r* is similar to or less than the critical dimension, as for crew or first responders, RADTRAN models the dose to the receptor as proportional to $1/r$. The dose calculated by the RADTRAN spherical model overestimates the measured dose by a few percent (Steinman et al., 2002).

2.3.2 Individual and Collective Doses

The dose to workers and the public from a cask during routine transportation depends on the time that the workers or public are exposed to the cask, the distance from the cask, the external radiation from the cask, and intervening shielding. When the vehicle carrying the cask is traveling along the route, the faster the vehicle goes, the less exposure to anyone along the vehicle's route. Therefore, an individual member of the public residing near the transport route receives the largest dose from a moving vehicle when he or she is as close as possible to the vehicle, and the vehicle is traveling as slowly as possible. For trucks and trains carrying spent fuel, a speed of 24 km per hour (15 mph) and distance of 30 meters (approximately 100 feet) are assumed for maximum exposure.[9] Table 2-2 shows the maximum dose to an individual member of the public under these conditions. The Rail-Lead cask has a higher dose than the Rail-Steel cask because it has a higher TI. The Truck-DU cask has a higher dose than the Rail-Lead cask (same TI) because it has a longer critical dimension, and therefore takes more time to go past a receptor. The transit speed used for both rail and truck transport in the calculation of the maximum individual dose is 24 kph (15 mph). These doses are about the same as one minute of average background: 6.9×10^{-9} Sv (6.9×10^{-4} mrem).

Comment [MF8]: Pervious comment, "This is for those not traveling the route but are stationary on the route. For those traveling on the route with the transport, matching speeds provides the maximum potential dose." does not appear to have been addressed.

Table 2-2. Maximum individual in-transit doses

Cask (mode)	Dose (Sv)
Rail-Lead (rail)	5.7E-09
Rail-Steel (rail)	4.3E-09
Truck-DU (truck)	6.7E-09

When a vehicle carrying a spent fuel cask travels along a route, the people who live along that route and the people in vehicles that share the route are exposed to the external radiation from the cask. Doses to groups of people are collective doses; the units of collective dose are person-Sv (person-rem). A collective dose, sometimes called a population dose, is essentially an average individual dose multiplied by the number of people exposed.[10] RADTRAN calculates collective doses along transportation routes by integrating over the width of a band along the route where the population resides (the *r* in Figure 2-2) and then integrating along the route. Collective doses to people on both sides of the route are included. The exposed population is in a band 770 meters (approximately 0.5 miles) on either side of the route: from 30 meters (100 feet) from the center of the route to 800 meters (0.5 miles). Figure 2-3 shows how these bands are defined with examples of distances within the bands.

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[9] Thirty meters is typically as close as a person on the side of the road can get to a vehicle traveling on an interstate highway.

[10] A detailed discussion of collective dose is in Appendix II.

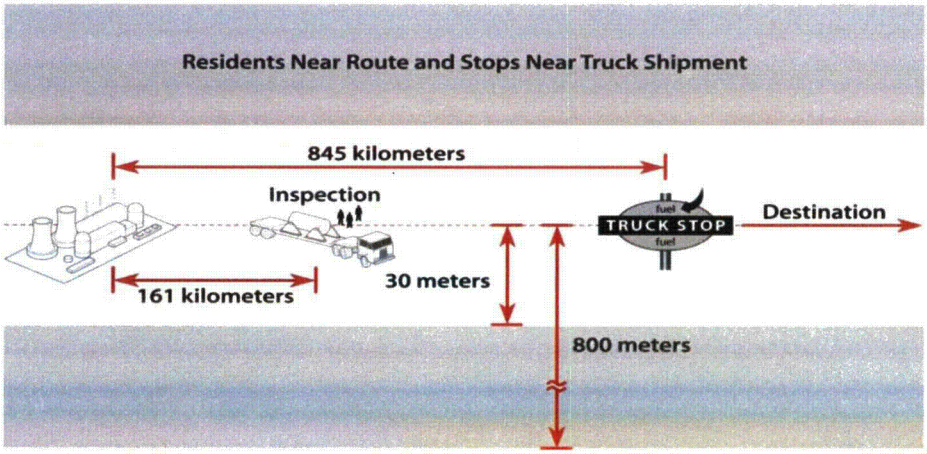

Figure 2-3. Diagram of a truck route as modeled in RADTRAN (not to scale).

Occupants of vehicles that share the route with the radioactive shipment also receive a radiation dose from the spent fuel cask. The collective dose to occupants depends on the average number of occupants per vehicle and the number of vehicles per hour that pass the radioactive shipment in both directions.

Any route can be divided into as many sections as desired for dose calculation, e.g., the dose to residents of a single house or city block. However, as a practical matter, routes are divided into rural, suburban, and urban segments according to the population per square mile (population density).
Table 2-3 summarizes the characteristics of each population type that is part of the dose calculation by RADTRAN. References for these parameter values are in the Table 2-3 footnotes.

Table 2-3. Characteristics of rural, suburban, and urban routes used in RADTRAN. Highway routes are interstate or other limited-access highways.

	Basis	Highway			Rail		
		Rural	Suburban	Urban	Rural	Suburban	Urban
Population density per km^2 (per mi^2)[a]	TRAGIS	0 to 54 (0 to 139)	54 to 1284 (139 to 3326)	>1284 (>3326)	0 to 54 (0 to 139)	54 to 1284 (139 to 3326)	>1284 (>3326)
Nonresident / resident ratio[b]	Urban Areas	NA	NA	6	NA	NA	6
Shielding by buildings[b]	Historic RADTRAN use	0 (outside)	13% (wood)	98.2% (concrete, brick)	0 (outside)	13% (wood)	98.2% (concrete, brick)
U.S. average vehicle speed[c] kph (mph)[c,d]	DOT	108 (67)	108 (67)	102(63)	40 (25)	40 (25)	24 (15)
U.S. average vehicles per hour[b,e]	DOT	1119	2464	5384	17	17	17
Occupants of other vehicles[b,f]	DOT	1.5	1.5	1.5	1	1	5

[a]Johnson and Michelhaugh, 2003, [b]Weiner, et al., 2009, [c]DOT, 2004a, [d]DOT, 2004b, Appendix D, [e,f]DOT, 2009; these are average railcars per hour, [f]DOT, 2008, Table 1-11.

Comment [MF9]: Did not use conversion provided by previous review. May have had a good reason for not doing so since other suggestions to the table were considered.

Each route clearly has a distribution of rural, urban, and suburban areas, as shown by the example of the truck route in Figure 2-4, which shows a segment of Interstate 80 through Salt Lake City, Utah. The broad stripe is the half-mile band on either side of the highway. The red areas are urban populations, the yellow areas are suburban, and the green areas are rural. Instead of analyzing each separate, rural, urban, and suburban segment of this stretch of highway, the rural, suburban, and urban areas are each combined for RADTRAN dose calculations. The routing code WebTRAGIS (Johnson and Michelhaugh, 2003) provides these combinations for each state traversed by a particular route.

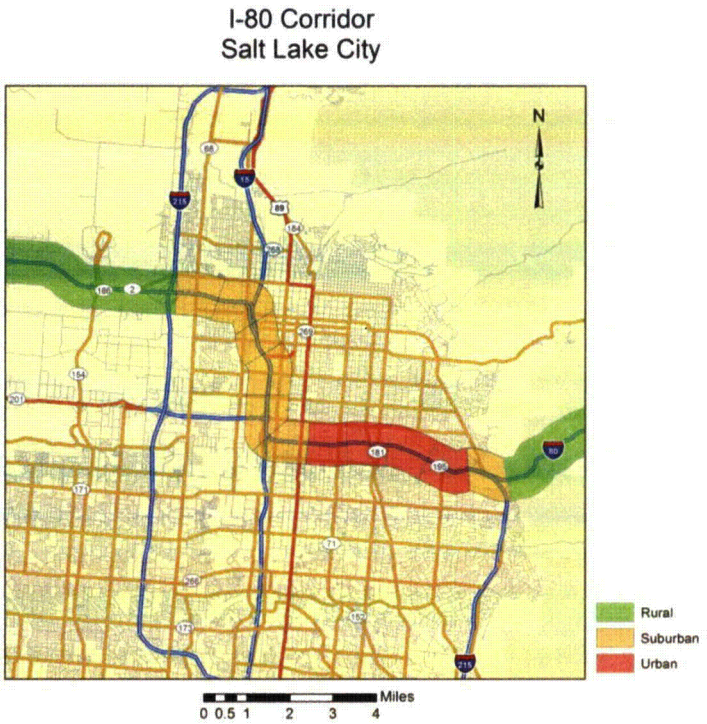

Figure 2-4. A segment of I-80 through Salt Lake City, Utah.

Table 2-4 shows this WebTRAGIS output for a rail route from Kewaunee Nuclear Plant (NP), WI to Skull Valley.

Comment [MF10]: Rail or road – not clear from text – see comment below.

Table 2-4. Rail route segment lengths and population densities, Kewaunee NP to Skull Valley. The route segment of Figure 2-4 is in bold.

Comment [MF11]: This states it is for a rail route, but references Figure 2-4 above that is a road map?

Comment [MF12]: Note: changed destination, so previous ORNL comments don't apply.

State	Kilometers (miles)			Persons/km^2 (persons/mi^2)[a]		
	Rural	Suburban	Urban	Rural	Suburban	Urban
Illinois	155 (96.4)	45 (28)	1.2 (0.7)	15.4 (40)	267 (691)	2049 (5301)
Iowa	394 (245)	95 (59.1)	5.1(3.2)	15.7 (41)	268 (693)	2185 (5653)
Nebraska	652 (405)	76 (47.2)	7 (4.4)	10 (26)	269 (696)	2401 (6212)
Utah	197 (123)	**38 (23.6)**	**15 (9.3)**	7.5 (19.4)	**407 (1053)**	**2412 (6240)**
Wisconsin	191 (119)	85 (52.8)	19.9 (12.4)	21.4 (55)	337 (872)	2660 (6882)
Wyoming	607 (377)	34 (21.1)	3.4 (2.1)	4.9 (13)	399 (1032)	1967 (5089)

[a]The populations density is a WebTRAGIS output, calculated by averaging the population density along the rural, suburban, or urban route length within each state.

The maps of Figure 2-5 through Figure 2-8 show the sixteen truck and sixteen rail routes analyzed in this report. These routes were selected as being representative of possible cross-country transport. No actual spent fuel transport has occurred from any of these plants to any of these destinations. The maps are adapted from the output of the routing code WebTRAGIS (Johnson and Michelhaugh, 2003).

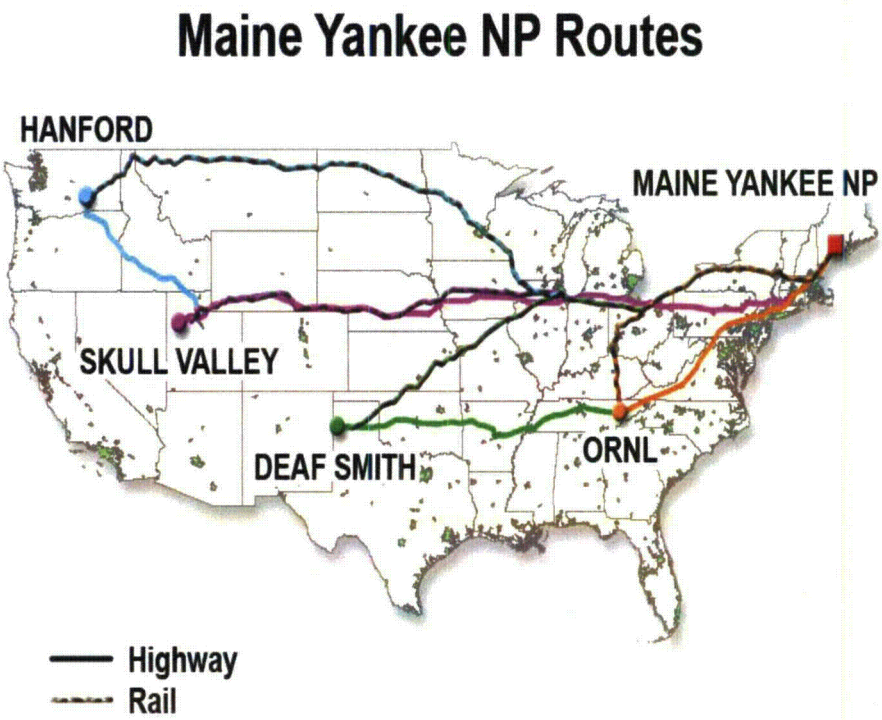

Figure 2-5. Highway and rail routes from Maine Yankee Nuclear Plant site.
(NP stands for Nuclear Plant and ORNL stands for Oak Ridge National Laboratory.)

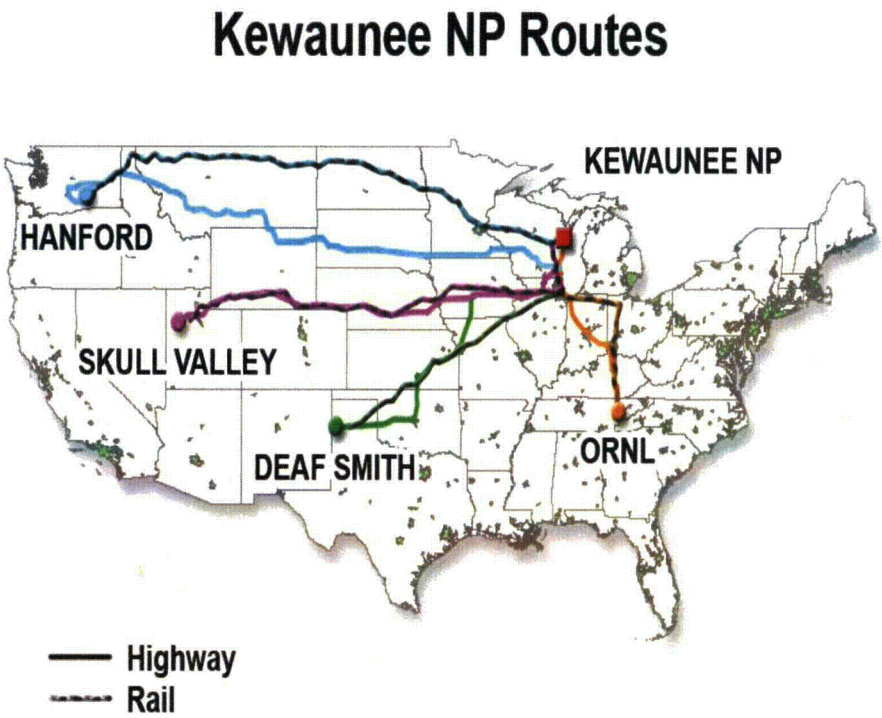

Figure 2-6. Highway and rail routes from Kewaunee Nuclear Plant.
(NP stands for Nuclear Plant and ORNL stands for Oak Ridge National Laboratory.)

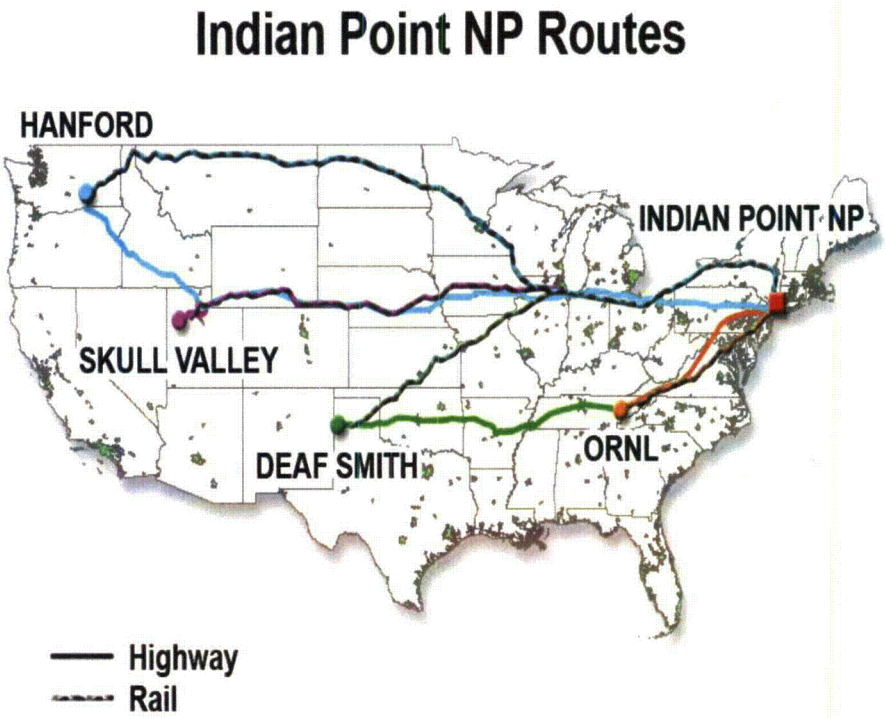

Figure 2-7. Highway and rail routes from Indian Point Nuclear Plant.
(NP stands for Nuclear Plant and ORNL stands for Oak Ridge National Laboratory.)

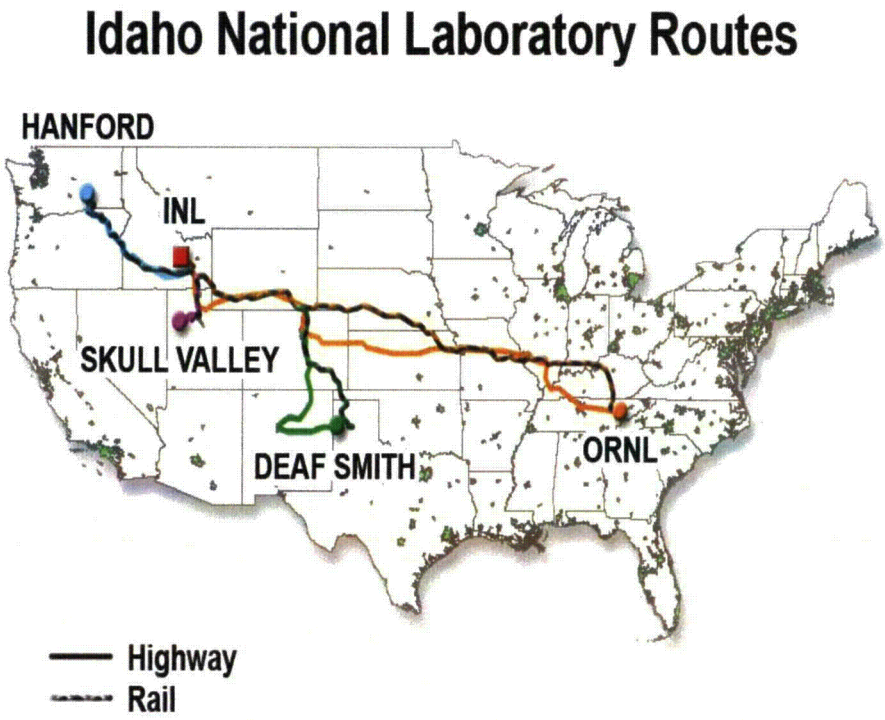

Figure 2-8. Highway and rail routes from Idaho National Laboratory.

(INL stands for Idaho National Laboratory and ORNL stands for Oak Ridge National Laboratory.)

The route segment lengths and population densities are entered into RADTRAN, which then calculates the collective doses to residents along these route segments. Collective doses, which depend on route length and on the populations along the route, were calculated for one shipment over each of 16 truck routes and 16 rail routes. Collective doses are reported as person-Sv.

The sites where the shipments originated include two nuclear generating plants (Indian Point and Kewaunee), a storage site at a fully decommissioned nuclear plant (Maine Yankee), and INL. The routes modeled are shown in Table 2-5. Both truck and rail versions of each route are analyzed.

Table 2-5. Specific routes modeled (urban kilometers are included in total kilometers)

Origin	Destination	Population within 800 m (1/2 mile)		Total Kilometers		Urban Kilometers	
		Rail	Truck	Rail	Truck	Rail	Truck
Maine Yankee Site, ME	Hanford, WA	1,647,190	1,129,685	5084	5013	355	116
	Deaf Smith County, TX	1,321,024	1,427,973	3362	3596	211	165
	Skull Valley, UT	1,451,325	1,068,032	4068	4174	207	115
	Oak Ridge, TN	1,146,478	1,137,834	2125	1748	161	135
Kewaunee NP, WI	Hanford, WA	476,914	423,163	3028	3453	60	52
	Deaf Smith County, TX	677,072	494,920	1882	2146	110	60
	Skull Valley, UT	806,115	505,226	2755	2620	126	58
	Oak Ridge, TN	779,613	646,034	1395	1273	126	92
Indian Point NP, NY	Hanford, WA	961,026	869,763	4781	4515	229	97
	Deaf Smith County, TX	1,027,974	968,282	3088	3074	204	109
	Skull Valley, UT	1,517,758	808,107	3977	3672	229	97
	Oak Ridge, TN	1,146,245	561,723	1264	1254	207	60
Idaho National Lab, ID	Hanford, WA	164,399	132,662	1062	959	20	15
	Deaf Smith County, TX	298,590	384,912	1913	2291	40	52
	Skull Valley, UT	169,707	132,939	455	466	26	19
	Oak Ridge, TN	593,680	569,240	3306	3287	75	63

These routes represent a variety of route lengths and populations. The routes include eastern U.S., western U.S., and cross country routes, are of varying lengths, and include a variety of urban areas. Two of the three nuclear plants chosen as origin sites: Kewaunee, WI, and Maine Yankee, ME, and two of the destination sites, Hanford, WA, and Skull Valley, TX, are origins and destinations used in NUREG/CR-6672 (Sprung et al., 2000). Indian Point Nuclear Plant, NY, involves a somewhat different set of cross-country and east coast routes than Maine Yankee, and is an operating nuclear plant, while Maine Yankee has been decommissioned and is now only a surface storage facility. Because this study could be utilized for both commercial nuclear power plant spent fuel shipments and Department of Energy spent fuel shipments, INL was included as an origin site. The destination sites include two proposed repository sites (Deaf Smith County, TX, and Hanford, WA) (DOE, 1986), the site of the proposed Private Fuel Storage facility (Skull Valley, UT), and ORNL. These routes were not intended to provide a "worst case" result, but were chosen to provide representative results over a broad range of conditions and large segments of the country.

Route segments and population densities are provided by WebTRAGIS. Population densities were updated from the 2000 census using the 2008 Statistical Abstract (U.S. Bureau of the Census 2008, Tables 13 and 21), though updates were made only when the difference between the 2006 and 2000 population densities was one percent or more. The collective doses reported in Table 2-6 and Table 2-7 are in units of person-Sv. Table 2-6 and Table 2-7 present collective doses for rail and truck, respectively, for the sixteen routes. State-by-state collective doses are tabulated in Appendix II.

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Table 2-6. Collective doses to residents near the route (person-Sv) per shipment for rail transportation (1 Sv = 10^5 mrem)

FROM/TO	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
MAINE YANKEE								
ORNL	1.5E-05	1.8E-04	9.0E-06	2.1E-04	1.2E-05	1.4E-04	6.8E-06	1.6E-04
DEAF SMITH	1.9E-05	2.2E-04	1.1E-05	2.5E-04	1.4E-05	1.7E-04	8.7E-06	1.9E-04
HANFORD	2.4E-05	2.6E-04	1.3E-05	2.9E-04	1.8E-05	2.0E-04	9.9E-06	2.3E-04
SKULL VALLEY	2.6E-05	2.7E-04	1.0E-05	2.9E-04	2.0E-05	2.0E-04	7.6E-06	2.2E-04
KEWAUNEE								
ORNL	1.0E-05	1.1E-04	6.7E-06	1.3E-04	7.9E-06	8.3E-05	5.1E-06	9.6E-05
DEAF SMITH	8.2E-06	9.5E-05	5.8E-06	1.1E-04	6.3E-06	7.2E-05	4.4E-06	8.3E-05
HANFORD	1.2E-05	9.3E-05	3.0E-06	1.1E-04	9.3E-06	7.1E-05	2.3E-06	8.3E-05
SKULL VALLEY	1.4E-05	1.2E-04	6.6E-06	1.4E-04	1.1E-05	9.0E-05	5.0E-06	1.1E-04
INDIAN POINT								
ORNL	7.5E-06	1.4E-04	1.4E-05	1.6E-04	5.7E-06	1.1E-04	1.1E-05	1.2E-04
DEAF SMITH	1.7E-05	1.8E-04	1.2E-05	2.0E-04	1.3E-05	1.3E-04	8.9E-06	1.5E-04
HANFORD	2.2E-05	2.1E-04	1.3E-05	2.5E-04	1.7E-05	1.6E-04	9.9E-06	1.9E-04
SKULL VALLEY	2.3E-05	2.0E-04	1.3E-05	2.4E-04	1.7E-05	1.5E-04	1.0E-05	1.8E-04
IDAHO NATIONAL LAB								
ORNL	1.8E-05	1.1E-04	3.7E-06	1.3E-04	1.4E-05	8.6E-05	2.8E-06	1.0E-04
DEAF SMITH	6.6E-06	5.8E-05	2.2E-06	6.7E-05	5.0E-06	4.5E-05	1.7E-06	5.2E-05
HANFORD	5.3E-06	3.0E-05	1.1E-06	3.6E-05	4.0E-06	2.3E-05	8.2E-07	2.8E-05
SKULL VALLEY	3.0E-06	2.5E-05	1.5E-06	3.0E-05	2.3E-06	1.9E-05	1.1E-06	2.2E-05

Table 2-7. Collective doses to residents near the route (person-Sv) for truck transportation per shipment (1 Sv = 10^5 mrem)

FROM	TO	Truck-DU				
		Rural	Suburban	Urban	Urban Rush Hour[a]	Total
MAINE YANKEE	ORNL	5.0E-06	8.9E-05	2.0E-06	4.5E-07	9.6E-05
	DEAF SMITH	1.0E-05	1.2E-04	2.1E-06	4.8E-07	1.4E-04
	HANFORD	1.4E-05	1.0E-04	1.5E-06	3.2E-07	1.2E-04
	SKULL VALLEY	1.1E-05	9.5E-05	1.5E-06	3.3E-07	1.1E-04
KEWAUNEE	ORNL	4.1E-06	4.6E-05	1.1E-06	2.5E-07	5.2E-05
	DEAF SMITH	6.6E-06	3.9E-05	7.6E-07	1.7E-07	4.7E-05
	HANFORD	9.1E-06	4.1E-05	7.0E-07	1.5E-07	5.1E-05
	SKULL VALLEY	7.3E-06	3.1E-05	6.7E-07	1.5E-07	3.9E-05
INDIAN POINT	ORNL	4.1E-06	6.4E-05	1.6E-07	1.6E-07	6.9E-05
	DEAF SMITH	1.3E-05	1.3E-04	6.9E-07	3.1E-07	1.4E-04
	HANFORD	1.3E-05	7.6E-05	2.6E-07	2.6E-07	8.9E-05
	SKULL VALLEY	1.0E-05	6.6E-05	2.7E-07	2.7E-07	7.7E-05
IDAHO NATIONAL LAB	ORNL	8.8E-06	5.3E-05	7.7E-07	1.7E-07	6.3E-05
	DEAF SMITH	4.6E-06	3.0E-05	6.9E-07	1.5E-07	3.7E-05
	HANFORD	5.5E-06	8.8E-06	1.1E-07	4.2E-08	1.4E-05
	SKULL VALLEY	1.2E-06	1.0E-05	2.7E-07	5.9E-08	1.2E-05

[a]During rush hour RADTRAN halves the truck speed and doubles the vehicle density to take into account traffic jams. Detailed data for the actual traffic speed and density on a city-by-city basis is not available. The rush-hour collective dose is in addition to the urban (non-rush-hour) collective dose, both are included in the total.

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Collective dose is best used in making comparisons, e.g., in comparing the risks of routine transportation along different routes, by different modes (truck or rail), or in different casks. Several such comparisons can be made from the results shown in Table 2-6 and Table 2-7.

- Suburban residents sustain the largest dose for all routes and shipment modes.
- Urban residents sustain a larger dose from a single rail shipment than from a truck shipment on the same state route, even though urban population densities are similar and the external dose rates from the cask are nearly the same. As shown in Table 2-5, most (though not all) rail routes have more urban miles than the analogous truck route. Train tracks go from city center to city center, while trucks carrying spent fuel must use interstates and bypasses. In several cases shown in Table 2-5, the rail route had twice as many urban miles as the corresponding truck route. Also, train speeds in urban areas are only ¼ the truck speeds.
- Overall, collective doses are larger for a single shipment on rail routes than truck routes because the rail routes are often longer, especially in the western U.S., where there is rarely a choice of railroads and train speeds are lower than truck speeds, especially in urban areas. However, the rail casks hold about six times as much spent fuel as the truck cask, so to move

a given amount of spent fuel would take six truck shipments for each rail shipment, making the total dose from shipping by truck higher.

- The collective doses shown in Table 2-6 and Table 2-7 are all very small. However, they are not the only doses the people along the route receive. Background radiation is 0.0036 Sv per year in the U.S., or 4.1×10^{-7} Sv/ hour. The contribution of a single shipment to the population's collective dose is illustrated by the following example of the Maine Yankee to ORNL truck route:
  - From Table 2-7 the total collective dose to residents for this route is 9.6×10^{-5} person-Sv.
  - From Table 2-5, there are 1,137,834 people within 800m of the route.
  - Background is 4.1×10^{-7} Sv/hour, which everyone is exposed to all the time, whether a shipment occurs or not.
  - A truck traveling at an average of 108 km per hour travels the 1748 km in 16 hours.
  - During those 16 hours, the 1,137,834 people will have received a collective background dose of 7.56 person-Sv, about 80,000 times the collective dose from the shipment.
  - The total collective dose during a shipment to these 1,137,834 people is not 9.6×10^{-5} person-Sv), but 7.560096 person-Sv.
  - The NRC recommends that collective dose be used only for comparative purposes (NRC, 2008).
  - The appropriate comparison between the collective dose from this shipment of spent fuel is not a comparison between 9.6×10^{-5} person-Sv from the shipment and zero dose if there is no shipment, but between 7.560096 person-Sv if there is a shipment and 7.560000 person-Sv if there is no shipment.

A more complete discussion of collective dose is in Appendix II, Section II.6.

2.3.3 Doses to Members of the Public Occupying Vehicles that Share the Route

Rail

Much of the United States rail is either double track or equipped with "passing tracks" that let one train pass another. When a train passes the train carrying the spent fuel cask, occupants of the passing train will receive some of the external radiation. The great majority of trains in United States carry freight, and the only occupants of the passing train are crew members. Only about one railcar in 60 has an occupant.

Comment [MF13]: Revised text suggested by ORNL not implemented.

The dose to occupants of other trains in this situation depends on train speed and the external dose rate from the spent fuel casks. Table 2-8 shows the collective dose to public passengers of trains sharing the route, assuming for calculation purposes that occupants of trains are represented by one person in each passing railcar in rural and suburban areas, and five people in

urban areas.[11] The rural and suburban collective doses are probably unrealistically large, since most freight rail going through rural and many suburban areas never encounters a passenger train. Data were not available to account for the occupancy of actual passenger trains, including commuter rail, that share rail routes with freight trains.

Table 2-8. Collective doses (person-Sv) per shipment to occupants of trains sharing the route

SHIPMENT ORIGIN/ DESTINATION	Rail-Lead Cask				Rail-Steel Cask			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
MAINE YANKEE								
ORNL	2.0E-05	1.2E-05	7.5E-06	4.0E-05	1.5E-05	9.3E-06	5.6E-06	3.0E-05
DEAF SMITH	3.8E-05	1.3E-05	9.7E-06	6.1E-05	2.9E-05	1.0E-05	7.4E-06	4.6E-05
HANFORD	6.2E-05	1.7E-05	1.6E-05	9.0E-05	4.7E-05	1.3E-05	1.2E-05	6.8E-05
SKULL VALLEY	4.8E-05	1.6E-05	9.6E-06	7.4E-05	3.6E-05	1.2E-05	7.3E-06	5.5E-05
KEWAUNEE								
ORNL	1.4E-05	7.0E-06	5.8E-06	2.7E-05	1.0E-05	5.3E-06	4.4E-06	2.0E-05
DEAF SMITH	2.4E-05	5.2E-06	5.1E-06	3.4E-05	1.8E-05	4.0E-06	3.9E-06	2.6E-05
HANFORD	4.2E-05	6.7E-06	2.8E-06	5.2E-05	3.2E-05	5.1E-06	2.1E-06	3.9E-05
SKULL VALLEY	3.5E-05	7.8E-06	5.8E-06	4.9E-05	2.7E-05	5.9E-06	4.4E-06	3.7E-05
INDIAN POINT								
ORNL	9.2E-06	8.1E-06	9.6E-06	2.7E-05	7.0E-06	6.1E-06	7.2E-06	2.0E-05
DEAF SMITH	3.6E-05	1.1E-05	9.4E-06	5.6E-05	2.8E-05	8.2E-06	7.1E-06	4.3E-05
HANFORD	6.0E-05	1.4E-05	1.1E-05	8.5E-05	4.6E-05	1.1E-05	8.0E-06	6.5E-05
SKULL VALLEY	4.8E-05	1.3E-05	1.1E-05	6.5E-05	3.6E-05	1.0E-05	8.0E-06	4.9E-05
INL								
ORNL	4.6E-05	7.1E-06	3.4E-06	5.7E-05	3.5E-05	5.4E-06	2.6E-06	4.3E-05
DEAF SMITH	2.7E-05	3.2E-06	1.9E-06	3.2E-05	2.1E-05	2.5E-06	1.4E-06	2.5E-05
HANFORD	1.5E-05	1.7E-06	9.3E-07	1.8E-05	1.2E-05	1.3E-06	7.0E-07	1.4E-05
SKULL VALLEY	5.5E-06	1.5E-06	1.2E-06	8.2E-06	4.2E-06	1.1E-06	9.0E-07	6.2E-06

Truck

Unlike the train situation, a truck carrying spent fuel shares the primary highway system with many cars, light trucks, and other vehicles, and is modeled in RADTRAN as shown in Figure

[11] The five persons per railcar in urban areas are assumed to include occupants of passenger trains. Passenger trains carry more than five per car, but the majority of railcars even in urban areas carry freight only. This estimate is consistent with estimates made in past studies.

2-9. The occupants of any car or truck that passes the spent fuel cask in either direction will receive a small radiation dose.

Deleted: -

The radiation dose to occupants of other vehicles depends on the exposure distance and time, the number of other vehicles on the road, and the number of people in the other vehicles. Occupants of the vehicles that share the route are closer to the cask than residents or others beside the route. Occupants of vehicles moving in the opposite direction from the cask are exposed to radiation from the cask for considerably less time because the vehicles involved are moving past each other. The exposure time for vehicles traveling in the same direction as the cask is assumed to be the time needed to travel the link at the average speed (Neuhauser et al., 2000). The number of other vehicles that share truck routes is very large; the average number of vehicles per hour on U.S. interstate and primary highways in 2004[12] (Weiner, et al., 2009, Appendix D) were:

- 1119 on rural segments, about 2 ½ times the 1977 vehicle density.
- 2464 on suburban segments, almost four times the 1977 vehicle density.
- 5384 on urban segments, about twice the 1977 vehicle density.

Each vehicle was assumed to have an average of one and a half occupants since the majority of cars and light trucks traveling on freeways have one or two occupants. State highway departments provide traffic count data but do not provide vehicle occupancy data. If two occupants are assumed, the collective doses are one-third larger.

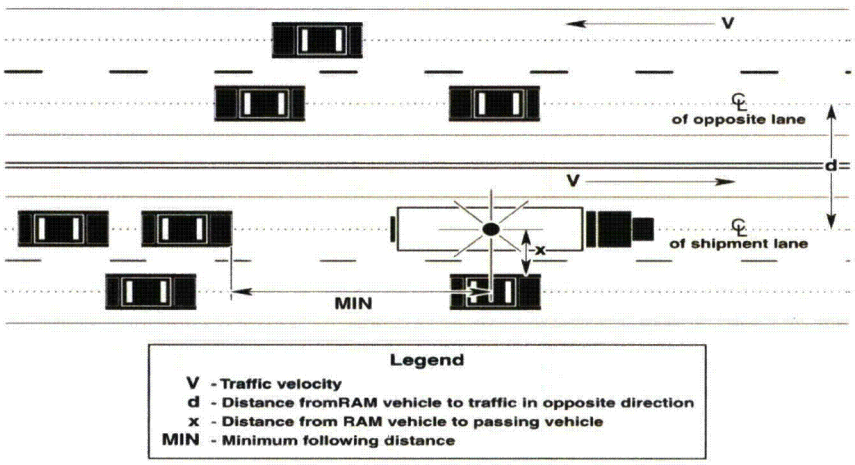


Figure 2-9. Diagram used in RADTRAN for calculating radiation doses to occupants of other vehicles (from Neuhauser et al., 2000).

Detailed discussion and state-by-state results are presented in Appendix II. The collective doses for truck traffic are shown in Table 2-9.

[12] 2004 is the most recent year for which data have been validated.

Table 2-9. Collective doses (person-Sv) per shipment to occupants of vehicles sharing truck routes

FROM	TO	Truck-DU				
		Rural	Suburban	Urban	Urban Rush Hour[a]	Total
MAINE YANKEE	ORNL	1.3E-04	2.4E-04	5.2E-05	4.8E-05	4.6E-
	DEAF SMITH	2.8E-04	3.3E-04	6.9E-05	6.4E-05	7.3E-
	HANFORD	4.5E-04	3.0E-04	4.3E-05	4.0E-05	8.3E-
	SKULL VALLEY	3.7E-04	2.5E-04	4.4E-05	4.5E-05	7.0E-
KEWAUNEE	ORNL	9.6E-05	1.4E-04	4.8E-05	4.4E-05	3.3E-
	DEAF SMITH	1.8E-04	8.9E-05	2.2E-05	2.0E-05	3.1E-
	HANFORD	3.4E-04	1.4E-04	3.3E-05	3.0E-05	5.4E-
	SKULL VALLEY	2.4E-04	8.6E-05	2.5E-05	2.3E-05	3.8E-
INDIAN POINT	ORNL	1.8E-04	2.1E-04	3.3E-05	3.0E-05	4.6E-
	DEAF SMITH	2.8E-04	3.1E-04	5.6E-05	5.2E-05	6.9E-
	HANFORD	4.2E-04	2.2E-04	4.8E-05	4.4E-05	7.2E-
	SKULL VALLEY	3.6E-04	2.2E-04	4.5E-05	4.1E-05	6.6E-
IDAHO NATIONAL LAB	ORNL	3.0E-04	1.5E-04	2.4E-05	2.2E-05	5.0E-
	DEAF SMITH	2.2E-04	7.3E-05	2.7E-05	2.5E-05	3.4E-
	HANFORD	1.0E-04	8.5E-05	9.5E-06	8.7E-06	2.0E-
	SKULL VALLEY	3.7E-05	3.2E-05	8.5E-06	7.8E-06	8.5E-

[a] During rush hour the truck speed is halved and the vehicle density is doubled.

Comment [MF14]: Previous ORNL comment, "Rush hour parameters need to be provided" appears to heave been ignored.

Comparing Table 2-6 to Table 2-8 it can be seen that for rail transport the collective dose to residents is generally larger (the exception is in rural areas) than the collective dose to people sharing the rail line. This is not surprising because of the low number of people sharing the rail line. In contrast to this, comparing Table 2-7 to Table 2-9 it can be seen that for all routes and population densities the collective dose to those sharing the highway is greater than the collective dose to nearby residents.

2.3.4 Doses at Truck and Train Stops

Both trucks and trains stop occasionally on long trips. Common carrier freight trains stop to exchange freight cars, to change crews, and, when necessary, to change railroads. The rail stops at the origin and destination of a trip are called "classification stops" and are 27 hours long. Spent fuel casks may be carried on dedicated trains as well as on regular freight trains although, in practice, previous spent fuel shipments have been carried on dedicated trains. A dedicated train is a train that carries a single cargo from origin to destination; coal unit trains are an example of dedicated trains. The analyses conducted here assume the casks are transported on dedicated trains, which eliminates the need for intermediate classification stops.

When a train is stopped, the dose to anyone nearby depends on the distance between that person and the cask and the time that the individual is exposed. The people exposed at a rail stop include:

- Railyard workers (including inspectors)
- Train crew (passenger trains do not typically enter railyards)
- Residents who live near the rail yard.

The semi-tractor trucks that carry Truck-DU casks each have two 300-liter (80-gallon) fuel tanks, and generally stop to refuel when half of the fuel is gone, approximately every 845 km (525 miles) (DOE, 2002). Trucks carrying spent fuel are also stopped at the origin and destination of each trip. Mandatory rest and crew changes are combined with refueling stops whenever possible.

The people likely to be exposed at a refueling truck stop are:

- The truck crew of two; usually one crew member at a time will fill the tanks.
- Other people who are using the truck stop, since these trucks stop at public truck stops.
- Residents of areas near the stop.

A number of states inspect spent fuel cask shipments when the trucks enter the state. Inspection stations may be combined with truck weigh stations, so that inspectors of both the truck carrying spent fuel and trucks carrying other goods can be exposed, as well as the crew from other trucks. When the vehicle is stopped, doses to receptors depend only on distance from the source and exposure time, so that any situation in which the cask and the receptor stay at a fixed distance from each other can be modeled as a stop. Such stop-like exposure situations include inspections, vehicle escorts, vehicle crew when the vehicle is in transit, and occupants of other vehicles near the stopped vehicle. Any of these situations can be modeled in RADTRAN. Details of the calculations performed for these situations in this analysis are found in Appendix II.

Figure 2-10 is a diagram of the model used to calculate doses at truck stops. The inner circle defines the area occupied by people who share the stop with the spent fuel truck, who are between the truck and the building, and who are not shielded from the truck's external radiation. People in buildings at the stop are shielded.

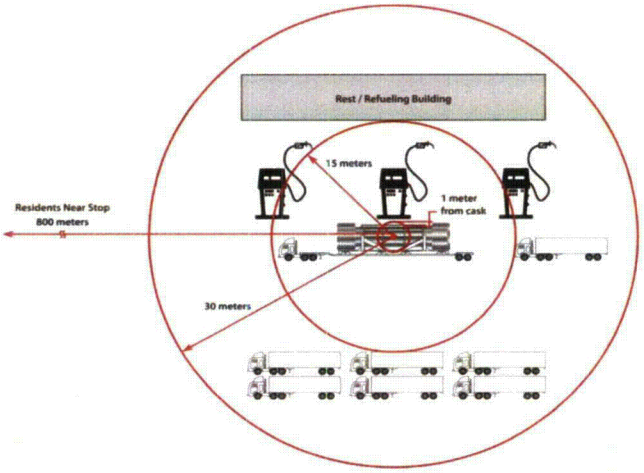


Figure 2-10. Diagram of truck stop model (not to scale).

Table 2-10 lists some sample input data used to calculate doses at truck and train stops.

Table 2-10. Sample input data for calculating doses at truck and train stops.

Data	Interstate Highway	Freight Rail
Minimum distance from nearby residents (m)	30	200
Maximum distance from nearby residents (m)	800	800
Stop time for rail classification (hours)	NA	27
Stop time in transit for railroad change (hours)	NA	0.5
Stop time at truck stops (hours)	0.83	NA
Minimum distance to people sharing the stop (m)	1[a]	NA
Maximum distance to people sharing the stop (m)	15[a]	NA

[a]From Griego et al., 1996

Comment [MF15]: ORNL comment, "Is this table used for anything and if so, how does it relate to table 2-11 and the other calculations?" not addressed.

Rail

Trains are stopped for classification for 27 hours at the beginning and end of a trip. The collective dose from the radioactive cargo to the railyard workers at these classification stops for the two rail casks studied is:

- For the Rail-Lead: 1.46×10^{-5} person-Sv (1.46 person-mrem)
- For the Rail-Steel: 1.09×10^{-5} person-Sv (1.09 person-mrem)

The average dose (calculated by dividing the collective dose by the number of exposed people) to an individual living between 200 and 800 meters from a classification yard is

- 3.5×10^{-7} Sv (0.035 mrem) from the Rail-Lead cask
- 2.7×10^{-7} Sv (0.027 mrem) from the Rail-Steel cask

Table 2-11 shows the doses at train stops to yard workers and residents near the stops for the Maine Yankee-to Hanford rail route. Because different routes have different in-transit stops and stop times for crew changes and inspections, a representative result is provided instead of presenting results for an entire route or for all sixteen rail routes.

Table 2-11. Collective doses at rail stops on the Maine Yankee-to-Hanford route (person-Sv)

Stop	Route type (R, S, U) and State	Time (hours)	Railyard Worker		Residents Near Stop	
			Rail-Lead	Rail-Steel	Rail-Lead	Rail-Steel
Classification, origin	S, ME	27	1.5E-05	1.1E-05	2.3E-05	1.8E-05
1	S, ME	4.0	2.2E-06	1.6E-06	3.4E-06	2.6E-06
2	R, NY	4.0	2.2E-06	1.6E-06	9.2E-07	6.9E-07
3	S, IL	2.0	1.1E-06	8.1E-07	1.2E-05	9.4E-06
Classification, destination	S, WA	27	1.5E-05	1.1E-05	1.9E-05	1.4E-05

Truck

Table 2-12 shows the collective doses to residents near stops for the rural and suburban segments of the 16 truck routes studied. Urban stops were not modeled because trucks carrying spent fuel casks are unlikely to stop in urban areas. A detailed discussion and example of the calculations performed to derive this table is provided in Appendix II.

Comment [MF16]: Previous ORNL comment, "Table 2-11 is a representative example. Provide reference to the actual results for all 16 routes (tables xx –x appendix II)." not enacted.

Comment [MF17]: Only a portion of pevious ORNL comment, "Is this table only to show as an example? Because the first stop in Maine is the origination point which should have a time of 27 hours, according to the text in the first paragraph of Section 2.3.4. What is Stop 1 listed as 4 hours?

Also, Stop 2 and 3 are locations where the rail shipment changes carriers. Table 2-9 states that this is a delay of 0.5 hours. Why is Stop 2 listed at 4 hours and Stop 3 at 2 hours?" has been addressed.

Comment [MF18]: Note numbers for 1, 2 and 3 did not change from previous edition. Also, the following comment may still need to be addressed, "As a quick check, the doses should vary by route with the different hours, but shouldn't the relative ratios of the doses between lead and steel cask be the same and in ratio with the 1 meter dose rates presented in Table 2-1? Beyond this difference in 1 meter dose rate, all other aspects of the calculations should be identical between the different casks. The ratio in table 2-1 is 1.03/1.4 =0.735. The values here range from 0.73 up to 0.78. Is this due to rounding or is there something fundamentally different? (see Results_Check spreadsheet)."

Comment [MF19]: Previous ORNL comment, "This is probably true, but it is not bounding or guaranteed. Since the number of stops here is sometimes on the order of only about 1, a single stop in an Urban area, with high population density would impact these values." not addressed.

Table 2-12. Collective doses to residents near truck stops (person-Sv)

Origin	Destination	Type	Persons/km^2	Number of Stops	Dose
MAINE YANKEE	ORNL	Rural	19.9	1.14	1.1 E-06
		Suburban	395	0.93	2.3 E-05
	Deaf Smith	Rural	18.6	2.47	1.5 E-06
		Suburban	371	1.6	1.7 E-05
	Hanford	Rural	15.4	4.33	2.2 E-06
		Suburban	325	1.5	1.4 E-05
	Skull Valley	Rural	16.9	3.5	1.9 E-06
		Suburban	332.5	1.3	1.2 E-05
KEWAUNEE	ORNL	Rural	19.8	0.81	5.2 E-07
		Suburban	361	0.59	6.0 E-06
	Deaf Smith	Rural	13.5	2.0	8.6 E-07
		Suburban	339	0.52	5.0 E-06
	Hanford	Rural	10.5	3.4	1.2 E-06
		Suburban	316	0.60	5.4 E-06
	Skull Valley	Rural	12.5	2.6	1.1 E-06
		Suburban	324.5	0.44	4.1 E-06
INDIAN POINT	ORNL	Rural	20.5	0.71	4.7 E-07
		Suburban	388	0.71	7.8 E-06
	Deaf Smith	Rural	17.1	2.3	1.3 E-06
		Suburban	370	1.2	1.3 E-05
	Hanford	Rural	13.0	4.1	1.8 E-06
		Suburban	338	1.1	1.1 E-05
	Skull Valley	Rural	14.2	3.3	1.5 E-06
		Suburban	351	0.93	9.3 E-06
IDAHO NATIONAL LAB	ORNL	Rural	12.4	3.1	1.3 E-06
		Suburban	304	0.72	6.3 E-06
	Deaf Smith	Rural	7.8	2.3	5.8 E-07
		Suburban	339	0.35	3.4 E-06
	Hanford	Rural	6.5	0.43	9.0E-08
		Suburban	200	0.57	3.2 E-06
	Skull Valley	Rural	10.1	0.42	1.4 E-07
		Suburban	343	0.11	1.1 E-06

Comment [MF20]: For the first two entries, the numbers of stops and dose changed from previous, then for the rest of the table the entries are the exact same as previous versions. Is this correct?

Also, for the first two entries for the number of stops this table is not in agreement with Table II-25 in Appendix II.

The rural and suburban population densities in Table 2-12 are the averages for the entire route. An analogous calculation can be made for each state traversed. However, in neither case can one determine beforehand exactly where the truck will stop to refuel. In some cases (e.g., INL to Skull Valley) the truck may not stop at all; the total distance from INL to the Skull Valley site is only 466.2 km (290 miles). The route from Indian Point to ORNL illustrates another situation. This route is 1028 km (639 miles) long, and would include one truck stop. This stop could be in

either a rural or a suburban area. The results shown in Table 2-12 are general average doses at stops.

2.4 Doses to Workers

Radiation doses to workers are limited in accordance with the regulations of 10 CFR Part 20 and the practice of ALARA: maintaining the worker exposure to ionizing radiation "as low as reasonably achievable." ALARA applies to occupational doses because workers are potentially exposed to much larger doses than the general public. For example, the cab of a truck carrying a loaded Truck-DU cask is shielded so that 63% of the radiation from the end of the cask is blocked.

Occupational doses from routine, incident-free radioactive materials transportation include doses to truck and train crew, railyard workers, truck-stop workers, inspectors, and escorts. Not included are workers who handle spent fuel containers in storage, loading and unloading casks from vehicles or during intermodal transfer, and attendants who would refuel trucks, because truck refueling stops in the U.S. no longer have such attendants.[13] Table 2-13 summarizes the occupational doses. All doses are reported per hour except for the truck stop worker (reported for the maximum truck stop time) and the rail classification yard workers. All doses are individual doses (Sv) except for the railyard worker collective doses.

Table 2-13. Occupational doses and dose rates from routine incident-free transportation

Cask and route type	Train crew in transit : 3 people; person-Sv/km	Truck crew in transit 2 people; person-Sv /km	Escort: Sv/hour[a]	Inspector: Sv per inspection	Truck stop worker: Sv per stop	Rail classification yard workers: person-Sv /stop
Rail-Lead rural/suburban	4.3E-07		5.8E-06			1.5E-05
Rail-Lead urban	7.2E-07		5.8E-06			[b]
Rail-Steel rural/suburban	3.3E-07		4.4E-06			1.1E-05
Rail-Steel urban	5.5E-07		4.4E-06			[b]
Truck - DU rural/suburban		3.8E-07	4.9E-09	3.7E-04	6.7E-06	
Truck - DU urban		3.6E-07	4.9E-09			

[a]The truck crew is shielded while in transit to sustain a maximum dose of 0.02 mSv/hour
[b]Even classification yards within metropolitan areas do not typically have urban population densities because of the large area occupied by the classification yard.

The doses to rail crew and rail escorts are similar. Spent fuel may be transported in dedicated trains so that both escorts and train crew are assumed to be within a distance of one railcar length

[13] The States of Oregon and New Jersey still require gas station attendants to refuel cars and light duty vehicles, but heavy truck crew do their own refueling.

of the railcar carrying the spent fuel. Escorts in the escort car are not shielded because they must maintain line-of-sight to the railcar carrying spent fuel. Train crew members are in a crew compartment and were assumed to have some shielding, resulting in an estimated dose about 25 percent less than the escort. The largest collective doses are to railyard workers. The number of workers in railyards is not a constant, and the number of activities that brings these workers into proximity with the shipment varies as well. This analysis assumes the dose to the worker doing an activity for each activity: inspection, coupling and decoupling the railcars, moving the railcar into position for coupling, etc. The differences between doses in the Rail-Lead case and the Rail-Steel case reflect the differences in cask dimensions and in external dose rate.

Truck crew members are shielded so that they receive a maximum dose of 2.0×10^{-5} Sv per hour. This regulatory maximum was imposed in the RADTRAN calculation. Truck inspectors generally spend about an hour within one meter of the cargo (Weiner and Neuhauser, 1992), resulting in a relatively large dose. An upper bound to the duration of a truck refueling stop is about 50 minutes (0.83 hours) (Griego, et al., 1996). The truck stop worker whose dose is reflected in Table 2-13 is assumed to be outside (unshielded) at 15 meters from the truck during the stop. Truck stop workers that are in concrete or brick buildings would be shielded from any radiation.

2.5 Chapter Summary

A summary of the results for the incident free transport of spent fuel in the three casks analyzed in this study are presented in Table 2-14, Table 2-15, and Table 2-16.

Table 2-14. Total Collective Dose in Person-Sv from Routine Transportation for Each Rail Route for the Rail-Lead Cask

Origin	Destination	Residents Along Route	Occupants of Vehicles Sharing Route	Residents near Stop	Railyard Crew and Escorts	Total
MAINE YANKEE	ORNL	2.1E-04	6.6E-05	6.6E-06	3.4E-04	6.2E-04
	Deaf Smith	2.5E-04	1.0E-04	6.6E-06	5.1E-04	8.7E-04
	Hanford	2.9E-04	1.6E-04	6.6E-06	7.6E-04	1.2E-03
	Skull Valley	2.7E-04	1.2E-04	6.6E-06	6.2E-04	1.0E-03
KEWAUNEE	ORNL	1.3E-04	4.6E-05	6.2E-06	2.3E-04	4.1E-04
	Deaf Smith	1.1E-04	5.5E-05	6.2E-06	3.0E-04	4.7E-04
	Hanford	1.1E-04	7.6E-05	6.2E-06	4.7E-04	6.6E-04
	Skull Valley	1.4E-04	7.7E-05	6.2E-06	4.3E-04	6.5E-04
INDIAN POINT	ORNL	1.6E-04	5.1E-05	1.7E-06	2.1E-04	4.3E-04
	Deaf Smith	2.0E-04	9.3E-05	1.7E-06	4.8E-04	7.7E-04
	Hanford	2.5E-04	1.3E-04	1.7E-06	7.2E-04	1.1E-03
	Skull Valley	2.4E-04	9.2E-05	1.7E-06	6.0E-04	9.3E-04
INL	ORNL	1.3E-04	8.4E-05	2.6E-06	5.1E-04	7.3E-04
	Deaf Smith	6.8E-05	4.8E-05	2.6E-06	3.1E-04	4.2E-04
	Hanford	3.6E-05	2.7E-05	2.6E-06	1.8E-04	2.5E-04
	Skull Valley	3.0E-05	1.3E-05	2.6E-06	9.5E-05	1.4E-04

Comment [MF21]: Values in this Table are from Tables 2-6, 2-8, 2-11, but they disagree.

Comment [MF22]: It is not intuitive that the residents near stop results would be the same for each origin regardless of destination, but different for each origin?

Table 2-15. Total Collective Dose in Person-Sv from Routine Transportation for Each Rail Route for the Rail-Steel Cask

Origin	Destination	Residents Along Route	Occupants of Vehicles Sharing Route	Residents near Stop	Railyard Crew and Escorts	Total
MAINE YANKEE	ORNL	1.6E-04	5.1E-05	5.0E-06	2.3E-04	4.5E-04
	Deaf Smith	1.9E-04	7.7E-05	5.0E-06	3.7E-04	6.4E-04
	Hanford	2.2E-04	1.2E-04	5.0E-06	5.6E-04	9.1E-04
	Skull Valley	2.1E-04	9.0E-05	5.0E-06	4.5E-04	7.5E-04
KEWAUNEE	ORNL	9.6E-05	3.6E-05	4.7E-06	1.5E-04	2.9E-04
	Deaf Smith	8.3E-05	4.3E-05	4.7E-06	2.1E-04	3.4E-04
	Hanford	8.1E-05	5.8E-05	4.7E-06	3.3E-04	4.8E-04
	Skull Valley	1.1E-04	5.9E-05	4.7E-06	3.0E-04	4.7E-04
INDIAN POINT	ORNL	1.2E-04	4.0E-05	1.3E-06	1.4E-04	3.1E-04
	Deaf Smith	1.6E-04	7.2E-05	1.3E-06	3.4E-04	5.7E-04
	Hanford	1.9E-04	1.0E-04	1.3E-06	5.2E-04	8.2E-04
	Skull Valley	1.8E-04	7.0E-05	1.3E-06	4.4E-04	6.9E-04
INL	ORNL	1.0E-04	6.4E-05	1.9E-06	3.6E-04	5.3E-04
	Deaf Smith	5.1E-05	3.7E-05	1.9E-06	2.1E-04	3.0E-04
	Hanford	2.7E-05	2.0E-05	1.9E-06	1.2E-04	1.7E-04
	Skull Valley	2.3E-05	1.0E-05	1.9E-06	5.0E-05	8.5E-05

Comment [MF23]: See comment above for values in this table.

Comment [MF24]: See comment above regarding residents near stops.

Table 2-16. Total Collective Dose in Person-Sv from Routine Transportation for Each Highway Route for the Truck Cask

Origin	Destination	Residents Along Route	Occupants of Vehicles Sharing Route	Residents near Stop	Persons Sharing Stop	Crew/ Truck Stop Worker	Total
MAINE YANKEE	ORNL	9.6E-05	4.6E-04	1.2E-05	4.4E-04	6.8E-04	1.7E-03
	Deaf Smith	1.4E-04	7.5E-04	1.8E-05	9.3E-04	1.4E-03	3.2E-03
	Hanford	1.2E-04	8.3E-04	1.4E-05	1.3E-03	1.9E-03	4.2E-03
	Skull Valley	1.1E-04	6.1E-04	1.4E-05	1.1E-03	1.6E-03	3.5E-03
KEWAUNEE	ORNL	5.2E-05	3.3E-04	6.6E-06	3.2E-04	4.9E-04	1.2E-03
	Deaf Smith	4.7E-05	3.1E-04	5.9E-06	5.7E-04	8.3E-04	1.8E-03
	Hanford	5.1E-05	5.4E-04	6.6E-06	9.2E-04	1.3E-03	2.9E-03
	Skull Valley	3.9E-05	4.5E-04	5.1E-06	7.0E-04	1.0E-03	2.2E-03
INDIAN POINT	ORNL	6.1E-05	4.6E-04	8.3E-06	3.3E-04	4.9E-04	1.3E-03
	Deaf Smith	1.0E-04	7.5E-04	1.4E-05	8.1E-04	1.2E-03	2.9E-03
	Hanford	9.0E-05	8.3E-04	1.2E-05	1.2E-03	1.7E-03	3.9E-03
	Skull Valley	7.8E-05	6.1E-04	1.1E-05	9.7E-04	1.4E-03	3.1E-03
INL	ORNL	6.3E-05	4.9E-04	7.5E-06	8.8E-04	1.3E-03	2.7E-03
	Deaf Smith	3.6E-05	3.4E-04	4.0E-06	6.1E-04	8.8E-04	1.9E-03
	Hanford	1.5E-05	2.0E-04	1.1E-06	2.6E-04	3.7E-04	8.5E-04
	Skull Valley	1.2E-05	1.3E-03	1.2E-06	1.2E-04	1.8E-04	1.6E-03

Comment [MF25]: Collective doses for truck transportation. The values in the columns of this Table are calculated in other Tables: First column, *Residents along route* is from Table 2-7, Second column, *Occupant of vehicles sharing the route* is from Table 2-9, third column, *Residents near stop* is from Table 2-12. But the values of the last two columns cannot be obtained clearly from other values/tables in the Chapter.
Some of the values in Table 2-16 do not agree with the values in the Tables where they are originated. Some discrepancies are small, one decimal, but others are large (from Table 2-12).

Comment [MF26]: These values differ for each destination as would be intuitive.

A code that estimates risk can never be completely precise because the input data are estimates and projections. To account for this imprecision, the risk assessment code, RADTRAN, employs

assumptions and values that overestimate doses. Actual measurements confirm that RADTRAN overestimates doses by a small margin. Therefore, the doses calculated in this chapter should be regarded as overestimates.

Both the individual and collective doses calculated are for a single shipment and, even though overestimated, they are uniformly very small. Individual doses are comparable to background doses and are less than doses from many medical diagnostic procedures. Collective doses are orders of magnitude less than the collective background dose, as shown in Figure 2-11 for an example shipment from Maine Yankee to ORNL. The NRC recommends that collective doses (average doses integrated over a population) be used only for comparisons (NRC, 2008). The proper comparison for collective doses is between the background collective dose plus the shipment dose and the background dose if there is no shipment. An important concept to understand is that the collective dose is *never* zero in the absence of a shipment.

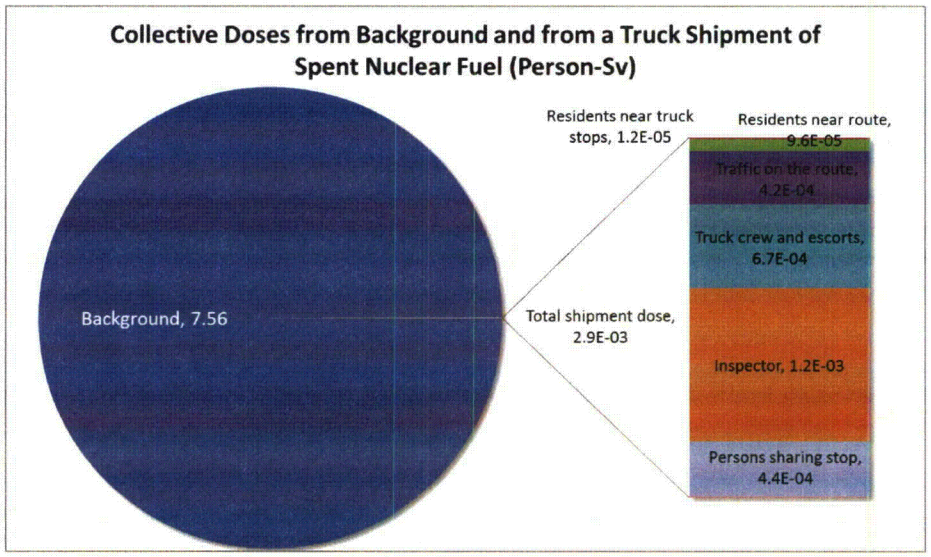


Figure 2-11. Collective doses from background and from Maine Yankee to ORNL truck shipments of spent nuclear fuel (person-Sv).

Comment [MF27]: Figure 2-11 and Table 2-16 the numbers for Traffic on the route (called occupant of vehicles sharing route in Table), Truck crew and escorts (called **Crew/ Truck Stop Worker in table [are these the same thing???]**, and total dose don't match. Column for Inspectors missing in Table 2-16 and also missing from appendix.

CHAPTER 3

CASK RESPONSE TO IMPACT ACCIDENTS

3.1 Introduction

Spent fuel casks are required to be accident resistant. During the certification process by the NRC the cask designer must demonstrate, among other things, that the cask would survive a free fall from a height of nine meters impacting onto a flat, essentially unyielding, target in the orientation that is most likely to damage the cask (10 CFR 71.73). The high standards and conservative approaches required by the NRC for this demonstration include the use of conservative (usually minimum) material properties in analyses, allowing only small amounts of yielding, and requiring materials with high ductility. These approaches ensure that the casks will not only survive impacts at the speed created due to the nine-meter drop, but will also survive much higher speed impacts.

In addition to the conservative designs assured by the certification process, there are two additional aspects of the nine-meter drop that provide safety when compared to actual accidents. The first of these is the requirement that the impact be onto an essentially unyielding target. This implies that all of the kinetic energy of the impact will be absorbed by the cask and none by the target. For impacts onto real surfaces, the kinetic energy is absorbed by both the cask and the target. The second aspect is the requirement that the vertical impact is onto a horizontal target. This requirement assures that at some point during the impact, the velocity of the cask will be zero, and all of the kinetic energy is converted into strain energy (absorbed by the cask). Most real accidents occur at an angle, and the kinetic energy of the cask is absorbed by multiple impacts instead of all in one impact. In this chapter, all three of these aspects will be discussed.

3.2 Finite Element Analyses of Casks

Previous risk studies have been carried out using generic casks. In the case of the Modal Study (Fischer et al., 1987) it was assumed any accident that was more severe than the regulatory hypothetical impact accident would lead to a release from the cask. In NUREG/CR-6672 (Sprung et al., 2000) the impact limiters of the generic casks were assumed to be unable to absorb more energy than the amount from the regulatory hypothetical impact accident (a nine-meter free fall onto an essentially rigid target). Modeling limitations at the time of the studies required both of these assumptions. In reality, casks and impact limiters each have excess capacity to resist impacts. In this study, three NRC-certified casks were used instead of generic casks, and the actual impact resistance capability of those cask designs was included in the analyses. However, for the truck cask no new finite element analyses were performed. This study relied upon analyses that were performed for other studies, some of which used a generic truck cask.

The response to impacts of 48, 97,145, and 193 kilometers per hour (kph)—equal to 30, 60, 90, and 120 mph—onto an unyielding target in the end, corner, and side orientations for the Rail-Steel and Rail-Lead spent fuel transportation casks were determined using the non-linear transient dynamics explicit finite element (FE) code PRESTO (SIERRA, 2009). PRESTO is a Lagrangian code, using a mesh that follows the deformation to analyze solids subjected to large, suddenly applied loads. The code is designed for a massively parallel computing environment

and for problems with large deformations, nonlinear material behavior, and contact. PRESTO has a versatile element library that incorporates both continuum (3D) elements and structural elements, such as beams and shells.

In addition to the detailed analyses of rail casks performed for this study, the response of the Truck-DU spent fuel transportation cask was inferred based upon the finite element analyses performed for the generic casks in NUREG/CR-6672. All analyses were performed with the direction of the cask travel perpendicular to the surface of the unyielding target. Figure 3-1 is a pictorial representation of the three impact orientations analyzed. In all of the analyses, the spent fuel basket and fuel elements were treated as a uniform homogenous material. The density of this material was adjusted to achieve the correct weight of the loaded basket. The overall behavior of this material was conservative (because it acts as a single entity that impacts the cask all at once instead of many smaller parts that impact the cask over a longer period of time) for assessing the effect the contents of the cask had on the behavior of the cask—the main focus of this chapter. Detailed response of the fuel assemblies was calculated using a sub-model of a single assembly.

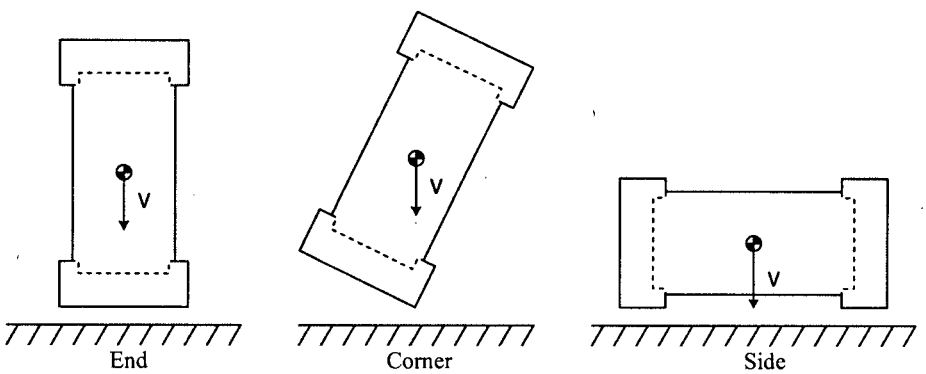


Figure 3-1. Impact orientations analyzed.

3.2.1 Rail-Steel Cask

Finite element model

Figure 3-2 shows the overall finite element model of the Rail-Steel cask depicted in Figure 1-3. This cask uses steel for its gamma-shielding material and transports 24 PWR assemblies in a welded multi-purpose canister. The impact limiters on each end of the cask are designed to absorb the kinetic energy of the cask during the regulatory hypothetical impact accident. They are made of an interior stainless steel support structure, aluminum honeycomb energy absorber, and a stainless steel skin. Figure 3-3 shows the finite element mesh of the closure end impact limiter (the one on the other end of the cask differs only in how it is attached to the cask). The aluminum honeycomb has properties that are direction-dependent. The strong direction of the honeycomb is oriented in the primary crush direction, requiring the finite element model to include the individual blocks of honeycomb material, rather than a single material for the entire impact limiter. The cask has a single solid steel lid that is attached with 54 1-5/8 inch diameter bolts and sealed with dual metallic o-rings. Figure 3-4 shows the finite element mesh of the

closure bolts (also shown are the bolts used to attach the closure end impact limiter) and the level of mesh refinement included in these important parts. Details of the finite element models, including material properties, contact surfaces, gaps, and material failure, are included in Appendix III.

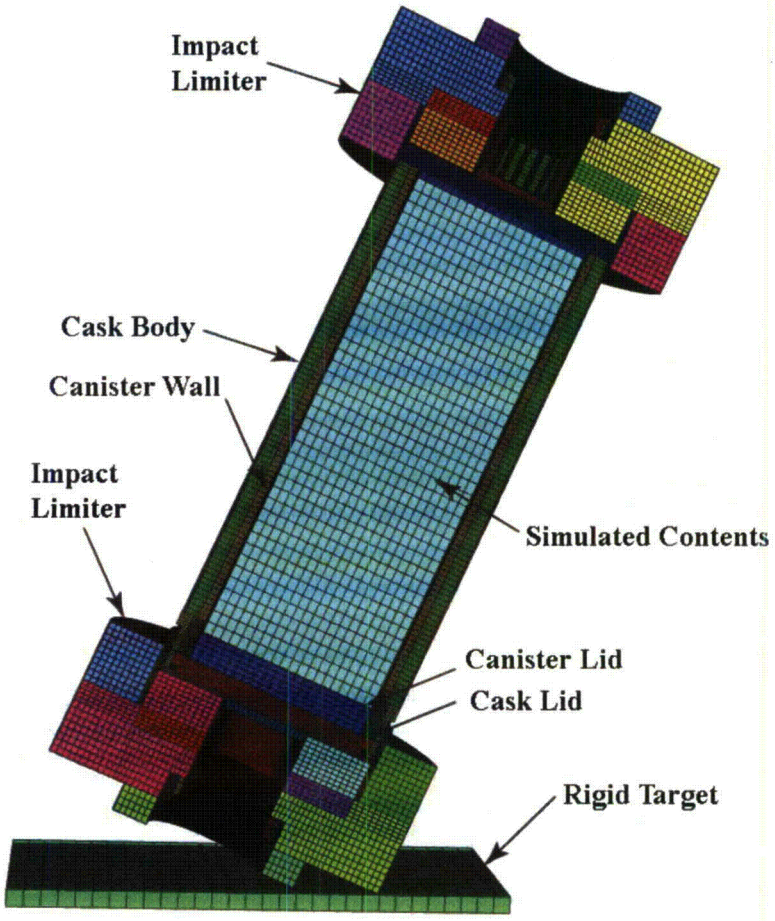
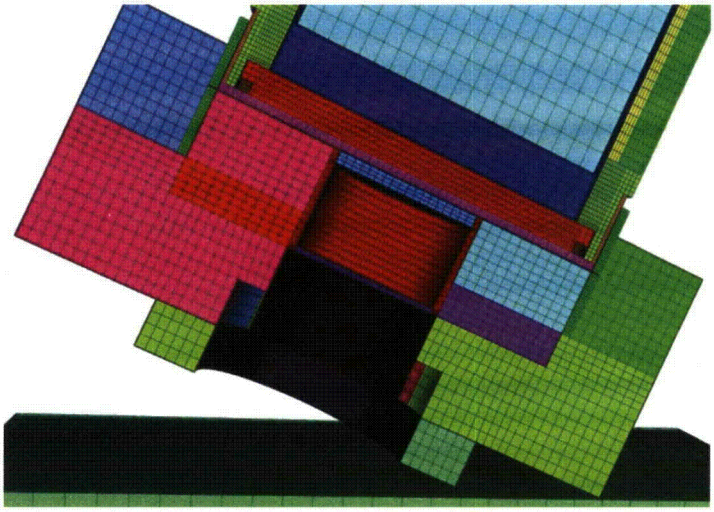
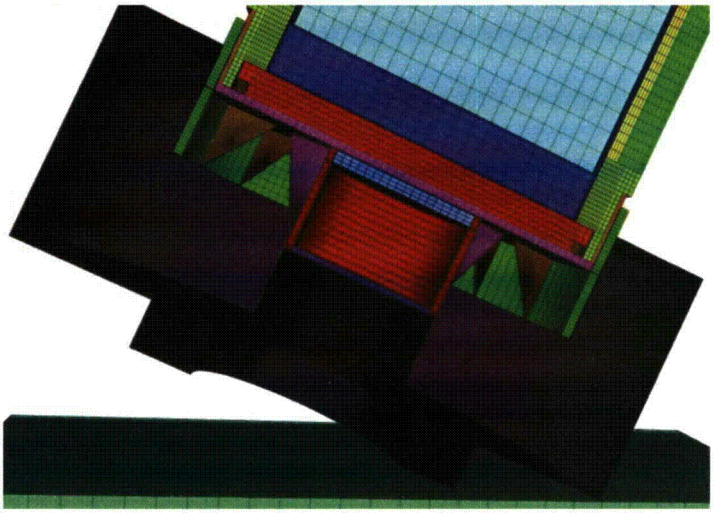


Figure 3-2. Finite element mesh of the Rail-Steel cask.



Impact limiter showing the various blocks of honeycomb



Impact limiter with the honeycomb removed to reveal the inner support structure

Figure 3-3. Details of the finite element mesh for the impact limiters of the Rail-Steel cask.

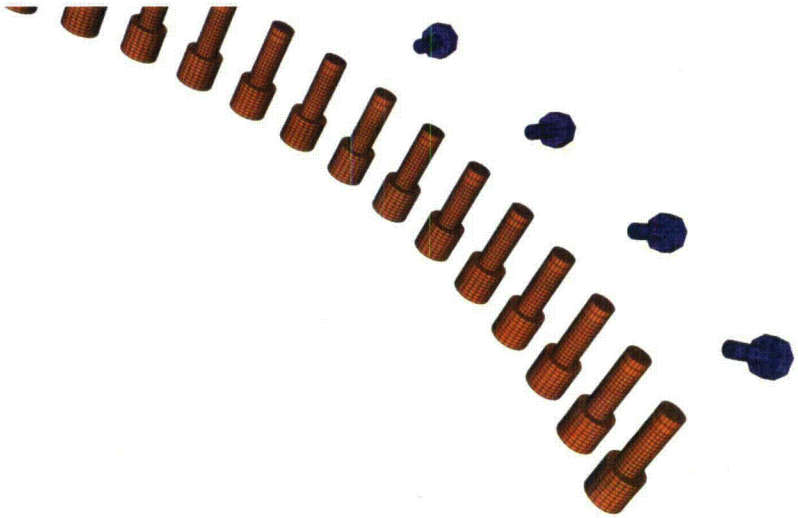


Figure 3-4. Finite element mesh of the Rail-Steel closure bolts and the closure end impact limiter attachment bolts. The highly refined mesh in these critical parts assures an accurate assessment of the closure response.

Analysis results

As expected, for all end, corner, and side impacts of the 48 kph (30 mph) impact analyses (the impact velocity from the regulatory hypothetical impact accident), the impact limiter absorbed almost all of the kinetic energy of the cask and there was no damage (permanent deformation) to the cask body or canister. As the impact velocity increases, there is first additional damage to the impact limiter, for all orientations, because it is absorbing more kinetic energy (this shows the margin of safety in the impact limiter design). At 97 kph there is still no significant damage to the cask body or canister. At an impact speed of 145 kph, damage to the cask and canister appears to begin. The impact limiter has absorbed all the kinetic energy it can, and any additional kinetic energy must be absorbed by plastic deformation in the cask body.

For the side impact at 145 kph, several of the lid bolts fail in shear (criteria for the failure model are included in Appendix III), but the lid remains attached. At this point the metallic seal no longer maintains the leak-tightness of the cask, but the spent fuel remains contained within the welded canister. Even at the highest impact speed, 193 kph, the welded canister remains intact for all orientations. Figure 3-5 shows the deformed shape and plastic strain in the canister for the 193 kph impact in a side orientation. This is the case that has the most plastic strain in the canister. The peak value of plastic strain (EQPS=Equivalent Plastic Strain, a representation of the magnitude of local permanent deformation) in this case is 0.7. The stainless steel material of the canister can easily withstand plastic strains greater than one (Blandford et al., 2007). These results demonstrate that no impact accident will lead to release of material from the Rail-Steel canister. Similar figures for the other orientations and speeds are included in Appendix III.

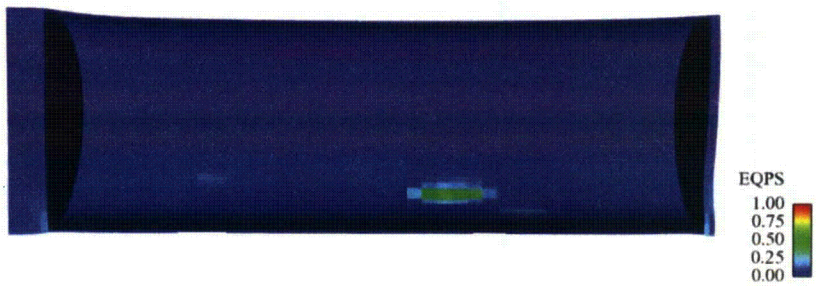


Figure 3-5. Plastic strain in the welded canister of the Rail-Steel for the 193 kph side impact case.

3.2.2 Rail-Lead Cask

Finite Element Model

Figure 3-6 shows the overall finite element model of the Rail-Lead cask depicted in Figure 1-2. This cask uses lead for its gamma-shielding material and transports either 26 directly loaded PWR assemblies or 24 PWR assemblies in a welded multi-purpose canister. The impact limiters on each end of the cask are designed to absorb the kinetic energy of the cask during the regulatory hypothetical impact accident. They are made up of redwood and balsa wood energy-absorbing material and a stainless steel skin. Figure 3-7 shows the finite element mesh of the closure end impact limiter (the impact limiter on the other end of the cask is identical). The cask has a dual lid system. The inner lid is attached with 42 1-1/2 inch diameter bolts and sealed with dual o-rings that are elastomeric if the cask is used only for transportation and metallic if the cask is used for storage before transportation case. The outer lid is attached with 36 1-inch diameter bolts and is sealed with a single o-ring that is elastomeric if the cask is used only for transportation and metallic if the cask is used for storage before transportation. Figure 3-8 shows the finite element mesh of the closure bolts and the level of mesh refinement included in these important parts. Details of the finite element models are included in Appendix III.

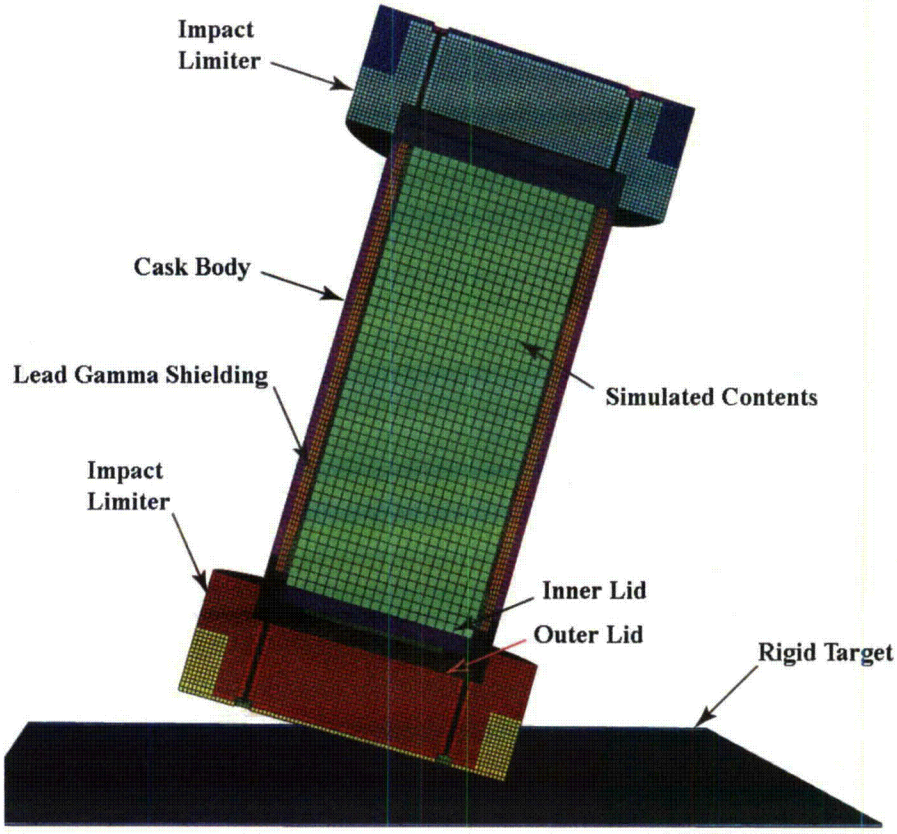
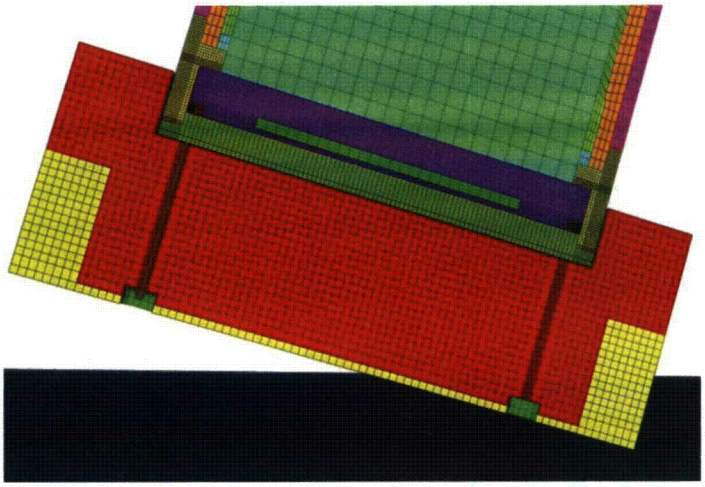
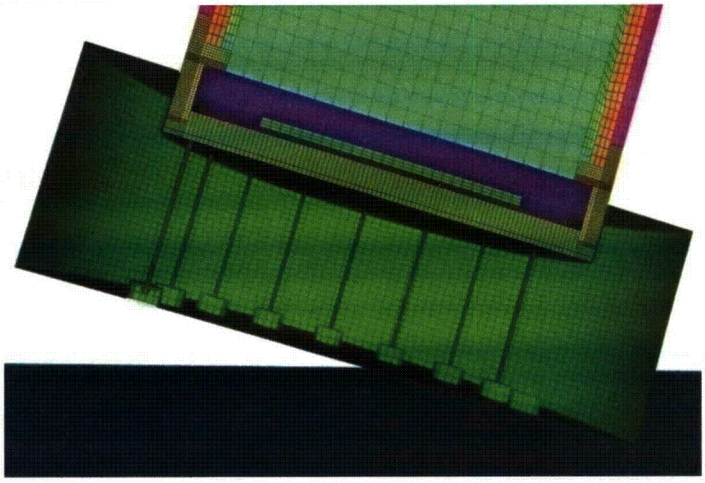


Figure 3-6. Finite element mesh of the Rail-Lead cask.



Impact limiter showing the two different types of wood. The yellow is balsa and the red is redwood.



Impact limiter with the wood removed to reveal the inner attachment bolts

Figure 3-7. Details of the finite element mesh for the impact limiters of the Rail-Lead cask.

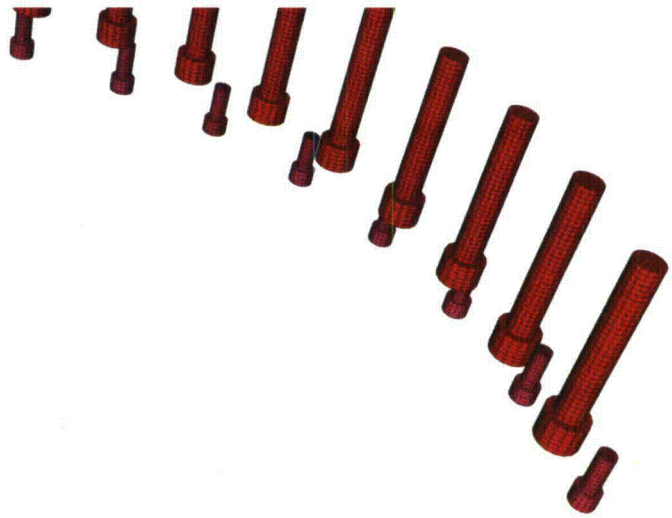


Figure 3-8. Finite element mesh of the Rail-Lead closure bolts for both the inner and outer lids. The longer bolts are for the inner lid and the shorter ones for the outer lid.

Analysis Results

For the 48 kph impact analyses (the impact velocity from the regulatory hypothetical impact accident) the impact limiter absorbed almost all of the kinetic energy of the cask, and there was no damage to the cask body. The response of the Rail-Lead cask is more complicated than that of the Rail-Steel cask. For the end orientation, as the impact velocity increases, there is initially additional damage to the impact limiter because it is absorbing more kinetic energy (this shows the margin of safety in the impact limiter design). At 97 kph there is no significant damage to the cask body or canister. At an impact speed of 145 kph, damage to the cask and canister appears to begin. The impact limiter has absorbed all the kinetic energy it can and any additional kinetic energy is absorbed by plastic deformation in the cask body. At this speed there is significant slumping of the lead gamma shielding material, resulting in a loss of shielding near the end of the cask away from the impact point (this is discussed in Chapter 5 and Appendix V). As the impact velocity is increased to 193 kph, the lead slump becomes more pronounced and there is enough plasticity in the lids and closure bolts to result in a loss of sealing capability. For the directly loaded cask (without a welded multi-purpose canister) there could be some loss of radioactive contents if the cask has metallic seals but not for the case with elastomeric seals. A more detailed discussion of leakage is provided later in this section. Figure 3-9 shows the deformed shape of the Rail-Lead cask following the 193-kph impact in the end-on orientation. The amount of lead slump from this impact is 35.5 cm, and the area without lead shielding is visible in Figure 3-9. Table 3-1 gives the amount of lead slump in each of the analysis cases.

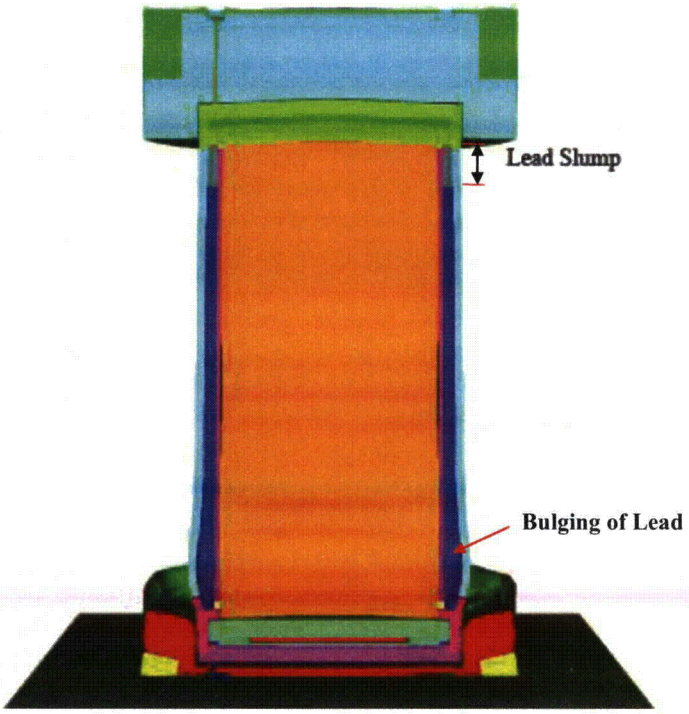


Figure 3-9. Deformed shape of the Rail-Lead cask following the 193-kph impact onto an unyielding target in the end-on orientation.

Table 3-1. Maximum lead slump for the Rail-Lead cask from each analysis case*

Speed (kph)	Max. Slump End (cm)	Max. Slump Corner (cm)	Max. Slump Side (cm)
48	0.64	0.17	0.01
97	1.83	2.51	0.14
145	8.32	11.45	2.09
193	35.55	31.05	1.55

*The measurement locations for each impact orientation are given in Appendix III.

For the corner impacts at 97 and 145 kph, there is some damage to the cask body, in addition to deformation of the impact limiter, that results in lead slump and closure bolt deformation. The amount of deformation to the closure in these two cases is not sufficient to cause a leak if the cask is sealed with elastomeric o-rings, but is enough to cause a leak if the cask is sealed with

metallic o-rings. For a corner impact at 193 kph there is more significant deformation to the cask, more lead slump, and a larger gap between the lid and the cask body. Figure 3-10 shows the deformed shape of the cask for this impact analysis. The deformation in the seal region is sufficient to cause a leak if the cask has metallic o-rings but not if it has elastomeric o-rings. The maximum amount of lead slump is 31 cm.

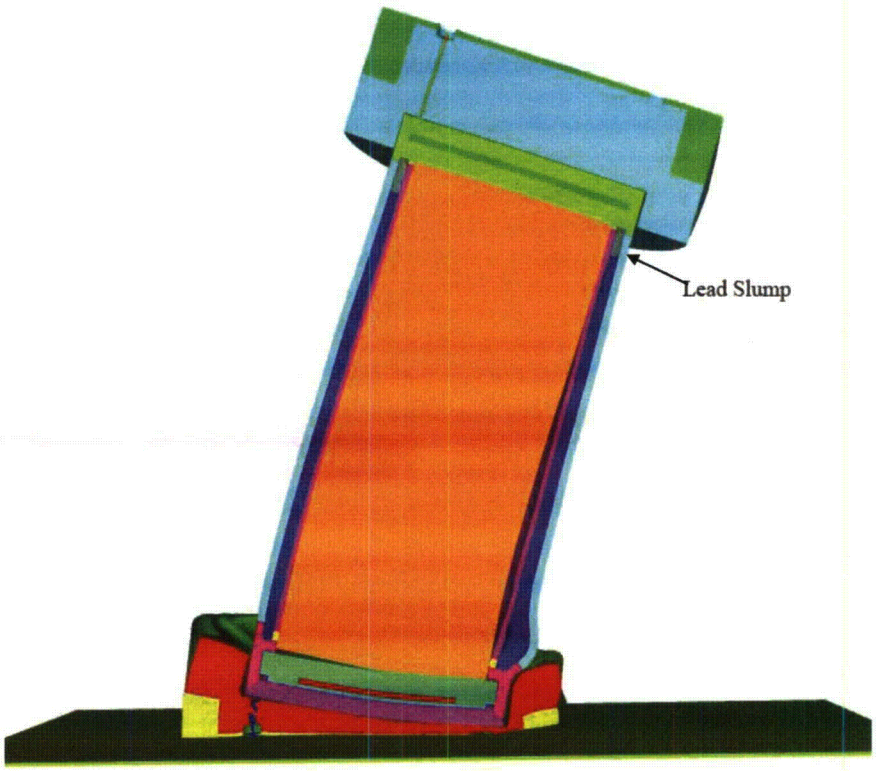


Figure 3-10. Deformed shape of the Rail-Lead following the 193 kph impact onto an unyielding target in the corner orientation.

In the side impact as the impact velocity increases from 48 kph to 97 kph, the impact limiter ceases to absorb additional energy and there is permanent deformation of the cask and closure bolts. The resulting gap in between the lids and the cask body is sufficient to allow leakage if there is a metallic seal, but not enough to leak if there is an elastomeric seal. This calculation of gap between the cask body and lid is conservative because the clamping force applied by bolt preload was neglected in the analysis (the clamping force acts to keep the lid and cask body together). When the impact speed is increased to 145 kph, the amount of damage to the cask increases significantly. In this case, many of the bolts from both the inner and outer lid fail in shear and there is a gap between each of the lids and the cask. This gap is sufficient to allow

leakage if the cask is sealed with either elastomeric or metallic o-rings. Figure 3-11 shows the deformed shape of the cask following this impact. The response of the cask to the 193 kph impact is similar to that from the 145 kph impact, except that the gaps between the lids and the cask are larger. Deformed shapes for all of the analysis cases are shown in Appendix III.

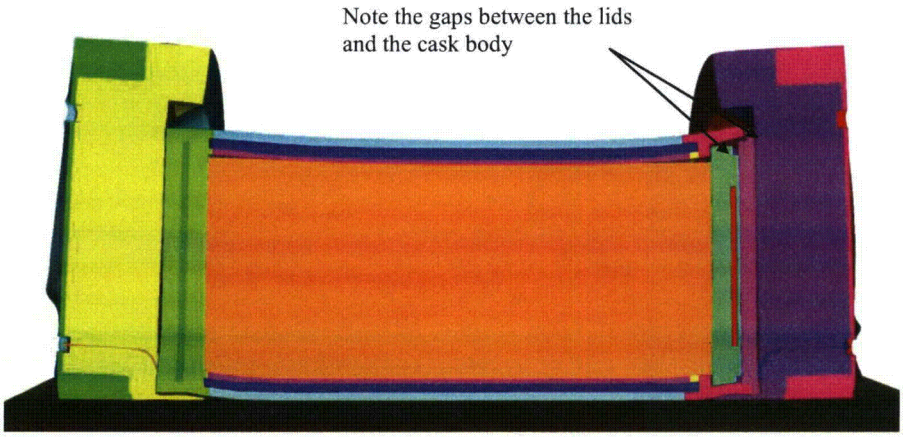


Figure 3-11. Deformed shape of the Rail-Lead cask following the 145 kph impact onto an unyielding target in the side orientation.

Leak Area

The Certificate of Compliance for the Rail-Lead cask allows transportation of spent fuel in three different configurations. The analyses conducted for this study were all for the direct-loaded fuel case, but the results can be applied to the case with an internal canister. The impact limiter and cask body are the same for that case. The addition of the internal canister adds strength and stiffness to the cask in the closure region (the canister has a 203-mm thick lid) that will inhibit the rotation of the cask wall and reduce any gaps between the closure lids and the cask. Figure 3-12 shows the deformation of the closure region for the 193 kph end impact. Gaps for the outer lid were measured as the shortest distance from Node A to the surface opposite it and gaps for the inner lid were measured as the shortest distance from Node B to the surface opposite it. None of the analyses show sufficient deformation into the interior volume of the cask to cause a failure of the internal welded canister. So for this cask, like the Rail-Steel cask, if the spent fuel is transported in an inner welded canister, there would be no release from any of the impacts.

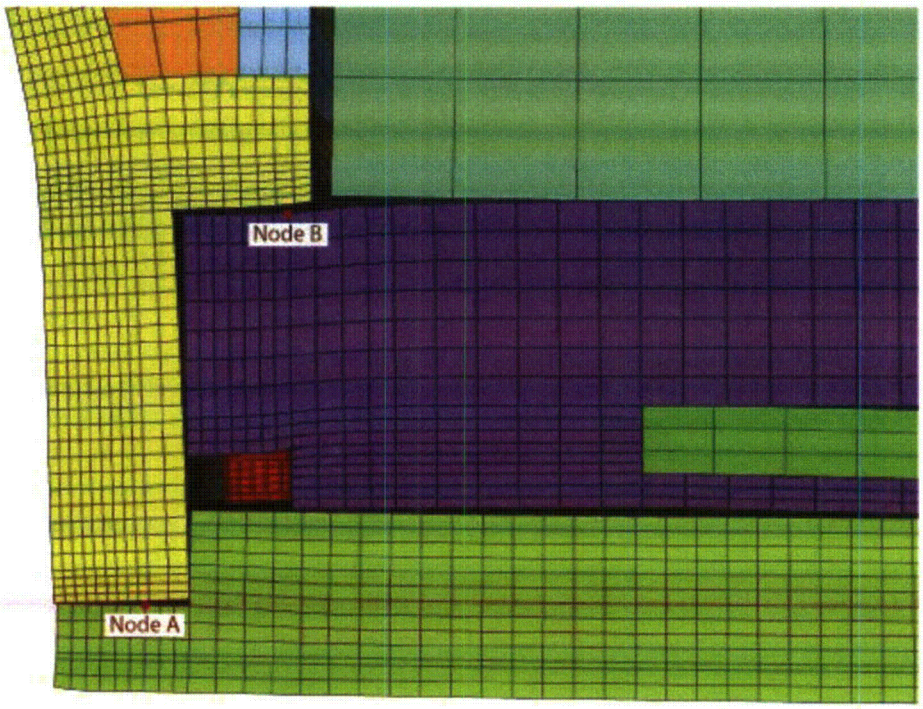


Figure 3-12. Measurement of closure gaps.

In the cases without an inner canister, the cask can be used for dry spent-fuel storage before shipment or to transport fuel that is removed from pool storage and immediately shipped. In the first of these two cases, metallic o-rings provide the seal between each of the lids and the cask body. This type of seal is less tolerant to movement between the lids and the cask, and a closure opening greater than 0.25 mm will cause a leak. If the cask is used for direct shipment of spent fuel, elastomeric o-rings provide the seal between each of the lids and the cask body. While no tests of the effect of gap on leak rates for the lids of this cask have been performed, it is assumed that this type of seal can withstand closure openings of 2.5 mm without leaking (Sprung et al., 2000). Table 3-2 gives the calculated axial gap in each analysis and the corresponding leak area for both metallic and elastomeric seals. The leak areas are calculated for the lid with the smaller gap because in order for there to be any leakage from the cask, both lids must leak.

Table 3-2. Available areas for leakage from the Rail-Lead cask

Orientation	Speed (kph)	Location	Lid Gap (mm)	Seal Type	Hole Size (mm^2)
End	48	Inner	0.226	Metal**	none
		Outer	0	Elastomer	none
	97	Inner	0.056	Metal	none
		Outer	0.003	Elastomer	none
	145	Inner	2.311	Metal	none
		Outer	0.047	Elastomer	none
	193	Inner	5.588	Metal	8796
		Outer	1.829	Elastomer	none
Corner	48	Inner	0.094	Metal	none
		Outer	0.089	Elastomer	none
	97	Inner	0.559	Metal	65
		Outer	0.381	Elastomer	none
	145	Inner	0.980	Metal	599
		Outer	1.448	Elastomer	none
	193	Inner	2.464	Metal	1716
		Outer	1.803	Elastomer	none
Side	48	Inner	0.245	Metal	none
		Outer	0.191	Elastomer	none
	97	Inner	0.914	Metal	799
		Outer	1.600	Elastomer	none
	145	Inner	8*	Metal	>10000
		Outer	25*	Elastomer	>10000
	193	Inner	15*	Metal	>10000
		Outer	50*	Elastomer	>10000

* Estimated. The method used to calculate the gaps for the other cases is explained in Appendix III. For these cases there was bolt failure and the gap was too large to measure using the standard method, but the resultant leak area is sufficiently large that any change to it would not change the cask release fraction.

**The metal seal for the Rail-Lead cask is installed only when the cask has been used for dry storage prior to transportation. Currently there are none of these casks being used for dry storage and there are no plans for using them in that way in the future.

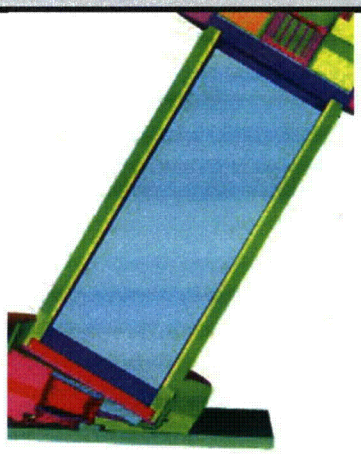
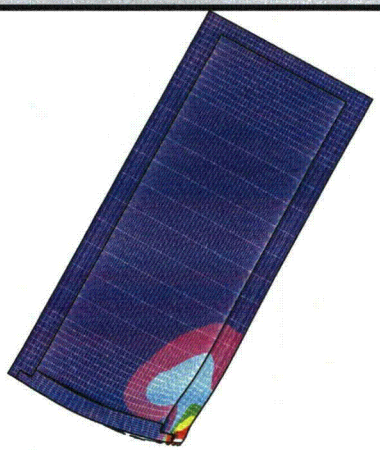
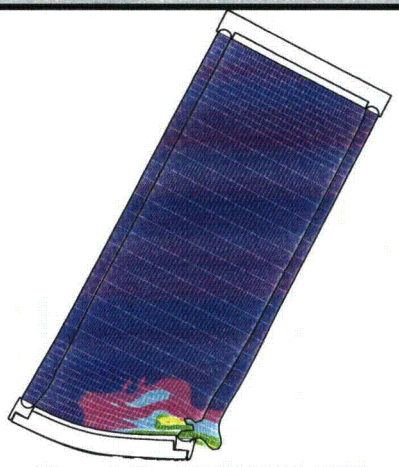
3.2.3 Truck-DU Cask

Detailed finite element analyses of the Truck-DU cask were not performed for this study because the response of the truck casks in NUREG/CR-6672 indicated no gaps between the lid and the cask body at any impact speed. Therefore, the results discussed here are based upon the finite

element analysis of the generic steel-DU-steel truck cask performed for NUREG/CR-6672. In general, the results from the analyses performed for this study have shown that the analyses performed for NUREG/CR-6672 were conservative (see Table 3-3), so the results discussed below are likely to be an overestimate of the damage to the Truck-DU cask from severe impacts. Figure 3-13 shows the deformed shape and plastic strain contours for the generic steel-DU-truck cask from Appendix A of NUREG/CR-6672 (Figures A-15, A-19, and A-22). None of the impacts caused strains that are great enough to fail the cask wall, and in all cases the deformation in the closure region was insufficient to cause seal failure. Figure 3-13. Deformed shapes and plastic strains in the generic steel-DU-steel truck cask from NUREG/CR-6672 (impact limiter removed) following 193 kph impacts in the (clockwise from top left) end-on, CG-over-corner, and side-on orientation.

Table 3-4 (extracted from Table 5.6 of NUREG/CR-6672) provides the deformation in the seal region for each case. For all of these cases there would be no release of radioactive contents.

Table 3-3. Comparison of analyses between this study and NUREG/CR-6672

Item/Cask	Rail-Steel	6672 Monolithic Steel
Deformed Shape 145 kph		 (Figure A-35 of NUREG/CR-6672)
Failed Bolts	No	Yes
Item/Cask	**Rail-Lead**	**6672 SLS Rail**
Deformed Shape 145 kph		 (Figure A-24 of NUREG/CR-6672)
Gap Size	Inner Lid - 0.980 mm Outer Lid – 1.448 mm	6.096 mm
Failed Bolts	No	Yes

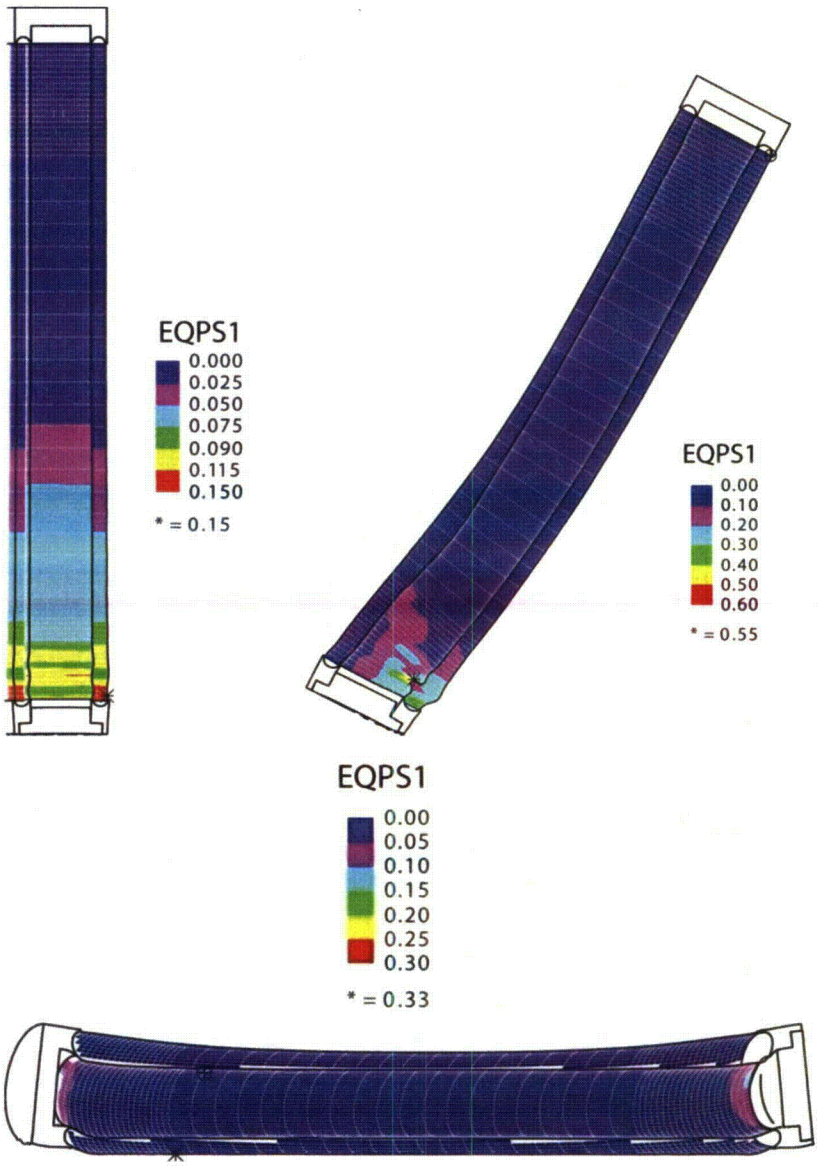


Figure 3-13. Deformed shapes and plastic strains in the generic steel-DU-steel truck cask from NUREG/CR-6672 (impact limiter removed) following 193 kph impacts in the (clockwise from top left) end-on, CG-over-corner, and side-on orientation.

Table 3-4. Deformation of the closure region of the steel-DU-steel truck cask from NUREG/CR-6672, in mm

Cask	Analysis Velocity	Corner Impact		End Impact		Side Impact	
		Opening	Sliding	Opening	Sliding	Opening	Sliding
Steel-DU-Steel Truck	48 kph	0.508	1.778	0.127-0.305	0.025-0.127	0.254	0.508
	97 kph	2.032	1.778	0.254-0.508	0.076-0.152	0.254	0.254
	145 kph	0.508	2.540	-	-	0.254	0.508
	193 kph	0.762	3.810	0.330	0.762	0.102	0.508

3.3 Impacts onto Yielding Targets

All of the analysis results discussed in Section 3.2 were for impacts onto an unyielding, essentially rigid, target. All real impact accidents involve targets that are to some extent yielding. When a cask impacts a real target, the amount of the impact energy that is absorbed by the target and the amount that is absorbed by the cask depend on the relative strength and stiffness of the two objects. For an impact onto a real target to produce the same amount of damage as the impact onto an unyielding target, the force applied to the cask has to be the same. If the target is not capable of sustaining that level of force, it cannot produce the corresponding level of damage in the cask.

For the Rail-Lead cask (the only one of the three investigated in this study that has any release), the peak force associated with each of the impact analyses performed is given in Table 3-5. In this table, the cases that have non-zero hole sizes from Figure 3-13. Deformed shapes and plastic strains in the generic steel-DU-steel truck cask from NUREG/CR-6672 (impact limiter removed) following 193 kph impacts in the (clockwise from top left) end-on, CG-over-corner, and side-on orientation.

Table 3-4 have bold text. It can be seen that in order to produce sufficient damage for the cask to release any material, the yielding target has to be able to apply a force to the cask greater than 146 million Newtons (MN), or 33 million pounds. Very few real targets are capable of applying this amount of force. The target type that is the closest to an unyielding target is hard rock. In this study, hard rock is defined as rock that requires blasting operations to remove. While not all classes of this type of rock are equally strong, all of them are assumed to absorb negligible energy during an impact and are thus treated as rigid.

If the cask hits a flat target, such as the ground, roadway, or railway, it will penetrate into the surface. The greater the contact force between the cask and the ground, the greater the penetration depth. Figure 3-14 shows the relationship between penetration depth and force for the Rail-Lead cask impacting onto hard desert soil. As the cask penetrates the surface, some of its kinetic energy is absorbed by the surface. The amount of energy absorbed by the target is equal to the area underneath the force vs. penetration curve of Figure 3-14. As an example, the end impact at 97 kph onto an unyielding target requires a contact force of 124 MN. A penetration depth of approximately 2.2 meters will cause the soil to exert this amount of force. The soil absorbs 142 million Joules (MJ) of energy in being penetrated this distance. Adding the energy absorbed by the soil to the 41 MJ of energy absorbed by the cask gives a total absorbed energy of

183 MJ. For the cask to have this amount of kinetic energy, it would have to be traveling at 205 kph. Therefore, a 205 kph impact onto hard desert soil causes the same amount of damage as a 97 kph impact onto an unyielding target. A similar calculation can be performed for other impact speeds, orientations, and target types. Table 3-6 provides the resulting equivalent velocities. Similar to Table 3-5, the cases that result in non-zero hole sizes have bold text. Where the calculated velocity is more than 250 kph, the value in the table is listed as ">250." No accident velocities are more than this. The concrete target used is a 23-cm-thick slab on engineered fill. This is typical of many concrete roadways and concrete retaining walls adjacent to highways. Details on the calculation of equivalent velocities are included in Appendix III.

Table 3-5. Peak contact force for the Rail-Lead cask impacts onto an unyielding target (bold numbers are for the cases where there may be seal leaks)

Orientation	Speed (kph)	Accel. (G)	Contact Force (Millions of Pounds))	Contact Force (MN)
End	48	58.5	14.6	65.0
	97	111.6	27.9	123.9
	145	357.6	89.3	397.1
	193	**555.5**	**138.7**	**616.8**
Corner	48	36.8	9.2	40.9
	97	**132.2**	**33.0**	**146.8**
	145	**256.7**	**64.1**	**285.1**
	193	**375.7**	**93.8**	**417.2**
Side	48	76.1	19.0	84.5
	97	**178.1**	**44.5**	**197.8**
	145	**411.3**	**102.7**	**456.7**
	193	**601.1**	**150.0**	**667.4**

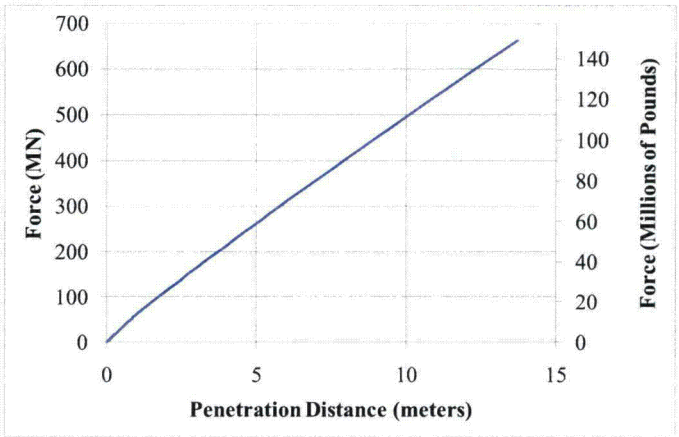


Figure 3-14. Force generated by the Rail-Lead cask penetrating hard desert soil.

Table 3-6. Equivalent velocities for impacts onto various targets with the Rail-Lead cask, kph

Orientation	Rigid (or hard rock)	Soil	Concrete
End	48	102	71
	97	205	136
	145	>250	>250
	193	**>250**	**>250**
Corner	48	73	70
	97	**236**	**161**
	145	**>250**	**>250**
	193	**>250**	**>250**
Side	48	103	79
	97	**246**	**185**
	145	**>250**	**>250**
	193	**>250**	**>250**

3.4 Effect of Impact Angle

The regulatory hypothetical impact accident requires the cask's velocity to be perpendicular to the impact target. All of the analyses were also conducted with this type of impact. During transport, the usual scenario is that the velocity is parallel to the nearby surfaces, and therefore, most accidents that involve impact with surfaces occur at a shallow angle (this is not necessarily true for impacts with structures or other vehicles).

Accident databases do not include impact angle as one of their parameters, so there is no information on the relative frequency of impacts at various angles. Given that vehicles usually travel parallel to the nearby surfaces, for this study a triangular distribution of impact angles was used. Figure 3-15 shows the assumed step-wise distribution of impact angle probabilities. For impacts onto hard targets, which are necessary to damage the cask, the component of the velocity that is parallel to the impact surface has very little effect on the amount of damage to the cask. This requires the accident speed to be higher for a shallow angle impact then a perpendicular one in order to achieve the same amount of damage. Figure 3-16 depicts an example of an impact at a shallow angle and the components of the velocity parallel and perpendicular to the surface. Table 3-7 provides the cumulative probability of exceeding an impact angle range and the accident speeds that are required to have the velocity component in the direction perpendicular to the target.

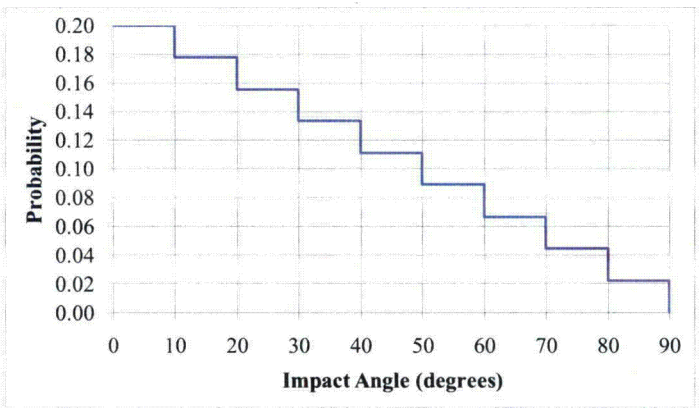


Figure 3-15. Probability distribution for impact angles.

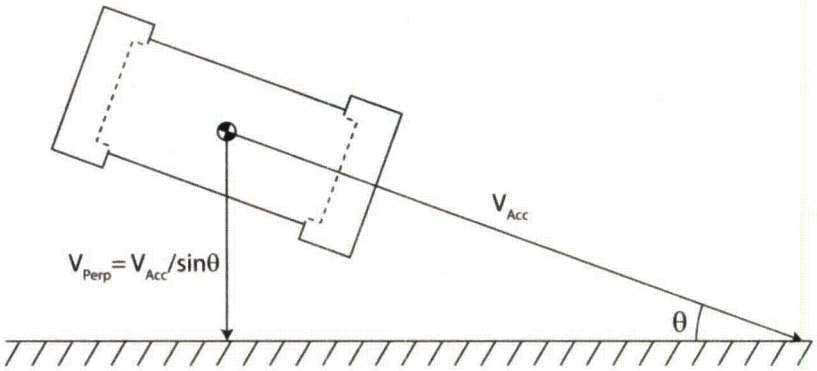


Figure 3-16. Influence of impact angle on effective velocity.

Table 3-7. Accident speeds that result in the same damage as a perpendicular impact, kph

Angle	Prob.	Cum. Prob.	V_{Acc} so V_{Perp} = 48 kph	V_{Acc} so V_{Perp} = 97 kph	V_{Acc} so V_{Perp} = 145 kph	V_{Acc} so V_{Perp} = 193 kph
0 - 10	0.2000	1.0000	278	556	834	1112
10 - 20	0.1778	0.8000	141	282	423	565
20 - 30	0.1556	0.6222	97	193	290	386
30 - 40	0.1333	0.4667	75	150	225	300
40 - 50	0.1111	0.3333	63	126	189	252
50 - 60	0.0889	0.2222	56	111	167	223
60 - 70	0.0667	0.1333	51	103	154	206
70 - 80	0.0444	0.0667	49	98	147	196
80 - 90	0.0222	0.0222	48	97	145	193

3.5 Impacts with Objects

The discussions in the preceding sections all dealt with impacts onto flat surfaces. A large number of impacts deal with surfaces that are not flat. These include impacts into columns and other structures, impacts by other vehicles, and, more rarely, impacts by collapsing structures. These types of impacts were not explicitly included in this study, but recent work by Sandia National Laboratories (NRC, 2003a, Ammerman and Gwinn, 2004, Ammerman et al., 2005) has shown the response of the GA-4 cask to some of these impacts. The result of an impact into a large, semi-circular, rigid column is shown in Figure 3-17 (NRC, 2003a). While this impact led to significant permanent deformation of the cask, the level of strain was not high enough to cause tearing of the containment boundary and there was no permanent deformation in the closure region and no loss of containment.

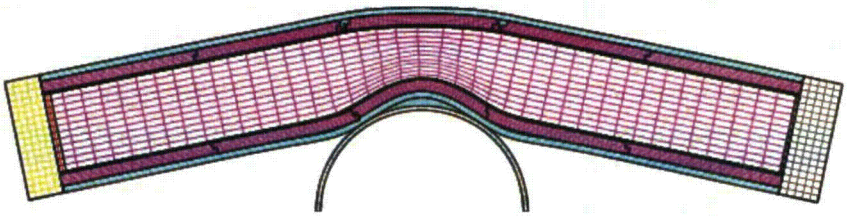


Figure 3-17. Deformations to the GA-4 truck cask after a 96 kph side impact onto a rigid semi-circular column, from (NRC, 2003b).

Another type of accident that could potentially damage a cask is the collision by a railroad locomotive. This is probably the most severe type of collision with another vehicle that is possible. Several different scenarios of this type of collision were investigated by Ammerman et al. (2005). The overall configuration of the general analysis case is shown in Figure 3-18. Most trains involve more locomotives and more trailing cars than were used in this analysis, but additional train mass has little effect on the force acting on the cask. The duration of impact is short and the coupling between the cars is flexible, so the impact is over before the inertia of more cars can have an influence on it. Variations on the general configuration included using the

two most common types of locomotives, having a level crossing (such that the tires of the truck and the wheels of the locomotive are at the same elevation), having a raised crossing where the bottom of the main beams of the trailer are at the same elevation as the top of the tracks, and having a skewed crossing so the impact is at 67° instead of at 90°. For all analyses, the truck was assumed to be stopped. Train velocities of 113 kph and 129 kph were considered.

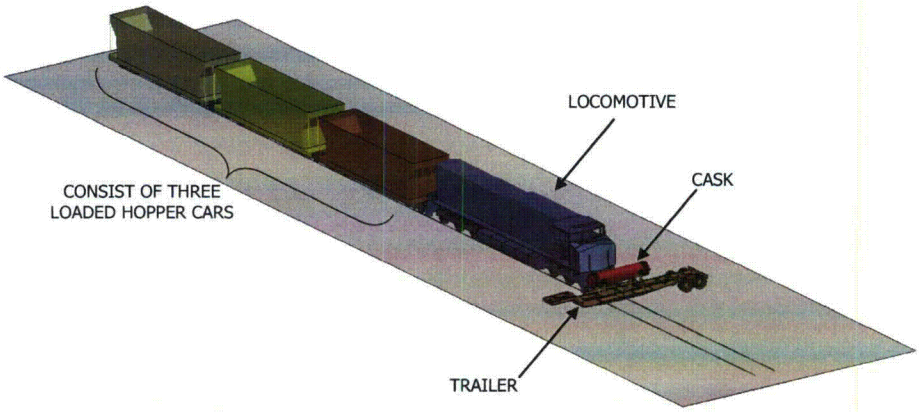


Figure 3-18. Configuration of locomotive impact analysis (Ammerman et al., 2005).

None of the analyses led to deformations that would cause a release of radioactive material from the cask and none of them resulted in cask accelerations that were high enough to fail the fuel rod cladding. Figure 3-19 shows a sequence of the impact. The front of the locomotive is severely damaged and the trailer is totally destroyed, but there is very little deformation of the cask—only minor denting where the collision posts of the locomotive hit the cask.

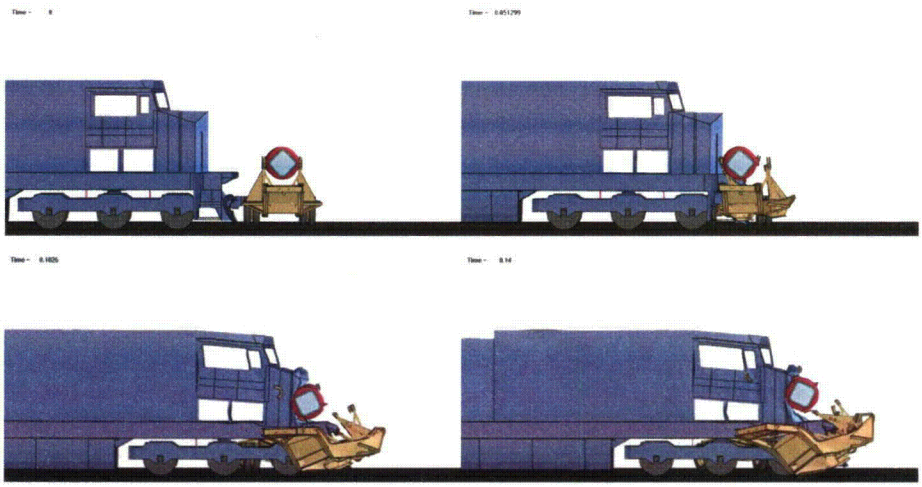


Figure 3-19. Sequential views of a 129 kph impact of a locomotive into a GA-4 truck cask (Ammerman et al., 2005).

A type of accident that occurs less frequently, but also has the potential to damage a cask is the collapse of a bridge onto the cask. This type of accident occurred when an elevated portion of the Nimitz Freeway collapsed during the Loma Prieta earthquake near San Francisco on October 17, 1989. This accident scenario was analyzed to determine if it would cause a release of spent fuel from the GA-4 truck cask (Ammerman and Gwinn, 2004). The analysis assumed the cask was lying directly on the roadway (neglecting the cushioning effect of the trailer and impact limiters) and one of the main beams of the elevated freeway fell and impacted the middle of the cask. The stresses in the cask and damage to the beam are shown in Figure 3-20. As in the other analyses for impacts with objects, there would be no loss of containment from this accident.

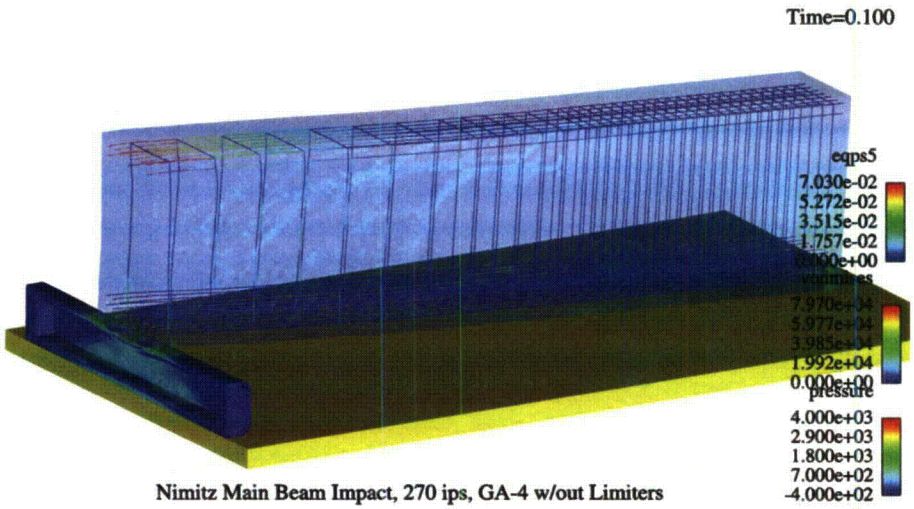


Figure 3-20. Results of a finite element simulation of an elevated freeway collapse onto a GA-4 spent fuel cask (Ammerman and Gwinn, 2004).

3.6 Response of Spent Fuel Assemblies

The finite element analyses of the casks in this study did not include the individual components of the spent fuel assemblies. Instead, the total mass of the fuel and its support structure were combined into an average material. To determine the response of individual components, a detailed model of a spent fuel assembly was developed (Kalan et al., 2005). Figure 3-21 shows this model. In the figure, the fuel rods are shown in yellow, the guide tubes in green, the spacer grids in red, the end plates in light blue and the impact surface in dark blue. The loads associated with a 100 G cask impact in a side orientation were then applied to this detailed model. Kalan et al., 2005, only analyzed side impacts of spent fuel assemblies because the strains associated with buckling of the rods during an end impact are limited by the constrained lateral deformations provided by the basket. The side impact results in forces in each fuel rod at their supports and in many of the fuel rods midway between the supports where they impact onto the rods above or below them. The response of the rod with the highest loads was determined by a detailed finite element model, shown in Figure 3-22. There is slight yielding of the rod at each support location and slightly more yielding where the rods impact each other.

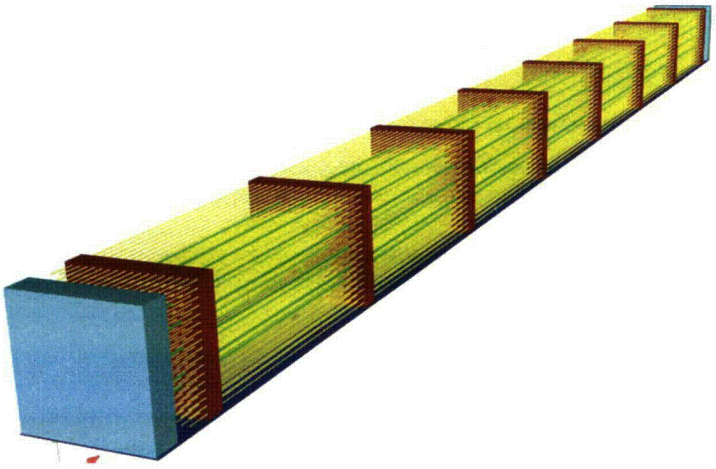


Figure 3-21. Finite element model of a PWR fuel assembly.

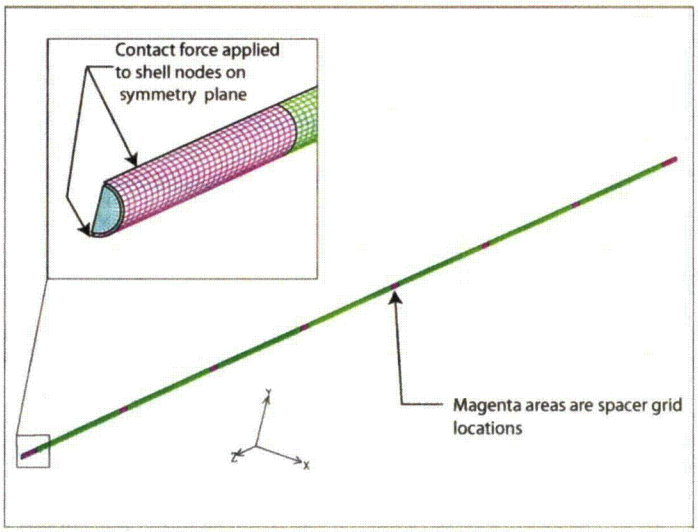


Figure 3-22. Detailed finite element model of a single fuel rod.

Figure 3-23 shows the maximum plastic strain at each location. The largest of these strains is slightly below two percent, which is half the plastic strain capacity of irradiated zircaloy at the

maximum burn-up allowed in the Rail-Lead cask (45,000 MWD/MTU) (Sanders et al., 1992), so fuel rods will not crack. For cladding to fail, the peak acceleration of the cask would have to be above 200 G. The only impacts that are severe enough to crack the rods are those with impact speeds onto an essentially unyielding target of 145 kph or higher. A detailed description of the fuel assembly modeling is included in Appendix III.

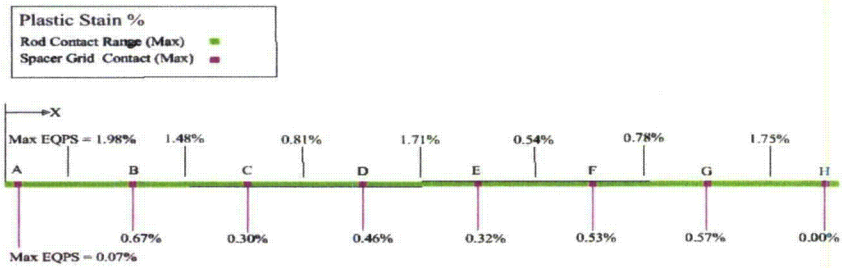


Figure 3-23. Maximum strains in the rod with the highest loads.

3.7 Chapter Summary

The detailed finite element analyses performed for two spent fuel transportation rail casks indicate that casks are very robust structures that are capable of withstanding almost all impact accidents without release of radioactive material. In fact, when spent fuel is transported within an inner welded canister or in a truck cask, there are no impacts that result in release. Even the rail cask without an inner welded canister can withstand impacts that are much more severe than the regulatory impact without releasing any material.

In the worst orientation (side impact) an impact speed onto a rigid target more than 97 kph is required to cause seal failure in a rail cask. (If the cask has an inner welded canister, even this impact will not lead to a release of radioactive material.) A 97 kph side impact onto a rigid target produces a force of about 200 MN (45 million pounds) and is equivalent to a 185 kph impact onto a concrete roadway or abutment or a 246 kph impact onto hard soil. For impacts onto hard rock, which may be able to resist these large forces, impacts at angles less than 30 degrees require a speed more than 193 kph in order to be equivalent.

Assessment of previous analyses performed for spent fuel truck transportation casks, including impacts onto flat rigid targets, impacts into cylindrical rigid targets, impacts by locomotives, and impacts by falling bridge structures, indicate that truck casks will not release their contents in any impact accidents.

In summary, the sequence of events that is needed for there to be the possibility of any release is: a rail transport cask with no welded canister travelling at an impact velocity greater than 97 kph. This cask would need to be impacted in a side orientation and the impact surface would need to be hard rock with an impact angle greater than 30 degrees.

CHAPTER 4

CASK RESPONSE TO FIRE ACCIDENTS

4.1 Introduction

Certified Type B casks are designed to withstand a fully-engulfing fire for thirty minutes while maintaining critical functions, including protecting the public from doses of radiation that exceed regulatory limits. Certification analyses of the hypothetical accident condition (HAC) fire specified in 10 CFR 71.73 generally impose a thermal environment on the cask that is similar to or more severe than the majority of the thermal environments a cask may be exposed to in actual transportation accidents that involve a fire (Fischer et al., 1988). Large open pool fires can burn at temperatures higher than the average temperature of 800°C (1475°F) specified in the regulations for the HAC fire. Actual fire plumes have location- and time-varying temperature distributions that vary from about 600°C (1112°F) to more than 1200°C (2192°F) (Koski, 2000; Lopez et al., 1998). Therefore, an evenly-applied 800°C (1475°F) fire environment used in a certification analysis could be more severe for cask seals and fuel rod response than the exposure to an actual fire.

For this risk study, computer codes capable of modeling both fire behavior and the thermal responses of objects engulfed in those fires in a realistic[14] fashion are used to analyze the response of the Rail-Steel and the Rail-Lead casks to three different fire configurations. These configurations are described in this chapter and the temperature responses of the casks are presented and discussed. An analysis of the thermal performance of the Truck-DU cask when exposed to a severe fire scenario is also presented.

The thermal response of each cask is compared to two characteristic temperature limits. These are the rated seal temperature [350°C (662°F) for elastomeric seals used in the Rail-Lead cask and the Truck-DU and 649°C (1200°F) for the metallic seal used in the Rail-Steel cask] and the fuel rod burst rupture temperature [750°C (1382°F) for all casks (Lorenz, 1980)]. The values selected for these temperature limits are the same as those used in NUREG/CR-6672 for the elastomeric seal and fuel rod burst temperature. The Rail-Steel cask seal temperature limit is obtained from Table 2.1.2 and Table 4.1.1 in the HI-STAR 100 SAR (Holtec International, 2004). Section 7.2.5.2 in NUREG/CR-6672 explains that 350°C (662°F) is a conservative temperature limit for elastomeric seals typically used in the SNF transportation industry. Section 7.2.5.2 of NUREG/CR-6672 also provides the rationale for the use of 750°C (1382°F) as the fuel rod burst rupture temperature. These temperature limits are used in this study to determine if the cask seals or fuel rods would be compromised under any of the accident scenarios analyzed. If seals are compromised coupled with fuel rod breach due to structural insults, or if fuel rods and seals are compromised, it is possible to release radioactive material to the environment and the consequences of that would need to be evaluated.

Comment [MF28]: These words added but not sure they are correct. The previous comment was not fully addressed.

[14] Computational fluid dynamics fire codes are capable of modeling flame behavior, soot formation, flow of hot gasses, and other physical phenomena found in actual fires.

4.2 Description of Accident Scenarios

4.2.1 Pool size

Three fire accident scenarios are analyzed for each rail cask and one for the truck cask. A hydrocarbon fuel pool that conforms to the HAC fire described in 10 CFR 71.73 is used as the basis for each scenario. This regulation specifies a hydrocarbon fuel pool that extends between one and three meters horizontally beyond the external surface of a cask. To ensure the large casks analyzed in this study are fully engulfed by the fire, all fuel pools extended three meters from the sides of the cask (for transportation packages smaller than those analyzed here, a pool fire that extends less than three meters can be sufficient to ensure full engulfment).

Comment [MF29]: Words were re-ordered but meaning is the same. Previous comment, "Not sure I understand this observation. What is it trying to say? Is it saying the assumption is non-conservative?" is still outstanding.

4.2.2 Fire Duration

The duration of the fires postulated for the rail cask analyses is based on the capacity of a large rail tank car. Typical large rail tank cars can carry about 113,562 liters (30,000 gallons) of flammable or combustible liquids (i.e., hydrocarbon based liquids). To estimate the duration of the fires, all the fuel in the tank car is released and assumed to form a pool with the dimensions of a regulatory pool fire for the rail casks that were analyzed. That is, fuel pools that extend horizontally three meters (ten feet) beyond the surfaces of the casks are used in the fire models. Provided that there are relatively small differences between the overall dimensions of the Rail-Steel cask and the Rail-Lead cask, these fuel pools are similar in size and are nominally 14 m × 9 m (46 ft × 29.5 ft). A pool of this size would need to be 0.9 m (3 ft) deep to pool 113,562 liters (30,000 gallons) of liquid fuel, a condition extremely unlikely to be met in any accident scenario. If all of the fuel in such a pool were to ignite and burn (i.e., none of the fuel runs off or soaks into the ground), this pool fire would burn for about 3 hours. This fire duration is estimated using a nominal hydrocarbon fuel recession (evaporation) rate of 5 mm (0.2 in) per minute, typical of large pool fires (SFPE, 2002; Lopez et al., 1998; Quintiere, 1998). Another way this large pool area could burn for up to three hours would be the even less likely case in which liquid fuel flows at exactly the right rate to feed and maintain the pool area for the duration of the fire. Provided that both of these pooling conditions are very difficult to obtain, the fire duration presented here is considered to be conservative. NUREG/CR-7034 corroborates the assertion that it is very difficult for a rail cask to be subjected to long duration, large fires (Adams et al., 2011a). Nevertheless, a three-hour fire that is not moving over time and is capable of engulfing a rail cask over the duration of the fire is conservatively used for the analysis of the two rail casks considered in this study.

In the case of the Truck-DU cask, the fire duration is based on the fuel capacity of a typical petroleum tank truck. About 34,070 liters (9,000 gallons) of gasoline can be transported on the road by one of these tank trucks. Provided that the overall dimensions of the Truck-DU cask are 2.3 m × 6 m (7.5 ft × 19.7 ft), a regulatory pool that extends horizontally 3 meters (10 feet) beyond the outer surface of the cask would be 8.3 m × 12 m (27.2 ft × 39.4 ft). To pool 34,070 liters (9,000 gallons) of gasoline in a pool of this area, the pool would need to be 0.3 m (1 ft) deep, a configuration that is difficult to obtain in an accident scenario and therefore unlikely to occur. Such a pool fire would burn for a little more than an hour. As discussed for the rail cask pool fire, the other possibility of maintaining a fire that can be engulfing and that can burn for that duration is if, for example, gasoline were to flow at the right rate to maintain the necessary

fuel pool conditions. This scenario is also very unlikely. As is the case for rail casks, NUREG/CR-7035 corroborates the assertion that it is very difficult for a truck cask to be subjected to long duration, large fires (Adams et al., 2011b). Nevertheless, one hour is used as the duration of a fire that is not moving over time for the conservative analysis of the Truck-DU cask.

4.2.3 Hypothetical Accident Configurations for the Rail Casks

Three fire accident scenarios that differ from the regulatory HAC fire configuration are analyzed in this study for the rail casks. These are:

1. Cask lying on the ground in the middle of (concentric with) a pool of flammable liquid (such as gasoline) as depicted in Figure 4-1. This scenario represents the case in which the liquid fuel spilled because of an accident flows to the location where the cask comes to rest following the accident and forms a large pool under (and concentric with) the cask.

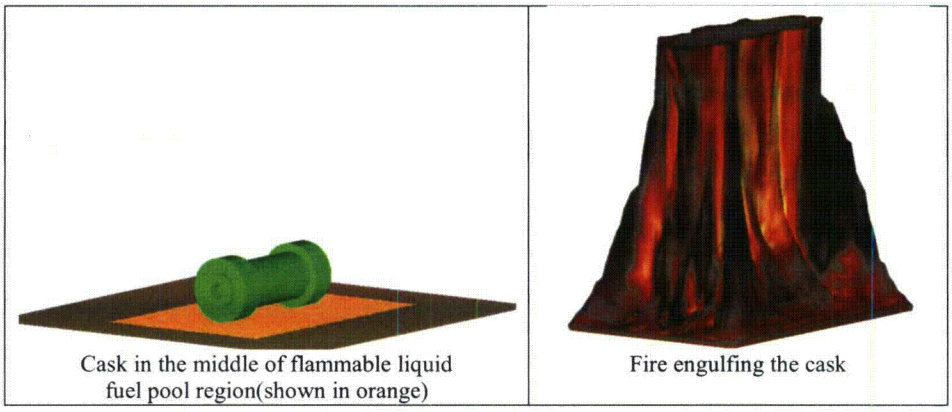


Figure 4-1. Cask lying on ground concentric with fuel pool.

2. Cask lying on the ground 3 meters (10 feet) away from the pool of flammable liquid (with the side of the cask aligned with the long side of the fuel pool) as depicted in Figure 4-2. This scenario represents the hypothetical case in which the fuel pool and the cask are separated by the width of one rail car. This could be the case in an accident in which the rail cars derail in an "accordion" fashion.

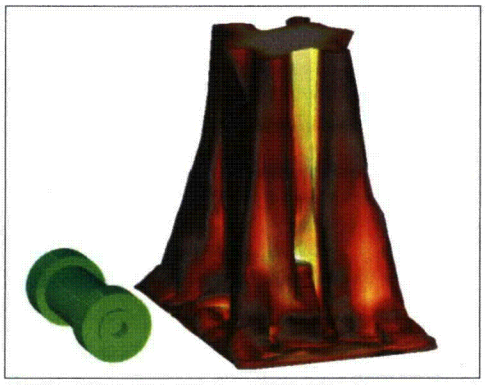


Figure 4-2. Cask lying on ground 3 meters from pool fire.

3. Cask lying on the ground 18 meters (60 feet) from the pool of flammable liquid (with the side of the package aligned with the long side of the fuel pool) as depicted in Figure 4-3. This scenario represents the hypothetical case in which the pool of flammable liquid and the cask are separated by the length of one rail car. This represents an accident in which the separation between a tank car carrying flammable liquid and the railcar carrying the SNF package is maintained (the distance of a buffer rail car, which is required anytime radioactive and flammable/hazardous liquids are transported on the same train[15]) after the accident. For this scenario, the most damaging cask position is assumed. That is, the side of the cask is assumed to face the fire.

[15] 49 CFR 174.85

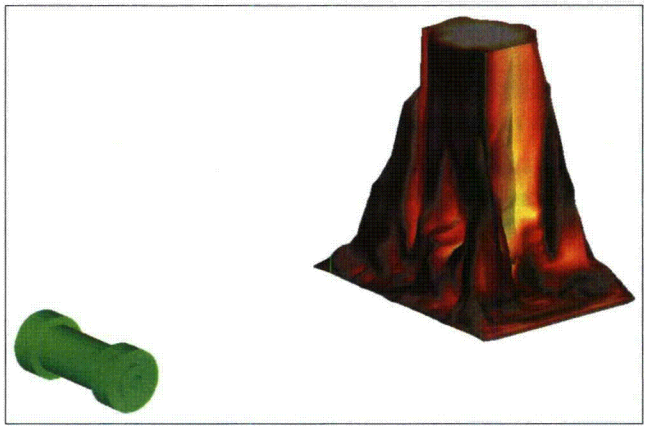
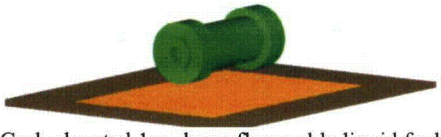


Figure 4-3. Cask lying on ground 18 meters from pool fire.

For each scenario, calm wind conditions (leading to a vertical fire) are assumed. Only the cask and the fuel pool are represented for the analysis. For conservatism, objects that would be present and could shield (protect) the cask from the fire (such as the conveyance or other rail cars) are not included. Decay heat from the cask content was included for all analyses.

Before these accident scenarios were analyzed, two additional 30-minute regulatory HAC fire analyses were performed for each rail cask based on the conditions described in 10 CFR 71.73. In the first analysis, a commercially-available FE heat transfer code is used to apply an 800°C (1475°F) uniform-heating fire condition to the casks. In the second analysis, a benchmarked computational fluid dynamics (CFD) computer model with radiation heat transfer is used.

In this model, each cask is positioned one meter above the fuel pool (as described in 10 CFR 71.73) and a realistic fire fully engulfs the cask as shown in Figure 4-4. The results from FE uniform heating analyses were compared to those in the safety analysis reports for the respective casks to ensure that the cask models used in these analyses are representative. The results from the CFD fire analyses are compared to the results obtained from the uniform-heating FE analyses to demonstrate that the realistic CFD fire does impose conditions that are similar to the uniform heating.



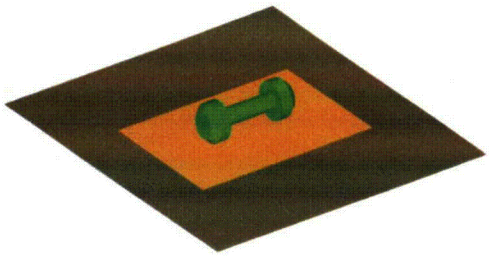
Cask elevated 1m above flammable liquid fuel pool region (shown in orange)

Regulatory fire engulfing the cask

Figure 4-4. Regulatory pool fire configuration.

4.2.4 Hypothetical Accident Configuration for the Truck Cask

In the case of the truck cask, only the most severe hypothetical accident configuration (in which the cask is assumed to be concentric with a flammable fuel pool and is fully engulfed by a fire) is analyzed because none of the temperature limits were reached in this case and the offset fire scenarios would be less severe. This hypothetical accident configuration is presented in Figure 4-5.



Cask in the middle of flammable liquid fuel pool region (shown in orange)

Fire engulfing the cask

Figure 4-5. Truck-DU cask lying on ground concentric with fuel pool.

4.3 Analysis of Fire Scenarios Involving Rail Casks

Advanced computational tools are employed to generate the data necessary for this risk study. For the hypothetical fire accidents, heat transfer from the fire to the cask body was simulated. To accomplish this, two computer codes including all the relevant heat transfer and fire physics are used in a coupled manner. This allows for the simultaneous detailed modeling of realistic external fire environments and heat transfer within the complex geometry of the cask. Brief descriptions of the models are presented in this section. Detailed information about the computer models including material properties, geometry, boundary conditions, and the assumptions used for model generation and subsequent analyses are presented in Appendix IV.

Results from the fire and heat transfer analyses that are performed on the Rail-Steel and the Rail-Lead casks are presented in this section. The scale in the temperature distribution plots of all the Rail-Steel cask analysis results are the same to make comparisons easier. The same is done for the Rail-Lead cask plots.

Results are presented in the following order:

1. 800°C (1475°F) uniform heating exposure for 30 minutes (based on 10 CFR 71.73)
2. 30-minute CFD pool fire using the Container Analysis Fire Environment (CAFE) code (based on 10 CFR 71.73)
3. 3-hour CAFE pool fire (cask on ground concentric with pool)
4. 3-hour CAFE pool fire (cask on ground 3 meters from pool)
5. 3-hour CAFE pool fire (cask on ground 18 meters from pool)

4.3.1 Simulations of the Fires

Fire simulations are performed with the CAFE code (Suo-Anttila et al., 2005). CAFE is a CFD and radiation heat transfer computer code that is capable of modeling fires realistically and is coupled to a commercially-available finite-element analysis computer code to examine the effects of fires on objects. CAFE has been benchmarked against large-scale fire tests specifically designed to obtain data for the calibration of fire codes (del Valle, 2009; del Valle et al., 2007; Are et al., 2005; Lopez et al., 2003). Appendix IV contains details of the benchmark exercises that were performed to ensure that proper input parameters are used to realistically represent the engulfing and offset fires assumed for this study.

4.3.2 Simulations of the Rail Casks

The heat transfer within the Rail-Steel and the Rail-Lead casks is modeled with the computer code MSC PATRAN-Thermal (P-Thermal) (MSC, 2008). This code is commercially available and may be used to solve a wide variety of heat transfer problems. P-Thermal has been coupled with CAFE, allowing for a refined heat transfer calculation within complex objects, such as spent fuel casks, with realistic external fire boundary conditions.

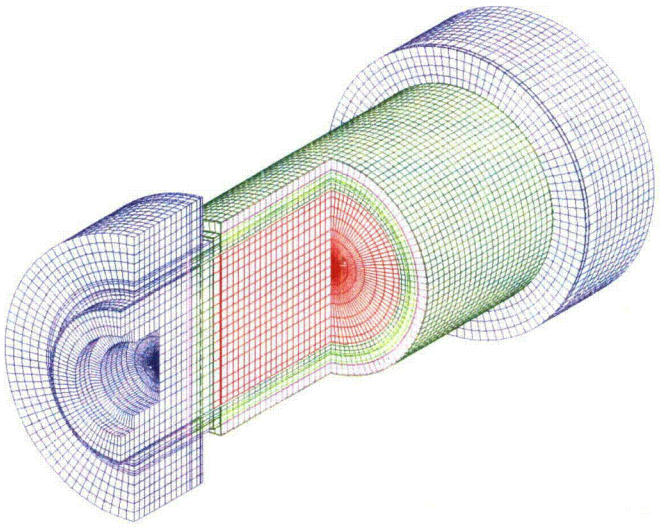
Both the Rail-Steel and the Rail-Lead casks have a polymeric neutron shield that is assumed to melt completely and be replaced by air at its operational temperature limit (see Appendix IV).

The Rail-Lead cask has a lead gamma shield that is allowed to change phase in the analyses upon reaching its melting temperature. Unlike the neutron shield, the thermal energy absorbed in the process of melting the gamma shield is included in the analyses. The effects of the thermal expansion of the lead are not included in the heat transfer calculations but are considered in the estimation of the reduction of the gamma shielding. Gamma shielding in the Rail-Steel cask is provided by the thick multi-layered carbon steel wall. Therefore, melting is not a consideration for this cask under any of the conditions to which it is exposed.

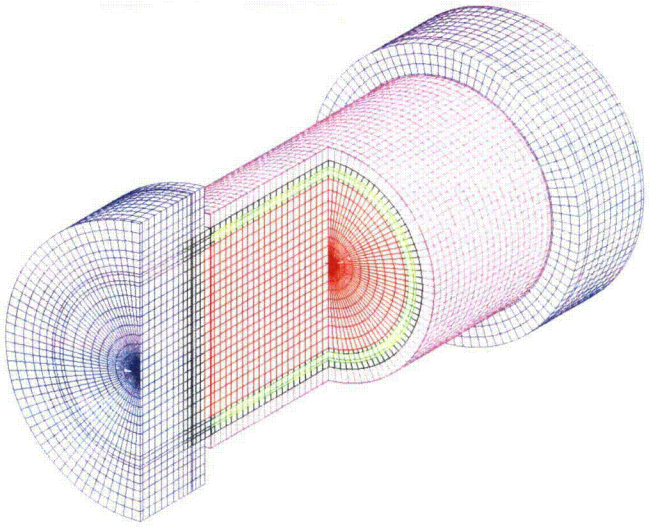
Impact limiters are modeled as undamaged (not deformed). The Rail-Steel cask has aluminum honeycomb impact limiters and the Rail-Lead cask has wood impact limiters. Spaces between components are explicitly modeled in both casks as these could have a significant effect on the thermal response of the cask. The finite element models of the two casks are shown in Figure 4-6. Cask modeling details are presented in Appendix IV.

4.3.3 Simulation of the Spent Nuclear Fuel Region

The interior of the package comprising the fuel basket and the fuel assemblies is not modeled explicitly. A homogenized SNF region, comprised of all materials and geometric features of the fuel basket and fuel assemblies, is represented as a solid cylinder inside the cask. The thermal response of the homogenized SNF region is similar to the overall response for the more detailed model of the fuel basket and assemblies reported in NUREG/CR-6886 (NRC, 2006) and provides enough resolution for the purposes of this study. The details of how the effective properties of the homogenized SNF region are determined and applied to the models are presented in Appendix IV.



Rail-Steel cask



Rail-Lead cask

Figure 4-6. Finite element models (cut views) of the two rail casks analyzed.

4.3.4 Rail-Steel Cask Results

The results for the Rail-Steel cask are presented in the order specified at the beginning of Section 4.3 in Figure 4-7 through Figure 4-21. Figure 4-7 through Figure 4-10 contain the temperature distribution and transient temperature response of key cask regions for the regulatory 800°C uniform heating and the regulatory CAFE fire.

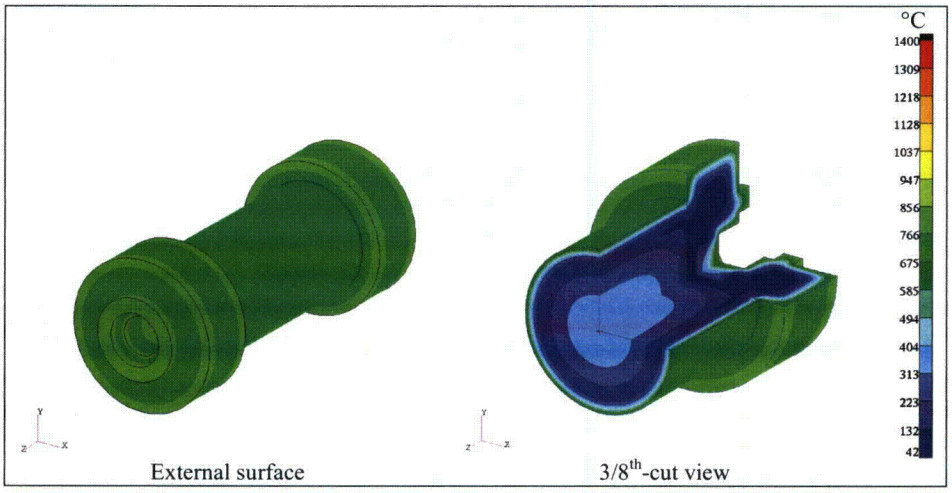


Figure 4-7. Temperature distribution of the Rail-Steel cask at the end of the 30-minute 800°C regulatory uniform heating.

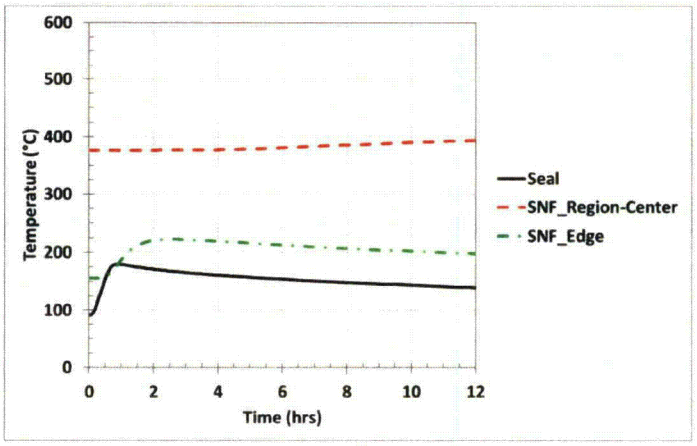


Figure 4-8. Temperature of key cask regions, Rail-Steel cask undergoing regulatory uniform heating.

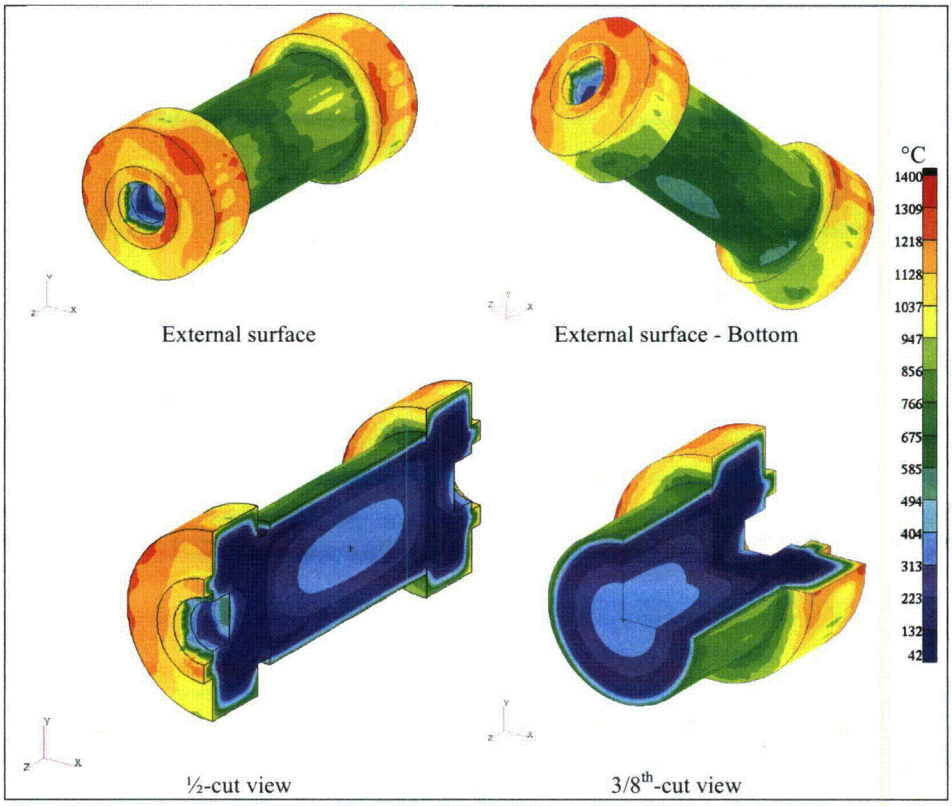


Figure 4-9. Temperature distribution of the Rail-Steel cask at the end of the 30-minute regulatory CAFE fire.

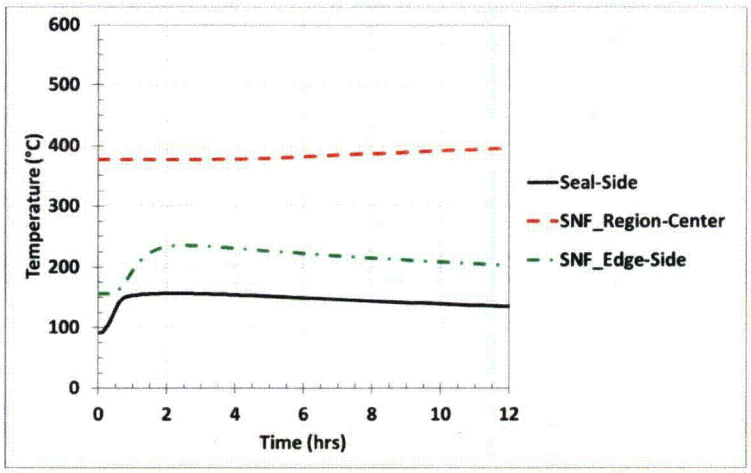


Figure 4-10. Temperature of key cask regions, Rail-Steel cask undergoing REGULATORY CAFE fire.

The uniform external heating, as modeled using FE, produces an even temperature response around the circumference of the cask. However, the realistic uneven fire heating of the exterior, as modeled using CAFE, produces temperatures that vary around the circumference. For comparison, the results obtained from the uniform (FE) regulatory fire simulation are plotted against the hottest regional temperatures obtained from the regulatory CAFE (non-uniform) fire simulation. This thermal response comparison is presented in Figure 4-11. This figure illustrates that the uniform heating thermal environment described in 10 CFR 71.73 heats up the seal region of the Rail-Steel cask more than a real fire may, even though a real fire can impart to the cask a temporary and localized thermal environment that is hotter than 800°C. A real fire applies a time- and space-varying thermal load to an object engulfed by it. In particular, large fires have an internal region where fuel exists in the form of gas, but not enough oxygen is available in that region for that fuel to burn. That region is typically called the "vapor dome." The lack of oxygen in the vapor dome is attributed to poor air entrainment in larger diameter pool fires, where much of the oxygen is consumed in the perimeter of the plume region. Since combustion is inefficient inside the vapor dome, this region stays cooler than the rest of the fire envelope. Thus, the presence of regions that are cooler than 800°C within a real fire makes it possible for fires with peak flame temperatures above 800°C to have an overall effect on internal temperatures of a thermally massive object that is similar to those obtained by applying a simpler heating condition such as the one specified in 10 CFR 71.73.

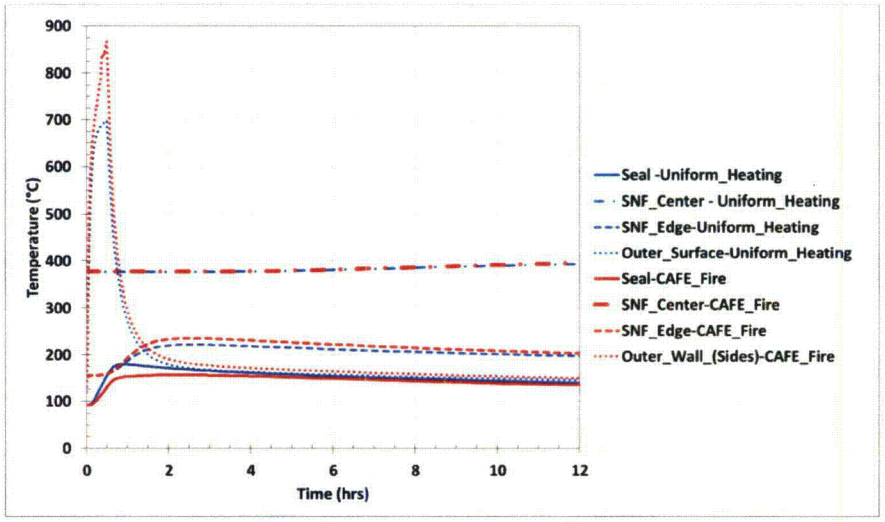


Figure 4-11. Comparison of regulatory fire analysis for Rail-Steel cask: Uniform heating vs. CAFE fire. The "Outer_Wall" CAFE curve is the average of the two "Outer_Surface" CAFE curves for the sides of the cask as presented in Appendix IV, Figure IV-11.

Deleted: Wall

The effects of the vapor dome on the temperature distribution within a fire and the concentration of unburned fuel available in the vapor dome for the CAFE regulatory analysis can be seen in Figure 4-12 and Figure 4-13.

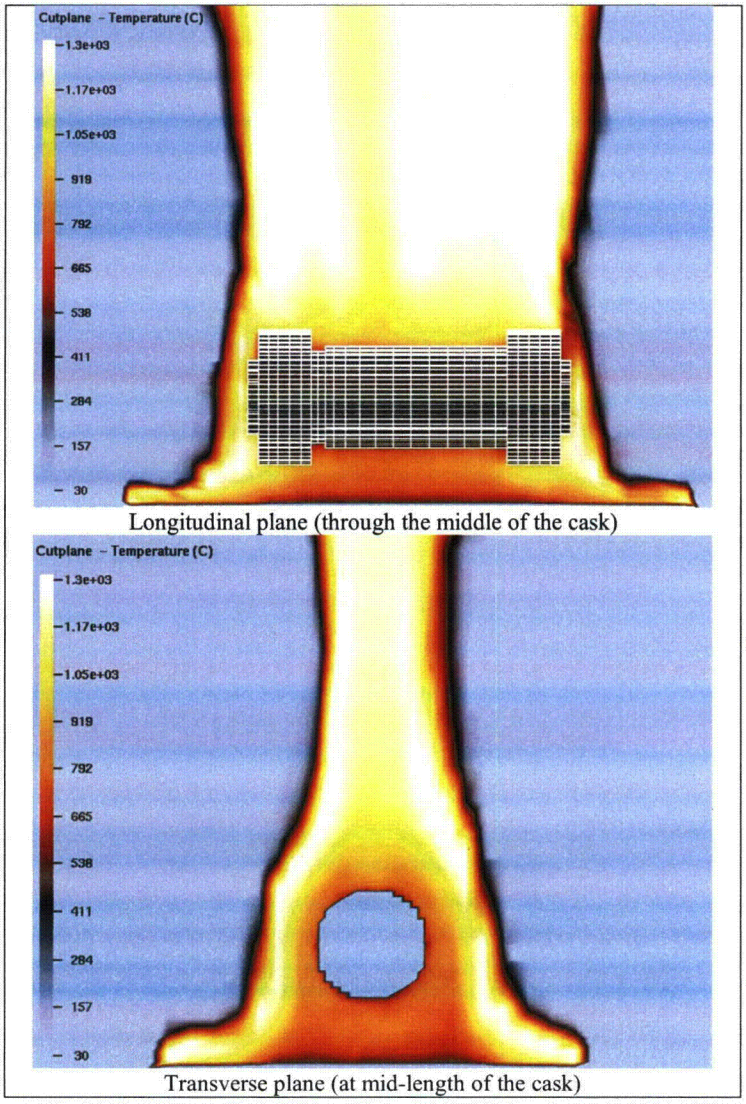


Figure 4-12. Gas temperature plots from the regulatory CAFE fire analysis.

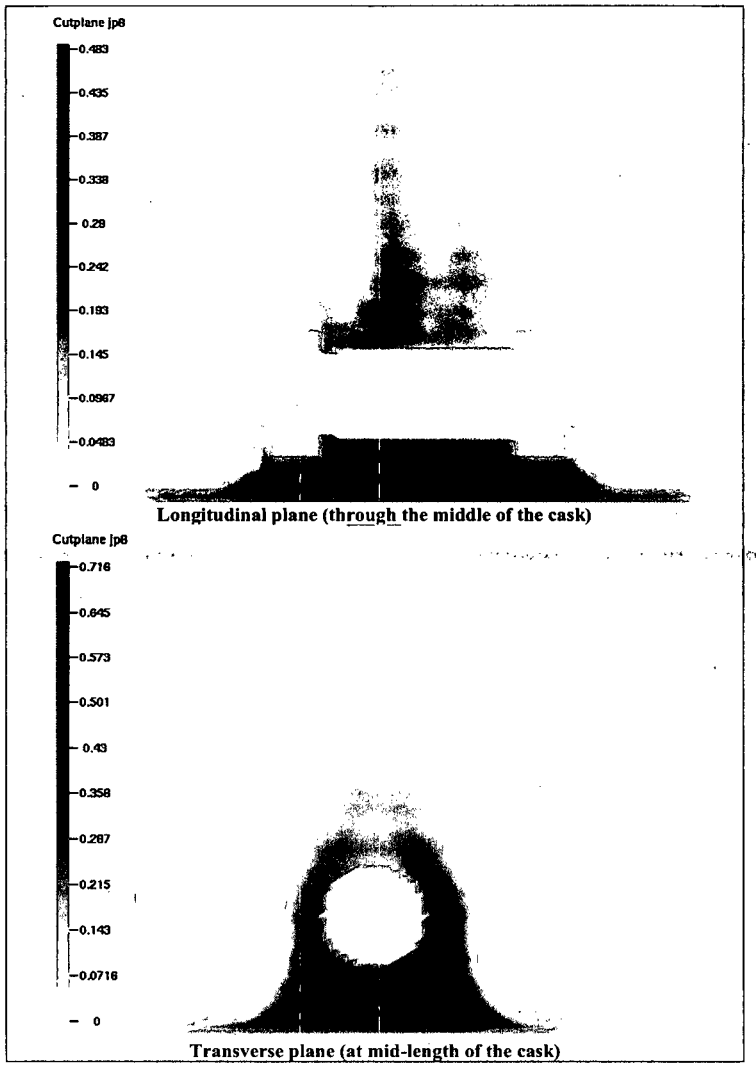


Figure 4-13. Fuel concentration plots from the regulatory CAFE fire analysis.

Note that the plots in Figure 4-12 and Figure 4-13 are snapshots of the distributions at an arbitrary time during the fire simulation. In reality, the fire moves slightly throughout the simulation causing these distributions to vary over time. Nevertheless, these plots show representative distributions for the cask and fire configuration shown.

Additional plots with more information about temperature distributions at different locations in the cask are shown in Appendix IV.

The results from the analysis of the cask lying on the ground and concentric with a pool fire that burns for three hours are presented in Figure 4-14 and Figure 4-15. As in the regulatory configuration, in which the cask is elevated 1 meter above the hydrocarbon fuel pool, the vapor dome had an effect on the temperature distribution of the cask in this case. This is evident by the cooler temperatures observed at the bottom of the cask. In this scenario, even after three hours in the fire, the temperatures at the bottom of the package are cooler than the temperatures observed in the regulatory configuration. However, the top of the cask in this configuration heats up more than the rest of the cask. This differs from what is observed in the regulatory configuration, in which the hotter regions are found on the sides of the cask.

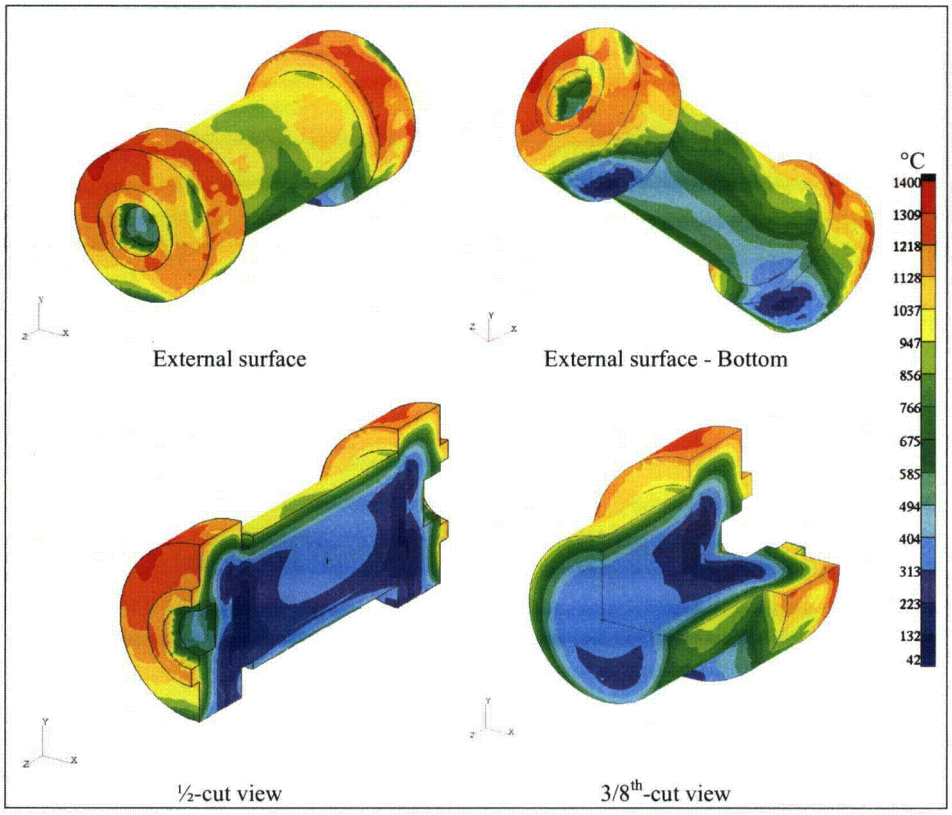


Figure 4-14. Temperature distribution of the Rail-Steel cask at the end of the 3-hour concentric CAFE fire with cask on ground.

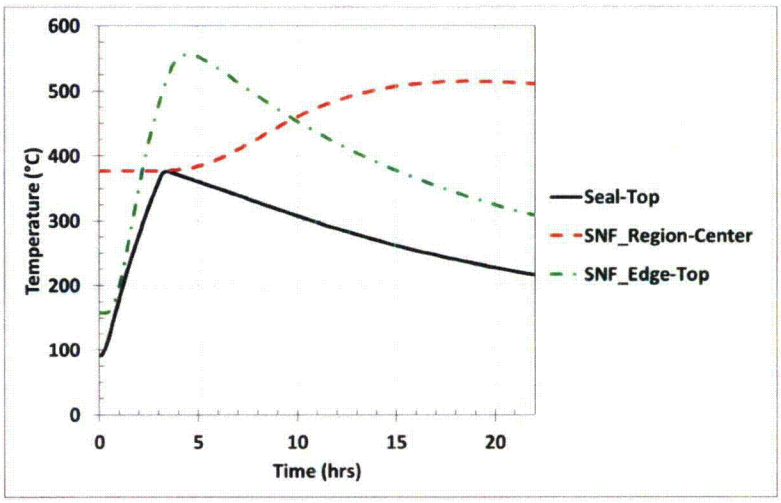


Figure 4-15. Temperature of key cask regions, Rail-Steel cask with cask on ground, concentric fire.

Figure 4-16 and Figure 4-17 are the fire temperature distribution and fuel concentration plots at an arbitrary time during the CAFE fire simulation of this scenario. In this case, the concentration of unburned fuel under the cask is high and therefore the temperature of the fire under the cask is lower than what is observed in the regulatory configuration.

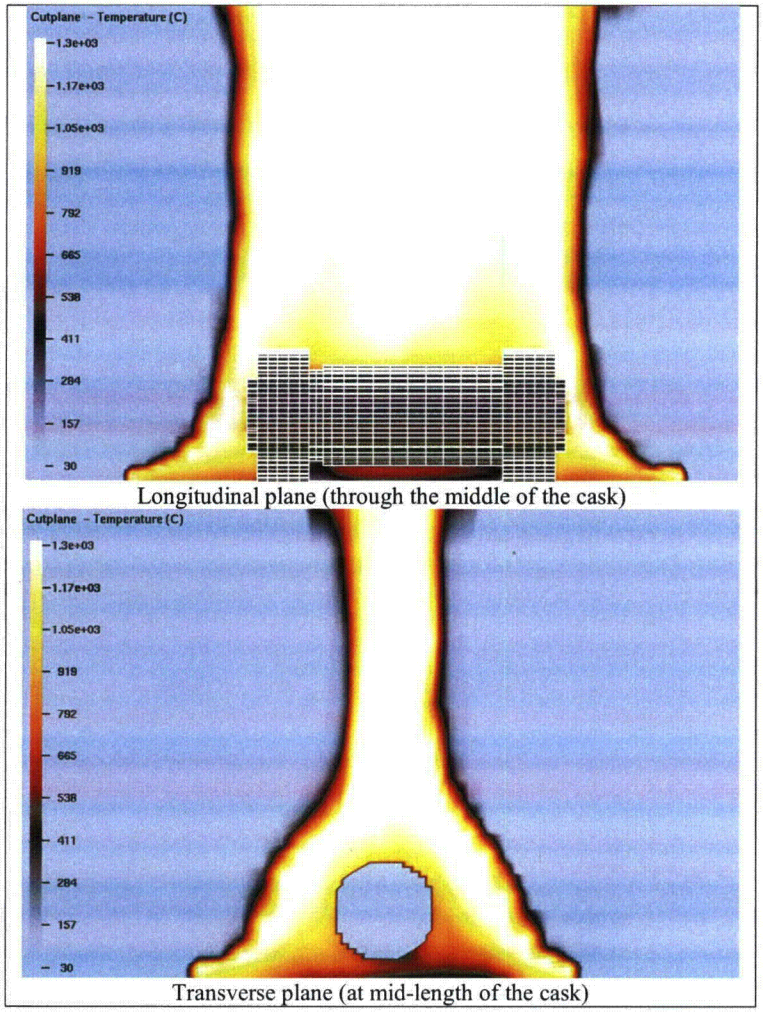


Figure 4-16. Gas temperature plots from the CAFE fire analysis of the cask on ground.

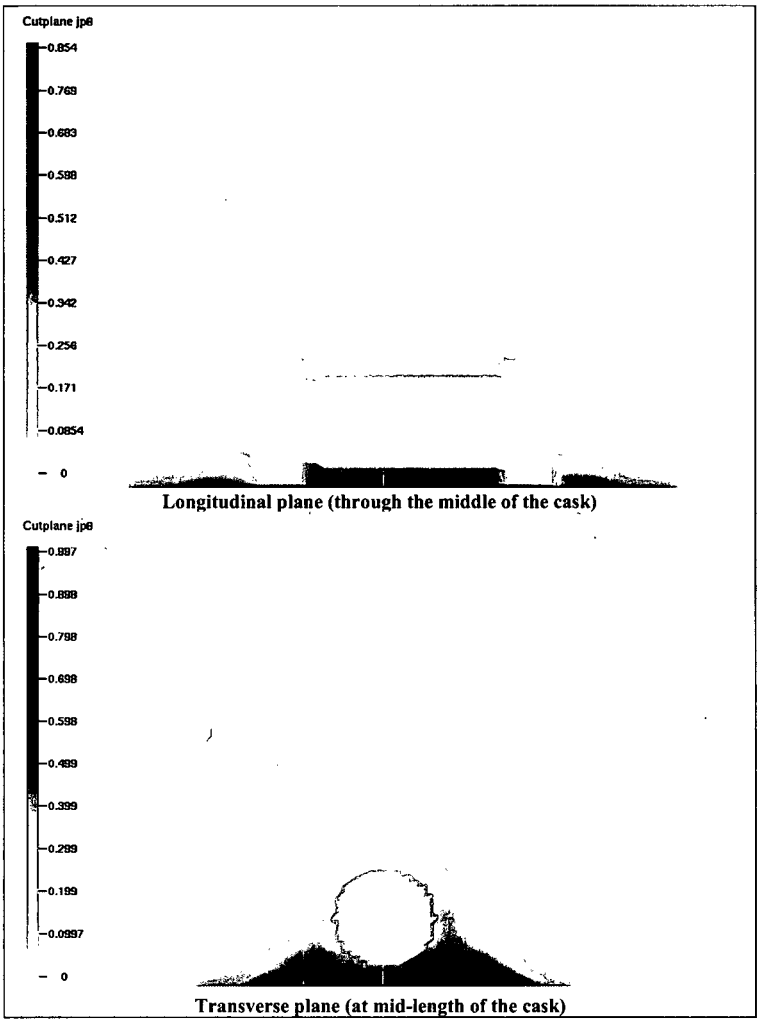


Figure 4-17. Fuel concentration plots from the CAFE fire analysis of the cask on ground.

The results of the offset fire analyses are summarized in Figure 4-18 through Figure 4-21. In the case of the 3-meter offset, the side of the cask facing the fire received heat by thermal radiation. The heat absorbed by the cask during the 3-hour exposure caused the temperature of the cask to rise as depicted in Figure 4-18 and Figure 4-19. Similarly, the 18-meter offset fire caused the cask temperature to rise as illustrated in Figure 4-20 and Figure 4-21. These results show that

offset fires, even as close to the cask as three meters, do not represent a threat to this thermally massive SNF transportation cask. The maximum temperatures observed in the seal and in the SNF region did not reach their temperature limits. Therefore, offset fire scenarios will not cause this package to release radioactive material.

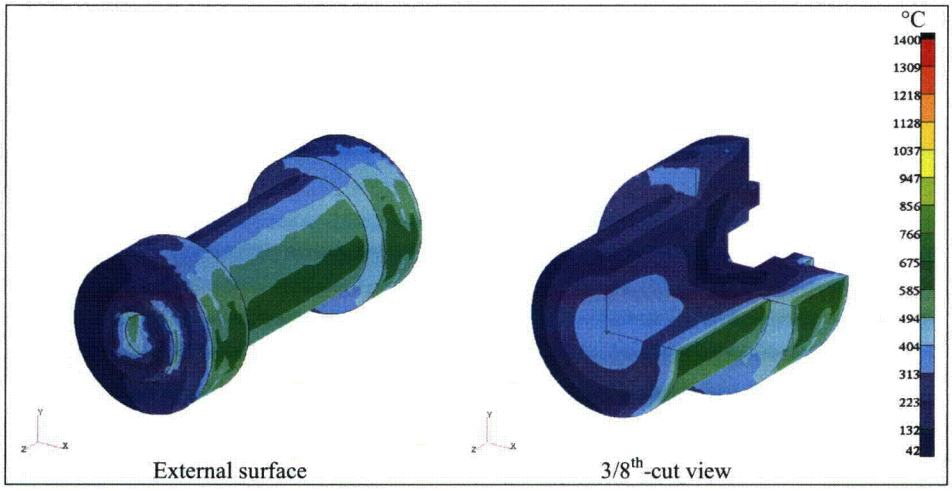


Figure 4-18. Temperature distribution of the Rail-Steel cask at the end of the 3-hour, 3m offset CAFE fire with cask on ground.

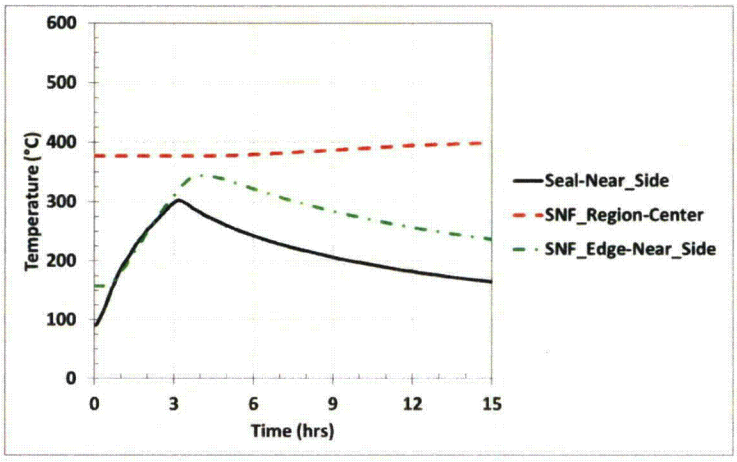


Figure 4-19. Temperature of key cask regions, Rail-Steel cask with Cask on ground, 3m offset fire.

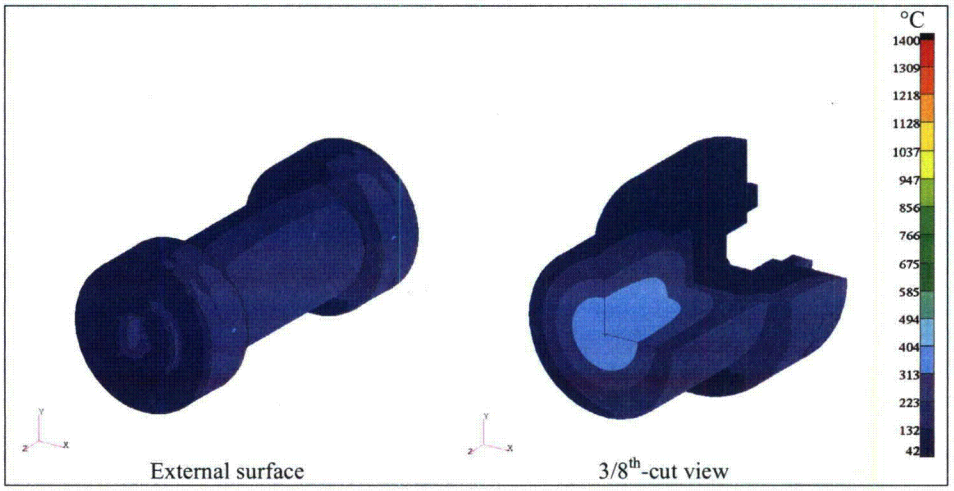


Figure 4-20. Temperature distribution of the Rail-Steel cask at the end of the 3-hour 18m offset CAFE fire with cask on ground.

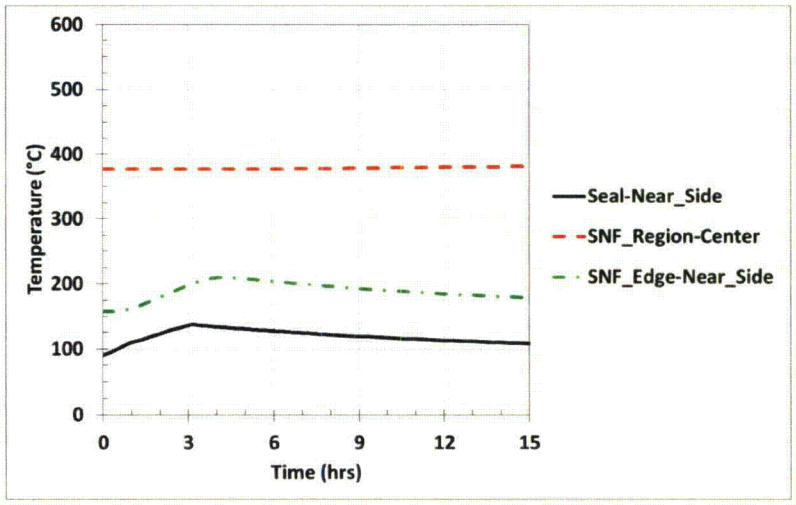


Figure 4-21. Temperature of key cask regions, Rail-Steel cask with cask on ground, 18m offset fire.

Summary of Rail-Steel Cask Analysis Results

The results presented here show that the Rail-Steel cask is capable of protecting the fuel rods from burst rupture and is also capable of maintaining containment when exposed to the severe fire environments that are analyzed as part of this study. That is, while the neutron shield material is conservatively assumed to be absent during the fire accident, the SNF region stays below 750°C (1382°F) and the seal region stayed under 649°C (1200°F) for all the scenarios that are considered. Furthermore, this cask uses a welded canister that will not be compromised under these thermal loads. This cask will not experience loss of gamma shielding because in this cask shielding is provided by the thick multi-layered carbon steel wall, which is not affected in a way that could reduce its ability to provide shielding.

4.3.5 Rail-Lead Cask Results

The thermal response of the Rail-Lead cask to the same fire environments discussed above for the Rail-Steel cask is presented in this section. The 30-minute regulatory fire results are summarized in Figure 4-22 through Figure 4-26.

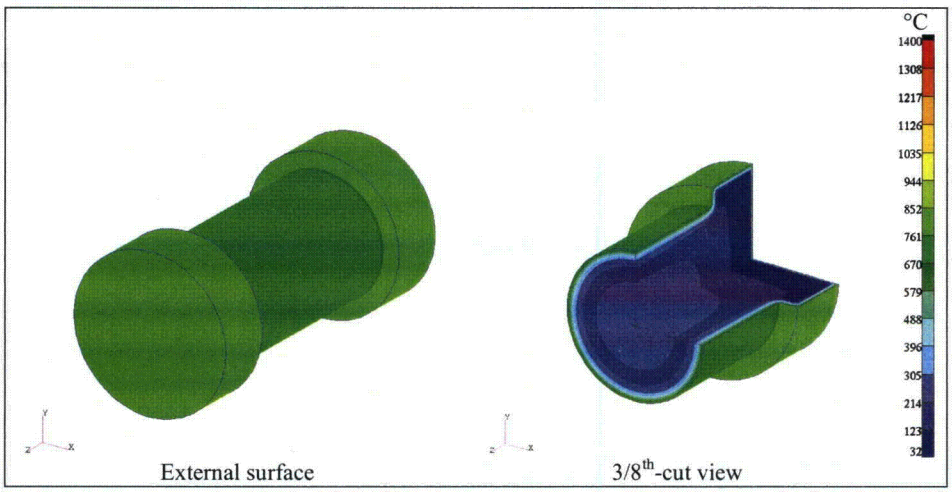


Figure 4-22. Temperature distribution of the Rail-Lead cask at the end of the 30-minute 800°C regulatory uniform heating.

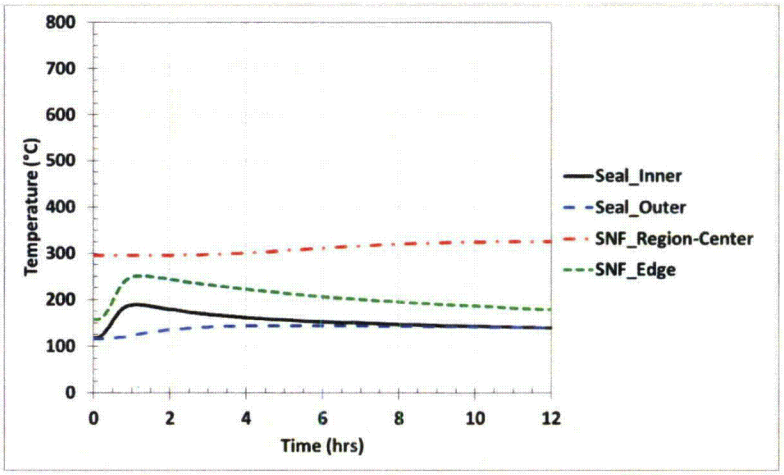


Figure 4-23. Temperature of key cask regions, Rail-Lead cask undergoing regulatory uniform heating.

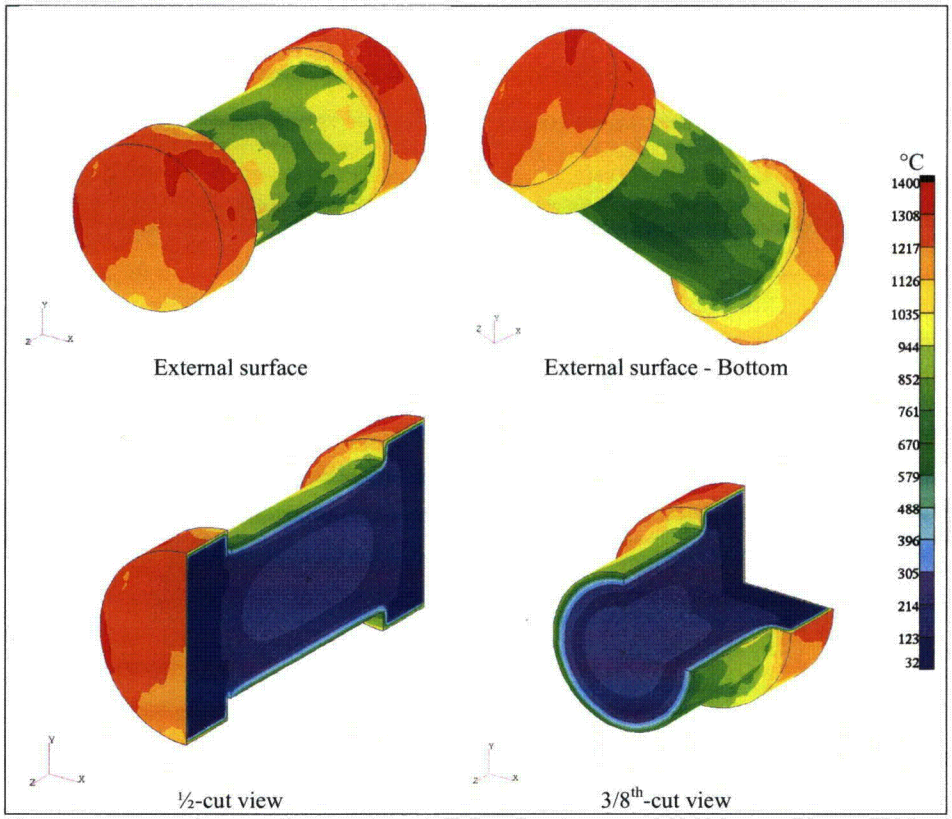


Figure 4-24. Temperature distribution of the Rail-Lead cask at the end of the 30-minute regulatory CAFE fire.

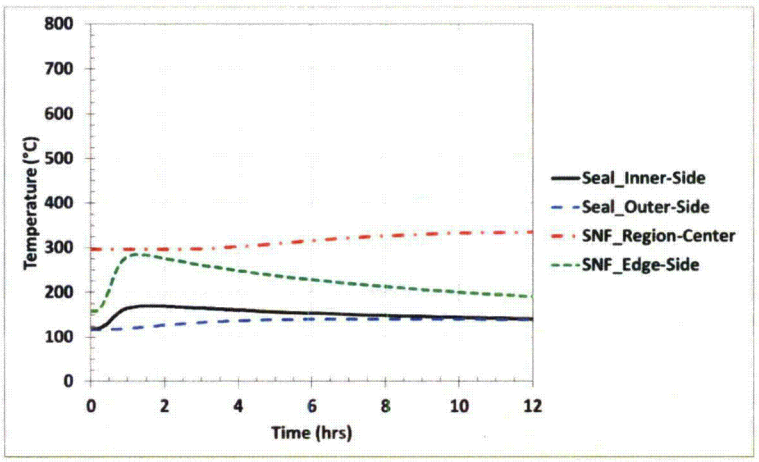


Figure 4-25. Temperature of key cask regions, Rail-Lead cask in regulatory CAFE fire.

The results obtained from the uniform regulatory fire simulation are plotted against the hottest regional temperatures obtained from the CAFE (non-uniform) regulatory fire simulation. This plot is shown in Figure 4-26. As with the Rail-Steel cask, this figure illustrates that the uniform heating thermal environment described in 10 CFR 71.73 heats the seal region of the Rail-Lead cask more than a non-uniform real fire may, even though a real fire may impart to the cask a localized thermal environment that is hotter than 800°C (1472°F).

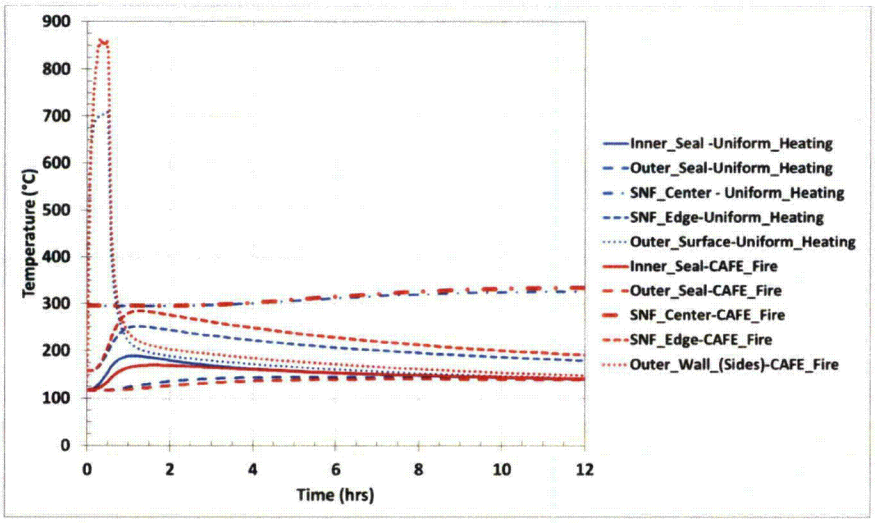


Figure 4-26. Comparison of regulatory fire analysis for Rail-Lead cask: Uniform heating vs. CAFE fire. The "Outer_Wall" CAFE curve is the average of the two "Outer_Surface" CAFE curves for the sides of the cask as presented in Appendix IV, Figure IV-21.

Deleted: Wall

The results of the analyses of the cask lying on the ground heated by the concentric and offset fires are summarized in Figure 4-27 through Figure 4-32. These plots show similar trends to those observed in the Rail-Steel cask for the same configurations.

Two of the scenarios that are analyzed show melting of the lead gamma shield in the Rail-Lead cask. Lead melts at 328°C (622°F) and during that process, it absorbs (stores) heat while maintaining its temperature relatively constant at 328°C. As a result, the heat-up rate of portions of the cask slows down while the lead melts. That is why the curve of the region inward from the gamma shield region (i.e., the edge of the SNF region) in Figure 4-28 and Figure 4-30 show a change in slope at about 328°C. This effect is more clearly seen in the slower heating case shown in Figure 4-30. Once the lead melting process is complete, the cask resumes heating up as before if the external source is still at a higher temperature. Note that a similar effect is observed when the lead solidifies at 328°C during the post fire cooling period. In this case, the cooling rate of portions of the cask slows down while the lead solidifies. This can also be clearly seen in Figure 4-30.

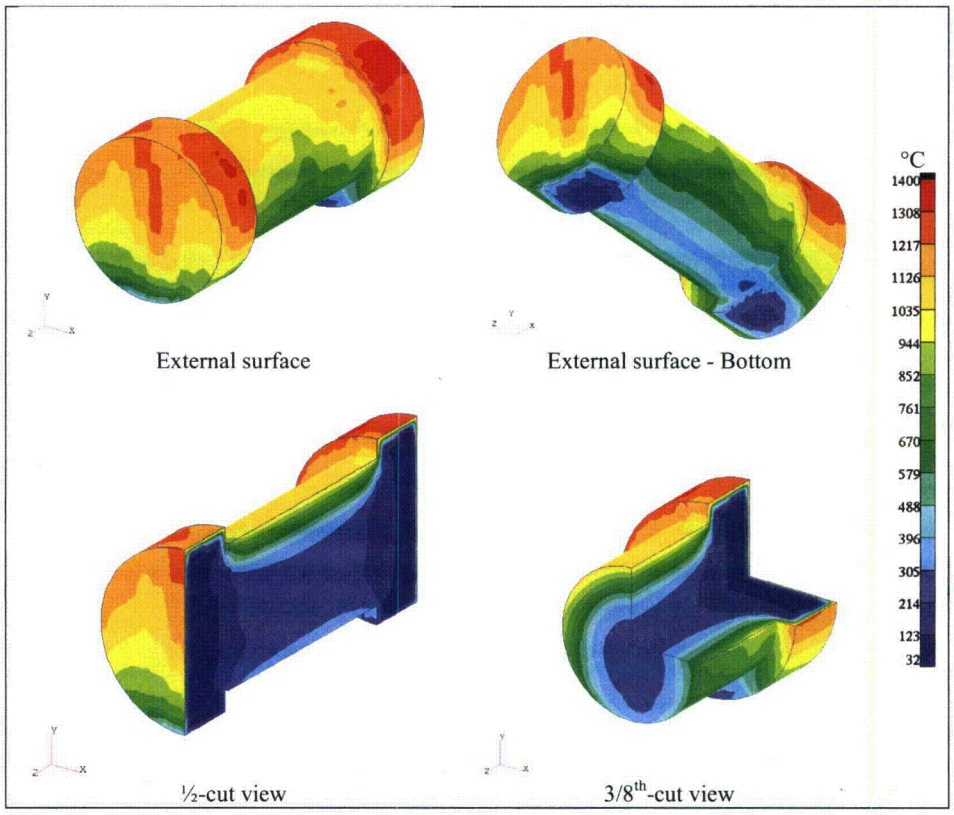


Figure 4-27. Temperature distribution of the Rail-Lead cask at the end of the 3-hour concentric CAFE fire with cask on ground.

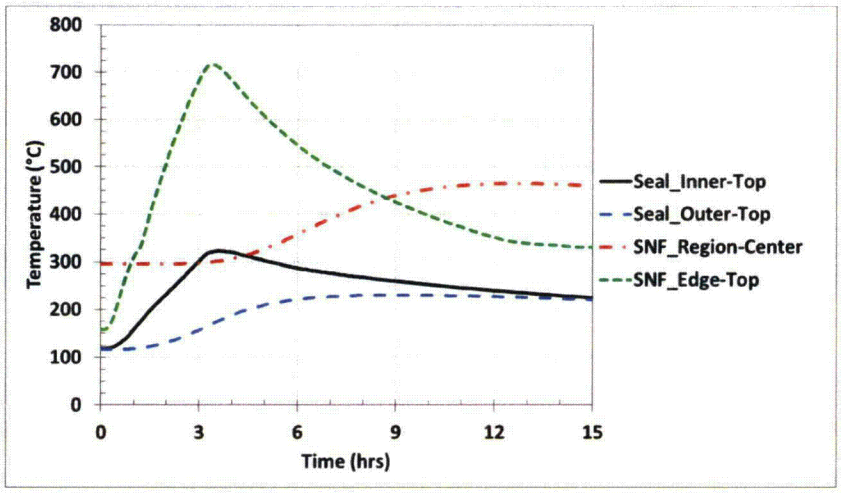


Figure 4-28. Temperature of key cask regions, Rail-Lead cask with cask on ground, concentric fire.

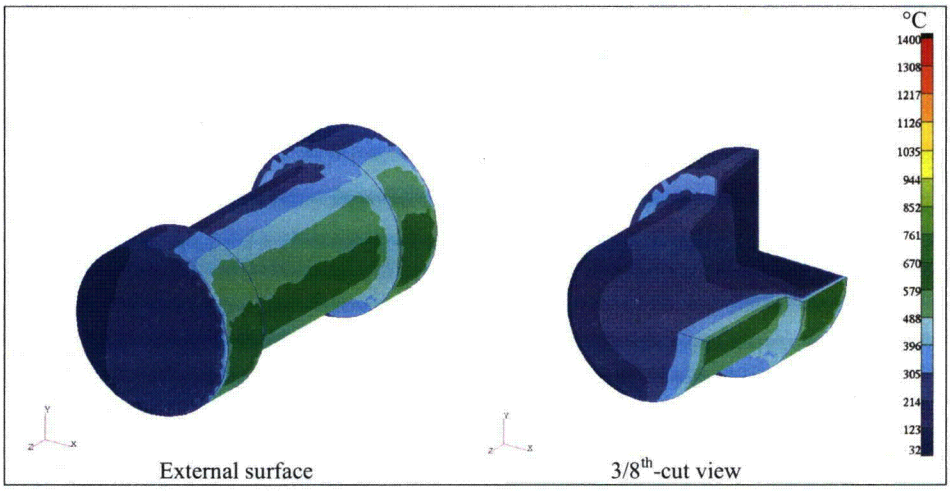


Figure 4-29. Temperature distribution of the Rail-Lead cask at the end of the 3-hour 3m offset CAFE fire with cask on ground.

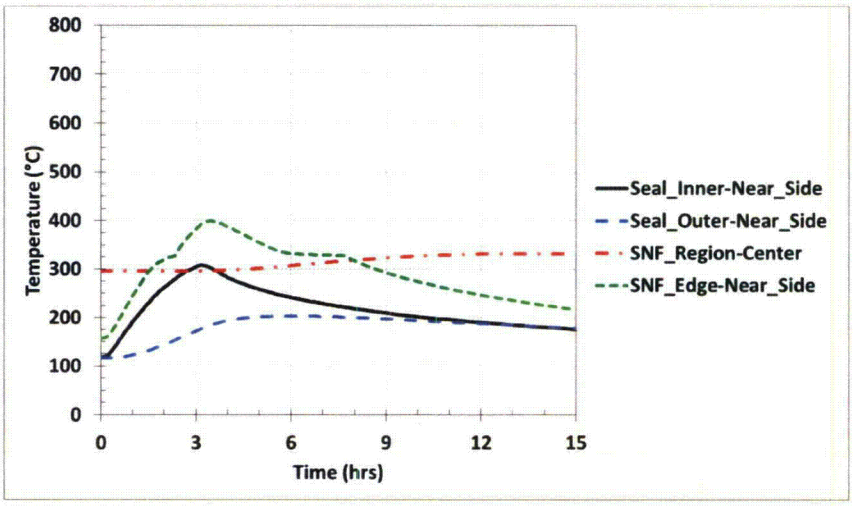


Figure 4-30. Temperature of key cask regions, Rail-Lead cask with Cask on ground, 3m offset fire.

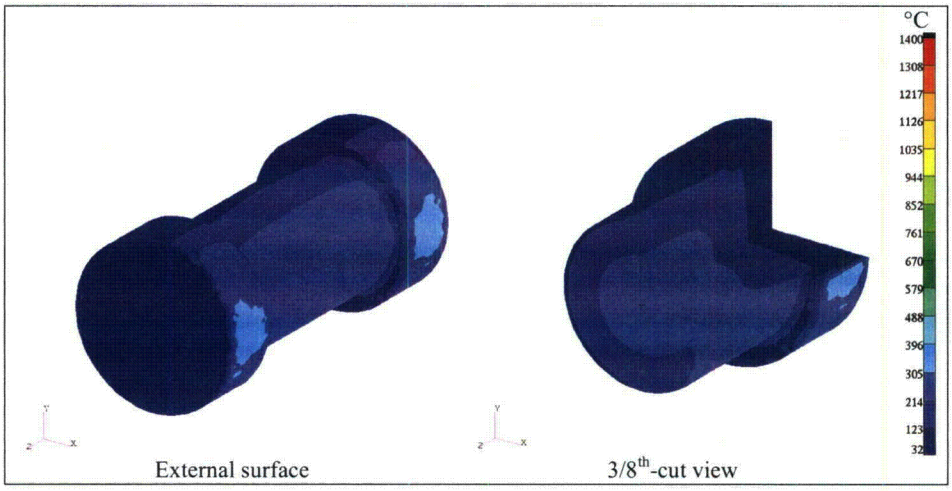


Figure 4-31. Temperature distribution of the Rail-Lead cask at the end of the 3-hour 18m offset CAFE fire with cask on ground.

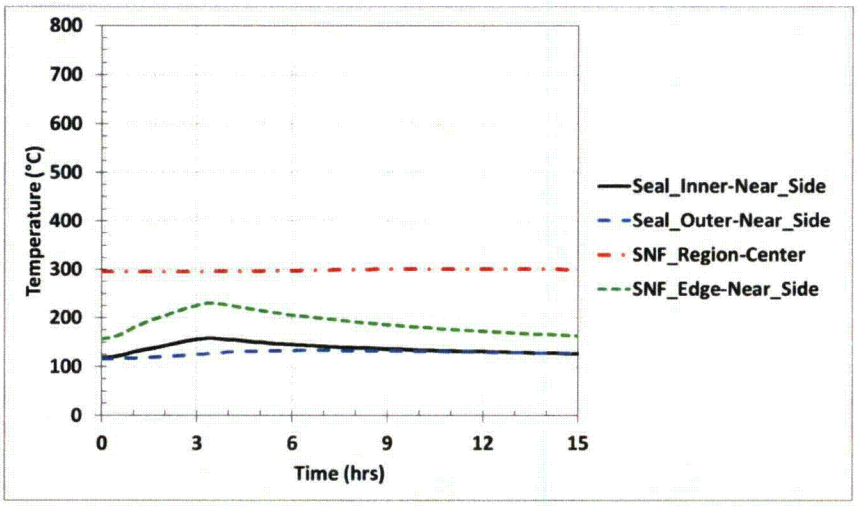


Figure 4-32. Temperature of key cask regions, Rail-Lead cask with Cask on ground, 18m offset fire.

Appendix IV contains additional plots with more information about temperature distributions at more locations in the cask. Another effect considered in the cases where lead melted is the gradual thermal expansion and contraction of the gamma shield region during the heating and cooling of the cask. This effect is discussed in the next subsection.

Melting of the Lead Gamma Shield

There are two cases in which a portion of the lead gamma shield melts. These are the three-hour concentric fire and the three-hour three-meter offset fire. The region of the lead gamma shield that melted for each case is shown in red in Figure 4-33 and Figure 4-34. Note that these two figures only show the portion of the cask wall that has lead. As shown in these figures, approximately 88% of the lead melts in the case of the three-hour concentric fire, whereas only about 30% of the lead melts in the case of the three-hour three-meter offset fire. Due to melting and thermal expansion of some of the lead gamma shield, some loss of shielding is observed, which translates to an increase in gamma radiation exposure. The width of the streaming path (gap created due to lead melt, expansion, and subsequent contraction as it solidifies) is estimated. For this estimate, the assumption is made that the thermal expansion of the lead permanently deforms (buckles) the interior wall of the cask, enabling the calculation of the gap in the lead gamma shield.

The gap in the lead region caused by the concentric fire case is assumed to appear on the top portion of the cask. That is, after the lead melts and buckles the interior wall of the cask due to its thermal expansion, molten lead is assumed to flow to the lower portions of the gamma shield

region of the cask, which allows a gap to be formed on the top portion of the cask. From a geometric analysis that considered the expansion and contraction of the lead and a conservative cask wall deformation, this gap is estimated to be about 0.5 m (20 inches), which translates to an 8.1% loss of shielding. In the case of the three-meter offset fire, the gap is assumed to form on the top portion of the molten lead region shown in Figure 4-34. For this case, the gap is estimated to be about 0.127 m (5 inches), which translates to a 2% loss of shielding. These gaps are estimated using geometric information and temperature-dependent density values of lead [*i.e.*, 11.35 g/cm^3 (0.41 lb/in^3) for solid lead and 10.6 and 10.3 g/cm^3 (0.38 lb/in^3 and 0.37 lb/in^3) for molten lead at temperatures of 384°C and 577°C (723°F and 1071°F), respectively]. The loss-of-shielding fractions reported in this section are used as part of the work presented in Chapter 5 to estimate the consequences.

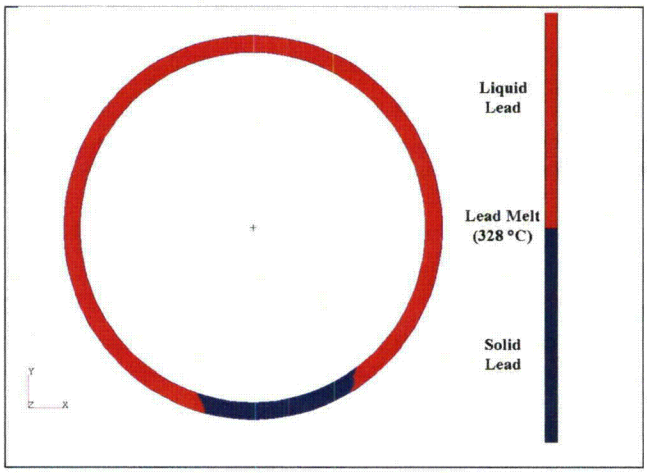


Figure 4-33. Rail-Lead cask lead gamma shield region – maximum lead melt at the middle of the cask. – Scenario: Cask on ground, 3-hour concentric pool fire.

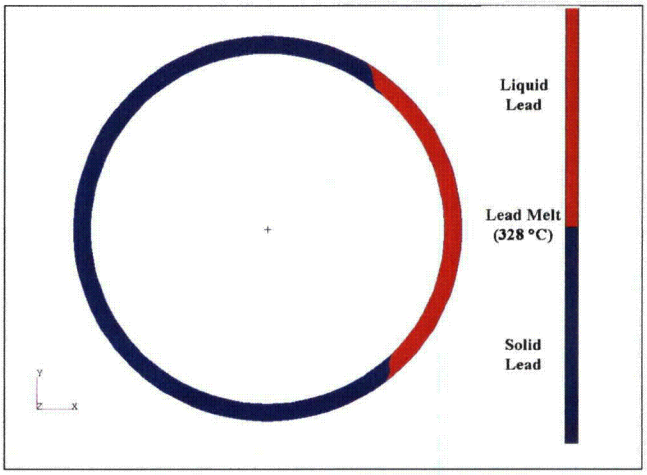


Figure 4-34. Rail-Lead cask lead gamma shield region – maximum lead melt at the middle of the cask. – Scenario: Cask lying on ground, 3-hour 3m offset pool fire.

Summary of Rail-Lead Cask Analysis Results

The results presented here show that the Rail-Lead cask is also capable of protecting the fuel rods from burst rupture and capable of maintaining containment when exposed to the severe fire environments that are analyzed as part of this study, even when the neutron shield material is conservatively assumed to be absent during the fire accident. However, some reduction of gamma shielding is estimated to occur in two cases. Partial loss of shielding is expected for the case in which the cask is exposed to an engulfing fire that burns for longer than 65 minutes and for the case in which the cask receives heat from a fire that is offset by three meters and burns for longer than two hours and 15 minutes. Nevertheless, no release of radioactive material is expected if this cask were to be exposed to any of these severe thermal environments, as the elastomeric seals did not reach their temperature limit. This ensures that the cask is capable of maintaining containment (*i.e.*, preventing any radioactive material from getting out of the package) under any of the fire environments that are analyzed.

4.4 Truck Cask Analysis

A three-dimensional analysis of the Truck-DU cask engulfed in a large fire is performed for this study. The cask is assumed to lie on the ground concentric with the hydrocarbon fuel pool fire. As explained in Section 4.2.2, the fire is assumed to last one hour. Results from the fire and heat transfer analyses that are performed on the Truck-DU cask is presented in this section.

4.4.1. Simulation of the Truck Cask

The heat transfer to and within the Truck-DU cask is modeled using P-Thermal/CAFE. The cask has a polymeric neutron shield that is assumed to melt completely and be replaced by air at its operational temperature limit (see Appendix IV). In this cask, gamma shielding is provided by a layer of DU found within the cask wall. Melting of the DU is not a concern for this cask under any of the conditions to which it is exposed. The aluminum honeycomb Impact limiters are modeled as undamaged (not deformed). Decay heat was included in the analysis. The finite element model of the cask is shown in Figure 4-35. Cask modeling details are presented in Appendix IV.

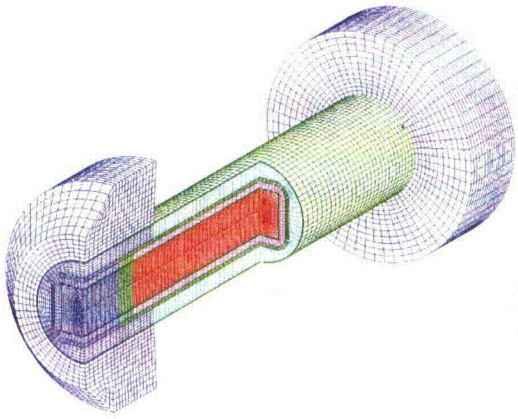


Figure 4-35. Finite element model (cut view) of the Truck-DU cask.

4.4.2. Simulation of the Spent Nuclear Fuel Region

As with the rail casks, the SNF region comprising the fuel basket and the fuel assemblies is not modeled explicitly for the Truck-DU cask. Instead, a homogenized SNF region is used. All materials and geometric features of the fuel basket and fuel assemblies are represented as a single solid inside the cask. The effective properties of the homogenized SNF region are presented in Appendix IV.

4.4.3. Truck-DU Cask Results

The results from the analysis of the cask lying on the ground and concentric with a pool fire that burns for one hour are presented in Figure 4-36 and Figure 4-37.

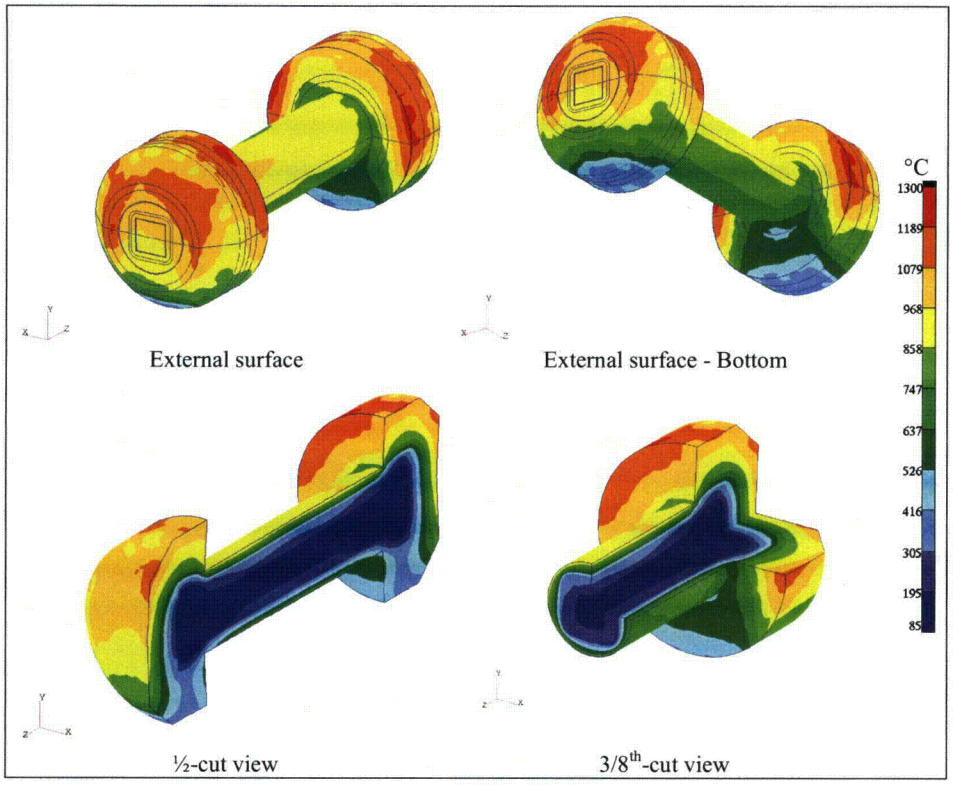


Figure 4-36. Temperature distribution of the Truck-DU cask at the end of the 1-hour concentric CAFE fire with cask on ground.

As observed with the rail casks, the vapor dome had an effect on the temperature distribution of the truck cask. This is evident by the cooler temperatures observed at the bottom of the cask. Even after one hour in the fire, the temperatures at the bottom of the cask are lowest and the temperatures at the top are highest.

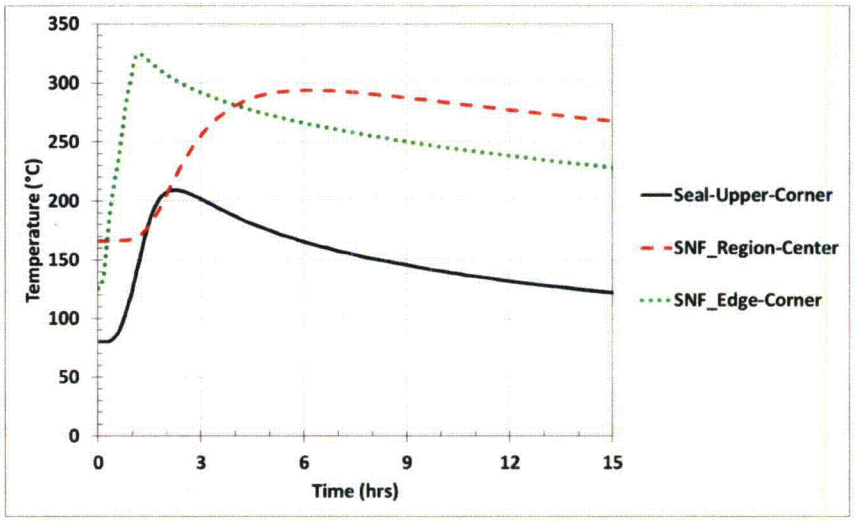
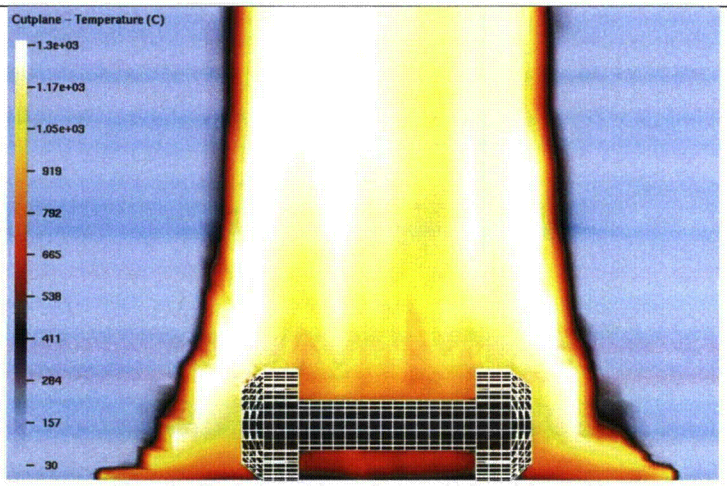
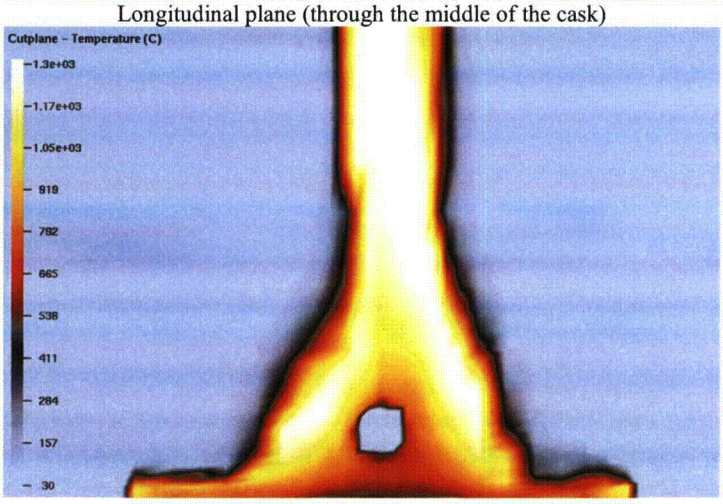


Figure 4-37. Temperature of key cask regions, Truck-DU cask with cask on ground, concentric fire.

Figure 4-38 and Figure 4-39 are the fire temperature distribution and fuel concentration plots at an arbitrary time during the CAFE fire simulation. Note that the concentration of unburned fuel under the cask is high. This means that poor combustion is occurring in that zone, leading to cooler temperatures of the lower region of the cask.



Longitudinal plane (through the middle of the cask)



Transverse plane (at mid-length of the cask)

Figure 4-38. Gas temperature plots. CAFE fire analysis of the truck cask on ground.

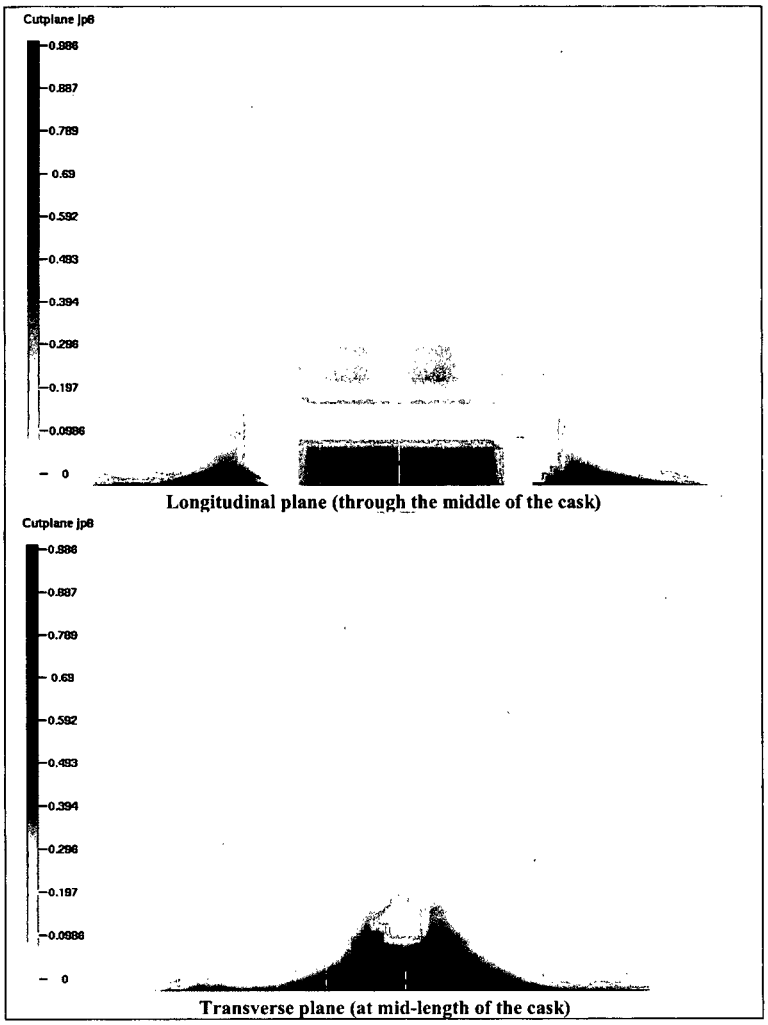


Figure 4-39. Fuel concentration plots. CAFE fire analysis of the Truck-DU cask lying on ground.

Summary of Truck-DU Cask Analysis Results

The results presented here show that the Truck-DU cask is capable of protecting the SNF rods from burst rupture and is also capable of maintaining containment when exposed to the severe fire environment analyzed in this study. That is, while the neutron shield material is conservatively assumed to be absent during the fire accident, the SNF region stays below 750°C (1382°F) and the seal region stayed under 350°C (662°F). This cask will not experience loss of gamma shielding because in this cask shielding is provided by a thick steel-DU wall, which is not affected in a way that could reduce its ability to provide shielding.

4.5 Chapter Summary

This chapter presents the realistic analyses of four fire accident scenarios. These are:

- the HAC fire described in 10 CFR 71.73,
- a cask on the ground concentric with a fuel pool sufficiently large to engulf the cask,
- a cask on the ground with a pool fire offset by the width of a rail car (3 meters), and
- a cask on the ground with a pool fire offset by the length of a rail car (18 meters).

Analyses of these four fire accident scenarios are performed for the Rail-Steel and the Rail-Lead casks. An analysis of a Truck-DU cask on the ground concentric with a hydrocarbon fuel pool sufficiently large to engulf the cask is also performed. Probable worst-case fire accident scenarios for a rail cask transported by railway and for a truck cask transported by roadway were represented within the cases analyzed. The neutron shield material of each cask analyzed was assumed to melt and flow out of the cask instantly at the beginning of the fire.

Results show that neither the Rail-Steel cask nor the Rail-Lead cask would lose the containment boundary seal in any of the accidents considered in this study. In addition, the SNF rods did not reach burst rupture temperature. However, some loss of gamma shielding is expected with the Rail-Lead cask in the event of a three-hour engulfing fire and a three-hour, three-meter offset fire. Nevertheless, because containment is not lost in any of the cases studied, no release of radioactive material is expected as a result of these hypothetical fire accidents. In the case of the Truck-DU cask, containment would be maintained in the one-hour fire accident considered in this study. These results demonstrate the adequacy of current regulations to ensure the safe transport of spent nuclear fuel. Furthermore, the results demonstrate that SNF casks designed to meet the current regulations will prevent the loss of radioactive material in realistic severe fire accidents.

CHAPTER 5

TRANSPORTATION ACCIDENTS

5.1 Types of Accidents and Incidents

The different types of accidents that can interfere with routine transportation of spent nuclear fuel are:

- Accidents in which the spent fuel cask is not damaged or affected.
  - Minor traffic accidents ("fender-benders," flat tires) resulting in minor damage to the vehicle. These are usually called "incidents."[16]
  - Accidents that damage the vehicle or trailer enough that the vehicle cannot move from the scene of the accident under its own power, but do not result in damage to the spent fuel cask.
  - Accidents involving a death or injury, but no damage to the spent fuel cask.
- Accidents in which the spent fuel cask is affected.
  - Accidents resulting in loss of lead gamma shielding or neutron shielding (or both), but there is no release of radioactive material.
  - Accidents in which there is a release of radioactive material.

Accident risk is expressed as "dose risk," a combination of the radiation dose that results from the accident and the probability of that dose. The units used for accident risk are dose units (Sv).

An accident happens at a particular spot on the route. When the accident happens, the vehicle carrying the spent fuel cask stops. Thus, there can be no more than one accident for a shipment: resumption of the shipment would essentially be a new shipment. Accidents can result in damage to spent fuel in the cask even if no radioactive material is released. While this would not result in additional exposure of members of the public, workers engaged in accident recovery operations, including unloading or subsequently opening the cask at a facility would be affected. Accidents damaging the fuel but not damaging the cask, and potential consequent risk to workers are not included in this study.

5.2 Accident Probabilities

Risk is the product of probability and consequence of a particular accident scenario. The probability, or likelihood, that a spent fuel cask will be in a particular type of accident is a combination of two factors:

- The probability that the vehicle carrying the spent fuel cask will be in an accident, and

[16] In Department of Transportation parlance, an "accident" is an event that results in a death, an injury, or enough damage to the vehicle that it cannot move under its own power. All other events that occur in non-routine transportation are "incidents." This document uses the term "accident" for both accidents and incidents.

- The conditional probability that the accident will be a certain type of accident. This is a conditional probability because it depends on the vehicle being in an accident.

The net accident probability is the product of the probability of an accident and the conditional probability of a particular type of accident. A few hypothetical examples are given in Table 5-1 to illustrate the probability calculation.

Table 5-1. Illustrations of net probability

Accident Probability for a 5000 km (3107-mile) Cross-Country Trip[a]	Accident Scenario	Conditional Probability[b]	Net Probability Of Accident
0.0099	Truck collision with a gasoline tank truck	0.82×0.003 = 0.00246	0.82×0.003×0.0099 = 0.0000244 = 2.44×10^{-5}
0.00066	Derailment into slope >80 kph (>50 mph), no fire	0.7355×0.9846×(0.06048+0.00005)×0.9887×0.0011 = 0.0000476	0.0000476×0.00066 = 3.14×10^{-8}
0.00066	Railcar accident on a bridge at 48-80 kph (30-50 mph), no fire	0.7355×0.9846×0.2665×0.0113 = 0.00218	0.00218×0.00066 = 1.44×10^{-6}

[a] Calculated from DOT, 2005, Table 1-32. [b] From event trees in Appendix V.

Accident probability is calculated from the number of accidents per kilometer (accident frequency) for a particular type of vehicle as recorded by the DOT and reported by the Bureau of Transportation Statistics. Large truck accidents and freight rail accidents are the two data sets used in this analysis. The DOT has compiled and validated national accident data for truck and rail from 1971 through 2007 (DOT, 2008), but the accident rates declined definitively between 1971 and the 1990s. For this analysis, rates from 1996 through 2007 are used: 0.0019 accidents per thousand large truck-km (0.0031 accidents per thousand large truck miles) and 0.00011 accidents per thousand railcar-km (0. 00018 accidents per thousand railcar miles).

Figure 5-1 shows the accidents per truck-km and per railcar-km for this period. The logarithmic scale is used on the vertical axis in order to show the entire range.

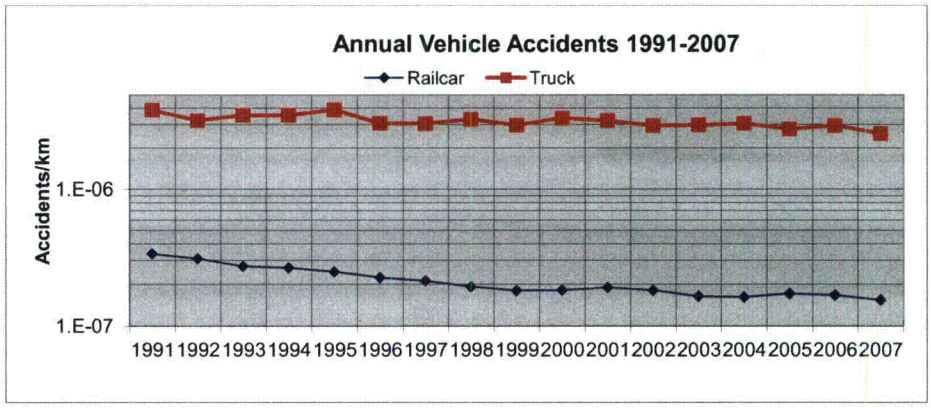


Figure 5-1. Accident frequencies in the U.S. from 1991 until 2007.

As Chapters 3 and 4 show, however, the only accidents that could result in either the loss of radiation shielding or release of radioactive material are rail accidents involving the Rail-Lead cask (when the fuel is directly loaded inside the cask, i.e. the fuel is not contained in a welded canister inside the cask). These are:

- Collisions with hard rock or equivalent at impact speeds greater than 97 km/hour (60 mph) that result in some loss of lead gamma radiation shielding or damage to the cask seals. Hard rock is not necessarily an unyielding target; however, collision of a cask with hard rock is the only type of collision along a transportation route that could damage the cask sufficiently to result in release of radioactive material or loss of lead shielding.
- Fires of long enough duration to compromise the lead shielding.

Whether or not these accidents happen depends on the likelihood (conditional probability) of the accident scenario as well as on the accident frequency. The event trees for truck and rail, Figures V-1 and V-2 of Appendix V, show some of the elements of accident scenarios in each branch of the respective event tree. The dependence on probability is illustrated by Figure V-6, which shows the sequence of events needed for a pool fire that can burn long enough to compromise the seals and the lead shielding.

Table 5-2 shows the conditional probabilities of accidents that could result in a radiation dose to a member of the public. The analysis that results in these conditional probabilities may be found in Appendix V, Sections V.3 to V.5. The calculation of these probabilities is made using the method that is typical for risk assessments, but because of the large degree of safety provided by spent fuel casks, only extremely low probability events have the possibility of leading to a radiation dose to the public. For these extremely low probability events, the results are reported to the precision of the calculation (to aid understanding of derivation of results) , but they should be considered accurate only to the order of magnitude.

Table 5-2. Scenarios and conditional probabilities of rail accidents involving the Rail-Lead cask

Accident Scenario for the Rail-Lead Cask	Conditional probability of gamma shield loss or radioactive material content release exceeding 10 CFR 71.51 quantities[a]
Loss of lead shielding from impact[b]	8.3×10^{-10}
Loss of lead shielding from fire[c]	10^{-14} to 10^{-10}
Radioactive materials release from impact[d]	5.1×10^{-10}
Radioactive materials release from fire	0

[a]More than 99.999999 percent of potential accidents would not result in either loss of lead shielding or a release of radioactive material.
[b]From the cases in Table V-2 of Appendix V with lead slump greater than 1%.
[c]From the fire event tree, Figure V-6 in Appendix V.
[d]From the sum of probabilities in the last row of Table 5-10 for the casks with metal seals. If the cask is shipped with elastomer seals the probability of release would be less.

5.3 Accidents with Neither Loss of Lead Shielding nor Release of Radioactive Material

The conditional probability that an accident involving a lead-shielded cask will be of the type with no release and no lead shielding loss is, as the footnote to Table 5-2 states, 99.999 percent. The only type of cask that could lose gamma shielding is a lead shielded cask like the Rail-Lead cask. The only type of cask that could release radioactive material in an accident is a cask carrying uncanistered spent fuel. Although the Truck-DU cask carries uncanistered fuel, this cask would not release any radioactive material under any scenario postulated in this report. The Rail-Steel cask carries only canistered fuel and would not release any radioactive material. Neither Truck-DU casks nor Rail-Steel casks are lead-shielded, therefore shielding loss would not occur.

The doses to emergency responders from an accident in which no material is released and there is no loss of lead gamma shield are shown in Table 5-3, and collective doses to the public from this type of accident are shown in Table 5-4 and Table 5-5. These radiation doses depend on:

- The external dose rate from the cask (Table 2-1).
- A ten-hour stop (DOE, 2002) at the scene of the accident, until the vehicle and/or cask can be moved safely. Ten hours is believed to overstate the stop time for most accidents.
- An average distance of five meters between the cask and the first responders and others who remain with the cask.
- For collective doses, the average rural, urban, and suburban population densities for each route.

The radiation doses in Table 5-3, Table 5-4, and Table 5-5 are the consequences of all Truck-DU accidents, all Rail-Steel accidents, and 99.999% of the Rail-Lead accidents.

Table 5-3. Dose to an emergency responder[17] from a cask in a no-shielding loss, no-release accident

Cask	Dose in Sv	Ten-hour allowed dose in Sv derived from the one-hour dose in 10 CFR 71.51
Truck-DU	1.0 E-03	1.0 E-01
Rail-Lead	9.2E-04	1.0 E-01
Rail-Steel	6.9E-04	1.0 E-01

Table 5-4 and Table 5-5 show collective doses in Sv for the ten-hour stop that follows the accident. Doses are shown for rural, suburban, and urban segments of each route, but an accident is only going to happen at one place on any route. Each listed dose is thus the collective dose that residents on that route segment could receive if the accident happened at any spot on that type of route segment.

[17] Includes police, incident command, fire fighters, EMTs, and any other emergency responders.

Table 5-4. Collective dose risks to the public from a no-shielding loss, no-release accident involving rail casks (person-Sv)

FROM/TO	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban[a]	Total	Rural	Suburban	Urban[a]	Total
MAINE YANKEE								
ORNL	3.1E-06	5.3E-05	6.6E-06	6.3E-05	2.3E-06	4.0E-05	5.0E-06	4.8E-05
DEAF SMITH	2.3E-06	5.7E-05	6.8E-06	6.6E-05	1.7E-06	4.3E-05	5.2E-06	5.0E-05
HANFORD	5.7E-06	5.2E-05	6.3E-06	6.4E-05	4.3E-06	3.9E-05	4.8E-06	4.8E-05
SKULL VALLEY	2.8E-06	5.1E-05	5.3E-06	6.0E-05	2.1E-06	3.9E-05	4.0E-06	4.5E-05
KEWAUNEE								
ORNL	3.1E-06	5.7E-05	7.2E-06	6.8E-05	2.3E-06	4.3E-05	5.4E-06	5.1E-05
DEAF SMITH	1.5E-06	6.1E-05	7.2E-06	6.9E-05	1.2E-06	4.6E-05	5.4E-06	5.2E-05
HANFORD	1.5E-06	5.3E-05	6.6E-06	6.1E-05	1.2E-06	4.0E-05	5.0E-06	4.6E-05
SKULL VALLEY	2.0E-06	6.2E-05	6.0E-06	7.0E-05	1.5E-06	4.7E-05	4.5E-06	5.3E-05
INDIAN POINT								
ORNL	2.6E-06	7.2E-05	8.7E-06	8.3E-05	2.0E-06	5.4E-05	6.6E-06	6.3E-05
DEAF SMITH	1.9E-06	5.9E-05	7.5E-06	6.9E-05	1.4E-06	4.5E-05	5.7E-06	5.2E-05
HANFORD	1.9E-06	5.6E-05	7.2E-06	6.5E-05	1.4E-06	4.3E-05	5.5E-06	5.0E-05
SKULL VALLEY	2.2E-06	6.0E-05	6.6E-06	6.9E-05	1.7E-06	4.6E-05	5.0E-06	5.2E-05
IDAHO NATIONAL LAB								
ORNL	1.9E-06	6.0E-05	5.8E-06	6.8E-05	1.4E-06	4.6E-05	4.4E-06	5.2E-05
DEAF SMITH	8.0E-07	6.0E-05	5.3E-06	6.6E-05	6.0E-07	4.6E-05	4.0E-06	5.0E-05
HANFORD	1.0E-06	6.0E-05	6.7E-06	6.8E-05	7.5E-07	4.6E-05	5.1E-06	5.2E-05
SKULL VALLEY	2.0E-06	5.9E-05	7.1E-06	6.8E-05	1.5E-06	4.4E-05	5.4E-06	5.1E-05
AVERAGE	2.3E-06	5.8E-05	6.7E-06	6.7E-05	1.7E-06	4.4E-05	5.1E-06	5.1E-05

[a]The urban dose is less than the suburban dose because urban residences are 83% shielded, while suburban residences are 13% shielded.

Table 5-5. Collective dose risks to the public from a no-shielding loss, no-release accident involving a truck cask (person-Sv)

FROM	TO	Truck-DU			
		Rural	Suburban	Urban[a]	Total
MAINE YANKEE	ORNL	4.2E-06	7.2E-05	9.1E-06	8.5E-05
	DEAF SMITH	3.9E-06	6.7E-05	8.4E-06	7.9E-05
	HANFORD	3.2E-06	5.9E-05	8.4E-06	7.1E-05
	SKULL VALLEY	3.5E-06	6.1E-05	8.6E-06	7.3E-05
KEWAUNEE	ORNL	4.1E-06	6.6E-05	8.3E-06	7.8E-05
	DEAF SMITH	2.8E-06	6.2E-05	8.4E-06	7.3E-05
	HANFORD	2.2E-06	5.8E-05	8.4E-06	6.9E-05
	SKULL VALLEY	2.6E-06	5.9E-05	8.6E-06	7.0E-05
INDIAN POINT	ORNL	3.6E-06	6.7E-05	8.2E-06	7.9E-05
	DEAF SMITH	3.6E-06	6.7E-05	8.2E-06	7.9E-05
	HANFORD	**2.7E-06**	**6.2E-05**	**8.4E-06**	7.3E-05
	SKULL VALLEY	3.0E-06	6.4E-05	8.5E-06	7.6E-05
IDAHO NATIONAL LAB	ORNL	2.6E-06	5.5E-05	7.9E-06	6.6E-05
	DEAF SMITH	1.6E-06	6.2E-05	6.8E-06	7.0E-05
	HANFORD	1.4E-06	3.6E-05	5.2E-06	4.3E-05
	SKULL VALLEY	2.1E-06	6.2E-05	8.4E-06	7.3E-05
AVERAGE		2.9E-06	6.1E-05	8.1E-06	7.2E-05

[a]The urban dose is less than the suburban dose because urban residences are 83% shielded, while suburban residences are 13% shielded

The average individual U.S. background dose for ten hours is 4.1×10^{-6} Sv. Average background doses during the 10-hour stop for the 16 truck routes analyzed are

- Rural: $(4.1\ 10^{-6}\ \text{Sv}) \times (16.8\ \text{persons/km}^2) \times \pi \times (0.8\ \text{km})^2 = 0.000138$ person-Sv
- Suburban: $(4.1\ 10^{-6}\ \text{Sv}) \times (463\ \text{persons/km}^2) \times \pi \times (0.8\ \text{km})^2 = 0.00382$ person-Sv
- Urban: $(4.1\ 10^{-6}\ \text{Sv}) \times (2682\ \text{persons/km}^2) \times \pi \times (0.8\ \text{km})^2 = 0.0221$ person-Sv

If the Truck-DU cask, for example, is in a no-shielding loss, no-release accident, the average collective dose (the sum of the background dose and the dose due to the accident) to residents for the 10 hours following the accident would be

- Rural: 0.000141 person-Sv
- Suburban: 0.003881 person-Sv
- Urban: 0.022108 person-Sv

The background and accident collective doses would be indistinguishable from the collective background dose. Any dose to an individual is well below the dose allowed by 10 CFR 71.51, as one would expect.

5.4 Accidental Loss of Shielding

The details of the calculation of doses from shielding losses are provided in Appendix V, Section V.3.1 (loss of gamma shielding) and Section V.3.2 (loss of neutron shielding).

5.4.1 Loss of Lead Gamma Shielding

Type B transportation packages are designed to safely carry radioactive material and need shielding adequate to meet the external dose regulation of 10 CFR Part 71. Spent nuclear fuel is extremely radioactive and requires shielding that absorbs both gamma radiation and neutrons. The sum of the external radiation doses from gamma radiation and neutrons should not exceed 0.0001 Sv per hour at two meters from the cask, as stipulated by 10CFR71.47.

Each spent-fuel transportation cask analyzed uses a different material to serve as the gamma shielding. Each may use different neutron shielding as well, but since no credit is taken for the neutron shield, it is not usually part of the accident analysis. The Rail-Steel cask has a steel wall thick enough to attenuate gamma radiation to acceptable levels. The Truck-DU cask uses metallic DU. Neither of these shields would lose their effectiveness in an accident. The Rail-Lead cask has a lead gamma shield which could have its effectiveness reduced in an accident. Lead is relatively soft compared to DU or steel, and melts at a considerably lower temperature (330°C) than either DU or steel.

In a hard impact, the lead shield will slump, and a small section of the spent fuel in the cask will be shielded only by the steel shells. Figure 5-2 and Figure 5-3 show the maximum individual radiation dose at various distances from the damaged cask for a range of gaps in the lead shield. In the figures, the dose estimates for the large, gaps are depicted on the left end of the graph, and the fraction of lead shield lost (gap size) increases from left to right. Figure 5-2 shows that doses larger than the external dose allowed by the regulation of 10 CFR 71.51 (0.01 Sv/hour at 1 meter from the cask) occur when the lead shielding gap is more than two percent of the shield.

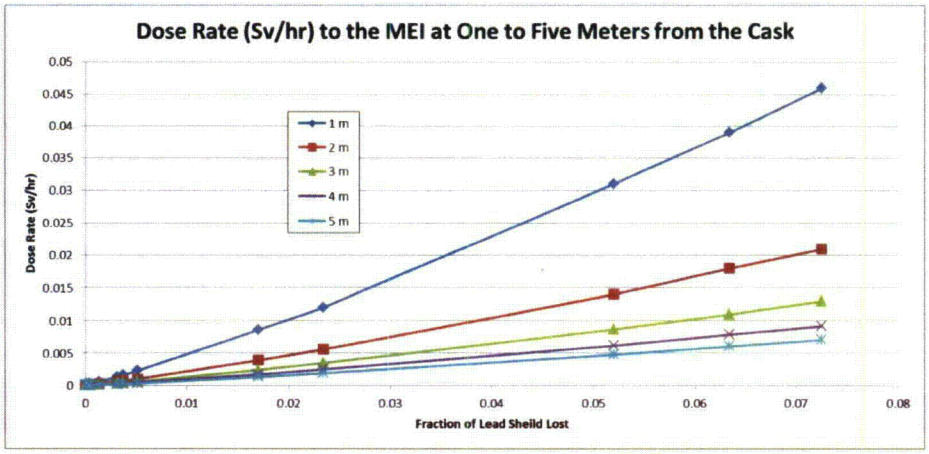


Figure 5-2. Radiation dose rates to the maximally exposed individual (MEI) from loss of lead gamma shielding at distances from one to five meters from the cask carrying spent fuel. The horizontal axis represents the fraction of shielding lost—the shielding gap.

Comment [MF30]: Figure 5-2, Figure 5-3 (also Figure V-4 and V-5 on page V-13) etc., the x-value label, the word shield is misspelled

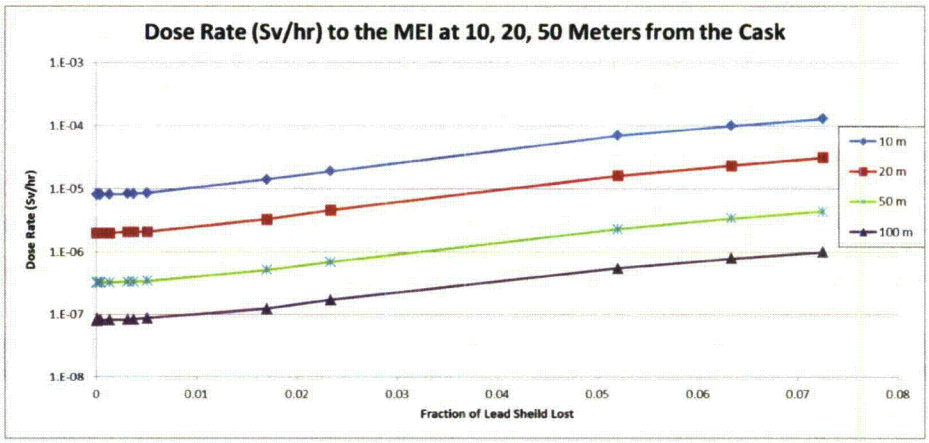


Figure 5-3. Radiation dose rates to the maximally exposed individual from loss of lead gamma shielding at distances from 10 to 100 meters from the cask carrying spent fuel. The vertical axis is logarithmic so that all of the doses can be shown on the same graph. The horizontal axis represents the fraction of shielding lost—the shielding gap.

Comment [MF31]:

One in a billion accidents (from the first row of Table 5-2) could be an impact accident that causes loss of lead shielding resulting in a dose rate that exceeds the regulatory dose rate from 10

CFR 71.51; the "one in a billion" is a conditional probability, conditional on an accident happening. The total probability of such an accident includes both this conditional probability and the probability that there will be an accident. The probability of an accident is shown in the right-hand column of Table 5-6. For example, the probability that an accident resulting in lead shielding loss leading to a dose rate greater than 0.01 Sv/hr will happen on the rail route from Maine Yankee Nuclear Plant site to Hanford is:

$$(8.3 \times 10^{-10})*(0.00214) = 1.74 \times 10^{-12}$$

or about twice in a trillion Maine Yankee to Hanford shipments.

This very small probability indicates that severe accidents, which are more traumatic to the cask than the tests shown in Figure 1-1, are not likely to happen. The conditions that can cause enough loss of lead shielding to result in radiation doses to the public above those allowed by 10 CFR 71.51 are extreme conditions.

Table 5-6. Average railcar accident frequencies and accidents per shipment on the routes studied

ORIGIN	DESTINATION	AVERAGE ACCIDENTS PER KM	ROUTE LENGTH (KM)	PROBABILITY OF AN ACCIDENT FOR THE TOTAL ROUTE
MAINE YANKEE	ORNL	6.5×10^{-7}	2125	0.00139
	DEAF SMITH	5.8×10^{-7}	3362	0.00194
	HANFORD	4.2×10^{-7}	5084	0.00214
	SKULL VALLEY	5.1×10^{-7}	4086	0.00208
KEWAUNEE	ORNL	4.3×10^{-7}	1395	0.00060
	DEAF SMITH	3.3×10^{-7}	1882	0.00062
	HANFORD	2.4×10^{-7}	3028	0.00073
	SKULL VALLEY	3.7×10^{-7}	2755	0.00103
INDIAN POINT	ORNL	8.8×10^{-6}	1264	0.0112
	DEAF SMITH	6.2×10^{-7}	3088	0.00192
	HANFORD	4.4×10^{-7}	4781	0.00212
	SKULL VALLEY	5.5×10^{-7}	3977	0.00217
INL	ORNL	3.6×10^{-7}	3306	0.0012
	DEAF SMITH	3.5×10^{-7}	1913	0.00067
	HANFORD	3.2×10^{-7}	1062	0.00034
	SKULL VALLEY	2.8×10^{-7}	455	0.00013

The overall collective dose risks to the resident population from a lead shielding loss accident on the sixteen rail routes studied are shown in Table 5-7. These include accidents whose resultant dose rates would be within regulatory limits. These doses are the totals of the rural, suburban, and urban doses from Table V-7. The expected dose to any member of the populations along the routes, at least 10 m. from the cask, is within the limits of 10 CFR 71.51. The Indian Point-to-ORNL collective dose risk is comparatively large because the suburban and urban populations

along this route are about 20 percent larger than along the other routes, and the rail accident rate per km is an order of magnitude larger.

Table 5-7. Collective dose risks per shipment in person-Sv for a loss of lead shielding accident involving a lead-shielded rail cask.

SHIPMENT ORIGIN	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	2.5E-13	2.7E-13	2.7E-13	2.6E-13
KEWAUNEE	1.0E-13	6.3E-14	5.4E-14	1.1E-13
INDIAN POINT	3.5E-12	2.4E-13	2.5E-13	2.7E-13
IDAHO NATIONAL LAB	9.9E-14	4.1E-14	2.1E-14	1.5E-14

The conditional probability that a gap in lead shielding will occur after a fire involving the cask is about 10^{-19}. The conditional probability is so small because the following has to happen before a fire is close enough to the cask, and burns hot enough, and long enough, to do any damage to the lead shield.

- The train must be in an accident that results in a major derailment or else the location of the fire will be too far removed from the cask to damage the lead shielding.
- There must be at least one tank car of flammable material involved in the accident (either on the train carrying the spent fuel cask or on another train involved in the accident).
- The derailment must result in a pileup. By regulation, railcars carrying spent fuel casks are required to have buffer cars and will never be located directly adjacent to a railcar carrying hazardous or flammable material.
- The flammable material must leak out so that it can ignite.
- The pileup must be such that the resulting fire is no further from the cask than a railcar length.

The probability of a pileup and the probability that the cask is within a railcar length from the fire are very small. Assessing the conditional probability without these two events, and considering only the more likely events, results in a conditional probability of about 10^{-10}, or about one in ten billion.

The event trees and probabilities for fire accident are discussed in detail in Appendix V.

5.4.2 Loss of Neutron Shielding

The type of fuel that can be transported in the three casks considered has relatively low neutron emission but does require neutron shielding. This is usually a hydrocarbon or carbohydrate polymer of some type that often contains a boron compound. All three of the casks studied have polymer neutron shields. Table 5-8 shows the neutron doses to individuals who are about five meters from a fire-damaged cask for ten hours. The dose allowed by 10 CFR 71.51 is provided for comparison. Neutrons are absorbed by air much better than is gamma radiation, so that

external neutron radiation would impact receptors close to the cask but not members of the general public.

Impacts caused by severe accidents, even those that cause breaches in the seals, will not damage the neutron shield significantly. However, the neutron shielding on any of the three casks is flammable and could be damaged or destroyed in a fire.

Table 5-8. Doses to an emergency responder or other individual five meters from the cask for 10 hours

Cask	Total Dose in Sv	Ten-hour allowed total dose in Sv from 10 CFR 71.51
Truck-DU	0.0073	0.1
Rail-Lead	0.0076	0.1
Rail-Steel	0.0076	0.1

Comment [MF32]: Section 5.4.2 and Table 5-8: Earlier comment asked if Table 5-8 column 2 was "Total Dose" or "Neutron Dose" since only "Dose" was noted in heading. It should be total dose to be comparable to column 3 and this version has noted 2nd column to be "Total Dose". However, text in the 1st para of the section (3rd sentence) indicates "Table 5-8 shows the neutron doses…." . So now there is an inconsistency between the text and the table. Need to confirm this is total dose in column 2 and modify text as appropriate.

The neutron doses do not exceed the allowed dose cited in the regulation following an accident. These doses could result from a regulatory fire accident. The conditional probability of this neutron dose is 0.0063 for a truck fire accident and 0.0000001 for a rail fire accident. The conditional probability of a fire for the Truck-DU cask is much higher than that for the two rail casks in part because truck accidents always include a potential source of fuel (the gas tanks of the truck) while many railcar accidents do not involve the locomotive, and in part because of the way the event trees were constructed. The truck event tree does not distinguish between minor fires and ones that are severe enough to damage the neutron shielding, while the rail event tree only considers severe fires. Therefore the conditional probability of a truck fire is quite conservative (overstated). Details are discussed in Appendix V Section V.3.2.

5.5 Accidental Release of Radioactive Materials

Radioactive materials released into the environment are dispersed in the air, and some deposit on the ground. If a spent fuel cask is in a severe enough accident, spent fuel rods can tear or be otherwise damaged, releasing fission products and very small particles of spent fuel into the cask. If the cask seals are damaged, these radioactive substances can be swept from the interior of the cask through the seals into the environment. Release to the environment requires that the accident be severe enough to damage the fuel rods and release the pressure in the rods, or there will be no positive pressure to sweep material from the cask to the environment.

The potential accidents that could result in such a release are discussed in Chapters 3 and 4. This chapter discusses the probability of such accidents and the consequences of releasing these radionuclides.

5.5.1 Spent Fuel Inventory

Spent nuclear fuel contains a great many different radionuclides. The amount of each fission product nuclide in the spent fuel depends on the type of reactor fuel and how much ^{235}U was in the fuel (the enrichment) when it was loaded into the reactor. The amount of each fission product

in the spent fuel also depends on how much nuclear fission has taken place in the reactor (the burnup). Finally, the amount of each radionuclide in the spent fuel depends on the time that has passed between removal of the fuel from the reactor and transportation in a cask (the cooling time) because the fission products undergo radioactive decay during this time. Plutonium, americium, curium, thorium, and other actinides produced in the reactor decay to a sequence of radioactive elements which are the progeny of the actinide. These progeny increase in concentration as the original actinide decays. However, there is never more radioactive material as a result of decay than there was initially.

The fuel studied in this analysis is PWR fuel that has "burned" 45,000 MWD/MTU and has been cooled for nine years[18].. The Rail-Lead cask, the only cask studied that could release radioactive material in an accident, is certified to carry 26 PWR assemblies.

The spent fuel inventory for accident analysis was selected by normalizing the radionuclide concentrations in the spent fuel by radiotoxicity (see Appendix V, Section V.4.1). The resulting inventory is shown in Table 5-9.

Table 5-9. Radionuclide inventory for accident analysis of the Rail-Lead cask (TBq)

Radionuclide	Terabecquerels (TBq) 26 Assemblies
^{240}Pu	7.82E+03
^{239}Pu	1.84E+02
^{137}Cs	4.38E+04
^{238}Pu	7.18E+01
^{243}Cm	2.50E+01
^{60}Co	5.56E+01
^{154}Eu	9.01E+02
^{134}Cs	4.03E+02
^{85}Kr	2.26E+03
^{241}Am	1.58E-01
^{242}Cm	1.00E+00
^{155}Eu	2.63E+02
^{231}Pa	3.12E-02
^{106}Ru	7.50E+00
^{236}U	1.92E-01
^{63}Ni	8.99E+02
^{233}U	5.75E-01
^{241}Pu	6.13E-01
^{113m}Cd	5.24E+00

[18] This was approximately the shortest time needed for the fuel to cool sufficiently to meet the thermal requirements of the cask certification.

The ^{60}Co inventory listed is not part of the nuclear fuel. It is the main constituent of a corrosion product, Chalk River unidentified deposits (CRUD), which accumulates on the outside of the rods, and is formed by corrosion of hardware in the fuel pool. It is listed here with the inventory because it is released to the environment under the same conditions that spent fuel particles are released.

5.5.2 Conditional Probabilities and Release Fractions

Seven accident scenarios involving the Rail-Lead cask, described in Chapter 3, could result in releases of material to the environment. The details of these scenarios that are important to calculating the resulting doses are shown in Table 5-10. A detailed description of the movement of radionuclide particles from fuel rods to the cask interior and from the cask interior to the environment is found in Appendix V, Section V.4.3.

Table 5-10. Parameters for determining release functions for the accidents that would result in release of radioactive material[a]

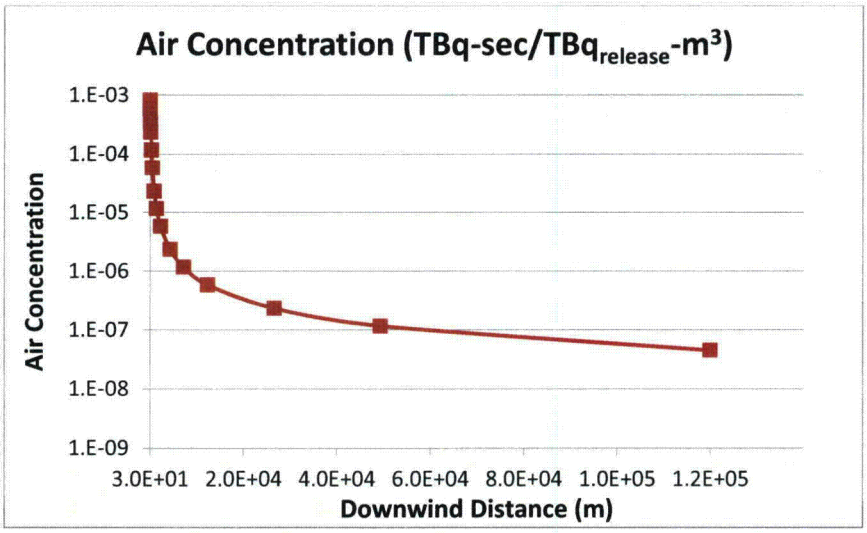
	Cask Orientation	End	Corner	Side	Side	Side	Side	Corner
	Rigid Target Impact Speed (kph)	193	193	193	193	145	145	145
	Seal	metal	metal	elastomer	metal	elastomer	metal	metal
Cask to Environment Release Fraction	Gas	0.800	0.800	0.800	0.800	0.800	0.800	0.800
	Particles	0.70	0.70	0.70	0.70	0.70	0.70	0.64
	Volatiles	0.50	0.50	0.50	0.50	0.50	0.50	0.45
	Crud	0.001	0.001	0.001	0.001	0.001	0.001	0.001
Rod to Cask Release Fraction	Gas	0.12	0.12	0.12	0.12	0.12	0.12	0.12
	Particles	4.80E-06	4.80E-06	4.80E-06	4.80E-06	4.80E-06	4.80E-06	2.40E-06
	Volatiles	3.00E-05	3.00E-05	3.00E-05	3.00E-05	3.00E-05	3.00E-05	1.50E-05
	Crud	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	Conditional Probability	5.96E-12	3.57E-11	1.79E-11	1.79E-11	3.40E-10	3.40E-10	1.13E-10

[a]Discussion of the values in this table is given in §V.4.3 of Appendix V.

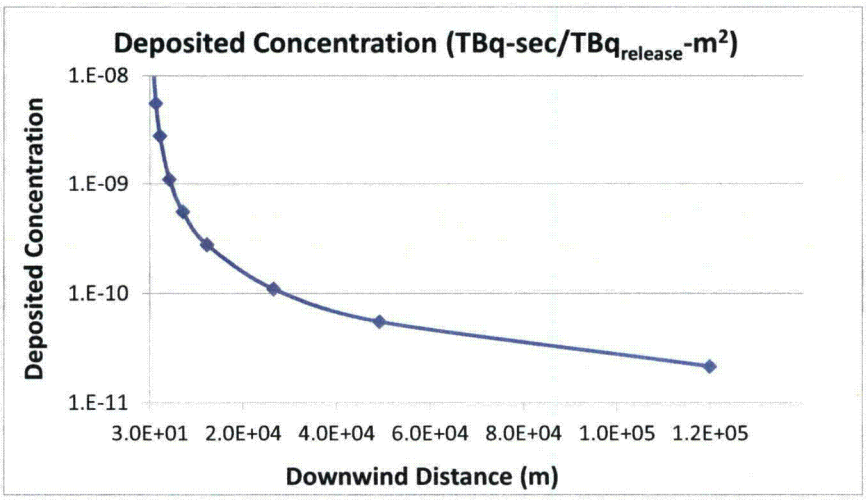
5.5.3 Dispersion

When material is swept from the cask and released into the environment, it is dispersed by wind and weather. The dispersion is modeled using the accident model in RADTRAN 6, which is a Gaussian dispersion model. The release would be at about 1.5 meters above ground level, since the cask is sitting on a railcar. The gas sweeping from the cask is warmer than ambient, so that the release is elevated. Under these conditions, the maximum ground level air concentration and deposition are 21 m downwind from the release. The dispersion was modeled using neutral weather conditions (Pasquill: stability D, wind speed 4.7 m/sec). It was repeated using very stable meteorology (Pasquill: stability F, wind speed 0.5 m/sec), but the difference was negligible because of the relatively low elevation of the release. The maximally exposed individual would be located directly downwind from the accident, 21 meters from the cask.

Figure 5-4 shows air and ground concentrations of released material as a function of downwind distance. The upwind side of the maximum concentration is short because the plume rise is very fast. Therefore the x-axis (downwind distance) is foreshortened so that the plume rise and gradual decay can be shown in the same graph. The concentrations shown are along the plume centerline and are the maximum concentrations in the plume. The figure shows the exponential decrease of airborne concentrations as the downwind distance increases. The ground (deposited) concentration also decreases in the downwind direction.



a. Airborne concentration of radioactive material released from the cask in an accident



b. Concentration of radioactive material deposited after release from the cask in an accident

Figure 5-4. Air and ground concentrations of radioactive material following a release.

5.5.4 Consequences and Risks from Accidents Involving Release of Radioactive Material

The dose from each of the accidents that would involve a release is shown in Table 5-11. Detailed discussion on how these values were obtained is provided in §V.4.3 of Appendix V.

Table 5-11. Doses (consequences) in Sv to the maximally exposed individual from accidents that involve a release

Cask Orientation	Impact Speed (kph)	Seal Material	Inhalation	Re-suspension	Cloud-shine	Ground-shine	Total
End	193	metal	1.6	0.014	8.8E-05	9.4E-04	1.6
Corner	193	metal	1.6	0.014	8.8E-05	9.4E-04	1.6
Side	193	elastomer	1.6	0.014	8.8E-05	9.4E-04	1.6
Side	193	metal	1.6	0.014	8.8E-05	9.4E-04	1.6
Side	145	elastomer	1.6	0.014	4.5E-06	3.6E-05	1.6
Side	145	metal	1.6	0.014	8.8E-05	9.4E-04	1.6
Corner	145	metal	0.73	0.0063	5.1E-05	9.0E-04	0.74

The doses listed in Table 5-11 are consequences, not risks. The dose to the maximally exposed individual is not the sum of the total doses. Each cask orientation is a different accident scenario and results in a set of internal (includes inhalation and resuspension) and external (includes cloudshine and groundshine) doses. Only one accident scenario can happen at a time. These doses would not result in either acute illness or death (Shleien et al., 1998, p. 15-3) and are in fact, similar to a single radiotherapy treatment for cancer. The internal and external doses are listed separately because they have different physiological effects. External doses are exactly that, and the receptor would receive a dose only as long as he or she is exposed to the deposited or airborne material. If people near the accident are evacuated, and evacuation can take as much as a day, then they only receive an external dose for a day. The most significant dose is the inhalation dose. All exposures to the dispersed material last until the end of the evacuation time, which for this analysis was 24 hours.

Inhaled radioactive particles lodge in the body and are eliminated slowly through physiological processes that depend on the chemical form of the radionuclide. The inhaled dose is called a "committed" dose, because the exposure is for as long as the radionuclide is in the body, though the activity of the nuclide decreases exponentially as it decays (this is shown in the Inhalation column of Table 5-11). The resuspension dose is also an inhaled dose. The NRC considers the total effective dose equivalent: the sum of the internal and external doses, which allows the doses to be added (the total is shown in the last column of Table 5-11).

A pool fire co-located with the cask and burning for a long enough time could damage the seals severely. None of the fires analyzed in this report caused sufficient seal damage to result in a release of radioactive material. The conditional probability of the series of events required to produce the most severe fire scenario analyzed is about 10^{-19}, as is discussed in detail in Appendix V, Section V.3.1.2, so analysis of a more severe fire is meaningless. Even a fire offset

from the cask but close enough to damage lead shielding has a conditional probability of between 10^{-14} and 10^{-10}.

The total collective dose risk from the universe of release accidents is shown in Table 5-12. The accident with the most severe consequence results in a release of 50 times the amount of radioactive material that can be transported in a non-accident resistant container and a collective dose of 2.18 person-Sv (direct output from RADTRAN) using a population density of 41.46 persons/km^2 (the US average minus Alaska). Of the three casks in this study, only the Rail-Lead cask could result in a release in each kind of accident considered.

Table 5-12. Total collective dose risk (person-Sv) for release accidents per shipment for each route

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	2.4E-09	1.5E-09	1.3E-09	6.4E-10
KEWAUNEE	1.0E-09	4.9E-10	4.8E-10	3.4E-10
INDIAN POINT	4.0E-08	1.6E-09	1.6E-09	5.1E-10
IDAHO NATIONAL LAB	2.5E-10	4.0E-10	1.1E-10	7.3E-10

These dose risks are negligible by any standard.

The total dose risks from loss-of-lead shielding accidents are shown in Table 5-13 (which is the same as Table 5-7, repeated here for ease of comparison), and the sum of the two is shown in Table 5-14.

Table 5-13. Total collective dose risk (person-Sv) for each route from a loss of lead shielding accident

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	2.5E-13	2.7E-13	2.7E-13	2.6E-13
KEWAUNEE	1.0E-13	6.3E-14	5.4E-14	1.1E-13
INDIAN POINT	3.5E-12	2.4E-13	2.5E-13	2.7E-13
IDAHO NATIONAL LAB	9.9E-14	4.1E-14	2.1E-14	1.5E-14

Table 5-14. Total collective dose risk (person-Sv) from release and loss of shielding accidents

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	2.4E-09	1.5E-09	1.3E-09	6.4E-10
KEWAUNEE	1.0E-09	4.9E-10	4.8E-10	3.5E-10
INDIAN POINT	4.1E-08	1.6E-09	1.6E-09	5.2E-10
IDAHO NATIONAL LAB	2.5E-10	4.1E-10	1.1E-10	7.3E-10

Table 5-15 shows the total collective dose risk for an accident involving the Rail-Lead shielded cask in which there is neither loss of lead shielding nor a release. Since the collective dose risk for this type of accident depends in the TI, the collective dose risk from an accident involving the truck cask would be the same. For the Rail-Steel cask carrying canistered fuel, the collective dose risk would be slightly less because the TI is smaller. For this analysis, the cask was assumed to be immobilized for ten hours.

Table 5-15. Total collective dose risk (person-Sv) from no-release, no-loss of shielding accidents involving the Rail-Lead cask (see Table 5-4).

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	6.3E-05	6.6E-05	6.4E-05	6.0E-05
KEWAUNEE	6.8E-05	6.9E-05	6.1E-05	7.0E-05
INDIAN POINT	8.3E-05	6.9E-05	6.5E-05	6.9E-05
IDAHO NATIONAL LAB	6.8E-05	6.6E-05	6.8E-05	6.8E-05

Table 5-16 shows the collective accident risk for the 16 rail routes from loss of neutron shielding for the Rail-Lead cask. This table is extracted from Table V-14 in Appendix V.

Table 5-16. Total collective dose risk (person-Sv) from loss of neutron shielding for accidents involving the Rail-Lead cask.

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	8.90E-14	1.16E-13	1.13E-13	1.12E-13
KEWAUNEE	3.48E-14	3.41E-14	3.72E-14	5.46E-14
INDIAN POINT	6.94E-13	1.13E-13	1.14E-13	1.22E-13
IDAHO NATIONAL LAB	5.88E-14	3.48E-14	1.09E-14	7.15E-15

5.6 Chapter Summary

The conclusions that can be drawn from the risk assessment as presented in this chapter, keeping in mind that these apply to the three types of casks studied, are:

- The sixteen truck and sixteen rail routes selected for study are an adequate representation of U.S. routes for spent nuclear fuel transportation, and there was relatively little variation in the risks per km over these routes.
- The overall collective dose risks are vanishingly small.
- The collective dose risks for the two types of extra-regulatory accidents, accidents involving a release of radioactive material and loss-of-lead-shielding accidents, are negligible compared to the risk from a no-release, no-loss-of-shielding accident. There is no expectation of any release from spent fuel shipped in inner welded canisters from any impact or fire accident analyzed.
- The collective dose risk from loss of lead shielding is comparable to the collective dose risk from a release, though both are very small. The doses and collective dose risks from loss of lead shielding are larger than were calculated in NUREG/CR-6672 as a result of better precision in the finite element modeling and a more accurate model of the dose from a gap in the lead shield.
- The conditional risk of either a release or loss of shielding from a fire is negligible.
- The consequences (doses) of some releases and some loss of shielding scenarios that occur with extremely low probability are larger than cited in the regulation of 10 CFR 71.51, and are significant, but are neither acute nor lethal.

Comment [MF33]: Section 5.6: Earlier comment on bullet 6 was that a discussion of the consequences should be a "point of discussion in preceding sections or give maximum value and compare to lethal values". I think discussion below Table 5-11 is improved. However, I fear many readers who jump to "summary" will want to know what "significant" is. I think it prudent to either reference Table 5-11, note value of 1.6 Sv in parentheses beside "significant", or indicate "significant" to mean dose similar to that received during cancer radiotherapy.

CHAPTER 6

FINDINGS AND CONCLUSIONS

The health and safety impacts of spent fuel transportation were first assessed by NRC as part of NUREG-0170, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes*, published in 1977. Based on NUREG-0170, NRC concluded that the regulations in force at the time of the environmental impact statement (1981) were "adequate to protect the public against unreasonable risk from the transport of radioactive materials" (46 FR 21629, April 13, 1981). The present document presents the most recent NRC assessment of the risks of transporting commercial spent nuclear fuel. Both NUREG-0170 and this document estimate the radiological impact for spent fuel transport conducted in compliance with the regulations of 10 CFR Part 71. Other NRC studies, including the Modal Study (Fischer, et al., 1987) and NUREG/CR-6672 (Sprung, et al., 2000), also provided spent fuel shipment risk assessments.

Regulations and regulatory compliance analyses are different from risk assessments. A regulation must be conservative because its purpose is to ensure safety, and 10 CFR Part 71, which regulates transportation, requires a conservative estimate (i.e., overestimate) of the damage to a cask in an accident and the radiation emitted from the cask during routine transportation. The original environmental assessment for 10 CFR Part 71, NUREG-0170, was also conservative, but for a different reason: only limited data were available to perform the assessment, so NUREG-0170 deliberately used conservative parameter estimates. The NRC's conclusion was that NUREG-0170 showed transportation of radioactive materials, even with conservative assumptions, to provide adequate public safety.

When an assessment is used to inform regulation, it should be as realistic as possible to provide information needed to confirm or revise the regulations it informs. Realistic assessment depends on the data availability and accurate and precise modeling techniques that have become increasingly available in the years since 1977. Consequently, the Modal Study and NUREG/CR-6672 made good progress in assessing transportation risks more realistically. As a result, both the calculated consequences and risks of radioactive materials transportation decreased. The decrease in risk means that the regulations provide for a greater level of safety than previously recognized.

The present study is a more accurate analysis than the previous analyses. Certified spent fuel casks are analyzed, rather than generic designs. Recent (2005 or later) accident frequency data and population data are used in the analyses, and the modeling techniques have been upgraded as well. This study, the Spent Fuel Transportation Risk Assessment, is another step in building a complete picture of spent nuclear fuel transportation radiological safety. Also, it represents the current state of the art for such analyses. The results of this study are compared with preceding risk assessments in the figures that follow.

6.1 Routine Transportation

Figure 6-1 and Figure 6-2 show results of routine truck and rail transportation of a single shipment of spent nuclear fuel using the single example route from NUREG-0170, the average of

the 200 routes from NUREG/CR-6672, and the average of the 16 truck or rail routes from this study. Figure 6-1 plots average collective radiation dose (person-Sv) from truck transportation, and Figure 6-2 plots average collective radiation dose from rail transportation. These average doses include the doses to the population along the route, doses to occupants of vehicles sharing the route, doses at stops, and doses to vehicle crew and other workers. Also shown are the doses without the crew and worker dose (labeled public only).

Collective doses from routine transportation depend directly on the population along the route and the number of other vehicles that share the route, and inversely on the vehicle speed. Doses to occupants of vehicles that share the route depend inversely on the square of the vehicle speed.

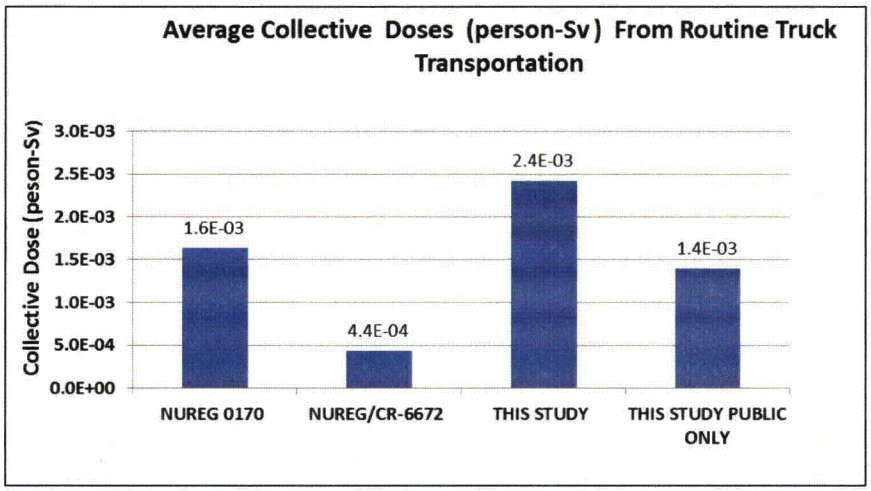


Figure 6-1. Collective doses (person-Sv) from routine truck transportation.

Comment [MF34]: Figure 6-1 (also Figure VI-3 on page VI-7) has the y-label person-Sv misspelled.

The NUREG-0170 results for truck transportation were based on a single long route, constant values of rural, suburban, and urban population densities, on different and conservative vehicle speeds on rural, urban, and suburban roads, on a fixed rate of vehicle stops, and on 1975 estimates of vehicle density (vehicles per hour), all of which led to conservative results. NUREG/CR-6672 used more realistic distributed route lengths, population densities, vehicle occupancy and density, vehicle dose rate and stop time and used the means of the distributions as parameters. As shown in Figure 6-1, the conservatism was decreased by over a factor of three.

The collective average dose in the present study is larger than the NUREG/CR-6672 result because present populations are generally larger, particularly along rural routes, and the vehicle densities are much larger (see Chapter 2). These increases were offset by the greater vehicle speeds used in the present study. The largest contributor to higher doses in this study are the parameters used for stops. In this study stops were assumed to occur every 845 km vs. 1290 km and last for 50 minutes vs. 30 minutes. The combination of these two factors results in a 2.5 times increases in the stop dose. This is especially significant because the greatest contributor to

the public collective dose is from people sharing the truck stops with the cask (56% of the collective dose). The second largest contributor is from people sharing the highway with the cask (38% of the collective dose). Residents along the route only receive 6% of the collective dose and residents near truck stops only 1%.

Figure 6-2 shows the differences between NUREG 0170, NUREG/CR-6672, and the present study for calculation of average doses to the public for routine rail transportation.

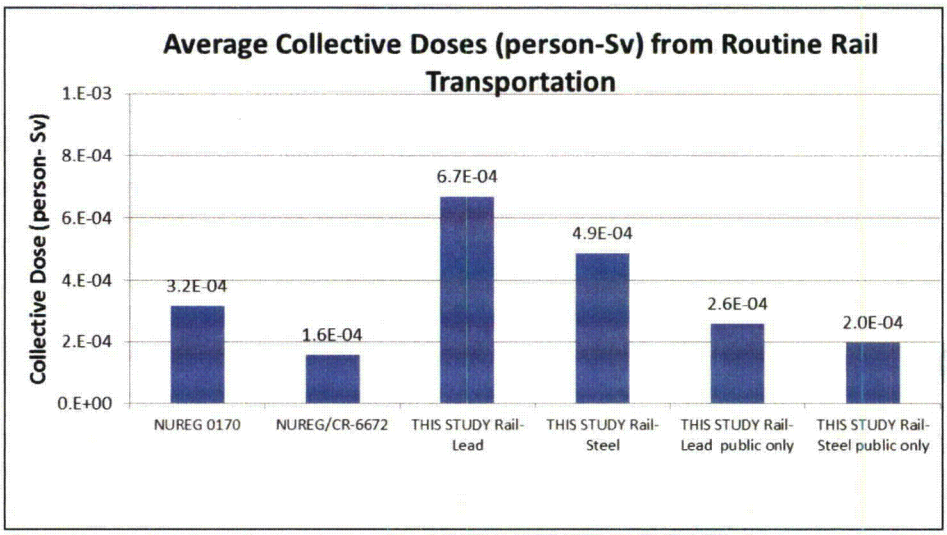


Figure 6-2. Collective doses (person-Sv) from routine rail transportation.

The difference in dose between the Rail-Lead cask and the Rail-steel cask occurs because the latter cask has a smaller external dose rate (Chapter 2). The differences in crew doses between the studies reflect the considerable difference between the methods used in the different studies.

The differences in the collective doses from routine transportation between the cited studies are not the result of differences in external radiation from the spent fuel casks. The 1975 version of 10 CFR Part 71[19] specified the same limit on external radiation (the transport index) as Part 71 specifies today. Instead, these differences reflect improvements to the modeling methods and the increase in population and traffic levels. Also the groups of people exposed that were considered in the various studies changed. For example, this study includes inspector doses that were not included in the other two studies.

The differences in results are due primarily to vehicle speed, population and vehicle densities, and differences in calculating train crew and railyard worker doses. These differences are summarized below.

[19] A copy is provided in NUREG-0170.

- *Differences in vehicle speed.* The faster the cask moves past a receptor, the less that receptor is exposed. NUREG-0170 and NUREG/CR 6672 used 80 kph (50 mph) for all truck routes and 64 kph (40 mph) on rural rail routes, 40 kph (25 mph) on suburban rail routes, and 24 kph (15 mph) on urban rail routes. The truck speeds used in this study are 108 kph (67 mph) on rural routes, 102 kph (63 mphh) on suburban routes, and 97 kph (60 mph) on urban routes, and the rail speed is 40 kph (25 mph) on rural and suburban routes and 24 kph (15 mph) on urban routes. The present speeds are based on data instead of the estimated values used in the previous studies.
- *Differences in populations along the routes.* NUREG-0170 used six persons per km^2 for rural populations, 719 per km^2 for suburban routes, and 3861 per km^2 for urban routes. NUREG/CR-6672 used 1990 census data provided by the codes HIGHWAY and INTERLINE and used the mean values of Gaussian distributions of population densities on 200 routes in the United States. This study uses 2000 census data provided by WebTRAGIS (Johnson and Michelhaugh, 2002), with some updates based on 2008 Bureau of Census data (U.S. Bureau of the Census, 2008), for the rural, suburban, and urban truck and rail route segments in each state traversed for each of the sixteen origin/destination pairs studied. The variation from the NUREG-0170 values is considerable.
- *Differences in vehicles per hour on highways.* NUREG-0170 and NUREG/CR-6672 both used the 1975 values of 470 vehicles per hour on rural routes, 780 on suburban routes, and 2800 on urban routes. This study used 2002 state vehicle density data for each state traversed. The national average vehicle density is 1119 vehicles per hour on rural routes, 2464 on suburban routes, and 5384 on urban routes. This large difference in vehicle density contributes to the difference in collective doses for routine truck transportation between NUREG/CR-6672 and this study.
- *Differences in calculating doses to rail crew.* NUREG-0170 calculated doses to rail and railyard crew by estimating the distance between the container carrying radioactive material and the crew member. NUREG/CR-6672 used the Wooden (1980) calculation of doses to railyard workers, and did not calculate a dose to the crew on the train. This study calculated all doses using the formulations in RADTRAN 6, calculated an in-transit crew dose, used an updated value for the time of a classification stop (27 hours instead of 30 hours), and used in-transit stop times from WebTRAGIS rather than the stop dose formula, pegged to total trip length, used in NUREG/CR-6672. The in-transit crew dose calculated in this study was small enough that it contributed a negligible amount to these doses.

Dose to the maximally exposed individual is a better indication of the radiological effect of routine transportation than collective dose. The same event results in different collective doses depending on the population affected, which varies both in location and considering rush hour. The dose to the maximally exposed individual is shown in Figure 6-3 for NUREG-0170 and for the three cask types of this study. NUREG/CR-6672 did not calculate this dose for routine transportation. The reduction is due to the higher speeds used in this study.

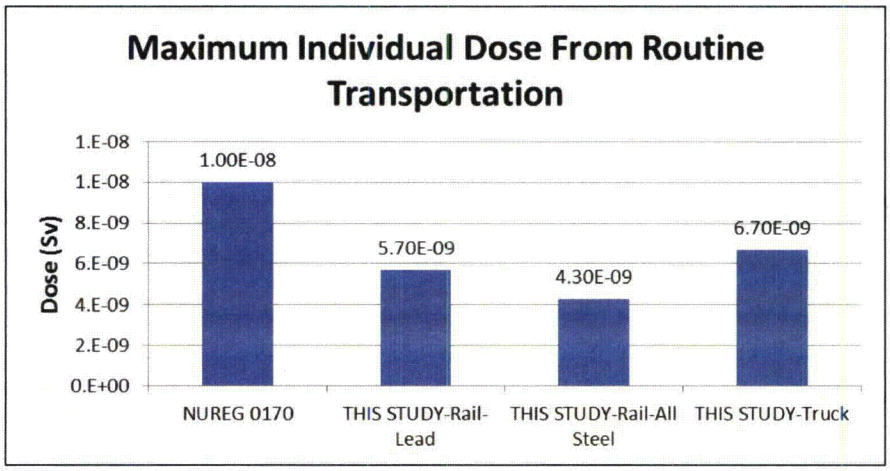


Figure 6-3. Maximum individual dose (Sv) from routine transportation.

6.2 Transportation Accidents

Radiological accident risk is expressed in units of "dose risk" that include the probability of an accident and the conditional probability of certain types of accidents. The units used are dose units (Sv) because probability is a unitless number. NUREG-0170, NUREG/CR-6672, and this study all used the version of RADTRAN available at the time of the study to calculate dose risk, but the input parameters differed significantly, primarily based on the detail and precision of the assessment of package performance,modeling improvements, and the availability of accident and population data.. In addition, improvements in RADTRAN and in other modeling codes described in earlier chapters resulted in a more accurate analysis of cask behavior in an accident.

The results shown in Figure 6-4 and Figure 6-5 for this study are averages over the 16 rail routes studied As was discussed in Chapters 3, 4, and 5, a lead-shielded rail cask, the Rail-Lead cask in this study, is the only cask type of the three studied that indicated either release of radioactive material or loss of lead gamma shielding in an accident.

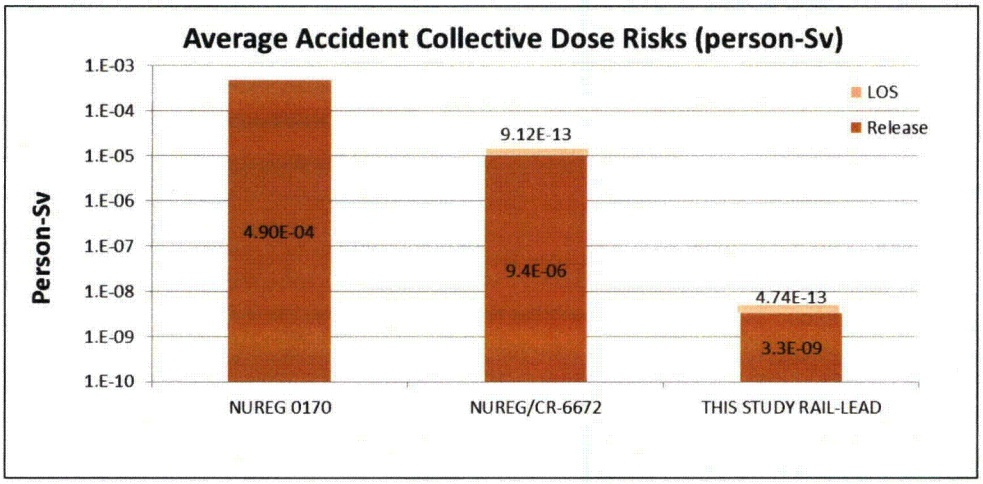


Figure 6-4. Accident collective dose risks from release and loss of shielding (LOS) accidents. The LOS bars are not to scale.

The results shown in Figure 6-4 reflect the different amounts of radioactive material released and different amounts of lead shielding lost as estimated in the respective studies. NUREG-0170 used a scheme of eight different accident scenarios, four of which postulated release of the entire releasable contents of the cask, two of which postulated no release, one postulated a ten percent release, and one postulated a one percent release. The range of conditional probabilities was from 1×10^{-5} for the most severe (100 percent release) accident to 80 percent for the two no-release accident scenarios. The NUREG-0170 "universe" of accidents and their consequences was based primarily on engineering judgment and was clearly conservative.

NUREG/CR-6672 analyzed the structural and thermal behavior of four generic cask designs—two truck and two rail casks—in great detail, and analyzed the behavior of the five groups that best describe the physical and chemical nature of the radioactive materials potentially released from the spent fuel through the casks. These five groups are particulate matter, semi-volatile substances, ruthenium, gas, and CRUD. The spent fuels considered were high burnup and low burnup PWR and BWR fuel. This analysis resulted in 19 truck accident scenarios and 21 rail accident scenarios, each with an attendant possibility, including a no-release scenario, with better than 99.99 percent probability.

The present study followed the analytical outline of the NUREG/CR 6672 analysis, but analyzed the structural and thermal behavior of a certified lead-shielded cask design loaded with fuel that the cask is certified to transport. Instead of the 19 truck scenarios and 21 rail scenarios that included potential releases of radioactive material, the current study resulted in only seven rail scenarios that included releases, as described in Chapters 3 and 5. The only parts of the cask structure that could be damaged enough to allow a release are the seals. Release could take place through the seals only if the seals fail and if the cask is carrying uncanistered fuel. No potential truck accident scenario resulted in seal failure, nor did any fire scenario. In the present study,

only the Rail-Lead cask response to extremely severe accident conditions resulted in a release. A comparison of the collective dose risks from potential releases in this study to both NUREG-0170 and NUREG/CR-6672 is appropriate, since the latter two studies considered only potential releases. The collective dose risks decrease with each succeeding study as expected, since the overall conditional probability of release and the quantity of material potentially released decreases with each successive study. The decrease in release is due primarily to the replacement of conservative estimates of cask performance in an accident with finite element analyses of cask performance in an accident. Basically, in succeeding studies, the calculated performance of the cask is better (it releases less) than estimated previously.

The collective dose risk from a release depends on dispersion of the released material, which then either remains suspended in the air, producing cloudshine, or is deposited on the ground, producing groundshine, or is inhaled. All three studies used the same basic Gaussian dispersion model in RADTRAN, although the RADTRAN 6 model is much more flexible than the previous versions and can model elevated releases. NUREG-0170 calculated only doses from inhaled and resuspended material. NUREG/CR-6672 included groundshine and cloudshine as well as inhaled material, but overestimated the dose from inhaled resuspended material. The combination of improved assessment of cask damage and improved dispersion modeling has resulted in the decrease in collective dose risk from releases shown in Figure 6-4.

Frequently, public interest in the transportation of spent nuclear fuel is focused solely on the consequences of possible accidents (without regard to the likelihood of the accident actually occurring). The maximum estimated consequence, based on average population density, from the accident with the largest release is 2.18 person-Sv. This consequence is orders of magnitude less than the 110 person-Sv in NUREG-0170 and the 9000 person-Sv estimated from Figure 8.27 in NUREG/CR-6672. The reduction in consequence is the result of using the actual spent fuel being shipped, a smaller release fraction, and improvements in the RADTRAN model. The maximum dose any person could receive from this accident is 1.6 Sv, about the same dose that is received in a single radiotherapy session by a cancer patient.

NUREG-0170 did not consider a loss of spent fuel cask lead shielding, which can result in a significant increase the dose from gamma radiation being emitted by the cask contents. NUREG/CR-6672 analyzed 10 accident scenarios in which the lead gamma shield could be compromised and calculated a fractional shield loss for each. An accident dose risk was calculated for each potential fractional shield loss.

The present study followed the same general calculation scheme, but with a more sophisticated model of gamma radiation from the cask due to the damaged shield and with 18 potential accident scenarios instead of 10. Much of the difference between the NUREG/CR-6672 dose risks from a loss of shielding and this study is the inclusion of accident scenarios that have a higher conditional probability (i.e., accidents that are more likely to happen) than any such scenarios in NUREG/CR-6672. The consequence of a loss of lead shielding estimated in NUREG/CR-6672 Table 8.13 is 41,200 person-Sv, about 100 times the 690 person-Sv estimated in this study because of the more conservative loss of shielding model used in NUREG/CR-6672 and the overestimation of the amount of lead slump in that study. Loss of lead shielding clearly affects only casks that have a lead gamma shield; casks using DU or thicker steel shielding would not be affected.

Comment [MF35]: p. 137, 5th line up from bottom is the statement, "The consequences of a loss of lead shielding estimated in ..... is ...about 100 times the 690 person-Sv estimated in this study.....". Comment on earlier version of report was that I could not find the 690 person-Sv number either manually or electronically the report. And I can not find that number in this version of the report (manually or electronically). Perhaps the number is correct and does exist within this report, but it would be nice to provide a reference to WHERE it exists (table, section, etc.) to aid the reader.

More than 99.999 percent of potential accident scenarios do not affect the cask at all and would result in neither a release of radioactive material nor an increased dose from loss of lead shielding. However, these accidents would result in an increased external radiation dose from the cask to the population near the accident because the cask would remain at the location of the accident until it could be moved. A nominal ten hour delay in moving the cask was assumed for this study. The resulting collective dose risk from this accident is shown in Figure 6-5 for all three of the cask types studied. Even including this additional consequence type, the accident collective dose risk from this study is less than that reported in either NUREG-0170 or NUREG/CR-6672.

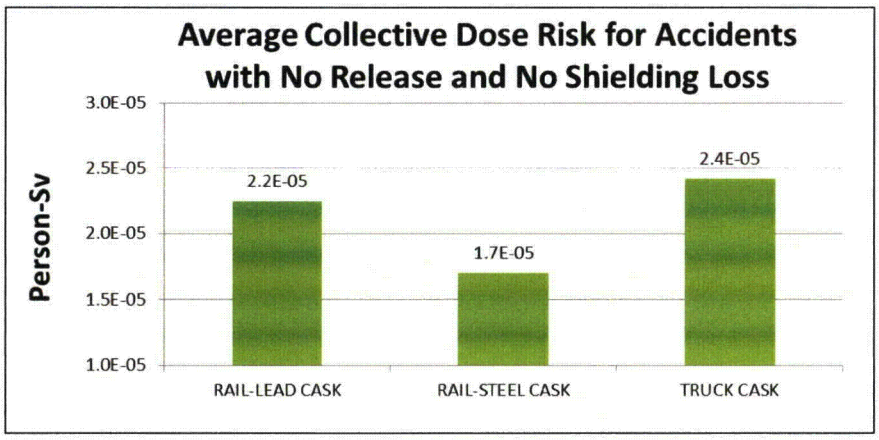


Figure 6-5. Average collective dose from accidents that have no impact on the cargo.

In conclusion, each of the three transportation risk assessments conducted for the NRC show that the NRC regulation of transportation casks ensures safety and health. The use of data in place of engineering judgment shows that accidents severe enough to cause a loss of shielding or release of radioactive material are improbable and the consequences of such unlikely accidents would require mitigation, but would not result in large radiation doses to even the maximally exposed individual. Moreover, these consequences depend on the size of the population exposed rather than on the radiation or radioactive material released. The consequences (doses) to the maximally exposed individual, 1.6 Sv to a member of the public from a release and 1.1 Sv from loss of lead shielding to a possible first responder, which is about the same as a single radiotherapy treatment for a cancer patient.

Comment [MF36]: P. 138, para below Fig 6-5, last sentence. In looking electronically and manually, I could never find the value reported "…1.1 Sv from loss of lead shielding to a possible first responder". Please confirm this is correct value and it reported somewhere in report. It would be beneficial to reference such figures as to WHERE they appear in report (table, section, etc.) to aid the reader who wants to explore more and is only looking at "Findings and Conclusions".

For the most probable accident, one that does not involve either loss of shielding or release of radioactive material, the most significant consequence, in addition to any non-radiological consequence of the accident itself, is the external dose from a cask immobilized at the accident location. Average collective doses from this type of accident for the 16 truck routes and 16 rail routes studied are shown in Figure 6-5. The most significant parameters contributing to this dose are the accident frequency and the length of time that the cask sits at the accident location. Even in this case, the significant parameter in the radiological effect of the accident is not the amount or rate of radiation released, but the exposure time.

6.3 Effect of Transportation of Higher Burnup Spent Nuclear Fuel

At the time the analyses for this report were completed, the maximum burnup for the spent fuel transported in any of the casks was 45 GWD/MTU. Current reactor operations result in spent fuel with burnup levels higher than this. A detailed examination of the effect of the higher burnup levels is outside the scope of this document, but this section provides some general insights on expected changes from transporting these higher burnup spent fuels.

The regulatory external dose rates must still be met, so there is no effect on incident free transport or on the results from accidents that do not result in damage to the cask. The higher burnup fuel will need to be cooled longer before being transported to meet the decay heat and dose rate limits of the cask, so the expected radiation emanating from the fuel should not substantially change. Therefore, the results from the loss of shielding accidents will not change. Higher burnup fuel is more likely to fail in an impact accident and the rod-to-cask release fraction may be higher (Table 5-10 gives the release fractions used in this study). In addition, the isotopic mixture of the higher burnup fuel that is cooled for a longer period of time will have more Pu and Am and less U (the amount of the fission product radioisotopes and Cm will be about the same). The combination of these effects will increase the consequence from the release accidents, but likely by less than a factor of 10. This increase does not alter the conclusions of this study.

Comment [MF37]: Section 6.3: On the surface, the majority of Section 6.3 is ok, but weak. The argument is somewhat misleading without explicitly indicating that the assumption is for placing >45 GWd/MTU fuel into one of the approved cask designs presented in this study and that the dose values of Table 2-1 (not necessarily the regulatory limits) are met. Also, if I design a package for higher burnups (simplistically add more shieding), but then lose the neutron shielding in an accident, the argument that the dose for a loss-of-shield event is not correct (the "exposed" fuel will give much higher dose in accident). Finally, if the release should increase by a factor of 10, the inhalation dose in Table 5-11 will likely exceed the lethal dose and thus significantly change the conclusions for those low-probability accidents. I very much question the value of including this section without more thought, discussion, and explanation. Happy to help/advise if desired.

6.4 Findings and Conclusion

The following findings are reached from this study:

- The collective dose risks from routine transportation are vanishingly small. Theses doses are about four to five orders of magnitude less than collective background radiation dose.
- The routes selected for this study adequately represent the routes for spent nuclear fuel transport, and there was relatively little variation in the risks per kilometer over these routes.
- Radioactive material would not be released in an accident if the fuel is contained in an inner welded canister inside the cask.
- Only rail casks without inner welded canisters would release radioactive material, and only then in exceptionally severe accidents.
- If there were an accident during a spent fuel shipment, there is only about one in a billion chance the accident would result in a release of radioactive material.
- If there were a release of radioactive material in a spent fuel shipment accident, the dose to the maximum exposed individual would be less than 2 Sv, about the dose given in a single radiotherapy treatment to cancer patients.
- The collective dose risks for the two types of extra-regulatory accidents (accidents involving a release of radioactive material and loss of lead shielding accidents) are negligible compared to the risk from a no-release, no-loss of shielding accident.
- The risk of loss of shielding from a fire is negligible.
- None of the fire accidents investigated in this study resulted in a release of radioactive material.

Based on these findings, this study reconfirms that radiological impacts from spent fuel transportation conducted in compliance with NRC regulations are low, in fact generally less than previous, already low, estimates. Accordingly, with respect to spent fuel transportation, the previous NRC conclusion that the regulations for transportation of radioactive material are adequate to protect the public against unreasonable risk is also reconfirmed by this study.