NUREG-xxxx

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
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ABSTRACT

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ACRONYMS

AMAD	activity median aerodynamic diameter
Bq	becquerel
BWR	boiling water [nuclear] reactor
CFR	Code of Federal Regulations
Ci	curie
CoC	Certificate of Compliance
DOE	[U.S.] Department of Energy
DOT	[U.S.] Department of Transportation
DU	depleted uranium
EPA	[U.S.] Environmental Protection Agency
EQPS	equivalent plastic strain
HLW	high-level [radioactive] waste
IAEA	International Atomic Energy Agency
ISCORS	[U.S.] Interagency Steering Committee on Radiation Standards
kph	kilometers per hour
LCF	latent cancer fatalities
MEI	maximally exposed individual
MPC	multi-purpose canister
mph	miles per hour
mrem	millirem
MTU	metric tons of uranium
MWD	megawatt-days
NRC	[U.S.] Nuclear Regulatory Commission
PWR	pressurized water [nuclear] reactor
rem	Roentgen equivalent man
Sv	sievert

CHEMICAL SYMBOLS

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

PUBLIC SUMMARY

Spent nuclear fuel is extremely radioactive. People are understandably concerned when spent fuel is moved in trucks and by rail over public roads and railroads. Thirty-five years ago the Nuclear Regulatory Commission responded to this concern by estimating what the radiation impact of transporting radioactive materials, including spent fuel, would be. The result was the Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, NUREG-0170, published in 1977, an environmental impact statement for transportation of all types of radioactive material by road, rail, air, and water. This EIS concluded that:

- The average radiation dose to members of the public from routine transportation of radioactive materials is a fraction of the background 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.

On the basis of this EIS, 1981 regulations were considered "adequate to protect the public against unreasonable risk from the transport of radioactive materials." The adequacy of these regulations was questioned, however, because the EIS was based mostly on estimates of radiation dose and accident rate, and not much data or information was available to support those estimates. Questions about "reasonable" risk and about accident consequences ("what if the accident does happen?") have also been raised.

Trucks and railcars carrying casks of spent nuclear fuel can have accidents like any other truck or railcar of similar size and weight. The Nuclear Regulatory Commission recognizes this, and requires that spent fuel casks be designed and built to withstand very severe accidents. Nonetheless, questions have been raised about accidents that are worse than what the cask is designed for. The NUREG 0170 and later studies of casks have considered accident conditions more severe than those the regulations require the cask to meet.

A 1987 study applied actual accident statistics to projected spent fuel transportation. This "Modal Study" also recognized that accidents could be described in terms of the strains they produced in the cask (for impacts) and the increase in cask temperature (for fires). Like NUREG-0170, the 1987 study based risk estimates on models, because spent fuel shipments had not had enough accidents to support projections or predictions. A 2000 study of two generic truck casks and two generic rail casks analyzed the cask structures using finite element modeling, a relatively sophisticated modeling technique. Truck and rail accident statistics were used because even by 2000 there had been too few accidents involving spent fuel shipments to provide statistically valid data.

The dispersion of material released from the cask in an accident was also modeled with increasing refinement. NUREG-0170 assumed that most very severe accidents would result in release of all of the releasable cask contents to the environment; the calculated risk then depended on accident rate. The 2000 study analyzed the physical properties of spent fuel rods in a severe accident, and revised estimates of material released to one percent or less of the NUREG-0170 estimates. Accordingly, risk estimates were revised downward. The 2000 study also verified that an accidental release of radioactive material could only be through the seals at the end of the cask: an accident could cause seal failure, but no breach in the cask body.

The present study models real casks and the commercial spent nuclear fuel that these casks are certified to transport. Two rail casks and a truck cask are evaluated. Measured external dose rates determine the radiation exposure from routine, incident free transportation, because all spent fuel casks emit some external radiation, as the regulations allow. The radiation dose from this external radiation to any member of the public during routine transportation, including stops, is barely discernible compared to natural background radiation. Figure PS-1 shows an example cask and the way the radiation to a member of the public is modeled.

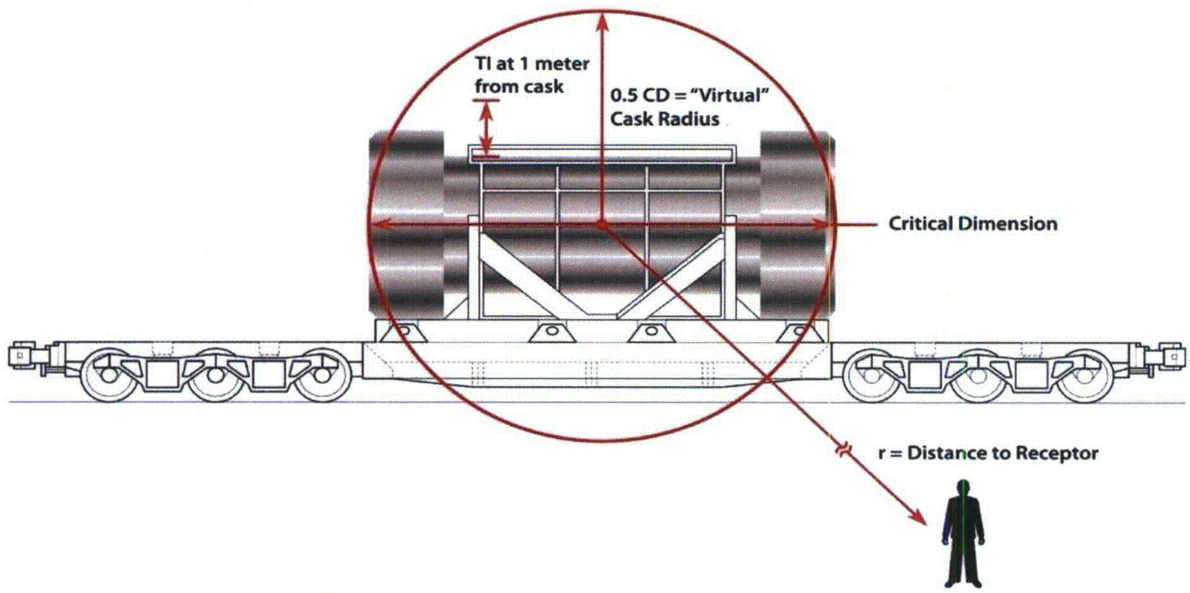

Figure PS-1. Model of a spent fuel cask in routine, incident-free transportation and radiation dose to a member of the public

The external radiation from the spent fuel cask results in a very small dose to each person along the route traveled by the cask. The collective dose from routine transportation is the sum of all of these doses. For this study, several example transportation routes were examined. Table PS-1 gives the total dose to all of the exposed people for one of these routes, the truck shipment from the Maine Yankee Nuclear Power Plant to Oak Ridge National Laboratory. The background radiation dose the exposed people experience during the time of the shipment is also included.

Table PS-1. Collective dose from routine transport for the truck route from Maine Yankee Nuclear Power Plant to Oak Ridge National Lab (person-Sv)

Exposed Population	Rural	Suburban	Urban	Urban Rush Hour	Total
Residents near route	7.9×10^{-6}	1.4×10^{-4}	2.9×10^{-6}	6.5×10^{-8}	1.5×10^{-4}
Traffic on the route	1.3×10^{-4}	2.3×10^{-4}	5.4×10^{-5}	5.0×10^{-6}	4.2×10^{-4}
Residents near truck stops	1.1×10^{-6}	2.3×10^{-5}	*	*	2.4×10^{-5}
Truck Crew	5.6×10^{-8}		4.8×10^{-9}		6.1×10^{-8}
Escort	4.7×10^{-8}		4.3×10^{-9}		5.1×10^{-8}
Inspectors (10 inspections)					3.2×10^{-8}
Truck stop workers					2.0×10^{-9}
Background					8.81

*Most truck stops are located in rural or suburban areas

The collective doses calculated for routine transportation have gone down with each successive study of the risks from spent fuel transportation. Figure PS-2 shows a comparison of the collective doses calculated in the three studies for truck transportation.

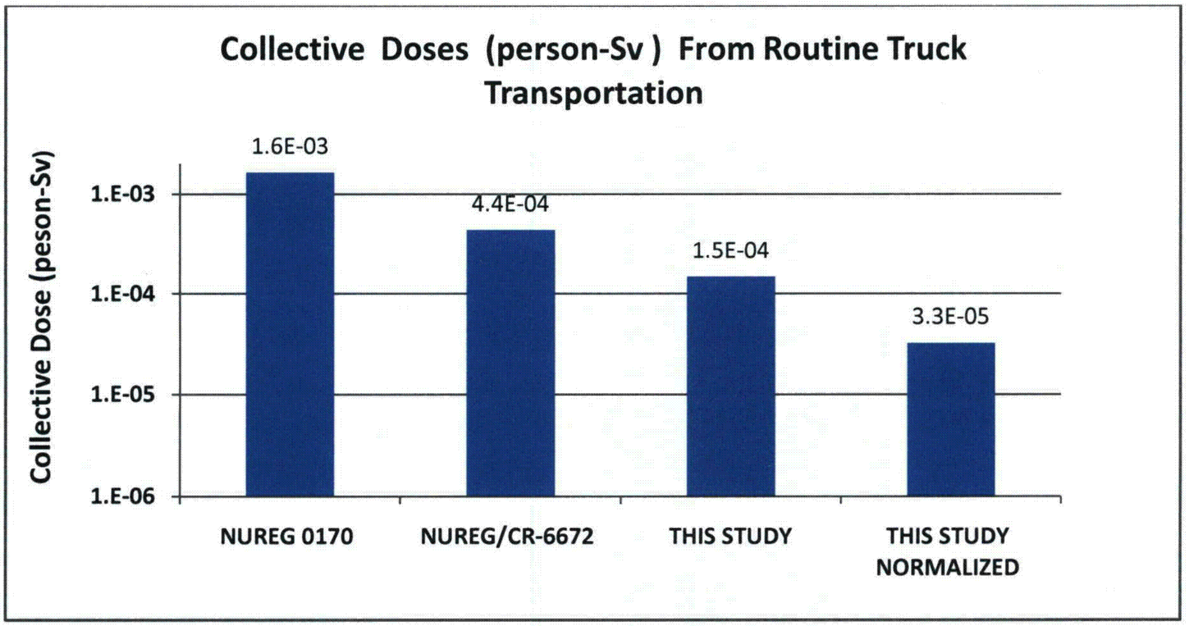

Figure PS-2. Collective doses (person-Sv) from routine truck transportation

The study uses current (2006 to 2008) truck and rail accident statistics to determine the probability of an accident and the severity of that accident. Detailed analyses are performed to evaluate how the cask responds to the accident. Figure PS-3 shows one impact scenario, a 97 kph (60 mph) corner impact onto a rigid target, and the resulting deformations. Almost all of the deformation is in the impact limiter, a device that is added to the cask to absorb energy, much like the bumper of a car. Similar analyses were performed for impacts at 48, 97,145, and 193 kilometers per hour (kph)—equal to 30, 60, 90, and 120 mph—in end-on, corner, and side-on orientations for two cask designs. These impact speeds encompass all accidents for truck and rail transportation. Figure PS-4 shows one fire scenario, a three-hour engulfing fire, and the resulting temperature distribution in the cask. Additional simulations were performed with the fire offset from the cask. These fires include all fire-related accidents in rail transportation. The longest duration for an engulfing fire during truck transportation is one hour, due to the smaller amount of fuel that is carried on board a tanker truck.

The detailed simulations were performed for two spent fuel casks that are intended for transportation by railroad, the NAC-STC and the HI-STAR 100. In addition, the results for a third cask, the GA-4, which is intended for transportation by truck, are inferred from earlier analyses.

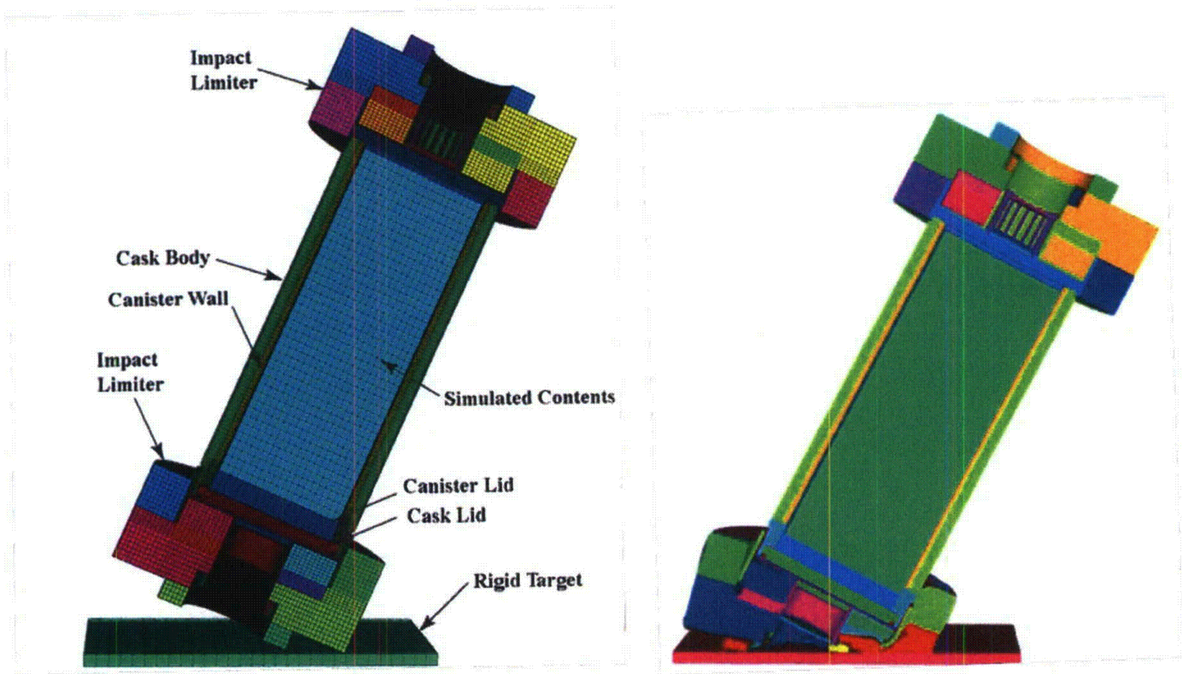

Figure PS-3. Corner impact onto a rigid target at 97 kph (60 mph) accident scenario for a spent fuel cask and the deformations produced by the impact.

Figure PS-4. Engulfing fire scenario and the temperature contours in the cask following a 3-hour fire duration. The transparency of the flames has been increased so the cask can be seen. In the actual fire simulation, and in a real fire, the flames are opaque.

The impact and thermal analysis results indicate that for the truck transportation cask no accident results in release of radioactive material or reduction in the effectiveness of the gamma shielding. The only radiological consequence of an accident is due to the long duration stop that is associated with the accident. During this stop emergency responders could be fairly close to the cask. Because there is no loss in effectiveness of the gamma shielding, the radiation dose to these responders is quite small.

For rail transport of spent fuel enclosed within an inner welded canister, the detailed impact and thermal analyses indicate there would be no release of radioactive material in any accident. For some very improbable impacts and some long duration fires there could be a small reduction in the effectiveness of the lead gamma shielding, leading to an elevated external radiation level. This loss of shielding results in a maximum dose to a person 20 meters from the cask of $2x10^{-5}$ Sv and a collective population dose risk of less than $6x10^{-9}$ person-Sv.

For rail transport of spent fuel that is not enclosed within an inner welded canister, there is the possibility for some release of radioactive material following exceptionally severe impacts. The maximum dose to an individual from this release is 1.6 Sv and the collective population dose risk is less than $5x10^{-7}$ person-Sv.

Similar to the routine transportation collective doses, the collective dose risk from accidents has decreased with each successive risk assessment. Figure PS-5 shows a comparison of collective doses from the three studies. This study considered accident doses from one source that was neglected in the prior studies, the dose that results from accidents where there is no release and no loss of shielding, but increased exposure to a cask that is stopped for an extended period of time. Considering this scenario is important because over 99.999 percent of all accident scenarios do not lead to release or loss of shielding.

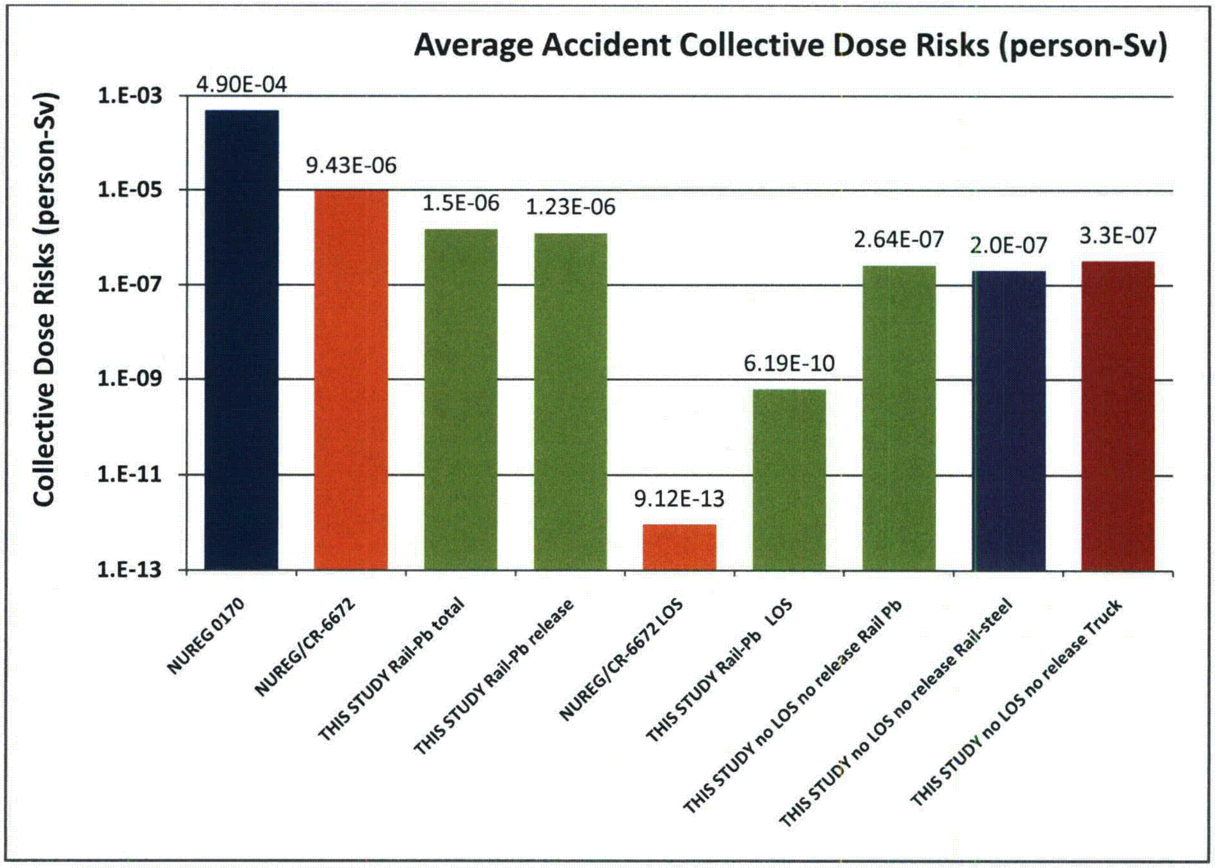

Figure PS-5. Accident collective dose risks

One other point of comparison between the studies is the maximum consequence of an accident. For NUREG-0170 this was about 110 person-Sv, for NUREG/CR-6672 it was about 9000 person-Sv, and for this study it is about 2 person-Sv. Not only is the estimated risk of spent fuel transportation exceedingly small, but the estimated maximum consequence is also very small.

CHAPTER 1

INTRODUCTION

1.1 History and Purpose of this Analysis

The purpose of this study is analysis of the radiological risks of transporting spent nuclear fuel, in both routine transportation and transportation accidents, using the latest available data and modeling techniques. This study is the latest in a series of such assessments and rounds out this series by analyzing the behavior of certified casks carrying real fuel of known isotopic composition and burnup. The studies that preceded this one were based on conservative and generic assumptions. The study is not intended to be a risk assessment for any particular transportation campaign, like transportation from reactors to a permanent repository, and does not include the consequences of malevolent acts nor does it attempt to ascribe a probability to them.

The Nuclear Regulatory Commission (NRC) licenses casks used to transport spent nuclear fuel under the regulation of Title 10 of the Code of federal Regulations Part 71 (10 CFR Part 71). Part of the technical basis for this regulation was NUREG-0170, the 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. The conclusions, drawn in part from this environmental assessment, were:

- The average radiation dose to members of the public from routine transportation of radioactive materials is a fraction of the background 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 (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 the imprecise models available at the time. 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 the spirit of continuing review, two comprehensive contractor reports dealing with radioactive materials transportation have been issued since 1977: the Modal Study (Fischer et al., 1987) and NUREG/CR-6672 (Sprung et al., 2000)[1]. Both were advances in transportation risk assessment. The Modal Study was the first intensive examination of vehicle accident statistics, and the first to organize the frequency of severe accidents by structural and thermal response of the cask. Using documented accident frequencies of large trucks and railcars, the Modal Study organized the probability of accidents by the structural and thermal response of the casks in the accident. The Modal Study concluded that the frequency of accidents severe enough to produce significant

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[11] "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.

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 matrix and recasting it as a matrix of impact speed and temperature. 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 capability available for NUREG/CR-6672 allowed analysis of the detailed structural and thermal damage to transportation casks. 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 GWD/MTU) and relatively high burnup (60 GWD/MTU) PWR and BWR fuel, although the transportability of the high burnup fuel was not considered. The particular characteristics of high-burnup fuel shown by Einziger (2007) and Einziger and Beyer (2007) were investigated after the publication of NUREG/CR-6673 and could not have been\ included in the NUREG/CR-6672 analysis. NUREG/CR-6672 studied the behavior of two generic truck casks and two generic rail casks which were each composites of several certified casks.

The results of the NUREG/CR-6672 risk assessment were several orders of magnitude less than the estimates of NURE 0170, and concluded that no radioactive material would be released in more than 99.99 percent of accidents involving spent fuel. These low risk estimates resulted from the use of refined and improved analytical and modeling techniques, exemplified by the finite element analyses of cask structure, and limited experimental data,

The present study analyzed the behavior of three currently certified casks carrying Westinghouse 17x17 pressurized water reactor (PWR) fuel with 45,000 megawatt-days per metric ton of uranium (MWD/MTU) burnup, the highest burnup that any of the three casks are certified to carry. The resulting radiological risks are less than those reported in NUREG/CR-6672. For routine transportation, the risks are slightly less than those estimated in NUREG/CR-6672, because the actual external dose rates are less than the regulatory maximum used in the other studies, and because of code and modeling improvement. For accidents, the radiological risks calculated in the current study are at least an order of magnitude less. The improvement of the risk estimates of NURGE-0170 and NUREG/CR-6672 is the result of new data and observations, and improved modeling techniques.

1.2 Risk

Risk provides understanding of events that might happen in the future. It is always a projection. Once an event happens, it is no longer a risk. Because risks are projections of potential future events, calculations of risk are based on estimates and approximations.

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. Therefore, both regulators and the public rely on risk estimates: projections of potential accidents and events, when and where they will happen, and how severe they will be. Risk estimates include estimating the likelihood and the severity of transportation accidents, as well as the likelihood of exposure to ionizing radiation from routine transportation.

Risk is usually defined by the risk triplet:

- What can happen (the scenario)?
- How likely is it (the probability)?
- What if it happens (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 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 (or we would hesitate to drive on the same freeway as a large truck). The probability of a traffic accident is about 1/100,000 per mile according to the Department of Transportation Bureau of Transportation Statistics (DOT, 2007), and the probability of a particular traffic accident scenario that includes vehicles carrying casks of radioactive material is much smaller still, as shown in the event trees in Appendix II of this document.

1.2.1 Accident Data

The only data available to estimate the future probability of a scenario is how often that scenario has occurred in the past. The frequency of the scenario can be considered the same as probability. In the case of transportation accidents, there must be enough accidents in the data that the accidents per kilometer can predict future accidents per kilometer 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] Even accidents in hazardous materials transportation do not provide a large enough data base for statistical validity. The database used in this study is the frequency of highway accidents involving large semi-detached trailer trucks and the frequency of freight rail accidents (DOT, 2007). Freight rail accident frequency is based on accidents per railcar-mile.

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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 and one Class 7 highway accident in the past ten years http://www.phmsa.dot.gov/staticfiles/PHMSA/DownloadableFiles/Files/tenyr_ram.pdf.

1.2.2 Radioactive Materials Transportation Scenarios

Transportation risk is defined in several scenarios, the most probable of which is routine transportation of a cargo 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 99.999% (see Chapter 5).
- What if it happens? The consequence is a radiation dose about 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:

- Accidents in which there is no release of radioactive material. These include minor traffic accidents (fender-benders) and severe accidents in which the vehicle is badly damaged but there is no release of radioactive material.
- Accidents in which there is loss of gamma shielding but no release of radioactive materials
- Accidents in which there is a release of radioactive material.

It is not the purpose of this study to analyze traffic accidents that do not involve radioactive material, Traffic accident statistics (accident frequencies) are used in the accident analysis to calculate risks from accidents. Traffic accident frequencies for large semi-detached trailer trucks are about five per 10,0000 highway kilometers and for freight rail, about four per thousand railcar kilometers. The net accident probability is the product of the traffic accident probability and the conditional probability. The conditional probability that, if an accident happens, it is an accident of a certain type or severity and results in a particular release. The conditional probability that there is no release is more than 99%.

The consequence of an accident scenario is a dose of ionizing radiation. The risk is the product of the net accident probability and the consequence, and is called "dose risk." Accident risks are discussed in Chapters 3, 4, and 5.

1.3 Regulation of Radioactive Materials Transportation

Transportation of radioactive materials on public rights of way is regulated by the NRC under 10 CFR Part 71 and by the DOT, as part of hazardous materials transport regulation, under 49 CFR Parts 173 to 178. The regulation of 10 CFR Part 20 are also relevant. In general, the DOT regulations apply to industrial packaging 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 for routine transportation and are not guaranteed to maintain their integrity in accidents, though many do. Type B packages are used to transport very 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.

Nuclear fuel that has undergone fission (" burned") in a reactor is both extremely hot and extremely radioactive when it is removed from the reactor. In order to cool thermally and to allow the very radioactive and very short-lived fission products in the fuel to decay away, the fuel is discharged from the reactor into a large pool of water. The fuel remains in this pool for at least three to five years, until it can be remotely handled safely. 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 can be transported to a storage site or other destination. Fuel is almost never transported before it has cooled for five years. The transportation casks used are rated for heat, and this rating determines the cooling time needed.

10 CFR Part 20

This section of the Code of Federal Regulations prescribes the largest radiation dose that a member of the public should 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 regulation. 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 for example, Table 2-12 shows, 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 is a useful measure of radiation exposure, since it affects everyone. This section also regulates occupational doses to:

- 0.05 Sv per year (5 rem per year) TEDE
- 0.15 Sv/year (15 rem/year) to the lens of the eye
- 0.5 Sv/year (50 rem/year) to the skin.

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 containers for very radioactive materials like spent nuclear fuel are therefore designed to maintain their integrity in severe accidents, so that no radioactive material is released. Containers that can meet this requirement are Type B containers, and include the casks considered in this analysis, the NAC-STC and Rail-Steel cask rail casks and the GA-4 overweight truck cask.

Type B containers are designed to pass the series of tests described in 10 CFR 71.73 and shown in the diagram of Figure 1-1.

- A 30-foot drop onto an unyielding surface. "Unyielding" means that the cask absorbs all the impact energy when it drops, and the surface does not absorb any impact energy. This drop is followed by
- A 40-inch drop onto a 6-inch-diameter steel cylinder, to test resistance to punctures. This test is followed by
- A 1475 °F fire that fully engulfs the cask for 30 minutes .
- Immersion for eight hours under three feet of water.

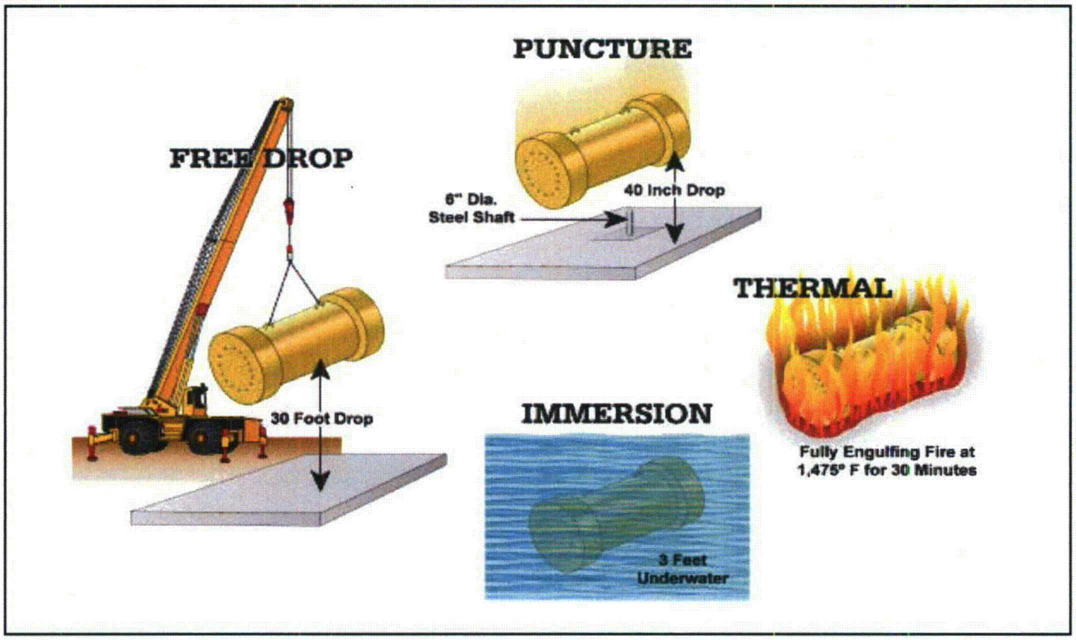

Figure 1-1. The four tests for Type B casks

Every cask used is not tested. The cask must withstand the tests without a leak or breach, but it would come through the test series dented, discolored and sooty from the fire, and mechanically weakened. The tests are designed to destroy the cask. Full-size and smaller scale prototype casks and critical features of the cask, like pressure welds, are tested. New cask design and structure are also compared to the design and structure of similar certified casks, and applicants for certification are required to show the results of tests with prototypes. New cask designs are modeled using models that have been benchmarked by physical tests; modeling is described in Chapters 3 and 4. Physical testing of prototypes and components, comparison with existing certified casks, and modeling by benchmarked thermal and structural models are all used to determine that a cask meets the test requirements

NRC regulations allow release of certain amounts of each radionuclide and certain radioactive emissions from Type B casks in the event of an accident. Releases of a number of radionuclides are allowed by 10 CFR 71.51. The regulation also allows an external radiation dose of 0.01 Sv per hour (one rem per hour).

1.4 Selection of Casks

Past generic risk assessments for the transportation of spent fuel have used generic casks with features similar to real casks, but generally without all of the conservatisms that are part of real cask designs. In this effort, we performed the generic risk assessment 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 sub-set of casks to be used was chosen. 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 chosen casks.

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 are no longer used, but still had valid certificates. Those below the heavy line are more modern and additional casks of these designs could be built. The casks for use in this study came from this last 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 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) will be used to evaluate truck shipments, but detailed finite element analyses of this cask will not be performed. The complete Certificates of Compliance (as of April 12, 2010) for each of these casks 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 are five casks in the group that use lead as their gamma shielding, of this group only the NAC-STC can transport fuel that is not contained within an inner welded canister. The HI-STAR 100 rail cask is chosen because it is the only all-steel cask in the group that is certified for transport of fuel in an inner welded canister. The GA-4 truck cask is chosen because it has a larger capacity than the NAC-LWT, and therefore is more likely to be used in any large transportation campaign.

The choice of rail casks allows 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.

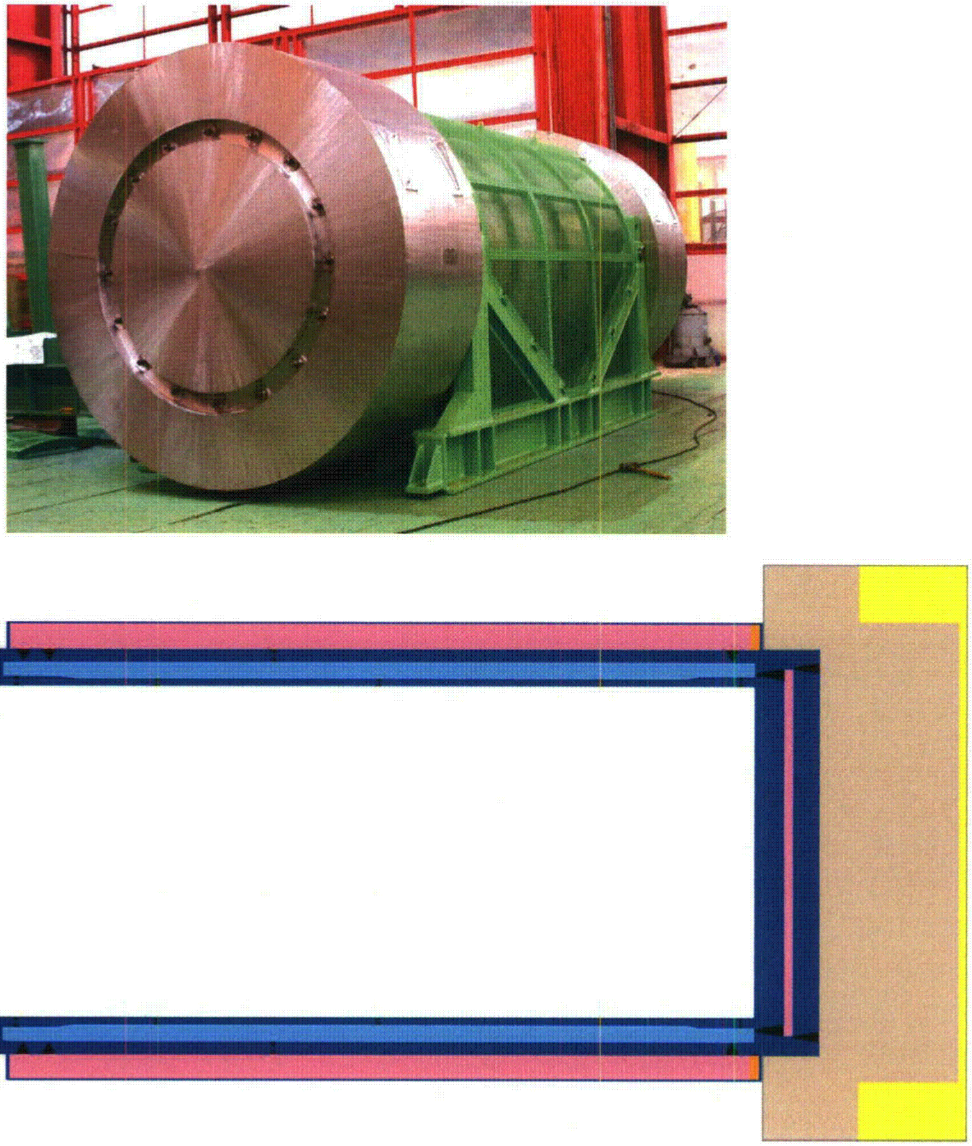

Figure 1-2. NAC-STC cask (courtesy of NAC International)

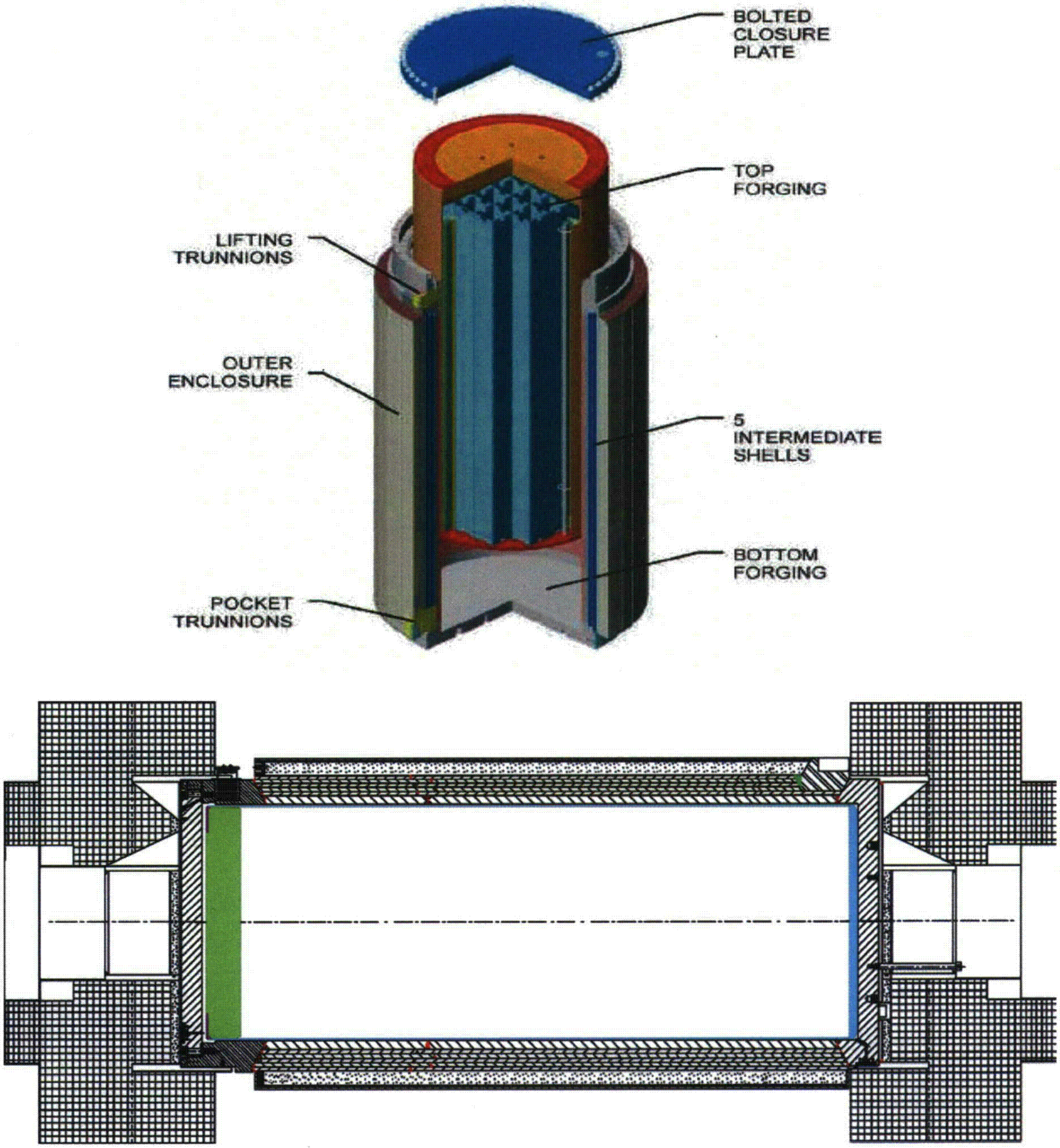

Figure 1-3. Basic layout of the HI-STAR 100 rail transport cask (from Haire and Swaney, 2005, and Holtec International, 2004)

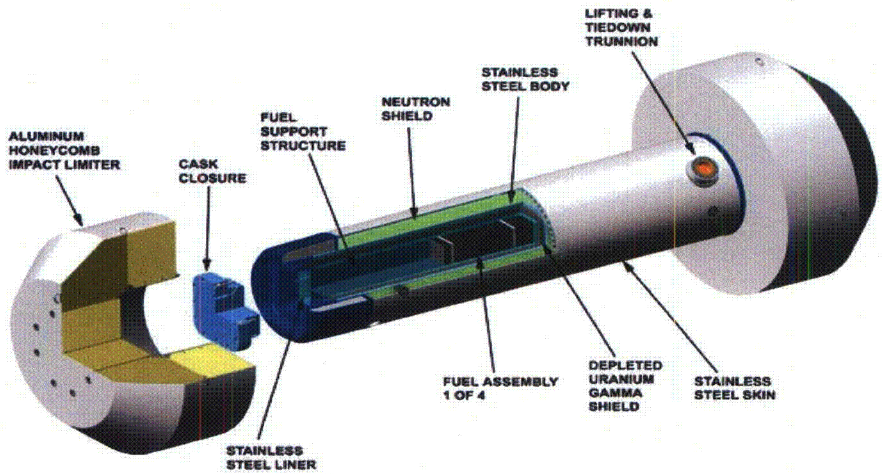

Figure 1-4. GA-4 cask (courtesy of General Atomics)

The NAC-STC rail cask is chosen because of the flexibility of its certificate of compliance. This cask can be used for both directly loaded fuel and for canistered fuel and is certified with either elastomeric o-rings or metallic ones. The HI-STAR 100 cask is chosen because it is the more modern of the two all-steel walled casks. The GA-4 cask is chosen because it has a higher capacity and has depleted uranium shielding. The chosen casks include all three of the most common shielding options; lead, depleted uranium, and steel

1.5 Organization of this Report

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.

1.5.1 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.

1.5.2 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 packages deliver an external dose, which is virtually entirely a gamma dose. In most studies to date the regulatory maximum dose rate, 10 mrem/hour at 2 meters from the cask, was assumed to be the external dose rate from every intact cask evaluated in the particular study. The present study uses the external dose rate from commercial certified casks as reported in the Safety Analysis Reports of those casks.

1.5.3 Chapter 3 and Appendix III

Chapter 3 and Appendix III address the structural analyses used to determine the cask response to these accidents and the parameters that determine loss of lead gamma shielding and releases of radioactive material. The results of detailed analyses of impacts 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.

1.5.4 Chapter 4 and Appendix IV

Chapter 4 and Appendix IV address the thermal analyses used to determine the cask response to these accidents and the parameters that determine loss of lead gamma shielding and 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.

1.5.5 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, and an essentially undamaged conveyance sits for many hours at the accident location.

1.5.6 Chapter 6 and Appendix VI

Chapter 6 summarizes the results of the analyses. Appendix VI includes a comparison between NUREG-0170 (NRC, 1977), the Modal Study (Fischer et al., 1987), NUREG/CR-6672 (Sprung et al., 2000) and this study.

1.5.7 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.

CHAPTER 2

RISK ANALYSIS OF ROUTINE TRANSPORTATION

2.1 Introduction

As described in Chapter 1, an ongoing study of the environmental impact of transporting radioactive materials is needed. NUREG-0170 (NRC, 1977) was the first comprehensive assessment of the environmental and health impact of transporting radioactive materials, and 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, not much 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[3] transportation of spent nuclear fuel. RADTRAN was initially developed by 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.[4]

RADTRAN 6.0, integrated with the input file generator RADCAT, (Neuhauser et al., 2000;[5] 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.

This chapter discusses the risks to the public and workers when the casks containing spent fuel are undamaged and the transportation of the fuel takes place without incident. Non-radiological vehicular accident risk, which is orders of magnitude larger than the radiological transportation risk, is not discussed 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.

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[3] The term "routine transportation" is used throughout this document to mean "incident-free" transportation, in order to avoid burdening the reader with RADTRAN terms of art.

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

[5] 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.

- Radiation doses from a single routine shipment to:

  Members of the public who live along the transportation route and near stops

  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. Like all radiation, ionizing radiation moves in straight lines. 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 extremely 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 exceedingly 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.

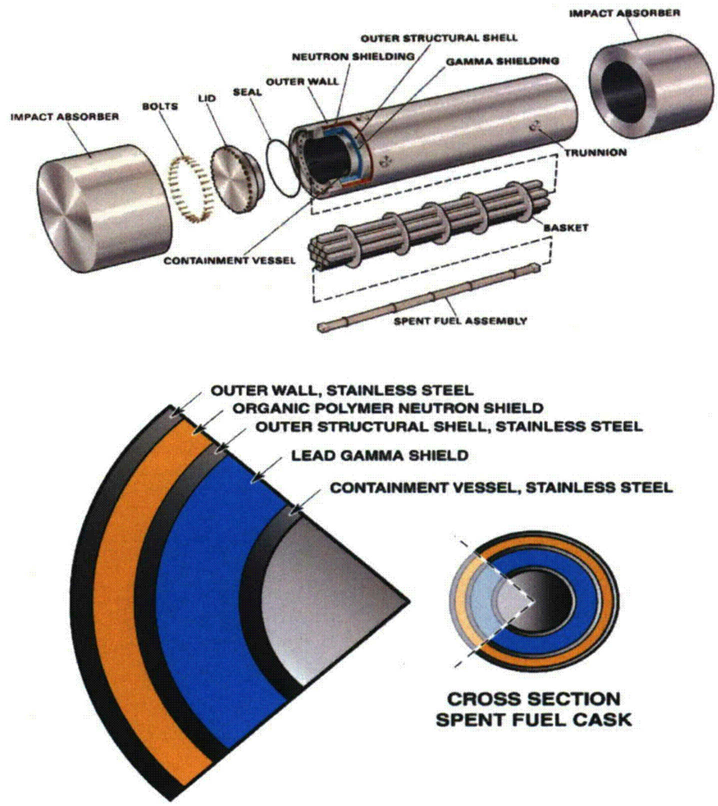

Figure 2-1. The upper figure is an exploded view of a generic spent fuel cask. The lower figure is a cross-section of the layers of the cask wall. (Sandia National Laboratories archive)

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 emission of gamma and neutron radiation at a very low dose rate. For spent uranium-based fuel, the allowed dose rate is almost entirely due to 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—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),[6] 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 average 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

[6] Recent increased diagnostic use of ionizing radiation, as in computerized tomography, has suggested increasing the average background to 0.006Sv (600 mrem).

71.4)), or about 0.00014 Sv/hour (14 mrem per hour) at one meter from a cask four to five meters long.

The measured external radiation doses from the casks in this study (Figures 1-3 to 1-5) are shown in Table 2-1. Measured values for the Truck-DU cask were not available, but it was assumed to meet the NRC standard of 10 CFR Part 71 (Holtec, 2004; NAC, 2004, General Atomics, 1998).

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

	Truck-DU	Rail-Pb	Rail-All 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 radiation dose to workers and members of the public from a routine shipment is based only on the external dose from the spent fuel cask, and not on the radioactive content of the spent fuel in the cask. Doses from the external radiation from the vehicle therefore depend on the distance of the receptor from the vehicle and on the exposure time, as well as on the external dose rate.

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 vehicle as a sphere with a radiation source at its center. The emission rate of the radiation source is the dose rate in Sv/hour (mrem/hour) at one meter from the cask, which NRC identifies as the transport index (TI). The TI is modeled as a virtual source at the center of the sphere, as shown in Figure 2-2. The diameter of this spherical model, called the "critical dimension," is the longest dimension of the actual spent fuel cask.

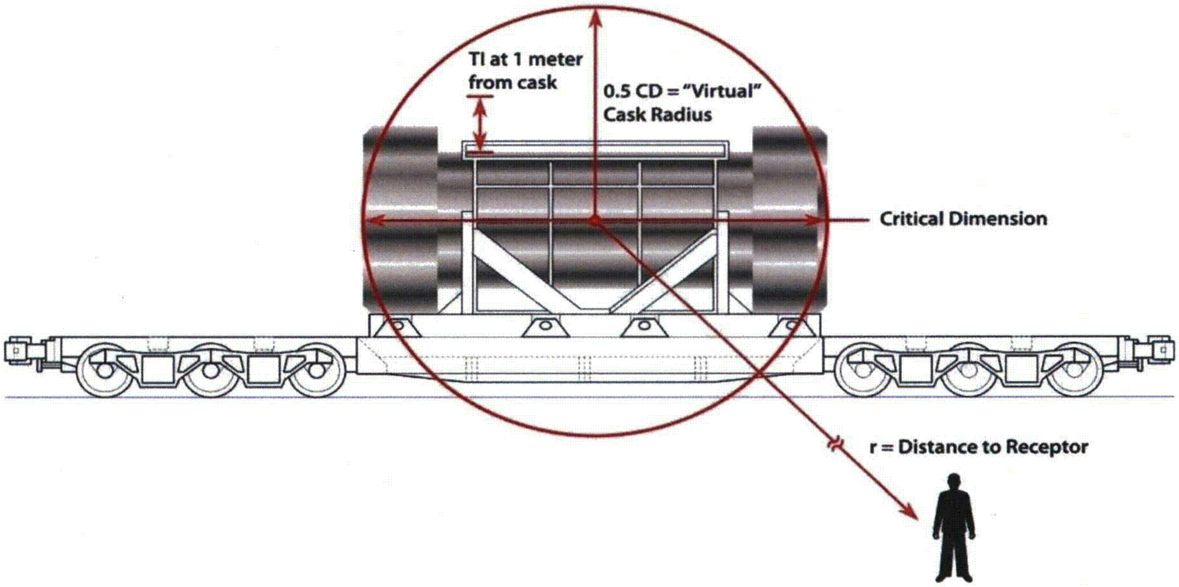

Figure 2-2. RADTRAN model of the vehicle in routine, incident-free transportation. The cask in this diagram is positioned horizontally, and the critical dimension is the cask length.

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 and the distance from the cask, as well as the cask's external radiation. When the vehicle carrying the cask is traveling along the route, the faster the vehicle goes, the less anyone along the vehicle's route is exposed. Therefore, an individual member of the public gets 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 (about 100 feet) are assumed for maximum exposure.[7] Table 2-2 shows the dose to an individual member of the public under these conditions. These doses are about the same as one minute of average background: 6.9×10^{-9} Sv (6.9×10^{-4} mrem).

[7] 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.

Table 2-1. Maximum individual in-transit doses

Package	Dose
Rail-Pb (rail)	5.7E-09 Sv (5.7E-04 mrem)
Rail-All Steel (rail)	4.3E-09 Sv (4.3E-04 mrem)
Truck - DU (truck)	6.7E-09 Sv (6.7E-04 mrem)

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-mrem). A collective dose, sometimes called a population dose, is essentially an average individual dose multiplied by the number of people exposed.[8] As shown in Figure 2-3, RADTRAN calculates collective doses along transportation routes by integrating over the width of a band along the route where the population resides and then integrating along the route (the *r* in Figure 2-2). Collective doses to people on both sides of the route are included. The exposed population is in a band 770 meters (about a half mile) on either side of the route: from 30 meters (10 feet) from the center of the route to 800 meters.

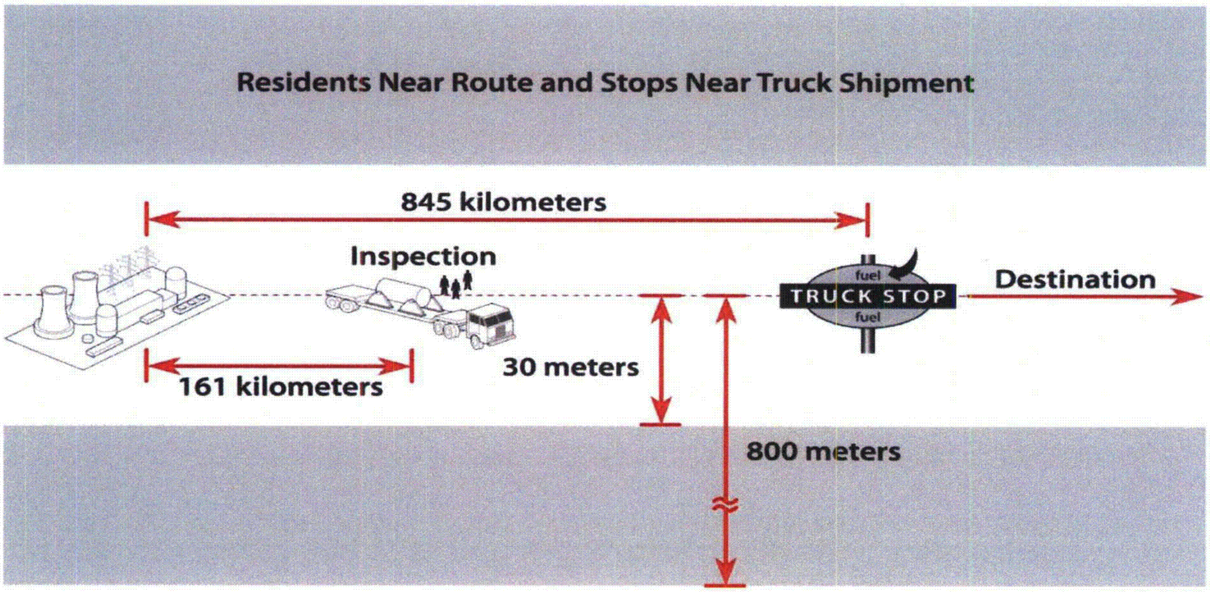

Figure 2-3. Diagram of a truck route as modeled in RADTRAN (not to scale)

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.

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

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 are part of the dose calculation by RADTRAN. References for these parameter values are in the Table 2-3 footnotes.

Table 2-2. Characteristics of rural, suburban, and urban routes used in RADTRAN

	Highway			Rail		
	Rural	Suburban	Urban	Rural	Suburban	Urban
Population density per km^2 (per mi^2)[a]	0 to 54 (0 to 139)	54 to 1286 (139 to 3326)	>1286 (>3326)	0 to 54 (0 to 139)	54 to 1286 (139 to 3326)	>1286 (>3326)
Nonresident/ resident ratio[b]	NA	NA	6	NA	NA	6
Shielding by buildings[b]	0	13%	98.2%	0	13%	98.2%
U.S. average vehicle speed[c] kph (mph)[c,d]	108 (67)	108 (67)	101(63)	40 (27)	40 (27)	24 (15)
U.S . average vehicles per hour[e,f]	1119	2464	5384	17	17	17
Occupants of other vehicles[e],[g]	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,

[e] Weiner, et al. 2009, Appendix D, [f]DOT, 2009; these are average railcars per hour, [g]DOT, 2008, Table 1-11.

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.

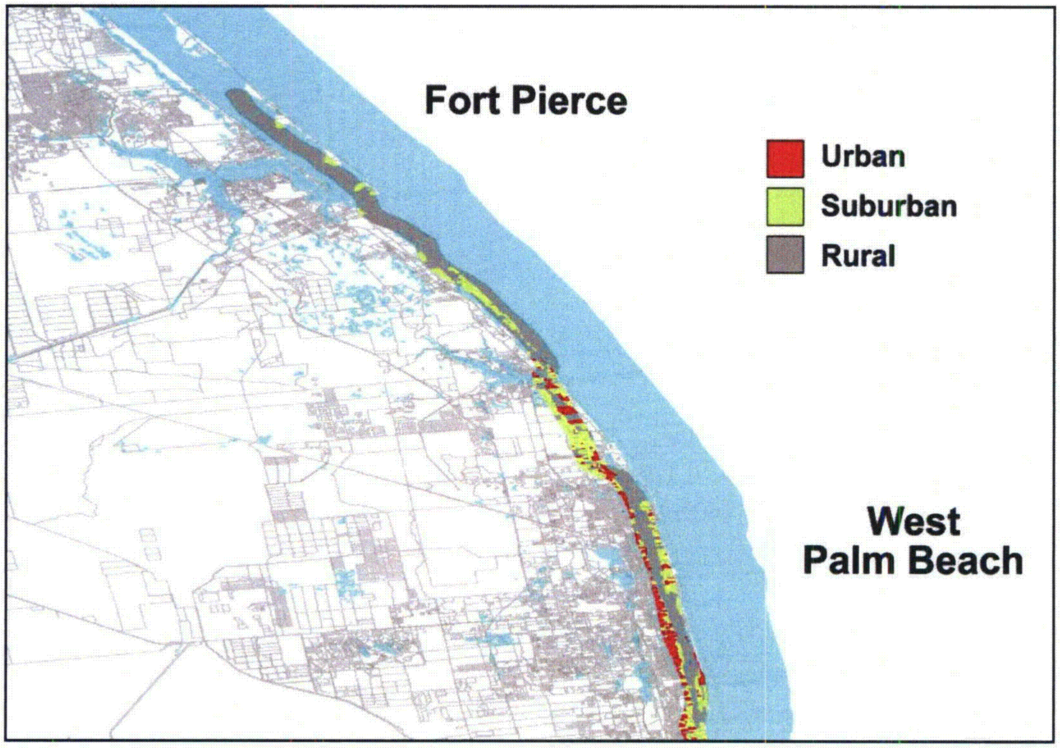


Figure 2-4. A segment of U.S.1 along the Florida coast (courtesy of G. Scott Mills)

Figure 2-4 shows a segment of U.S.1 along the Florida coast from West Palm Beach to Fort Pierce. The broad stripe along the coastline is the half-mile band on either side of the coastal highway. The red areas are urban populations, the blue areas are suburban, and the gray 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. Table 2-4 shows this WebTRAGIS output for a rail route from Kewaunee Nuclear Plant, WI to Oak Ridge National Laboratory.

Table 2-3. Route segment lengths and population densities, Kewaunee NP to ORNL

State	Kilometers (miles)			Persons/km^2 (persons/mi^2)		
	Rural	Suburban	Urban	Rural	Suburban	Urban
Illinois	12 (7.5)	63 (39)	45 (28)	26 (67)	504 (1305)	2593 (6710)
Indiana	171 (106)	51 (32)	11 (6.6)	17 (44)	351 (909)	2310 (5977)
Kentucky	254 (158)	84 (52)	13 (7.8)	17(45)	312 (806)	2532 (6551)
Ohio	201 (125)	117 (73)	29 (18)	15 (38)	402 (1041)	2243 (5802)
Tennessee	56 (35)	23 (14)	1 (0.6)	17 (44)	330 (855)	2084 (5392)
Wisconsin	148 (92)	92 (57)	28 (17)	18 (46)	434 (1124)	2410 (6234)

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 are reported as person-Sv.

Collective doses were calculated for one shipment over each of sixteen routes. These routes represent a variety of route lengths and populations, and are typical of routes that these trucks travel. Collective doses depend on route length and on the populations along the route.

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 a National Laboratory (Idaho National Laboratory). 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 a National Laboratory site (Oak Ridge, TN; ORNL). The routes modeled are shown in Table 2-5. Both truck and rail versions of each route are analyzed.

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 2008 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 2-4. Specific routes modeled. Urban kilometers are included in total kilometers.

Origin	Destination	Population within 800 m (1/2 mile)		Kilometers		Urban Kilometers	
		Rail	Truck	Rail	Truck	Rail	Truck
Maine Yankee Site, ME	Hanford, WA	1,146,479	980,355	5051	5011	235	116
	Deaf Smith County, TX	1,321,023	1,248,079	3360	3593	210	164
	Skull Valley, UT	1,199,091	934,336	4248	4173	235	115
	Oak Ridge, TN	1,119,154	1,336,208	2124	1747	161	135
Kewaunee NP, WI	Hanford, WA	779,613	419,951	3026	3451	60	57
	Deaf Smith County, TX	677,072	418,424	1881	2145	110	60
	Skull Valley, UT	472,098	354,911	2753	2619	125	51
	Oak Ridge, TN	806,116	522,128	1394	1272	126	92
Indian Point NP, NY	Hanford, WA	1,146,246	751,189	4779	4512	228	97
	Deaf Smith County, TX	1,027,974	376,259	3071	3071	204	207
	Skull Valley, UT	956,210	705,170	3975	3671	229	97
	Oak Ridge, TN	1,517,759	464,070	1263	1254	207	60
Idaho National Lab, ID	Hanford, WA	593,681	107,325	1062	958	20	15
	Deaf Smith County, TX	298,589	310,351	1912	2290	40	52
	Skull Valley, UT	164,399	102,341	454	466	26	19
	Oak Ridge, TN	169,707	494,068	3304	3286	74	62

Table 2-6 and Table 2-7 present collective doses for trail and truck, respectively, for the sixteen routes. State by state collective doses are tabulated in Appendix II.

Table 2-5. Collective doses (person-Sv) for rail transportation

FROM	TO	Rail-Pb			Rail-All Steel		
		Rural	Suburban	Urban	Rural	Suburban	Urban
MAINE YANKEE	ORNL	2.5E-05	3.0 E-04	1.4 E-04	1.9 E-05	2.2 E-04	1.1 E-05
	DEAF SMITH	3.3 E-05	3.9 E-04	1.9 E-04	2.3 E-05	2.8 E-04	1.4 E-05
	HANFORD	3.8 E-05	4.1 E-04	2.1 E-04	2.9 E-05	3.0 E-04	1.5 E-05
	SKULL VALLEY	4.2 E-05	4.2 E-04	1.1 E-04	3.2 E-05	2.9 E-04	1.1 E-05
KEWAUNEE	ORNL	1.7 E-05	1.7 E-04	1.1 E-04	1.0 E-05	1.1 E-04	8.1E-06
	DEAF SMITH	1.3 E-05	1.5 E-04	9.2E-05	8.0E-06	9.6 E-05	7.0E-06
	HANFORD	1.6 E-05	1.5 E-04	4.7E-05	1.2 E-05	2.7 E-05	4.0 E-06
	SKULL VALLEY	2.3 E-05	1.9 E-04	1.1 E-04	1.5 E-05	9.3 E-05	8.0E-06
INDIAN POINT	ORNL	1.2 E-05	2.3 E-04	2.3 E-04	9.0E-06	1.7 E-04	1.7 E-05
	DEAF SMITH	2.7 E-05	2.8 E-04	1.9 E-04	2.0 E-05	2.0 E-04	1.4 E-05
	HANFORD	3.2 E-05	3.4 E-04	2.1 E-04	2.4 E-05	1.3 E-04	1.4 E-05
	SKULL VALLEY	3.6 E-05	3.2 E-04	2.1 E-04	2.9 E-05	2.1 E-04	2.1 E-05
IDAHO NATIONAL LAB	ORNL	2.8 E-05	1.8 E-04	6.0 E-05	2.2 E-05	1.4 E-04	4.5E-06
	DEAF SMITH	1.1 E-05	9.4 E-05	4.0 E-05	3.6 E-05	7.1 E-05	2.8 E-06
	HANFORD	8.5 E-06	4.7 E-05	1.9 E-05	1.7 E-05	5.6 E-06	1.4 E-06
	SKULL VALLEY	4.9 E-06	4.1 E-05	2.5 E-05	2.3 E-05	2.4 E-05	1.8 E-06

Table 2-6. Collective doses (person-Sv) for truck transportation (1 Sv = 10^5 mrem)

FROM	TO	Truck - DU			
		Rural	Suburban	Urban	Urban Rush Hour[a]
MAINE YANKEE	ORNL	7.9E-06	1.4E-04	2.9E-06	6.5E-08
	DEAF SMITH	1.4E-05	1.9E-04	3.3E-06	7.3E-08
	HANFORD	2.2E-05	1.7E-04	2.3E-06	5.2E-08
	SKULL VALLEY	1.8E-05	1.5E-04	2.3E-06	5.2E-08
KEWAUNEE	ORNL	6.5E-06	7.4E-05	1.8E-06	4.0E-08
	DEAF SMITH	1.1E-05	6.3E-05	1.2E-06	2.6E-08
	HANFORD	1.4E-05	6.6E-05	1.1E-06	2.5E-08
	SKULL VALLEY	1.2E-05	5.0E-05	1.1E-06	2.3E-08
INDIAN POINT	ORNL	6.1E-06	8.9E-05	1.2E-06	2.6E-08
	DEAF SMITH	1.1E-05	1.1E-04	1.6E-06	3.4E-08
	HANFORD	2.1E-05	1.2E-04	1.9E-06	4.2E-08
	SKULL VALLEY	1.7E-05	1.1E-04	1.9E-06	4.2E-08
IDAHO NATIONAL LAB	ORNL	1.4E-05	8.4E-05	1.2E-06	2.7E-08
	DEAF SMITH	7.3E-06	4.9E-05	1.1E-06	2.5E-08
	HANFORD	4.2E-06	2.0E-05	3.0E-07	6.6E-09
	SKULL VALLEY	2.0E-06	1.6E-05	4.3E-07	9.5E-09

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

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 Tables 2-6 and 2-7.

- Urban residents sustain a slightly 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.
- 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. In several cases shown in Table 2-5, the rail route had twice as many urban miles as the corresponding truck route.
- Any particular shipment campaign will need fewer rail shipments than truck shipments, because rail casks hold about six times as many fuel assemblies as truck shipments.

The collective doses shown in Table 2-6 and Table 2-7 are all very small. Moreover, they are not the only doses the people along the route receive. Background radiation is 0.0036 Sv per year

(360 mrem per year) in the U.S., or 4.1 x 10^{-7} Sv/ hour. The effect of a single shipment on the collective dose is illustrated by the following example of the Maine Yankee to ORNL truck route:

- From Table 2-7 the total collective dose for this segment 1.5 x 10^{-5} person-Sv
- From Table 2-5, there are 1.34 million people within a half mile of the route.
- Background is 4.1 x 10^{-7} Sv/hour. Everyone is exposed to this background all the time. .
- A truck traveling at an average of 108 kph travels the 1747 km in 16 hours.
- During those 16 hours, the 1.34 million people will have received a collective background dose of 8.81 person-Sv, about 600,000 times the collective dose from the shipment.
- The net collective dose to this 1.34 million people is not 1.5 x 10^{-5} person-Sv), but 8.81015 person-Sv.
- The NRC recommends that collective dose be used only in comparisons (NRC, 2008).
- The appropriate comparison between the collective dose from this shipment of spent fuel is thus not a comparison between 1.5 x 10^{-5} person-Sv from the shipment and zero dose if there is no shipment, but between 8.81015 person-Sv if there is a shipment and 8.81000 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.

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 passenger trains are represented by one persons in each passing railcar in rural and suburban areas, and five people in urban areas. The rural and about half of the 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 light rail, that share rail routes with freight trains.

Table 2-7. Collective doses (person-Sv) to occupants of trains sharing the route

SHIPMENT ORIGIN	SHIPMENT DESTINATION	Rail-Pb			Rail-All Steel		
		Rural	Suburban	Urban	Rural	Suburban	Urban
MAINE YANKEE	ORNL	5.3E-06	1.6E-05	1.1E-04	4.0E-06	1.2E-05	7.6E-06
	DEAF SMITH	1.0E-05	1.8E-05	1.4E-04	7.7E-06	1.4E-05	9.9E-06
	HANFORD	1.5E-05	2.2E-05	1.5E-04	1.2E-05	1.7E-05	1.1E-05
	SKULL VALLEY	1.3E-05	2.4E-05	1.2E-04	9.9E-06	1.9E-05	8.5E-06
KEWAUNEE	ORNL	3.7E-06	9.4E-06	8.5E-05	2.8E-06	7.1E-06	5.9E-06
	DEAF SMITH	6.4E-06	7.0E-06	7.4E-05	4.8E-06	5.3E-06	5.2E-06
	HANFORD	6.7E-06	9.0E-06	4.1E-05	5.0E-06	6.9E-06	2.8E-06
	SKULL VALLEY	9.4E-06	1.0E-05	8.5E-05	7.2E-06	7.9E-06	5.9E-06
INDIAN POINT	ORNL	2.5E-06	1.1E-05	1.4E-04	1.9E-06	8.2E-06	9.7E-06
	DEAF SMITH	9.8E-06	1.4E-05	1.4E-04	7.4E-06	1.1E-05	9.6E-06
	HANFORD	1.2E-05	1.9E-05	1.5E-04	8.8E-06	1.5E-05	1.1E-05
	SKULL VALLEY	5.9E-06	4.2E-05	7.1E-05	4.4E-06	3.2E-05	2.7E-05
IDAHO NATIONAL LAB	ORNL	4.0E-06	5.3E-05	5.5E-05	3.0E-06	4.0E-05	3.8E-06
	DEAF SMITH	7.3E-06	4.4E-06	2.7E-05	5.6E-06	3.3E-06	1.9E-06
	HANFORD	4.1E-06	2.3E-06	1.4E-05	3.1E-06	1.8E-06	9.4E-07
	SKULL VALLEY	1.5E-06	2.0E-06	1.7E-05	1.1E-06	1.5E-06	1.2E-06

Truck

Unlike the train situation, a truck carrying spent fuel shares the primary highway system with many cars, light trucks, and other vehicles, as shown in Figure 2-5, a model of the RADTRAN calculation. The occupants of any car or truck that passes the spent fuel cask in either direction will sustain a small radiation dose.

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. Although occupants of the vehicles that share the route are closer to the cask than residents or others beside the route, they are exposed to radiation from the cask for considerably less time because the vehicles involved are moving past each other.

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[9] (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.

---

[9] 2004 is the most recent year for which data have been validated.

• 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 had been assumed, the collective doses would have been one-third larger.

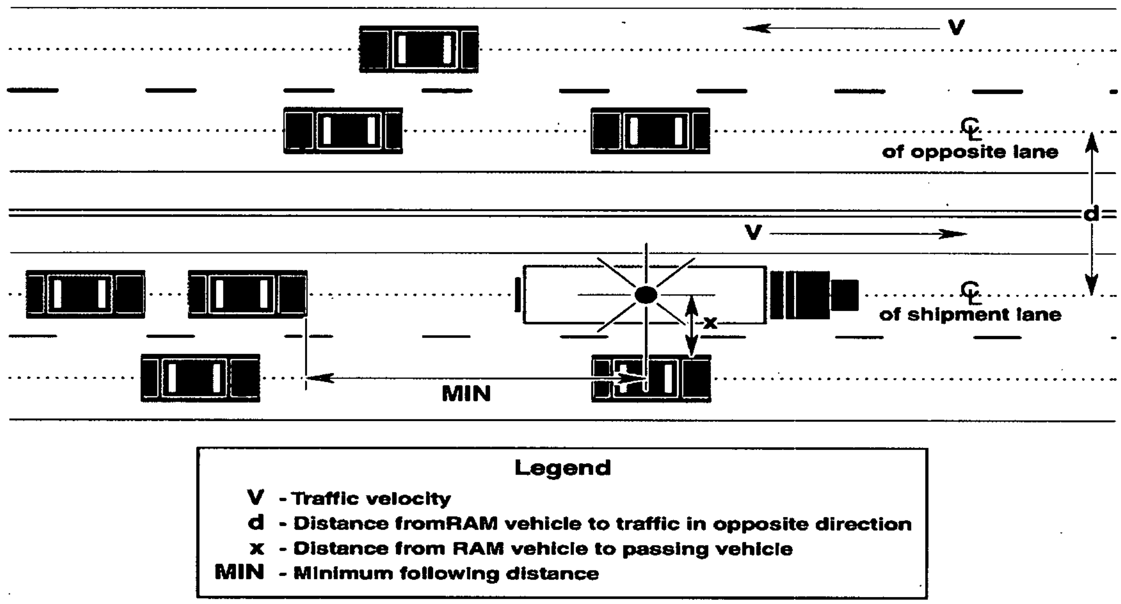


Figure 2-5. Diagram for calculating radiation doses to occupants of other vehicles (from Neuhauser et al., 2000)

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

Table 2-8. Collective doses (person-Sv) to occupants of vehicles sharing truck routes

ORIGIN	DESTINATION	TRUCK - DU			
		Rural	Suburban	Urban	Urban Rush Hour
MAINE YANKEE	ORNL	1.3E-04	2.3E-04	5.4E-05	5.0E-06
	DEAF SMITH	2.9E-04	3.6E-04	7.5E-05	1.5E-05
	HANFORD	4.4E-04	2.9E-04	4.1E-05	4.0E-06
	SKULL VALLEY	5.0E-04	2.8E-04	4.3E-05	4.0E-06
KEWAUNEE	ORNL	9.6E-05	1.4E-04	4.8E-05	4.0E-06
	DEAF SMITH	1.8E-04	8.9E-05	2.2E-05	2.0E-06
	HANFORD	3.4E-04	1.4E-04	3.3E-05	3.0E-06
	SKULL VALLEY	2.5E-04	8.6E-05	2.5E-05	1.0E-05
INDIAN POINT	ORNL	1.8E-04	2.1E-04	3.3E-05	3.0E-06
	DEAF SMITH	2.8E-04	3.1E-04	5.6E-05	5.0E-06
	HANFORD	4.2E-04	2.2E-04	4.8E-05	4.0E-06
	SKULL VALLEY	3.6E-04	2.2E-04	4.5E-05	4.0E-06
IDAHO NATIONAL LAB	ORNL	3.0E-04	1.5E-04	2.4E-05	2.0E-06
	DEAF SMITH	2.2E-04	7.3E-05	2.7E-05	1.8E-05
	HANFORD	1.0E-04	8.5E-05	9.0E-06	1.0E-06
	SKULL VALLEY	3.7E-05	2.3E-05	8.0E-06	1.0E-06

The collective doses to vehicle occupants on rural routes (Table 2-9), are an average of 25 times the collective doses to residents in rural areas (Table 2-7), even though the population in vehicles sharing the route is about the same as the resident population. The difference in collective dose is that vehicles on the road are much closer to the shipment than residents, particularly in rural areas. For suburban areas, the relationship is more complex. Suburban residents are further from the shipment, like rural residents, but the population in vehicles sharing the suburban routes is only one-tenth the resident population. Thus, as expected, the collective doses differ only by a smaller factor.

Collective doses to vehicle occupants in urban areas are about 25 times the collective doses to residents, except for rush hour. During rush hours, the doses to vehicle occupants, as modeled, are about 100 times the doses to residents, rather than 25 times. This factor of four reflects the way rush hours are modeled, which is:

- Ten percent of the time spent on urban routes is assumed to be during rush hours.
- Vehicle density during rush hours is assumed to be twice the vehicle density in urban areas at other times.
- Vehicle speed during rush hours is assumed to be half of the vehicle speed in urban areas at other times.

That is, rush hour traffic is assumed to be twice as heavy and traveling at half the speed, so that the collective dose is increased by a factor of four. Even with such extremely conservative assumptions, collective doses during rush hours are 10 percent of the urban collective doses or less. Collective doses to occupants of other vehicles are still very small.

2.3.4 Doses at 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 generally 20 to 30 hours long. Spent fuel casks may be carried on dedicated trains as well as on regular freight trains. A dedicated train is a train that carries a single cargo from origin to destination; coal unit trains are a good example of dedicated trains.

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
- Residents who live near the rail yard.

The semis that carry TRUCK - DU casks each have two 80-gallon fuel tanks, and generally stop to refuel when half of the fuel is gone, approximately every 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 casks 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 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 calculation may be found in Appendix II.

Figure 2-6 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.

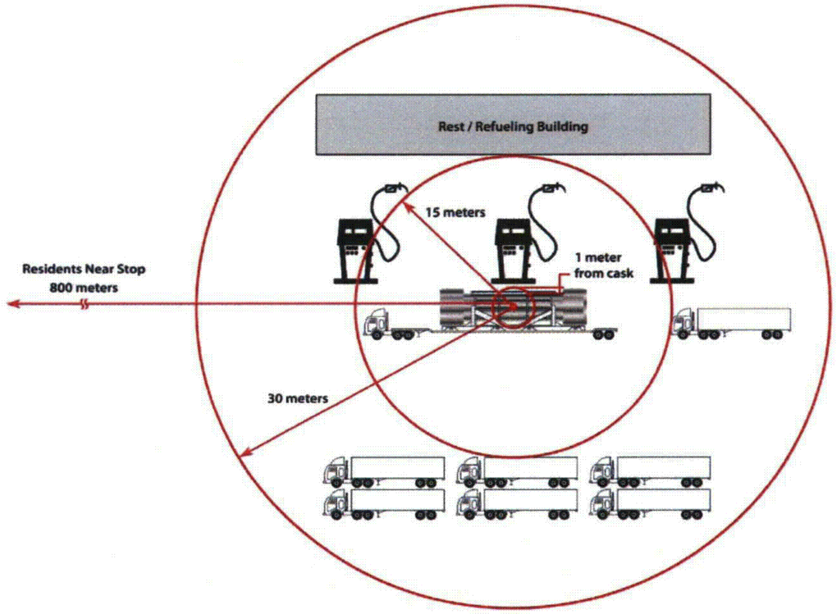


Figure 2-6. Diagram of truck stop model (not to scale)

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

Table 2-9. Some sample data for calculating doses at 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

Rail

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

- For the Rail-Pb: 1.5 x 10^{-5} person-Sv (1.5 person-mrem)
- For the Rail-All Steel: 1.1 x 10^{-5} person-Sv (1.1 person-mrem)

The average dose to an individual living 200 to 800 meters from a classification yard, as calculated by RADTRAN, is

- 0.35 x 10^{-5} Sv (0.35 mrem) from the Rail-Pb
- 0.27 x 10^{-5} Sv (0.27 mrem) from the Hi-STAR 100

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

Table 2-10. Collective doses at rail stops on the Maine Yankee-to-Hanford route (person-Sv)

Stop	Route type (R,S,U)	Time (hours)	Railyard worker		Residents near stop	
			Rail-Pb	HISTAR	NAC-STC	HISTAR
1	S, ME	4.0	2.2 E-05	1.6 E-05	3.4 E-05	2.6 E-05
2	R, NY	4.0	2.2 E-05	1.6 E-05	9.2 E-06	6.9 E-06
3	S, IL	2.0	1.1 E-05	8.1 E-06	1.2 E-04	9.4 E-05

Truck

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

Table 2-11. Collective doses to residents near truck stops (person-Sv)

Origin	Route	Type	Persons/km^2	Number of stops	Dose
MAINE YANKEE	ORNL	Rural	19.9	1.73	1.1E-06
		Suburban	395	2.09	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	1361.0	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

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 thus include one truck stop, which 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. In addition, the time that a truck crew can spend in the vehicle with a loaded cask is limited.

Occupational doses from routine, incident-free radioactive materials transportation include doses to truck and train crew, railyard workers, inspectors and escorts. Workers who handle spent fuel containers in storage, loading and unloading casks from vehicles or during intermodal transfer are not addressed in this analysis. Truck refueling stops in the U.S. no longer have attendants who refuel trucks.[10] Gas station and truck stop workers are in concrete or brick buildings and would be shielded from the radiation with the same shielding as in urban housing (83% shielded).

Table 2-13 summarizes the occupational doses.

Table 2-12. Occupational doses per shipment from routine incident-free transportation

Cask and route type	Rail crew: 3 people (person-Sv/hour)	Truck crew: 2 people; (person-Sv/hour)	Escort: (Sv/hour)	Inspector (Sv per inspection)	Truck stop worker (Sv/hour per stop)	Rail classification yard workers: (person-Sv) (see p. 16)
Rail-Pb rural/suburban	5.4E-09		5.8 E-06			1.5E-05
Rail-Pb urban	9.1E-08		5.8 E-06			
Rail-All Steel rural/suburban	4.1E-09		4.4 E-06			1.1E-05
Rail-All Steel urban	6.8E-09		4.4 E-06			
TRUCK - DU rural/suburban		3.8E-09	3.2E-09	3.2E-09	2.0E-09	
TRUCK - DU urban		3.6E-09	3.2E-09			

[10] The State of Oregon still requires gas station attendants to refuel cars and light duty vehicles, but heavy truck crew do their own refueling.

2.5 Unit Risk

RADTRAN, the model used for the calculation of transportation risk, multiplies numbers. The only calculation that RADTRAN makes which is not a simple multiplication is calculating emissions from the spherical model shown in Figure 2-2. All other parameters multiply the result of this calculation. RADTRAN can be programmed to calculate the collective dose from a passing vehicle for a population density of one person per square kilometer and one kilometer of a route. This type of calculation is called a unit risk calculation. The result may then be multiplied by the population per square kilometer and the route length in kilometers (if the area along the route is 800 meters wide on either side of the route), and divided by the vehicle speed.

2.6 Conclusions

As Chapter 1 states, risk is a projection, and a code that estimates risk can never be precise because the input data are themselves estimates and projections. The risk assessment code RADTRAN overestimates doses, and no estimate of dose can substitute for an actual measurement. 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. They are comparable to background and less than doses from many medical diagnostic procedures. The NRC recommends that collective doses (average doses integrated over a population) only 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. The collective dose is not zero in the absence of a shipment.

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 minimum material properties, allowing only small amounts of yielding, and requiring materials with high ductility. These approaches ensure that the casks will not only survive a 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 factors 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 excess capacity of those cask designs was included in the analyses.

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-Pb spent fuel transportation casks were determined using the non-linear transient dynamics explicit finite element 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 elements and structural elements, such as beams and shells.

In addition to the detailed analyses 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 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 study. Detailed response of the fuel assemblies was calculated using a sub-model of a single assembly.

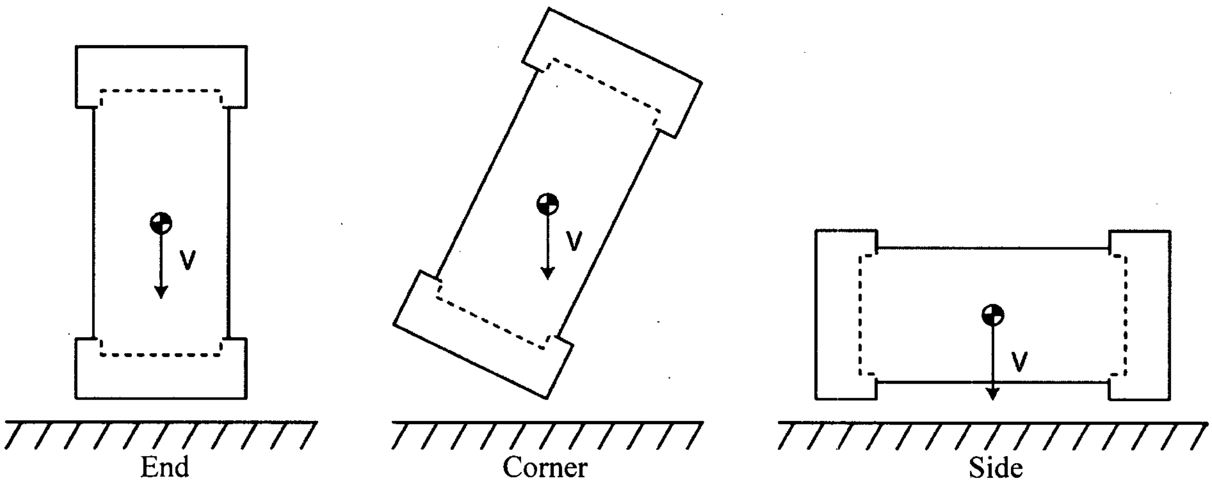


Figure 3-1. Impact orientations analyzed

3.2.1 Rail-Steel Cask

Finite element model

Figure 3-2 shows the overall finite element model of the Rail-Steel cask. 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 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.

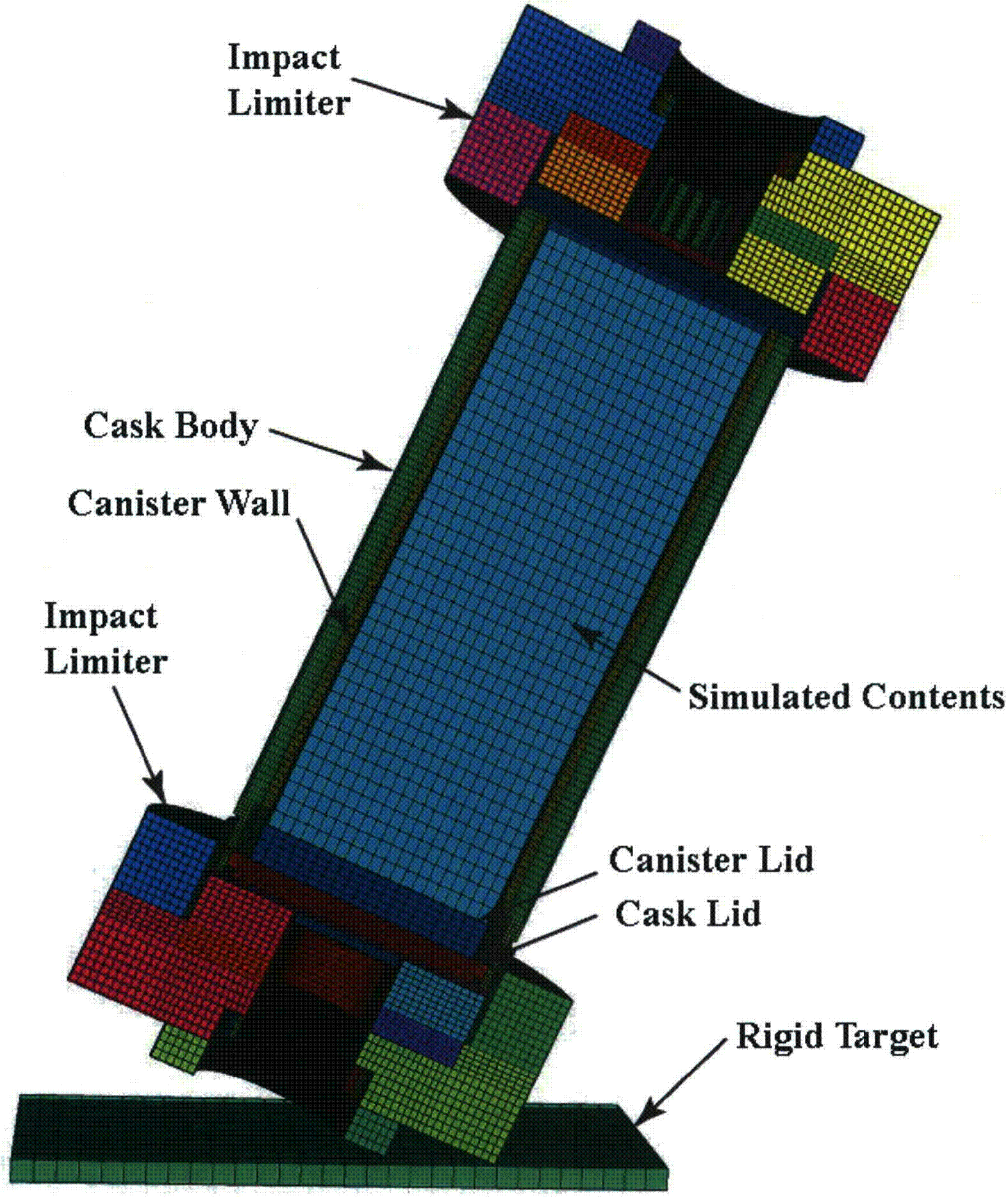
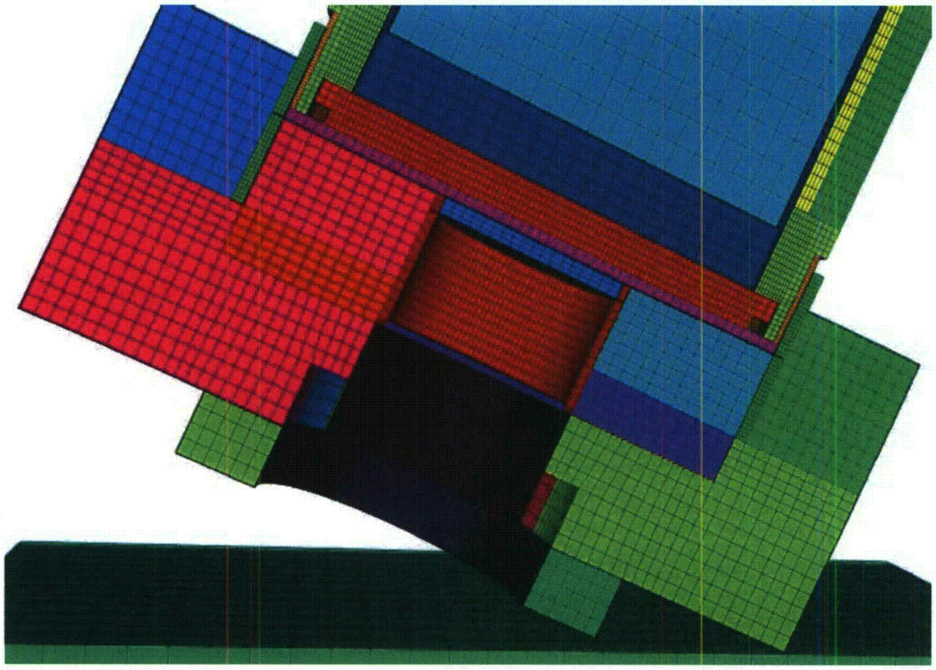
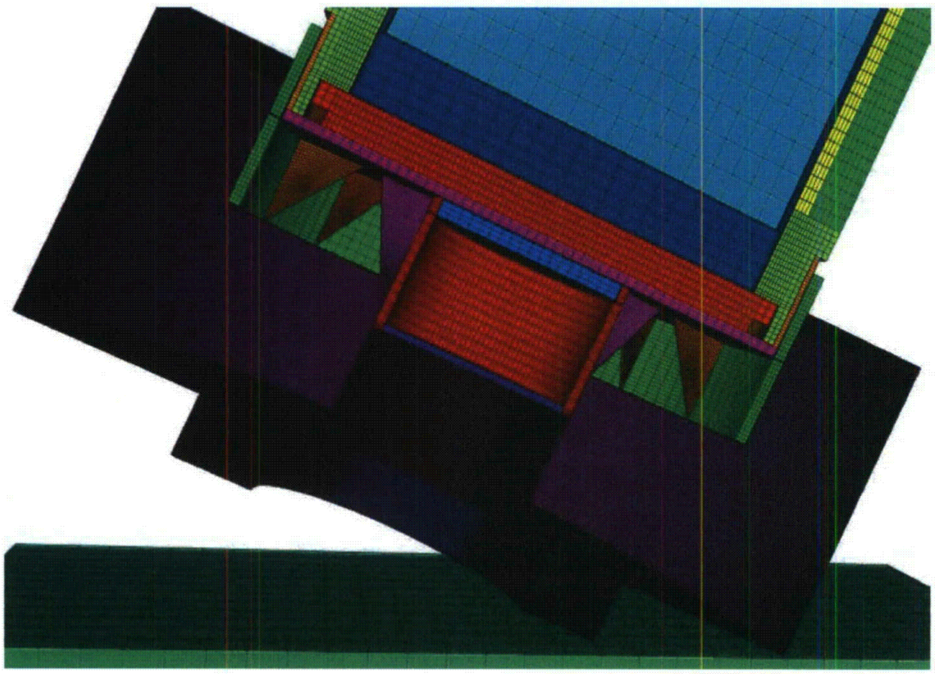


Figure 3-2. Finite element mesh of the Rail-Steel cask



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

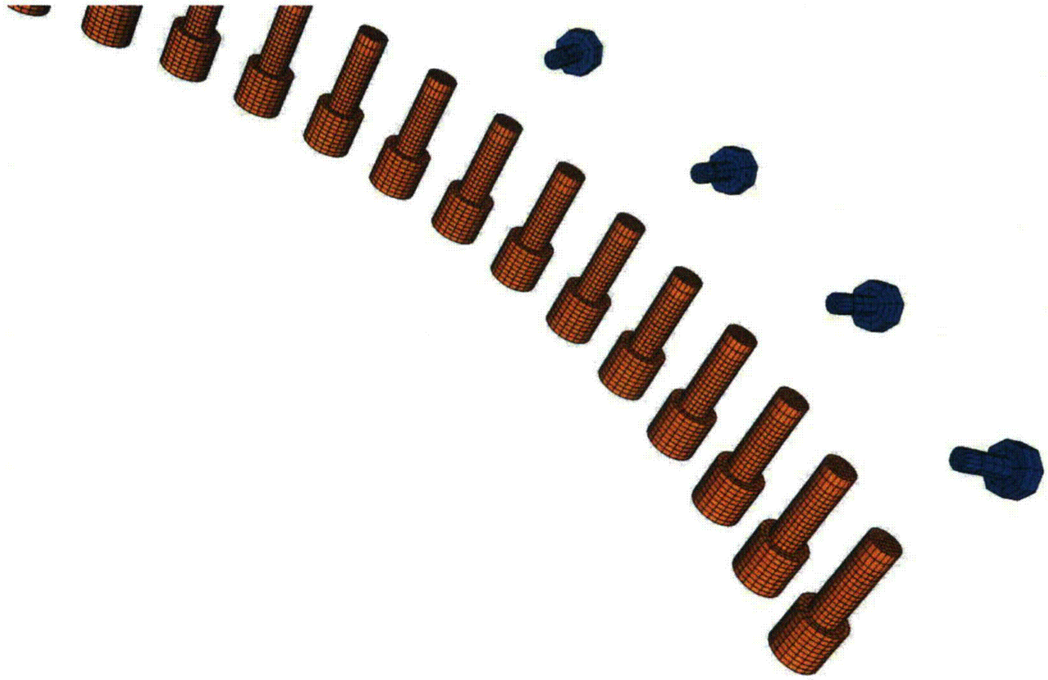


Figure 3-4. Finite element mesh of the Rail-Steel closure bolts and the closure end impact limiter attachment bolts. The highly refined mesh in these critical parts assures an accurate assessment of the closure response.

Analysis results

As expected, for all of 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 (permanent deformation) to the cask body or canister. As the impact velocity increases there is first 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 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 (discussion of the failure model is 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. 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. 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.

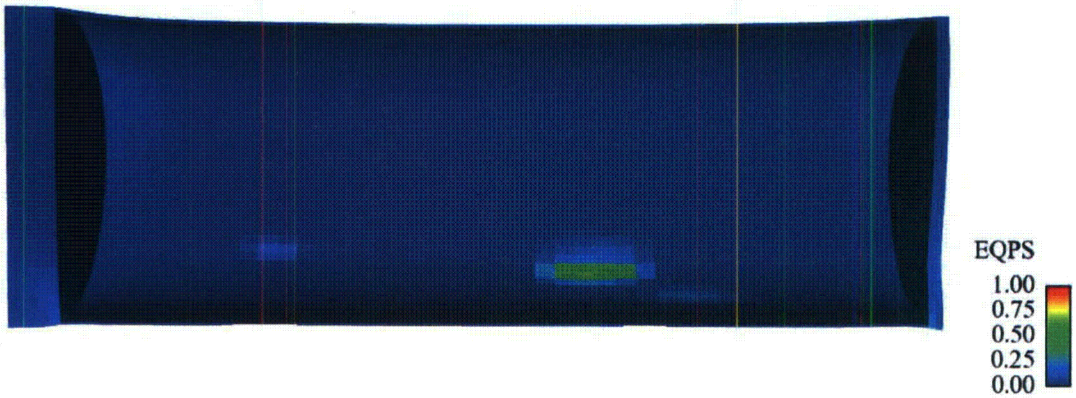


Figure 3-5. Plastic strain in the welded canister of the Rail-Steel for the 193 kph side impact case

3.2.2 Rail-Pb Cask

Finite Element Model

Figure 3-6 shows the overall finite element model of the Rail-Pb cask. 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 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 case . 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.

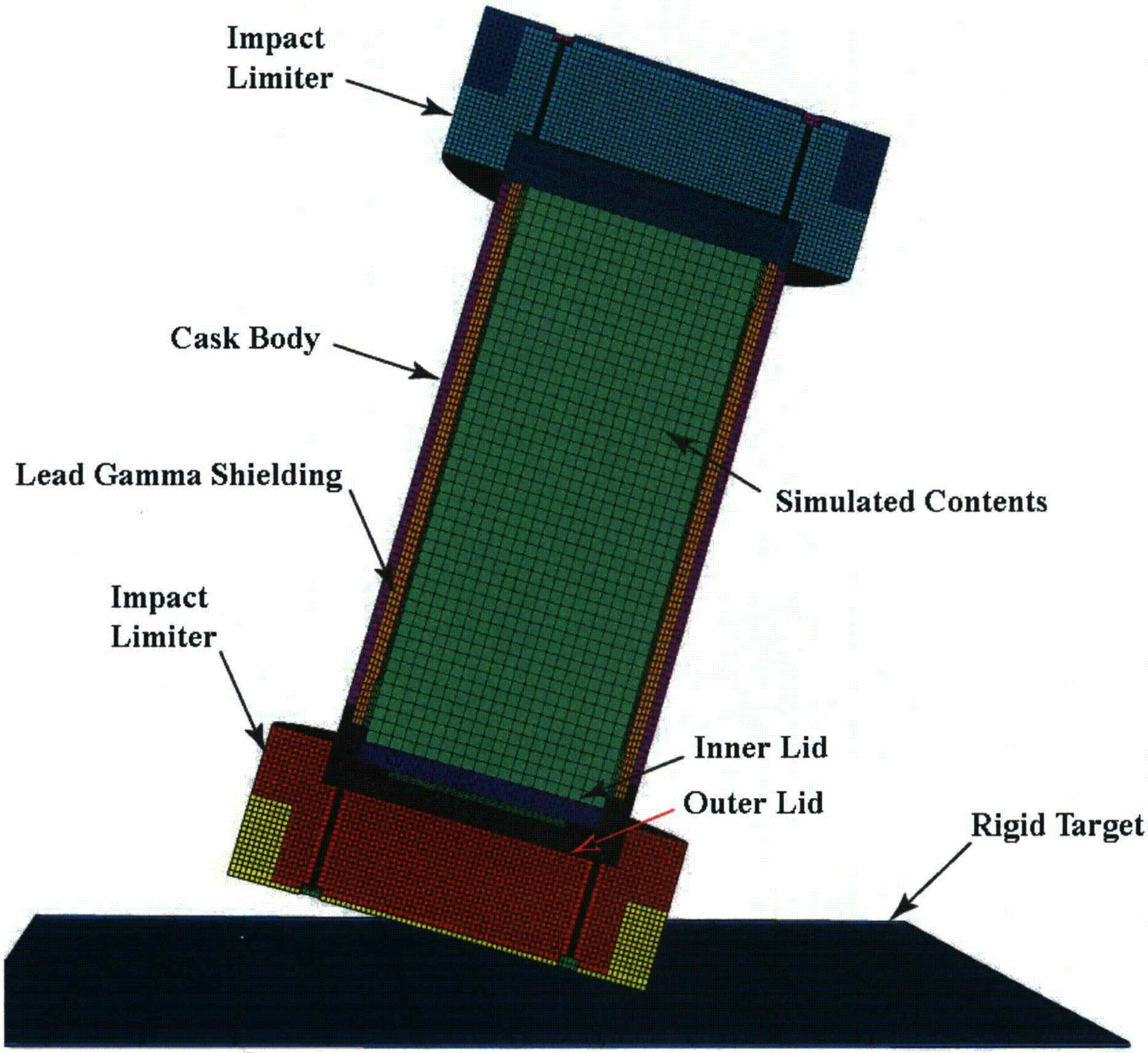
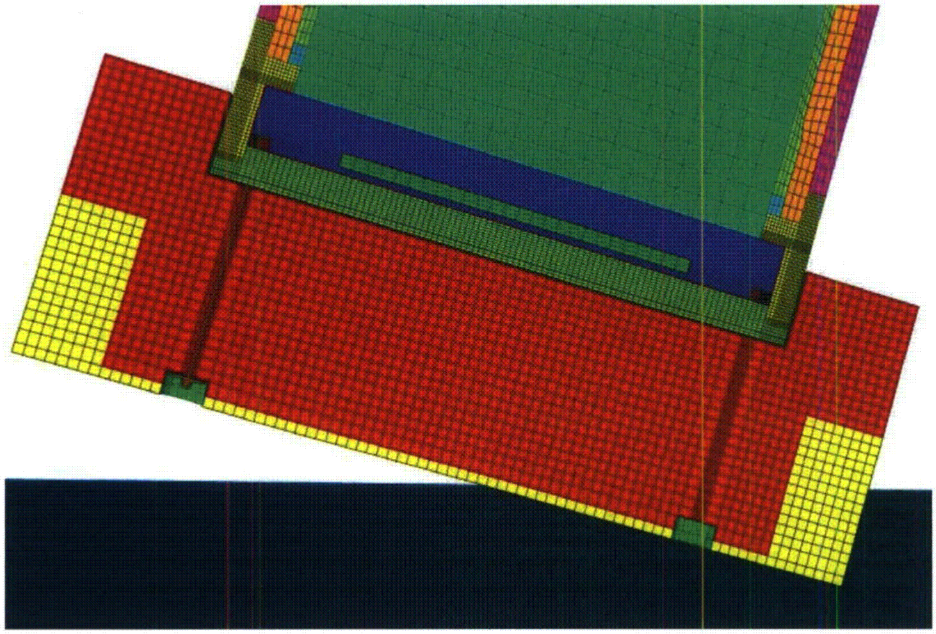
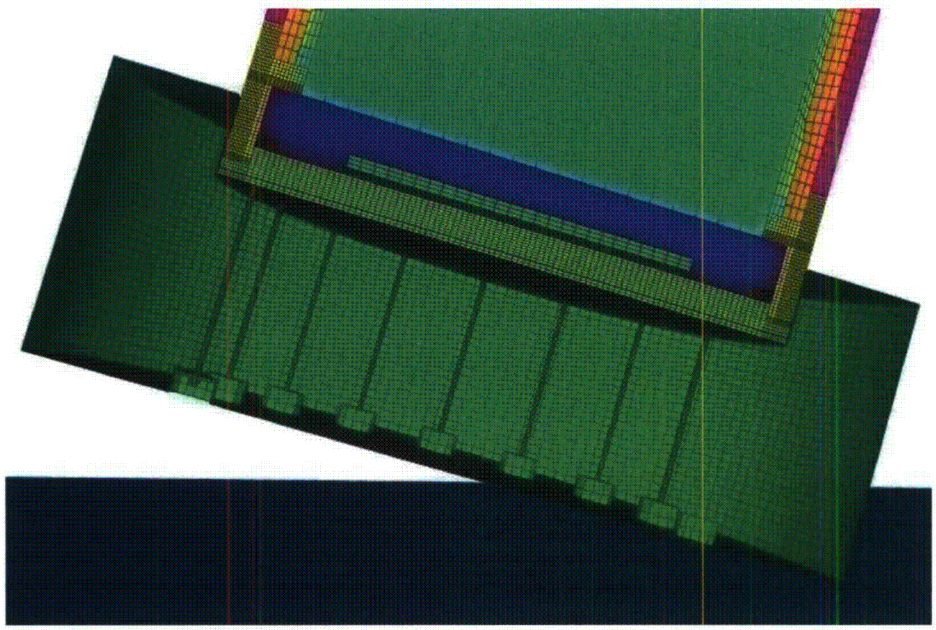


Figure 3-6. Finite element mesh of the Rail-Pb cask



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-Pb

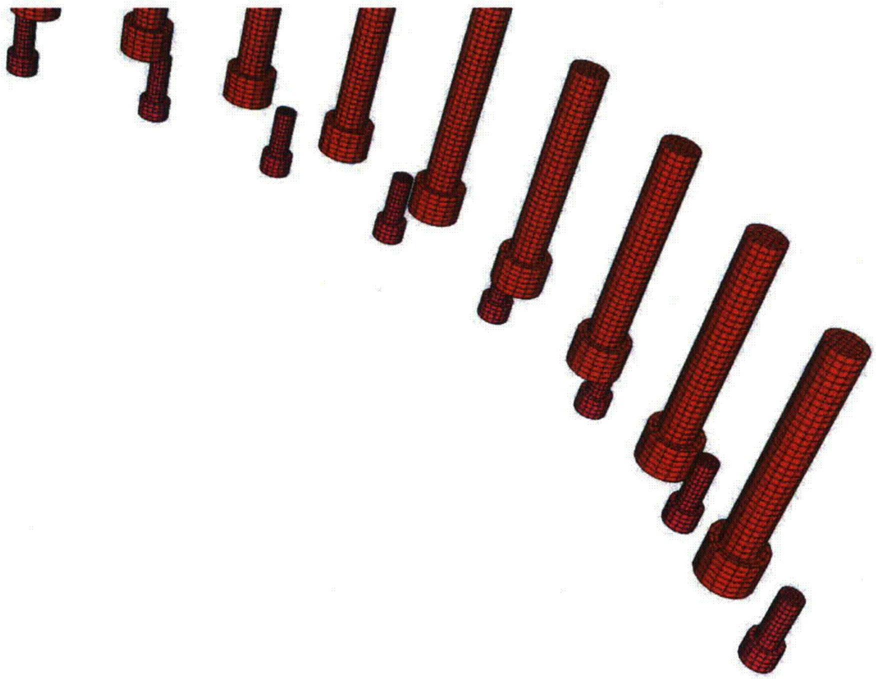


Figure 3-8. Finite element mesh of the Rail-Pb closure bolts for both the inner and outer lids. The longer bolts are for the inner lid and the shorter ones for the outer lid.

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-Pb cask is more complicated than that of the Rail-Steel cask. For the end orientation, as the impact velocity increases there is first 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-Pb 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.

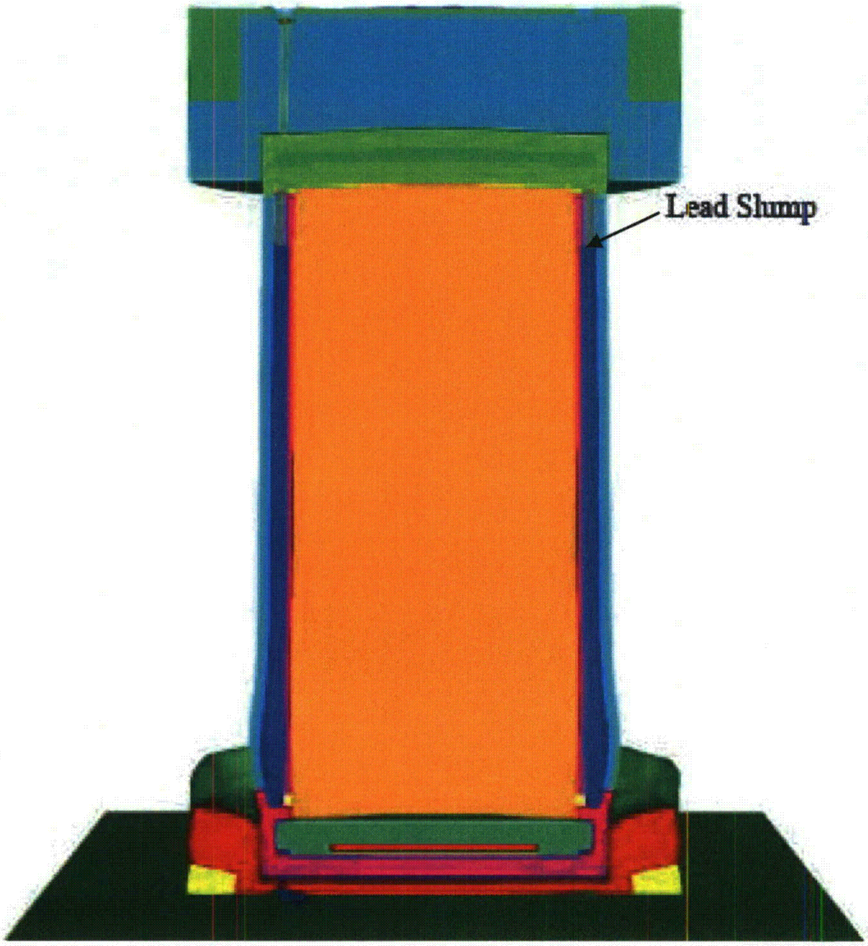


Figure 3-9. Deformed shape of the Rail-Pb cask following the 193 kph impact onto an unyielding target in the end-on orientation

Table 3-1. Maximum lead slump for the Rail-Pb 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.

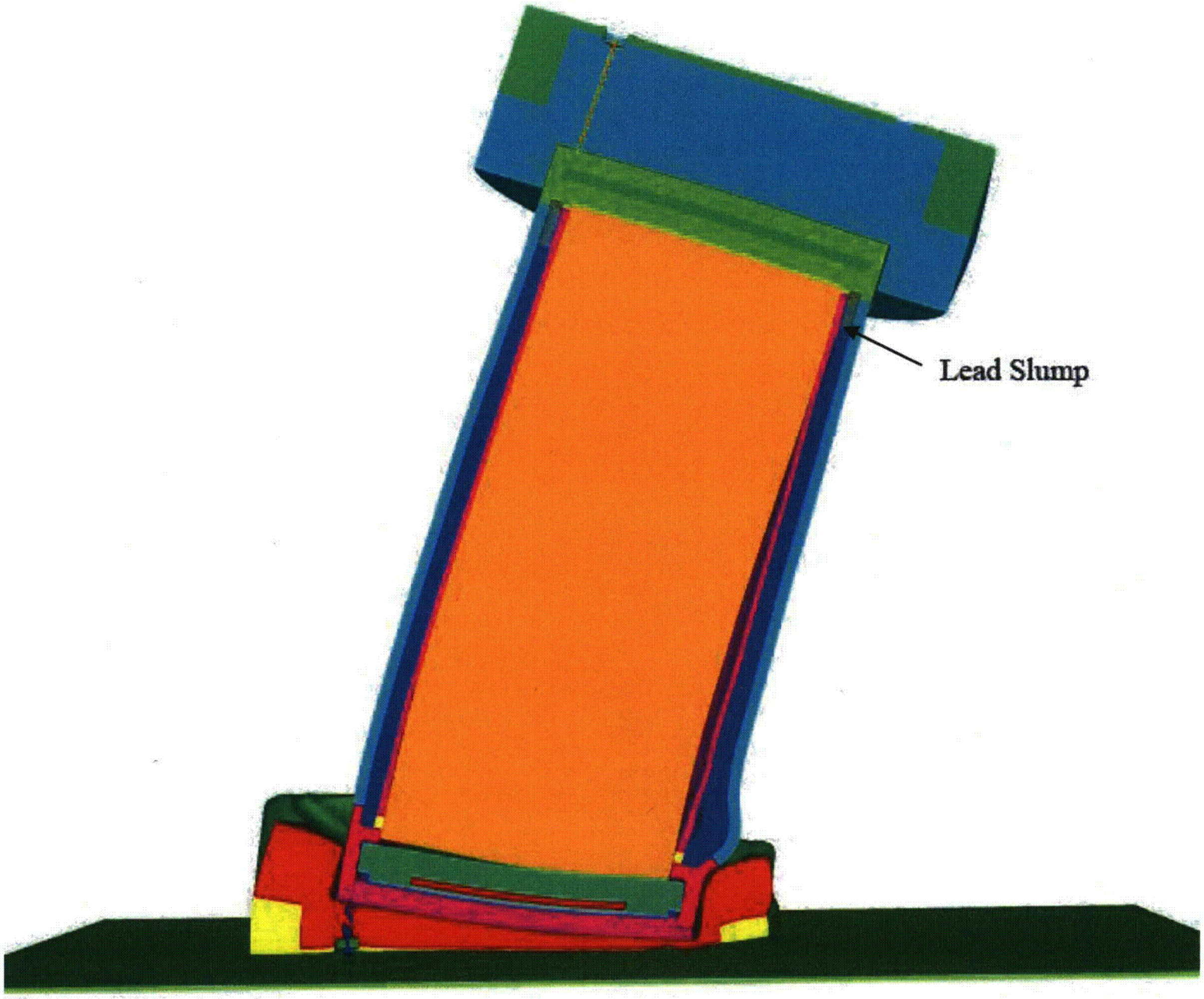


Figure 3-10. Deformed shape of the Rail-Pb following the 193 kph impact onto an unyielding target in the corner orientation

In the side impact as the impact velocity increases from 48 kph to 97 kph, the impact limiter ceases to absorb 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. 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, only the gaps between the lids and the cask are larger. Deformed shapes for all of the analysis cases are shown in Appendix III.

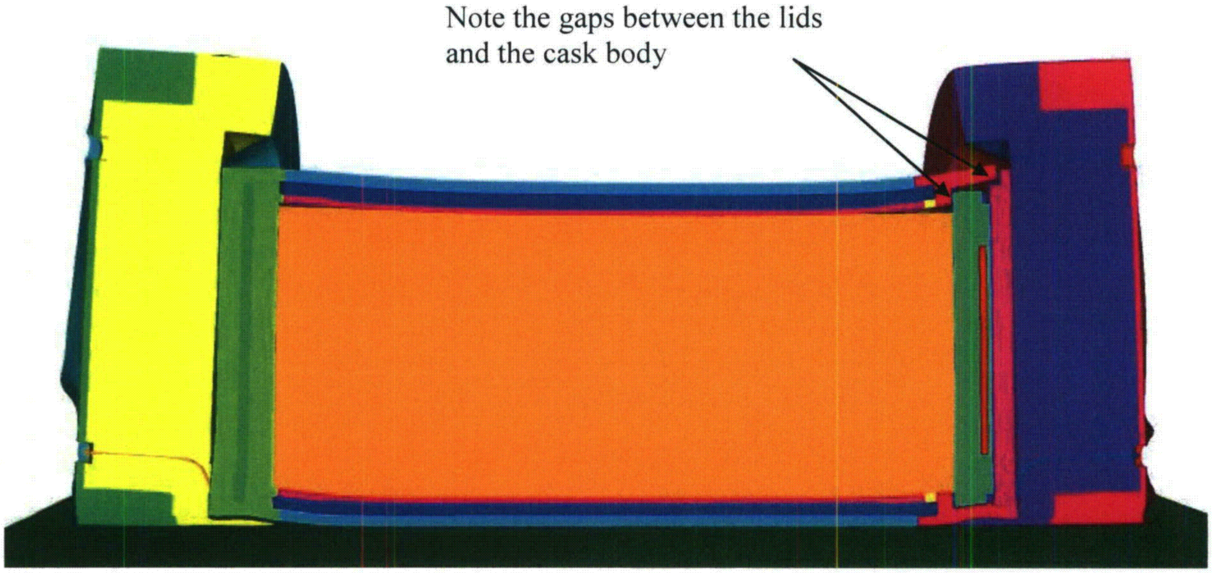


Figure 3-11. Deformed shape of the Rail-Pb following the 145 kph impact onto an unyielding target in the side orientation

Leak area

The Certificate of Compliance for the Rail-Pb 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. 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.

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 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. 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.

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

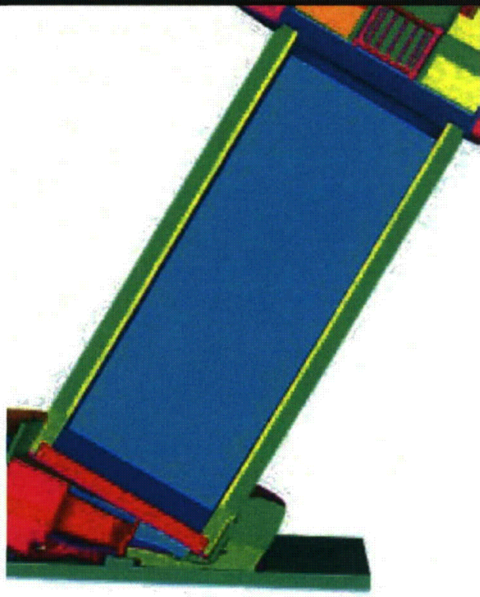
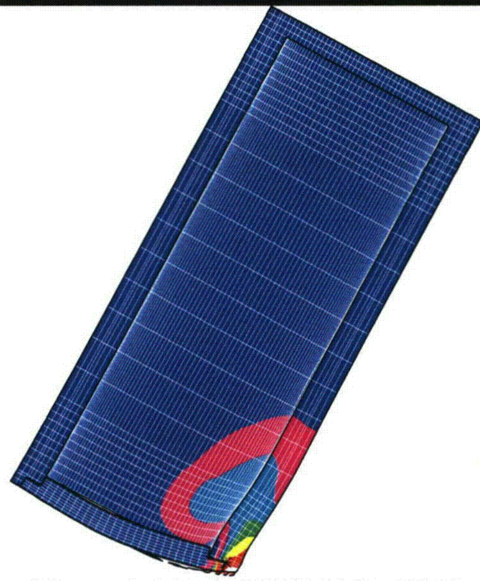
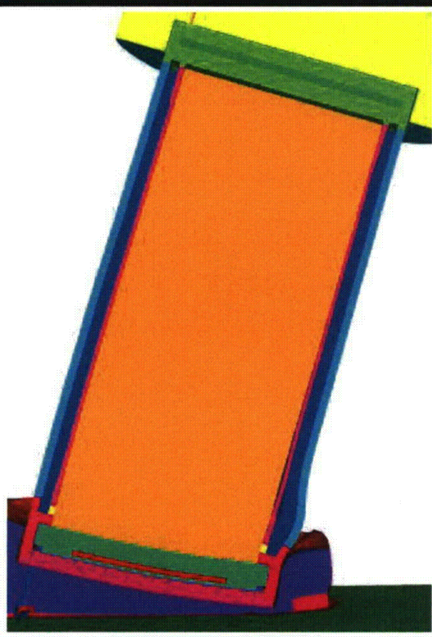
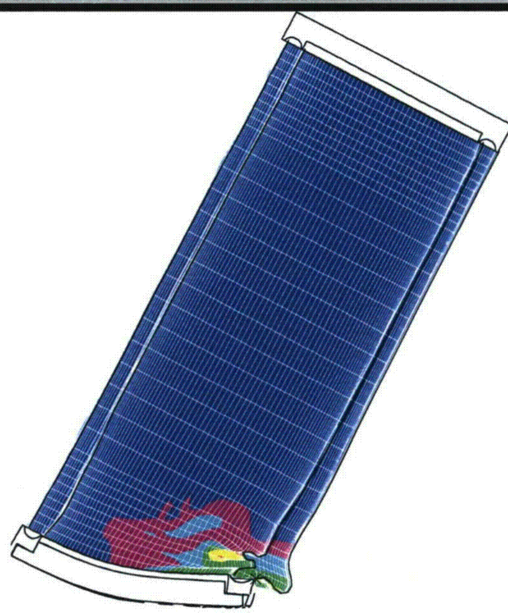
*Estimated; the method used to calculate the gaps for the other cases is explained in Appendix III.

**The metal seal for the Rail-Pb cask is only installed 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 3-12 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. 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-Pb**	**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

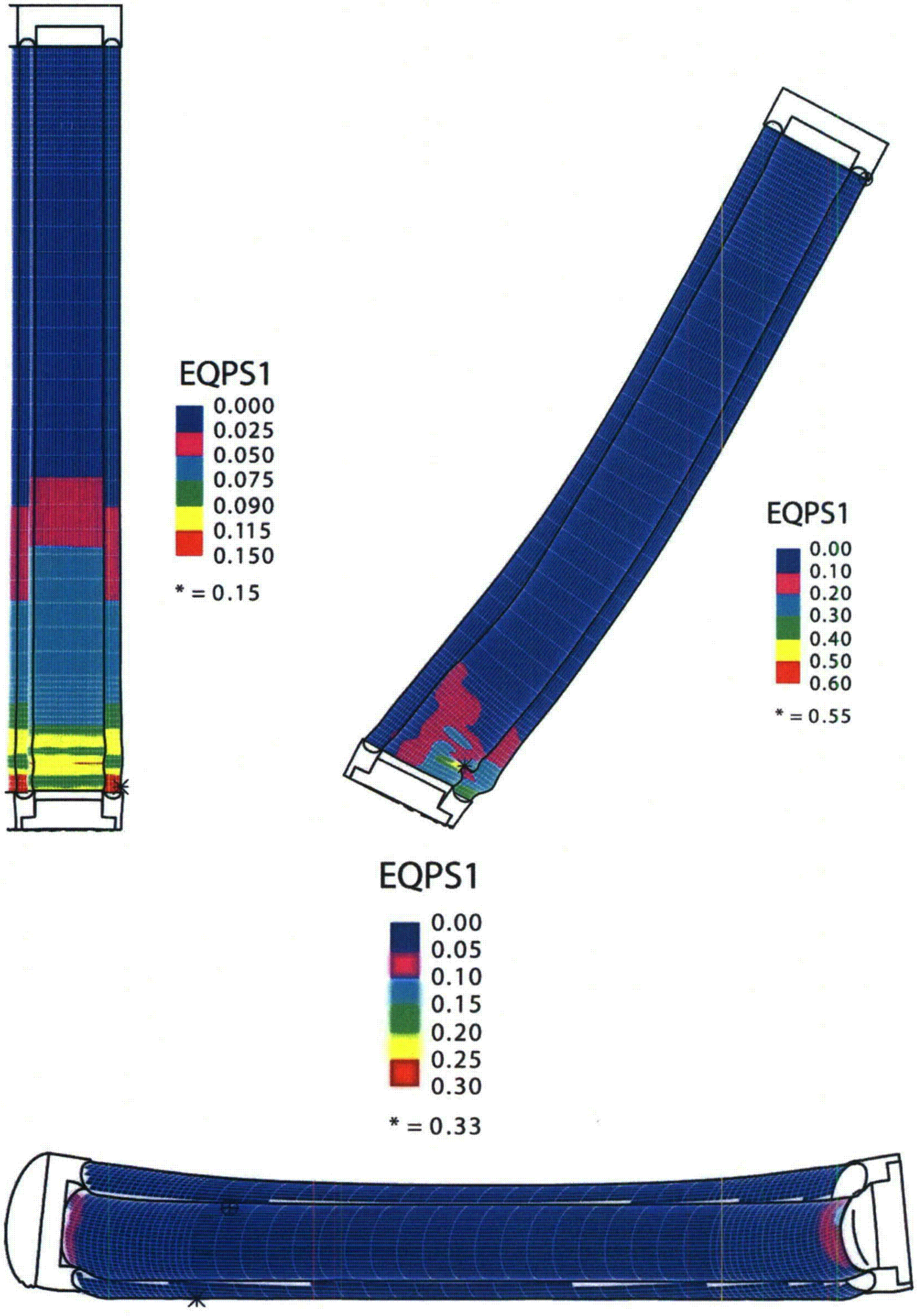


Figure 3-12. Deformed shapes and plastic strains in the generic steel-DU-steel truck cask from NUREG/CR-6672 (impact limiter removed)

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 Opening	Corner Impact Sliding	End Impact Opening	End Impact Sliding	Side Impact Opening	Side Impact 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-Pb 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 3-18. In this table the cases that have non-zero hole sizes from Table 3 5 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 MN (33 million pounds). Very few real targets are capable of applying this amount of force.

If the cask hits a flat target, such as the ground, roadway, or railway, it will penetrate into the surface. The greater the penetration depth, the more force the target can exert on the cask. Figure 3-13 shows the relationship between penetration depth and force for the Rail-Pb 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 3-13. As an example, the end impact at 97 kph onto an unyielding target requires a contact force of 123.9 MN. A penetration depth of approximately 2.2 meters will cause the soil to exert this amount of force. The soil absorbs 142 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. Where the calculated velocity is more than 250 kph the value in the table is listed as greater than 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-Pb 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**

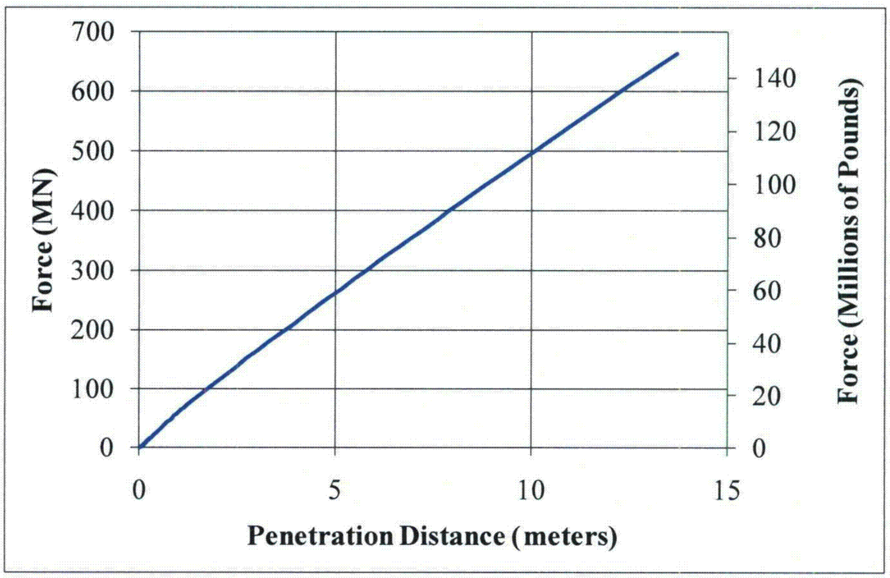


Figure 3-13. Force generated by the Rail-Pb cask penetrating hard desert soil

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

Orientation	Rigid	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 3-14 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 3-15 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.

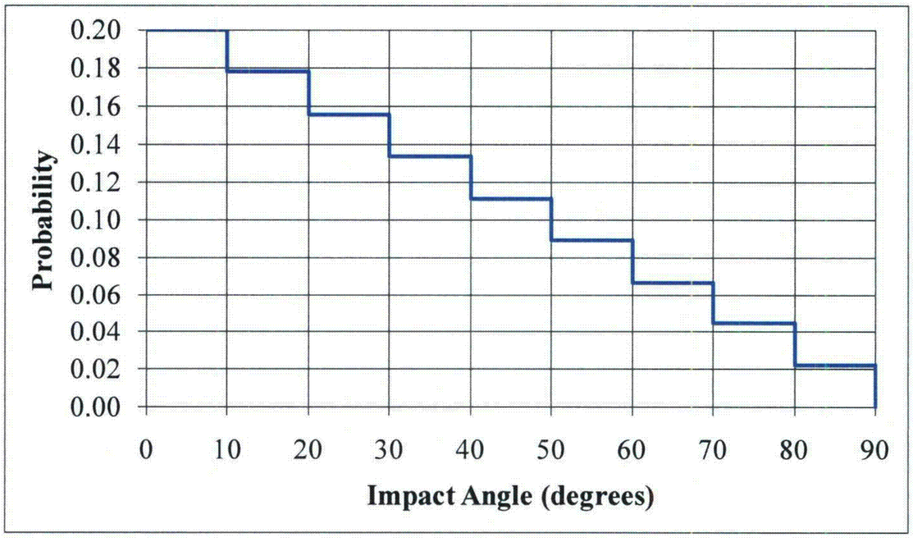


Figure 3-14. Probability distribution for impact angles

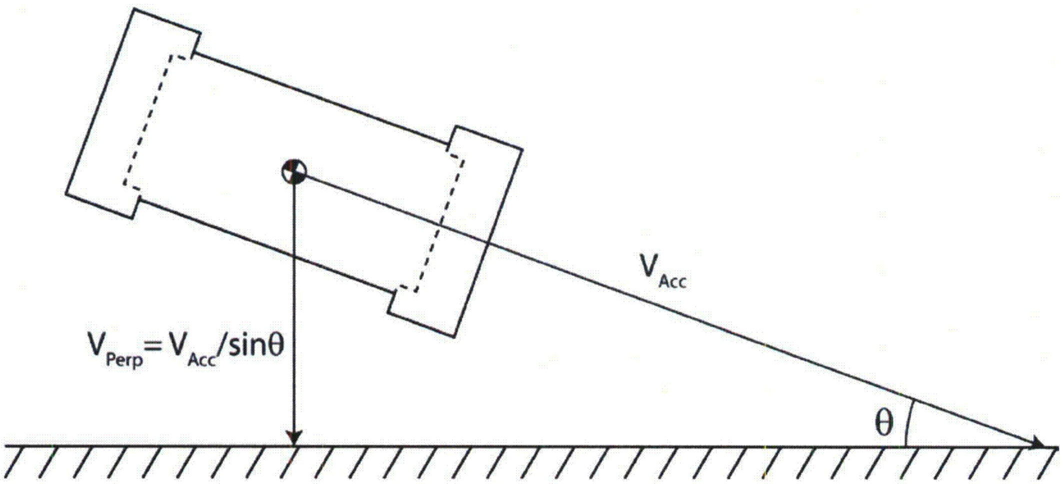


Figure 3-15. Influence of impact angle on effective velocity

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, 2003b, 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 3-16 (NRC, 2003b). 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.

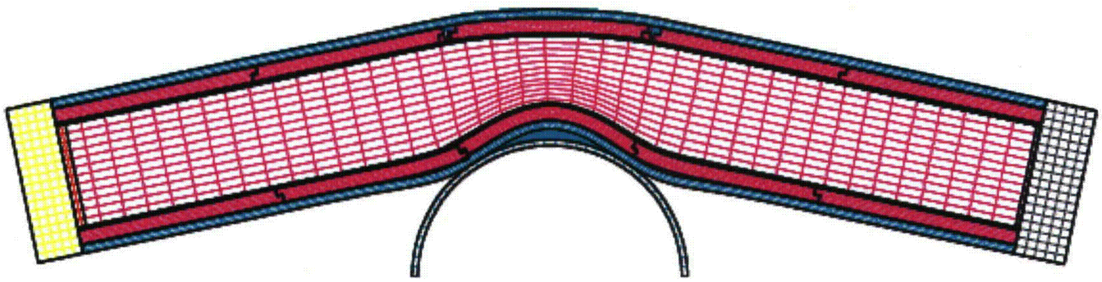


Figure 3-16. Deformations to the GA-4 truck cask after a 96 kph side impact onto a rigid semi-circular column, from (NRC, 2003b).

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 3-17. 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 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. 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 3-18 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.

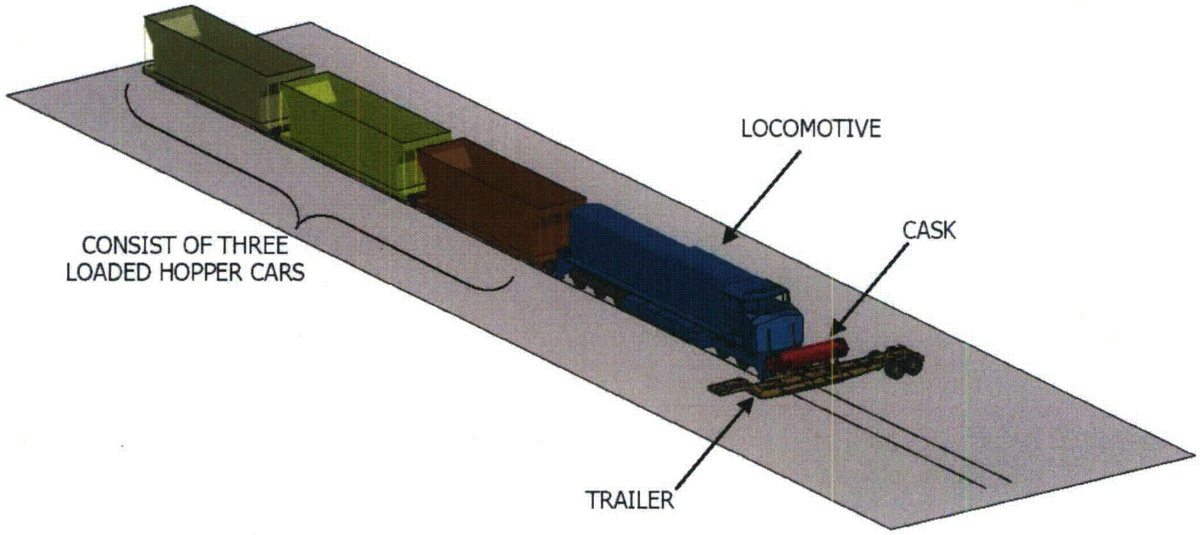


Figure 3-17. Configuration of locomotive impact analysis (from Ammerman et al., 2005)

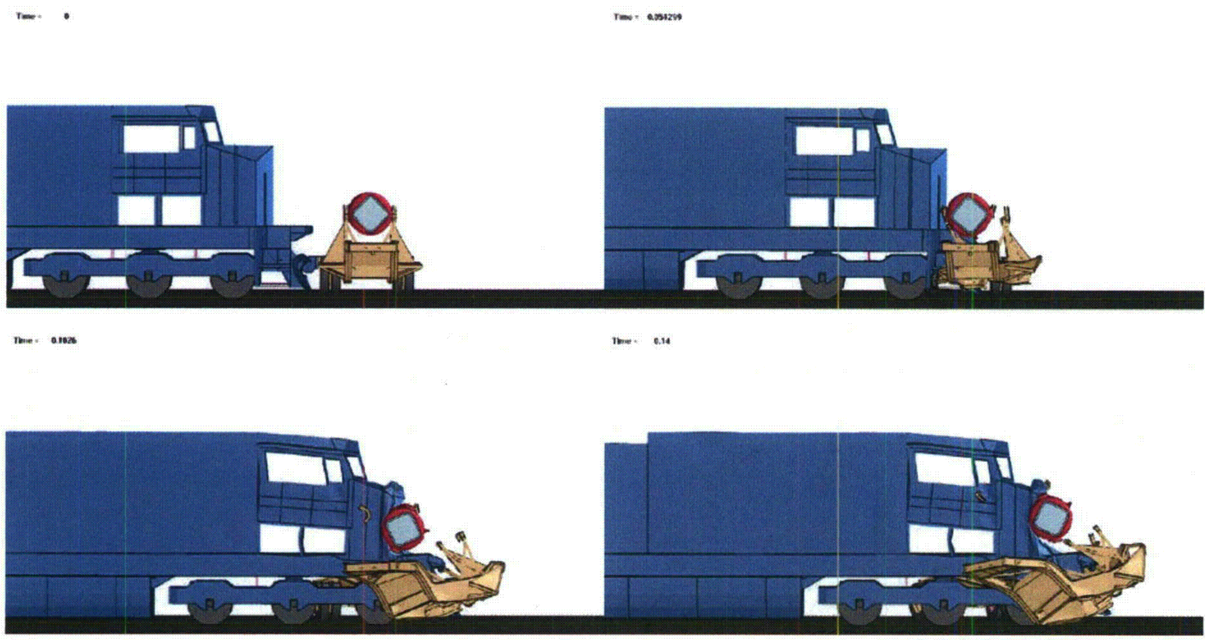


Figure 3-18. Sequential views of a 129 kph impact of a locomotive into a GA-4 truck cask (from Ammerman et al., 2005)

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. 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 3-19. As in the other analyses for impacts with objects, there would be no loss of containment from this accident.

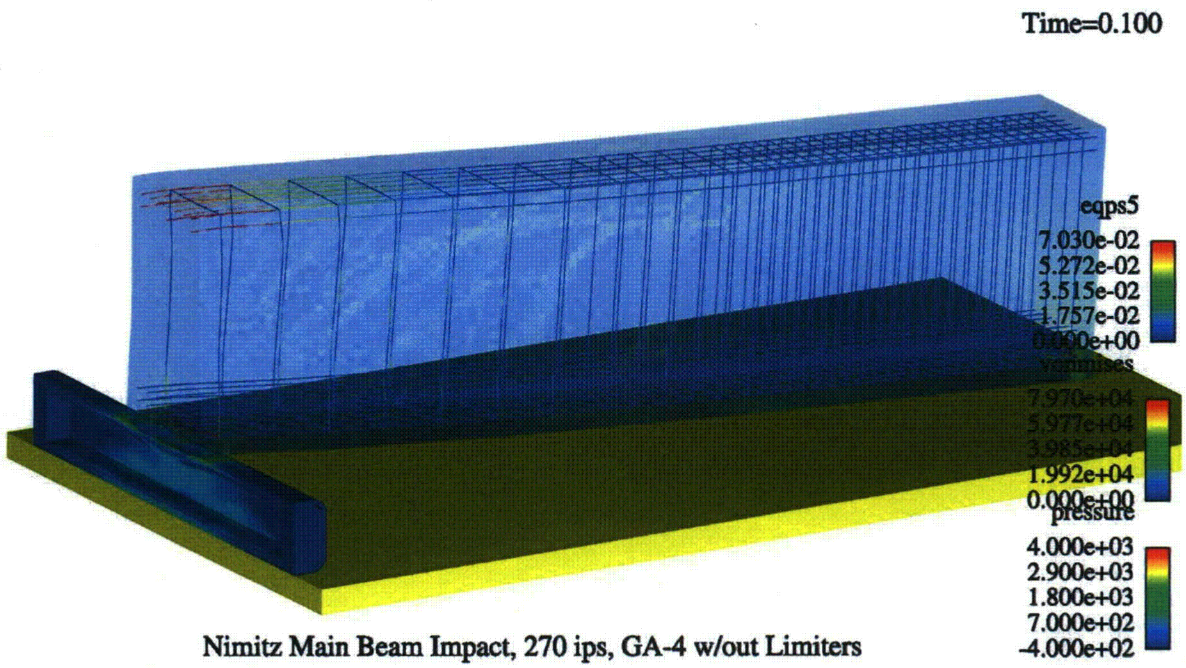


Figure 3-19. Results of a finite element simulation of an elevated freeway collapse onto a GA-4 spent fuel cask (from Ammerman and Gwinn 2004)

3.6 Response of Spent Fuel Assemblies (Kalan et al., 2005)

The finite element analyses 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 . Figure 3-20 shows this model. The loads associated with a 100 G cask impact in a side orientation were then applied to this detailed model. Kalan et al. only analyzed side impacts 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 3-21. There is slight yielding of the rod at each support location and slightly more yielding where the rods impact each other. Figure 3-22 shows the maximum plastic strain at each location. The largest of these strains is slightly below 2%, which is half the plastic strain capacity of irradiated zircaloy at the maximum burn-up allowed in the Rail-Pb 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 200G. 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.

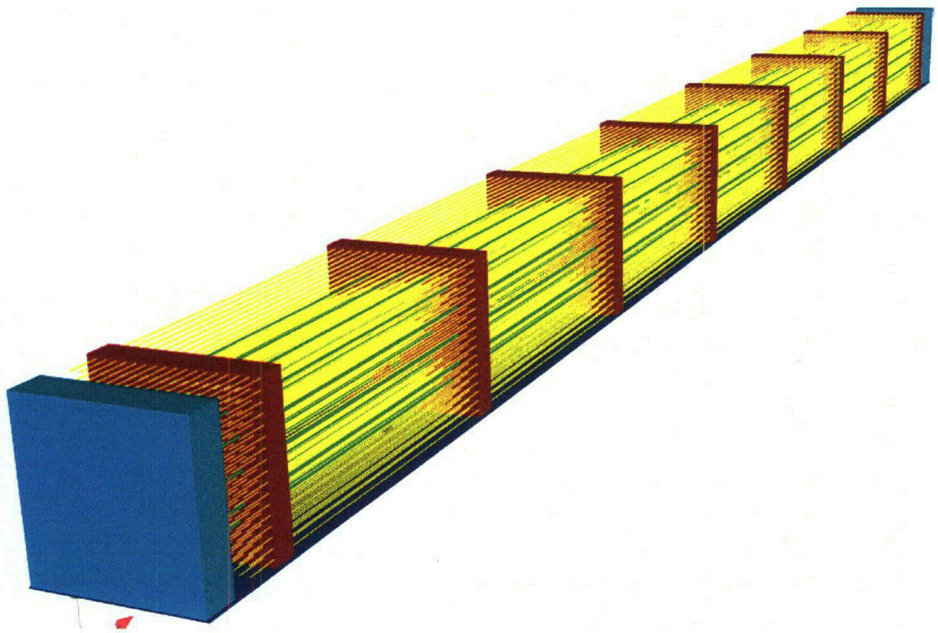


Figure 3-20. Finite element model of a PWR fuel assembly.

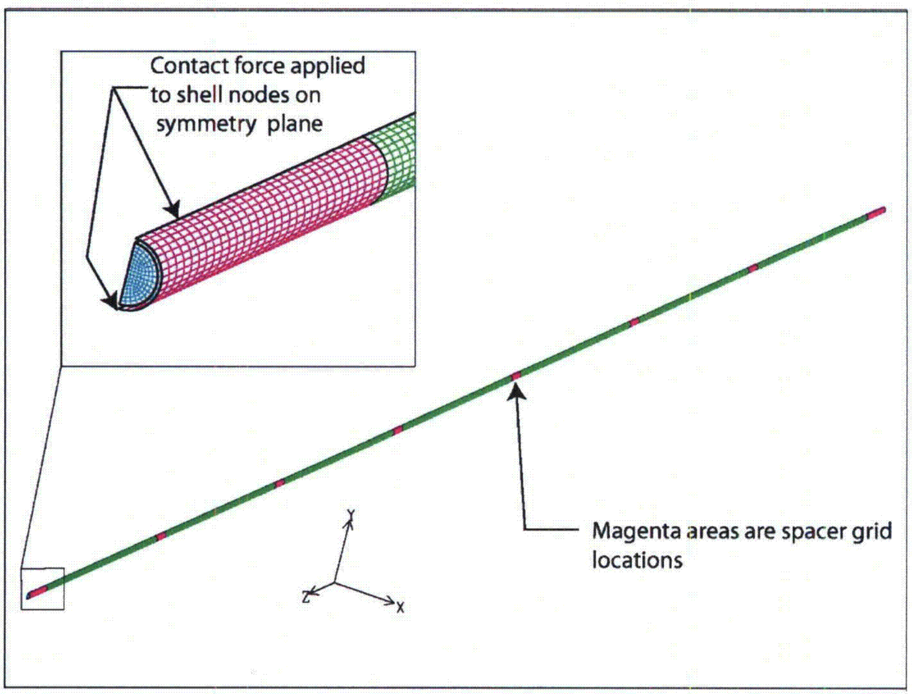


Figure 3-21. Detailed finite element model of a single fuel rod.

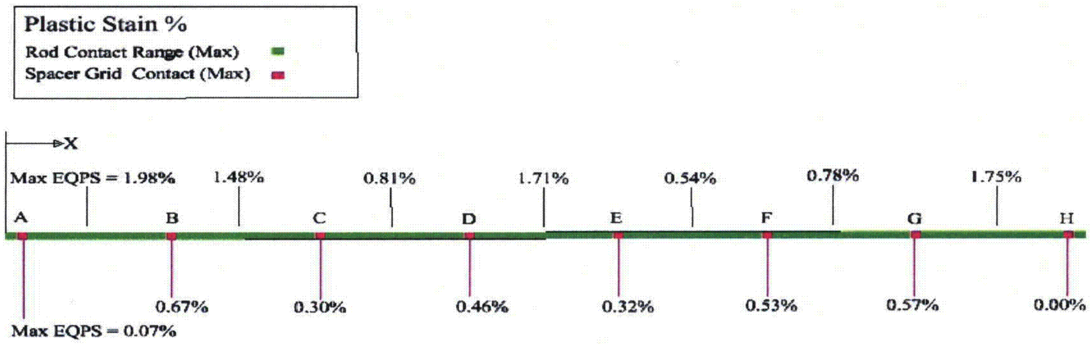


Figure 3-22. Maximum strains in the rod with the highest loads.

3.7 Conclusions

The finite element analyses performed 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. 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. In summary, the sequence of events that is needed for there to be the possibility of any release is: a rail transport cask, no welded canister, an impact velocity greater than 97 kph, the cask impacting in a side orientation, the impact surface being hard rock, and the impact angle being greater than 30 degrees.

CHAPTER 4

CASK RESPONSE TO FIRE ACCIDENTS

4.1 Introduction

Certified Type B casks are designed to survive a fully-engulfing fire for thirty minutes. This is more severe than the majority of the thermal environments a cask may be exposed to in an accident that results in a fire. Certification analyses of the hypothetical fire environment specified in 10 CFR 71.73 impose on the cask a similar or more severe thermal environment than a real fully-engulfing fire. Large open pool fires can burn at temperatures higher than the 800°C specified in the regulations. Real fire plumes have location- and time-varying temperature distributions that vary from about 600°C to more than 1200°C (Koski, 2000; Lopez et al., 1998). Therefore, the evenly-applied 800°C fire environment used in certification analysis can be more severe for seal and fuel rod response than a real fire.

For this risk study, computer codes capable of modeling fires and the thermal response of casks exposed to fires in a realistic fashion are used to analyze the response of the Rail-Steel and the Rail-Pb casks to three different hypothetical fire configurations. These configurations are described in this chapter and the temperature responses of the casks are presented and discussed. A discussion of the thermal performance of the Truck-DU cask exposed to a severe hypothetical fire is also presented, using the analyses in NUREG/CR-6672 (Sprung et al., 2000).

The thermal response of each cask is compared to two characteristic temperature limits. These are the seal failure temperature (350°C for elastomeric seals used in the Rail-Pb cask and the Truck-DU and 649°C for the metallic seal used in the Rail-Steel cask) and the fuel rod burst rupture temperature (750°C for all casks). 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 is a conservative temperature limit for elastomeric seals typically used in the spent nuclear fuel transportation industry. NUREG/CR-6672 also provides the rationale for the use of 750°C 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, allowing release of radioactive material under any of the accident scenarios analyzed.

4.2 Description of Accident Scenarios

4.2.1 Pool size

Three hypothetical fire accident scenarios are analyzed for each rail cask and one for the truck cask. A fuel pool that conforms to the regulatory requirement in 10 CFR 71.73 is used as the basis for each scenario. This regulation specifies a fuel pool that extends between one and three meters horizontally beyond the external surface of a cask. In this study, all fuel pools were assumed to extend three meters from the sides of each package analyzed to ensure the casks would be fully engulfed.

4.2.2 Fire duration

The duration of the hypothetical fires for the rail cask analysi is based on the capacity of a large rail tank car. Typical large rail tank cars can carry about 30,000 gallons of flammable liquid. To estimate the duration of the fires, this amount of fuel is 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 beyond the surfaces of the casks are used in the computer models. Provided that there are relatively small differences between the overall dimensions of the Rail-Steel cask and the Rail-Pb cask, these fuel pools are similar in size and are nominally 14m x 9m. A pool of this size would need to be 0.9m deep to pool 30,000 gallons of liquid fuel, a condition that is extremely unlikely to be met in an accident scenario. If the fuel in such pool were to ignite, this pool fire would burn for about 3 hours. This fire duration is estimated using a nominal hydrocarbon fuel recession (evaporation) rate of 5mm 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 more unlikely case in which 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. Nevertheless, a 3-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 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 2.3m x 6m, a regulatory pool that extends horizontally 3 meters beyond the outer surface of the cask would be 8.3m x 12m. To pool 9,000 gallons of gasoline in a pool of this area, the pool would need to be 0.3m 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 gasoline were to flow at the right rate to maintain the necessary fuel pool conditions. Again, this scenario is very unlikely also. However, 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 hypothetical fire accident scenarios different from the regulatory configuration are analyzed in this study for the rail casks. These are:

1. Cask on the ground and concentric with the fuel pool as depicted in Figure 4-1.

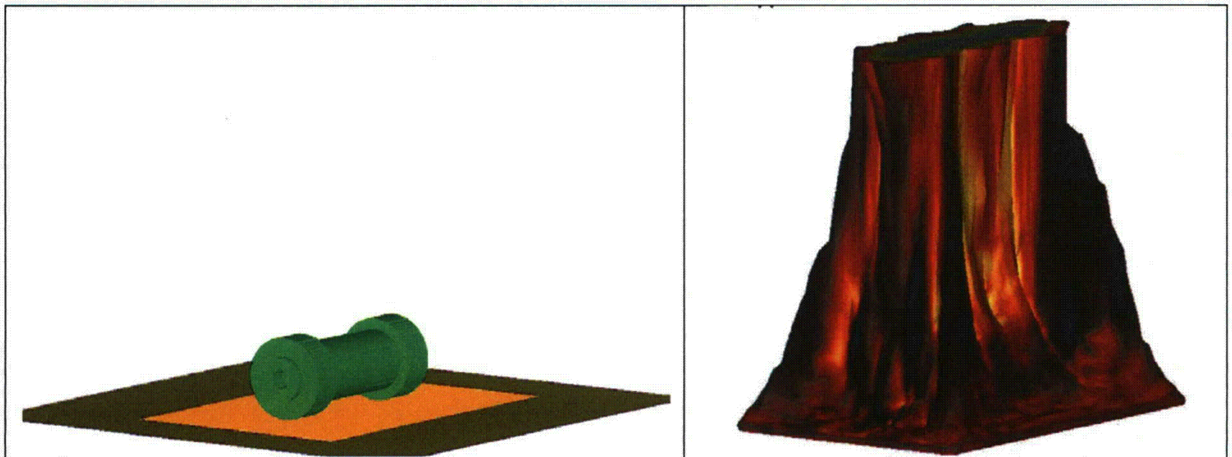
   This scenario represents the hypothetical case in which the liquid fuel spilled as a consequence of the accident flows to the location where the cask came to rest after the accident and forms a large pool under (and concentric with) the cask.

2. Cask on the ground with the fuel pool offset three meters (side of cask to side of 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 one rail car width. This could be the case of an accident in which the train jackknifes or a pile-up accident.

3. Cask on the ground with the fuel pool offset 18 meters (side of cask to side of fuel pool) as depicted in Figure 4-3.

   This scenario represents the hypothetical case in which the fuel pool and the cask are separated by one rail car length. This represents an accident in which the tank car carrying the flammable liquid maintains the distance of a buffer rail car 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.



Cask and fuel pool set-up; pool region in orange | Fire engulfing the cask

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

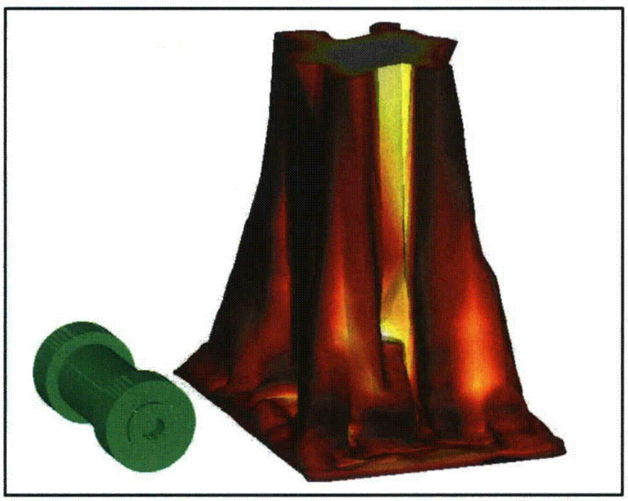


Figure 4-2. Cask on ground and the 3m offset pool fire

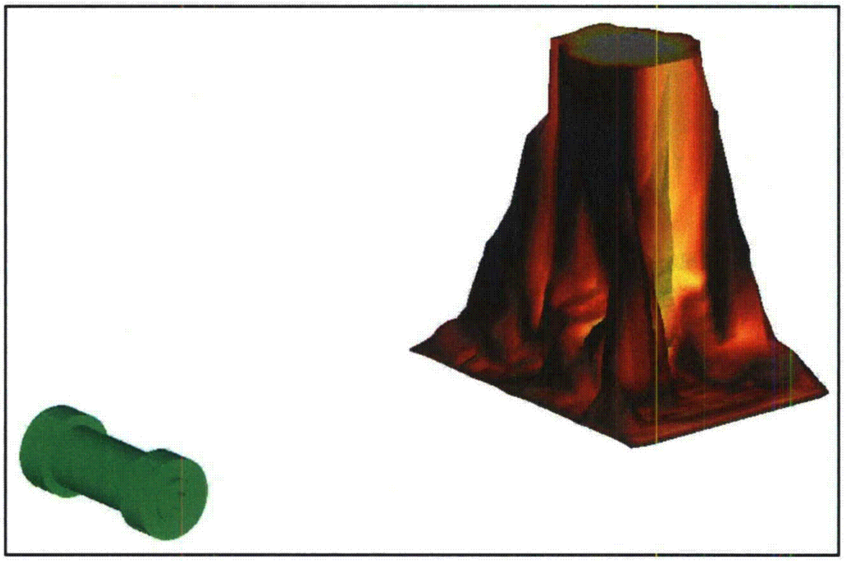
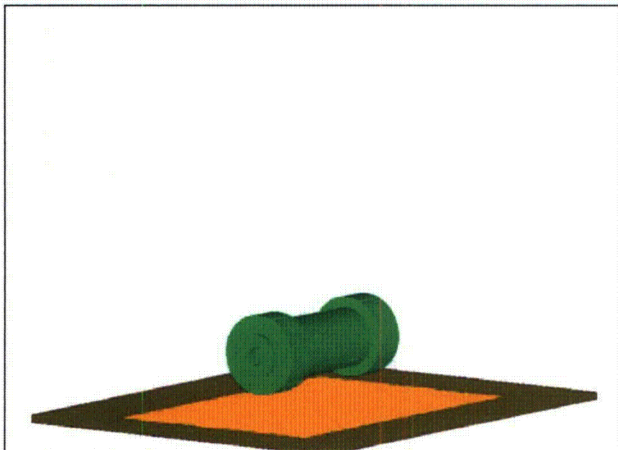


Figure 4-3. Cask on ground and the 18m offset pool fire

In each scenario, only the cask and the fuel pool are present. There are no other objects (such as other rail cars) that are likely to be present that could shield (protect) the cask from the fire. Calm wind conditions are also assumed.

In addition to these hypothetical accident scenarios, two 30-minute regulatory fire analyses are performed as described in 10 CFR 71.73. In the first analysis a commercially-available finite element (FE) heat transfer code is used to apply an 800°C uniform-heating fire condition to the casks. In the second analysis, a benchmarked computational fluid dynamics (CFD) and radiation heat transfer computer model are used. In this model, the cask is positioned one meter above the fuel pool and the fire is realistically modeled as shown in Figure 4-4.



Cask and fuel pool set-up; cask elevated 1m above orange fuel pool region

Regulatory fire engulfing the cask

Figure 4-4. Regulatory pool fire configuration.

4.3 Analysis of Fire Scenarios Involving Rail Casks

Advanced computational tools are employed to generate the data necessary for this risk study by simulating the heat transport in the fire and the cask body. Two computer codes that include 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 complex casks. Brief descriptions of the models are presented in this section. Detailed information of 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-Pb casks are presented in this section. The temperature range of the legends in the temperature distribution plots of all the Rail-Steel cask analysis results are made the same to make relative comparisons easier. The same is done for the Rail-Pb cask plots. However, the temperature scale for the Rail-Steel cask differs slightly from the scale for the Rail-Pb cask.

Results of the analyses are presented in the following order:

1. Regulatory 800°C uniform heating (30 minutes)
2. Regulatory CAFE fire (30-minute fire)
3. Cask on the ground and concentric with a three-hour pool fire
4. Cask on the ground with a three-hour pool fire offset three meters
5. Cask on the ground with a three-hour pool fire offset 18 meters

4.3.1 Simulations of the fires

Fire simulations are performed with the Container Analysis Fire Environment (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 that has been successfully coupled to commercially-available finite-element analysis computer codes. It can be used for the design and risk analysis of packages for the transportation of radioactive material (RAM). 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 the 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.

As described in Section 4.2.3, in addition to the regulatory configuration, three other hypothetical fire scenarios are analyzed. These are: 1) a cask on the ground engulfed by a pool fire for three hours, 2) a cask on the ground offset from a fire by 3-meters for three hours, and 3) a cask on the ground offset from a fire by 18-meters for three hours. Calm wind conditions are assumed for all cases.

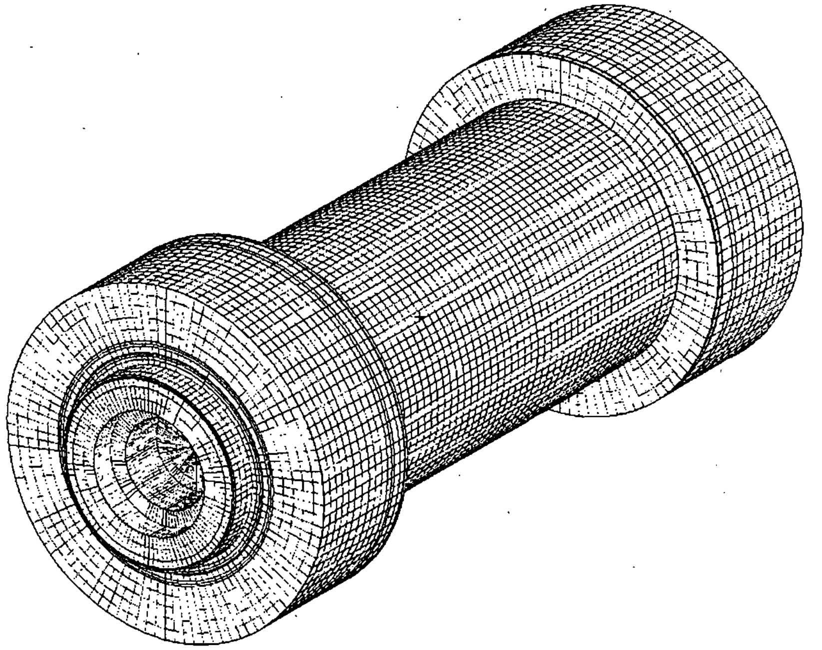
4.3.2 Simulations of the rail casks

The heat transfer within the Rail-Steel and the Rail-Pb casks is modeled with the computer code MSC PATRAN-Thermal (P-Thermal) (MSC, 2008). This code is commercially available and is 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.

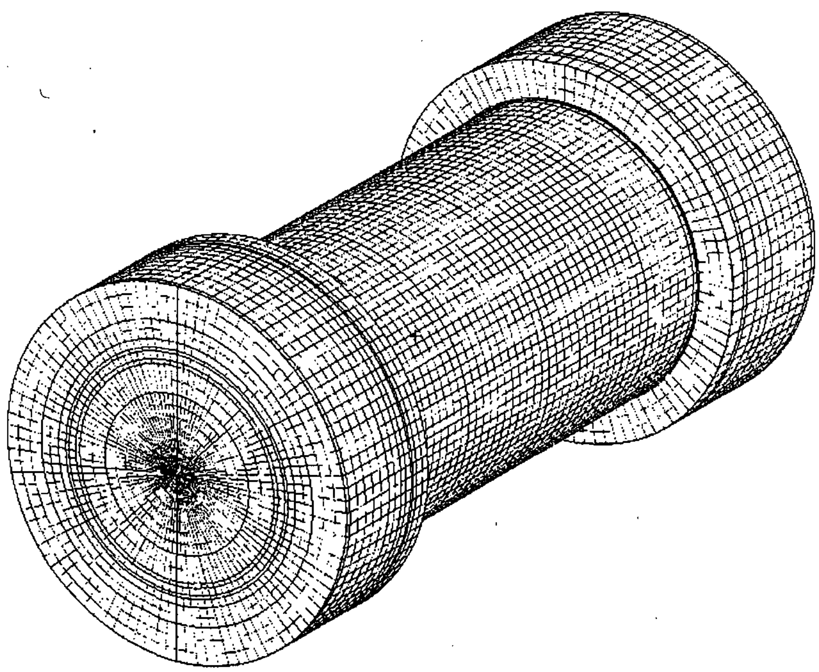
Both the Rail-Steel cask and the Rail-Pb cask have a polymeric neutron shield that is assumed to disintegrate completely and be replaced by air at its operational temperature limit. (see Appendix IV).

The Rail-Pb cask has a lead gamma shield that is allowed to melt if it reached 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-Pb cask has wood impact limiters. Large gaps within cask 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 4-5. Cask modeling details are presented in Appendix IV.



Rail-Steel cask



Rail-Pb cask

Figure 4-5. Finite element models of the two rail casks. The left figure is the Rail-Steel and the right figure is the Rail-Pb.

4.3.3 Simulation of the spent fuel region

The fuel region comprising the fuel basket and the fuel assemblies is not modeled explicitly. Instead, a homogenized fuel region is used. All materials and geometric features of the fuel basket of the casks that are analyzed are represented as a solid cylinder inside the cask. The thermal response of the homogenized fuel region is very similar to the overall response of the actual fuel region and provides sufficient information for this study. The details of how the effective properties of the homogenized fuel region are determined and applied to the model are presented in Appendix IV.

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 4-5 through Figure 4-19. Figure 4-5 through Figure 4-8 contain the temperature distribution and transient temperature response of key cask regions for the regulatory 800°C uniform heating and the regulatory CAFE fire. The uniform external heating produces an even temperature response around the circumference of the seal. However, the realistic uneven fire heating of the exterior produces temperatures at the seal that vary around the circumference. For comparison, the results obtained from the uniform 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 4-9. 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 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 in the form of gas exists but sufficient oxygen for that fuel to burn is not available. This region is typically called the vapor dome. The lack of oxygen in the vapor dome is attributed to poor air entrainment in larger diameter 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 envelop. 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.

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 4-11 and Figure 4-12. Note that these plots 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.

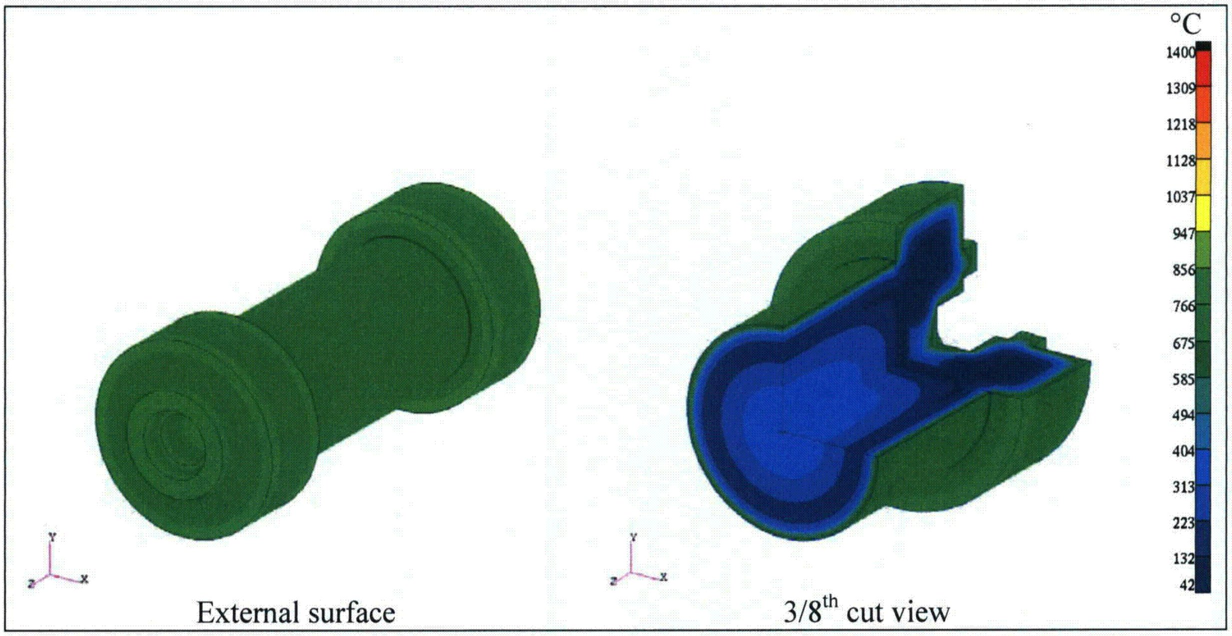


Figure 4-6. Temperature distribution of the Rail-Steel cask at the end of the 30-minute 800°C regulatory uniform heating

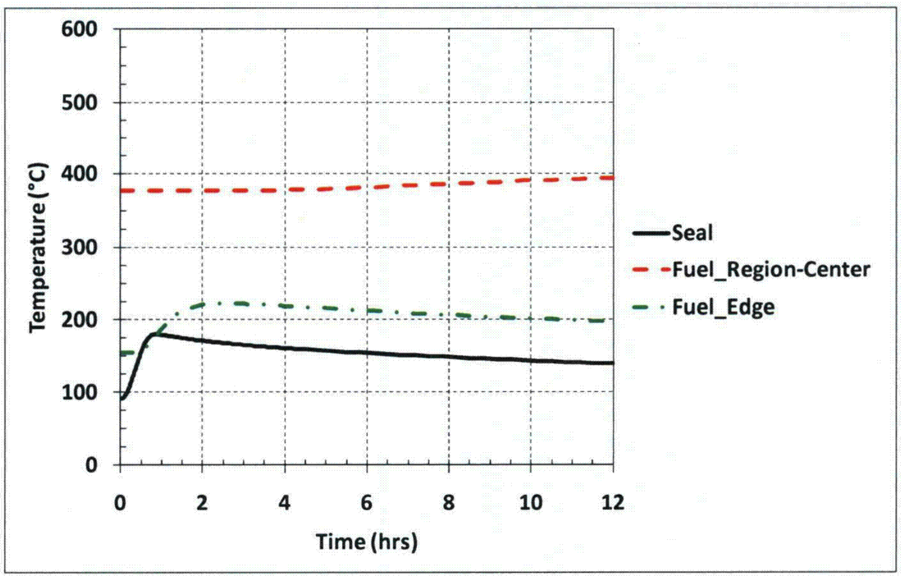


Figure 4-7. Temperature of key cask regions, Rail-Steel cask – Regulatory uniform heating

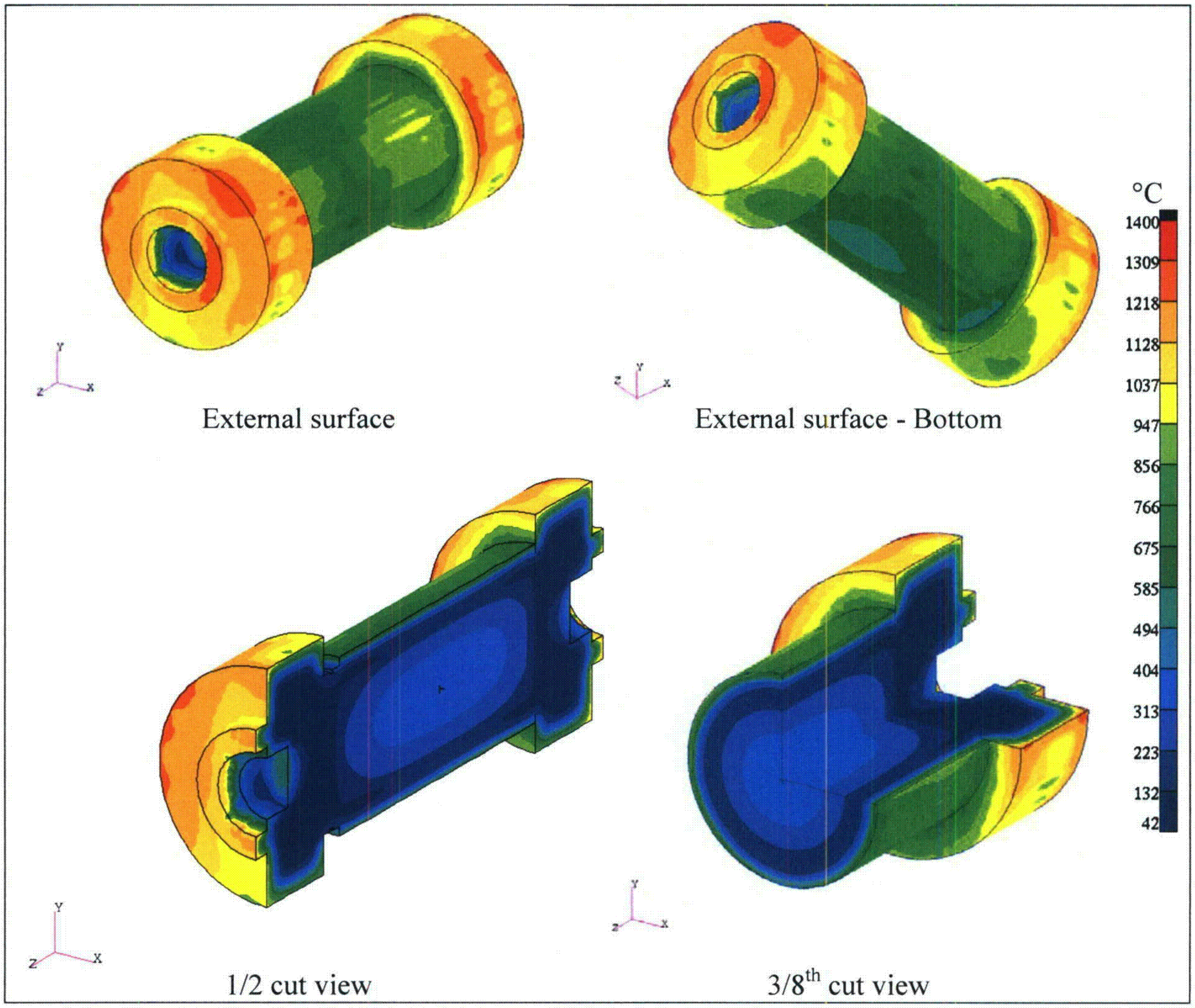


Figure 4-8. Temperature distribution of the Rail-Steel cask at the end of the 30-minute regulatory CAFE fire

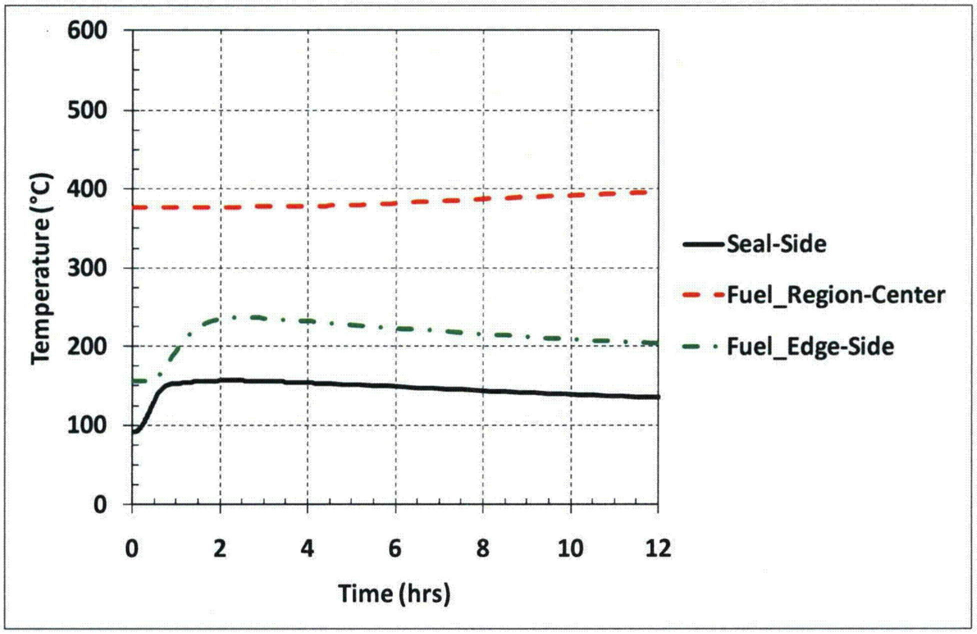


Figure 4-9. Temperature of key cask regions, Rail-Steel cask – Regulatory CAFE fire

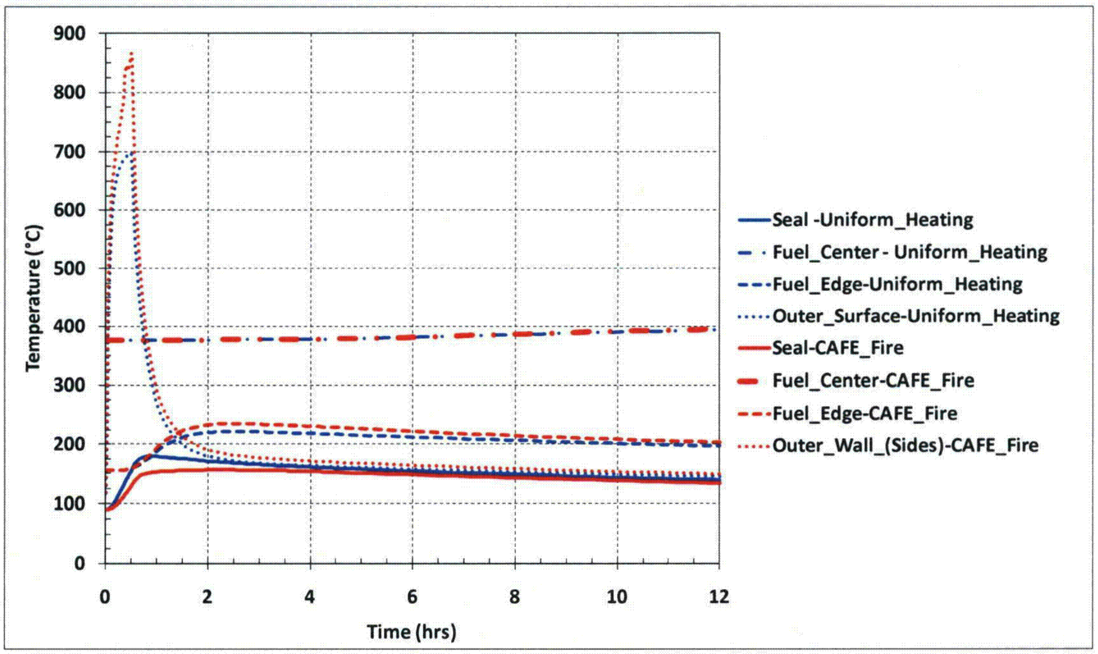
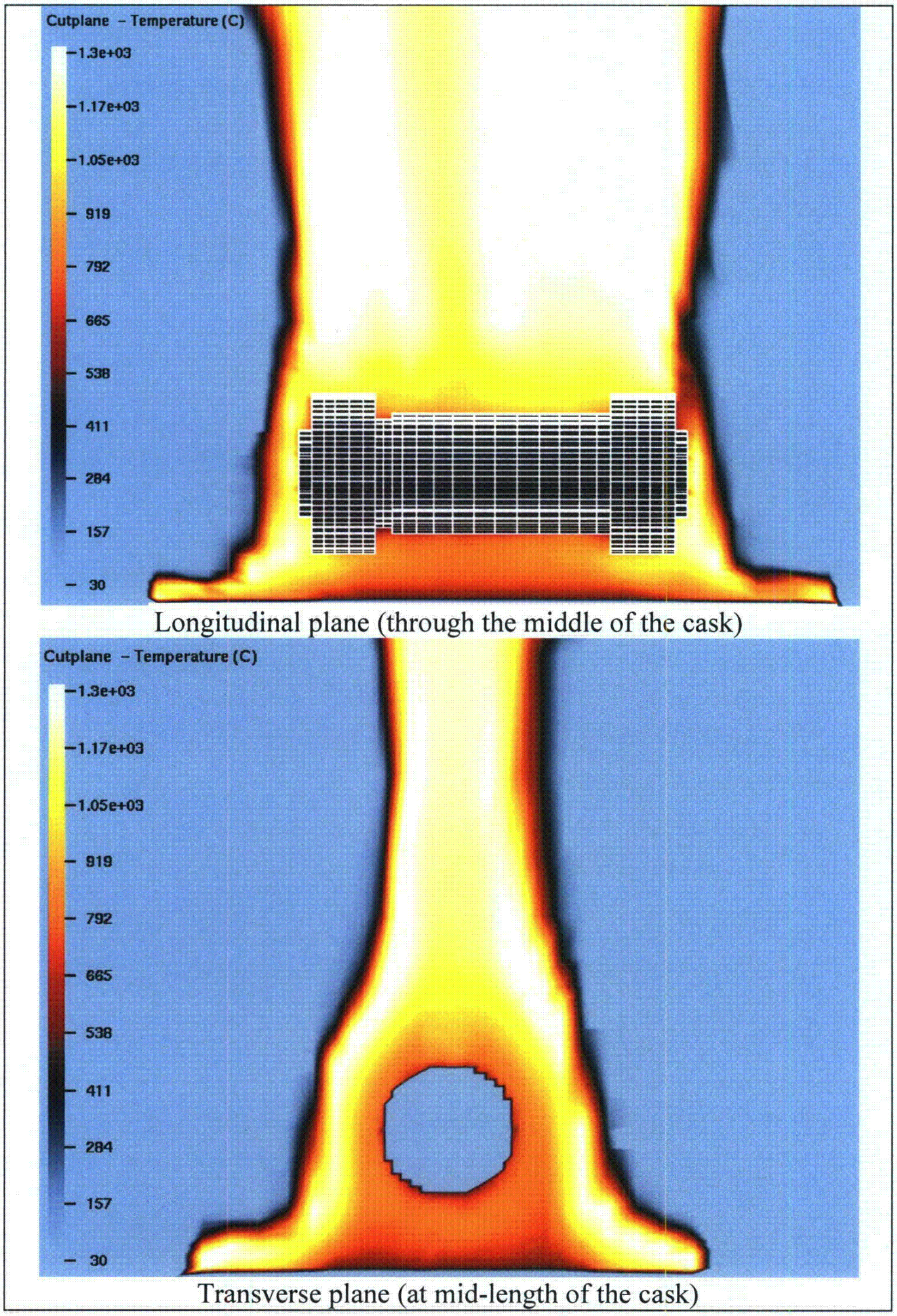


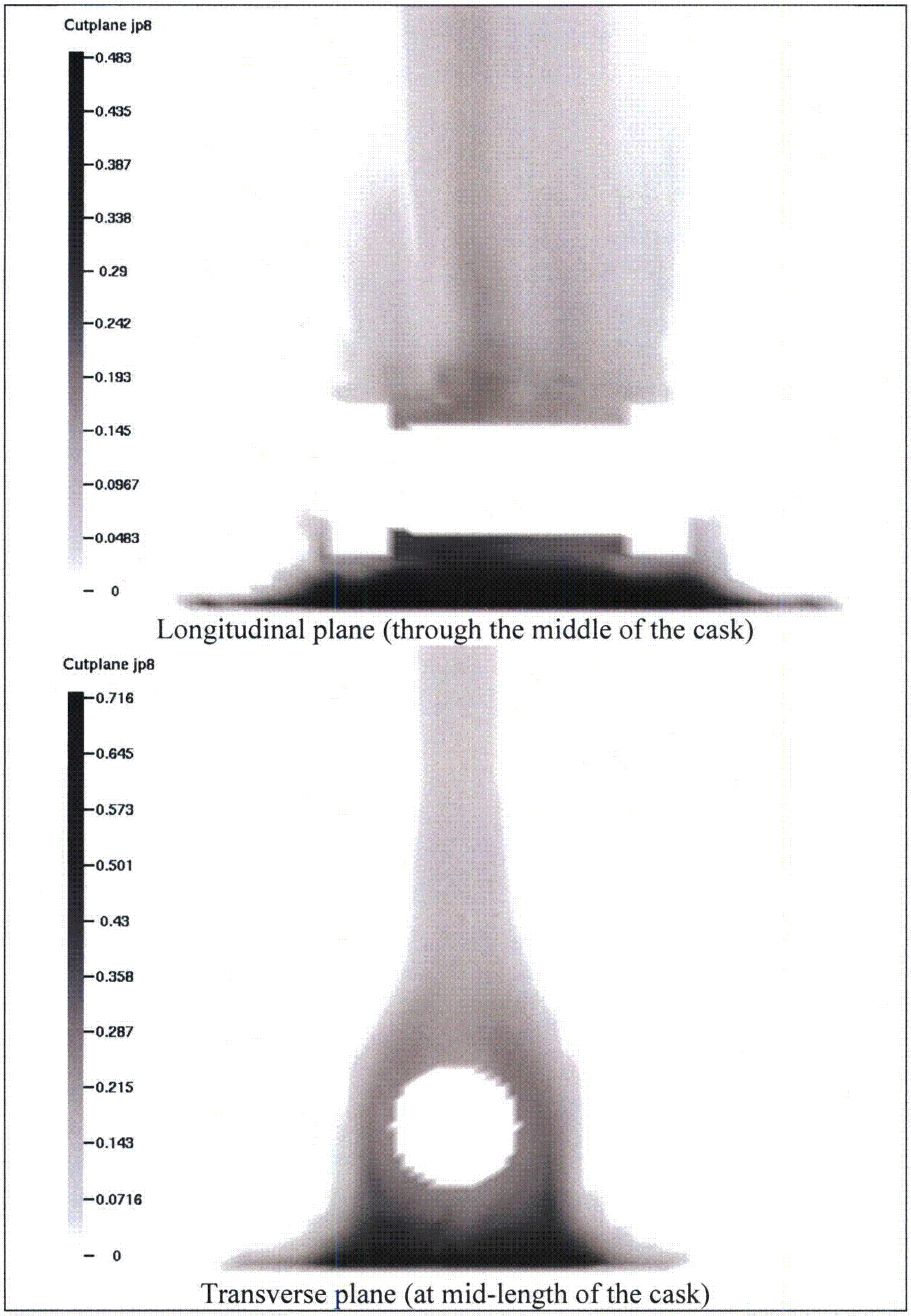
Figure 4-10. Comparison of regulatory fire analysis - Rail-Steel cask: Uniform heating vs. CAFE fire



Longitudinal plane (through the middle of the cask)

Transverse plane (at mid-length of the cask)

Figure 4-11. Gas temperature plots from the regulatory CAFE fire analysis



Longitudinal plane (through the middle of the cask)

Transverse plane (at mid-length of the cask)

Figure 4-12. Fuel concentration plots from the regulatory CAFE fire analysis

The results from the analysis of the cask on the ground and concentric with a pool fire that burns for three hours are presented in Figure 4-13 and Figure 4-14. As in the regulatory configuration, in which the cask is elevated 1 meter above the 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 is different from what is observed in the regulatory configuration, in which the hotter regions are found on the sides of the cask. Figure 4-15 and Figure 4-16 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.

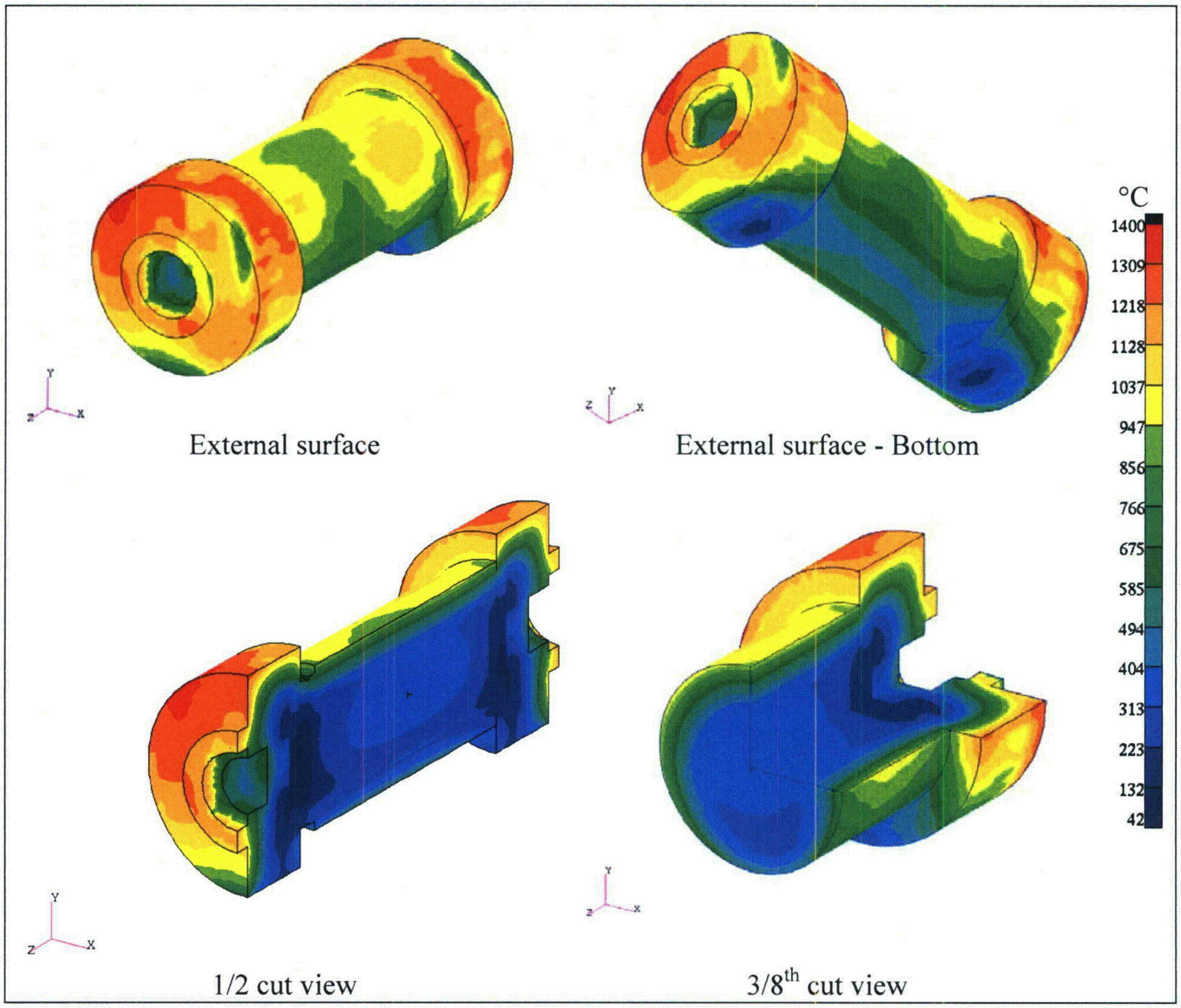


Figure 4-13. Temperature distribution of the Rail-Steel cask at the end of the 3-hour concentric CAFE fire - cask on ground

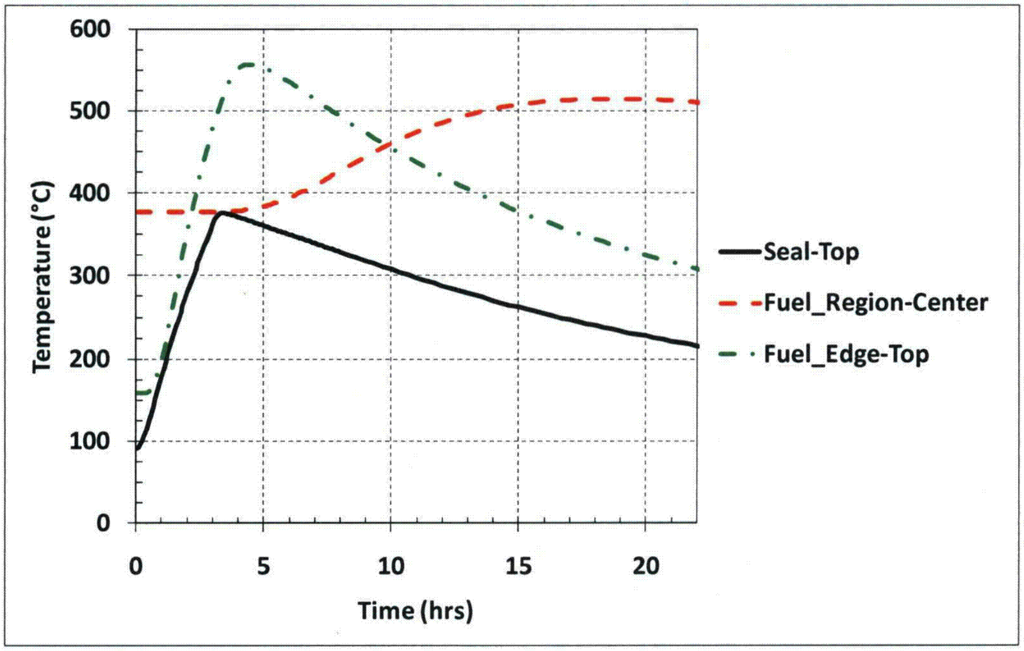


Figure 4-14. Temperature of key cask regions, Rail-Steel cask – Cask on ground, concentric fire

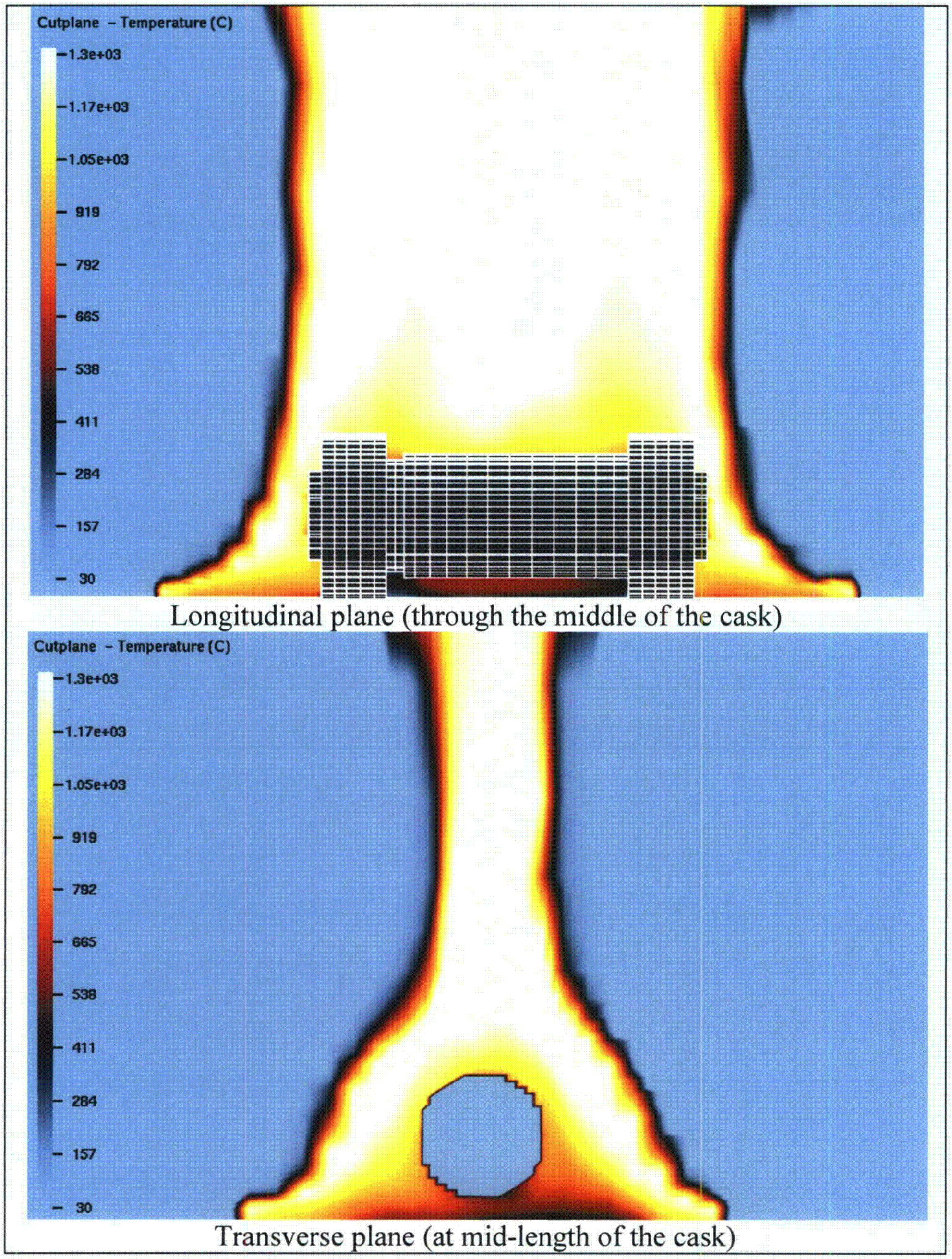
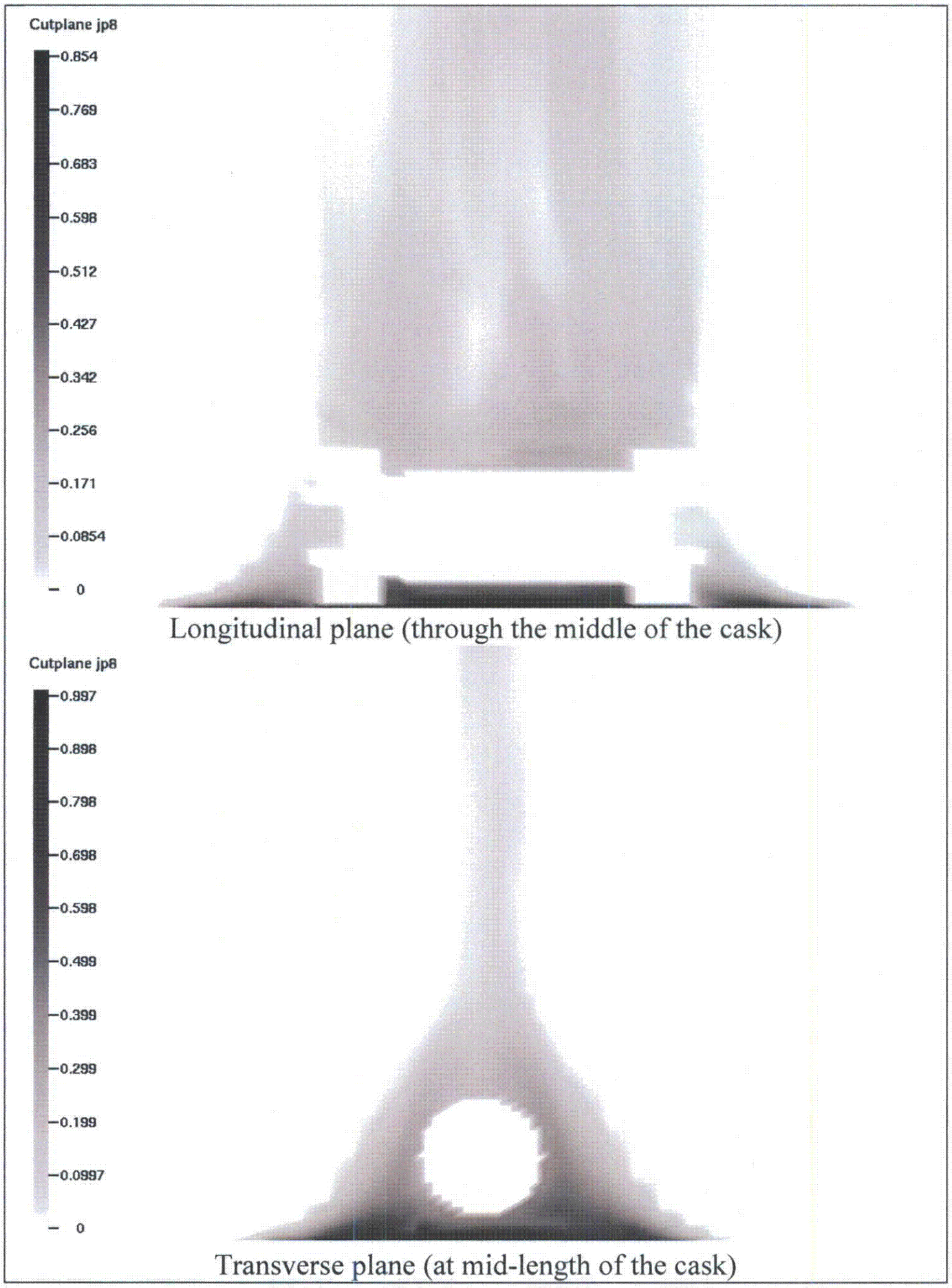


Figure 4-15. Gas temperature plots from the CAFE fire analysis of the cask on ground



Longitudinal plane (through the middle of the cask)

Transverse plane (at mid-length of the cask)

Figure 4-16. Fuel concentration plots from the CAFE fire analysis of the cask on ground

The results of the offset fire analyses are summarized in Figure 4-17 through Figure 4-20. In the case of the three-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 4-17 and Figure 4-18. Similarly, the 18-meter offset fire caused the cask temperature to rise as illustrated in Figure 4-19 and Figure 4-20. These results show that offset fires, even as close to the cask as three meters, do not represent a threat to this thermally-massive spent nuclear fuel transportation cask. The maximum temperatures observed in the seal and fuel region did not reach the temperature limits discussed at the beginning of this chapter. Therefore, offset scenarios will not cause this package to fail.

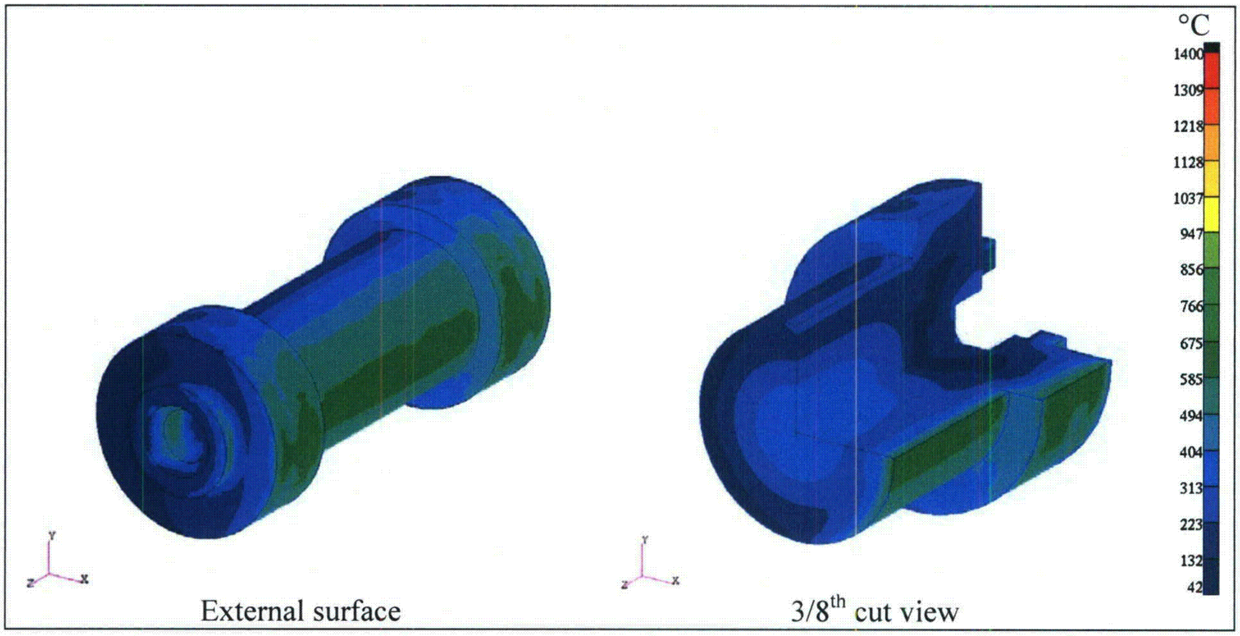


Figure 4-17. Temperature distribution of the Rail-Steel cask at the end of the 3-hour 3m offset CAFE fire - cask on ground

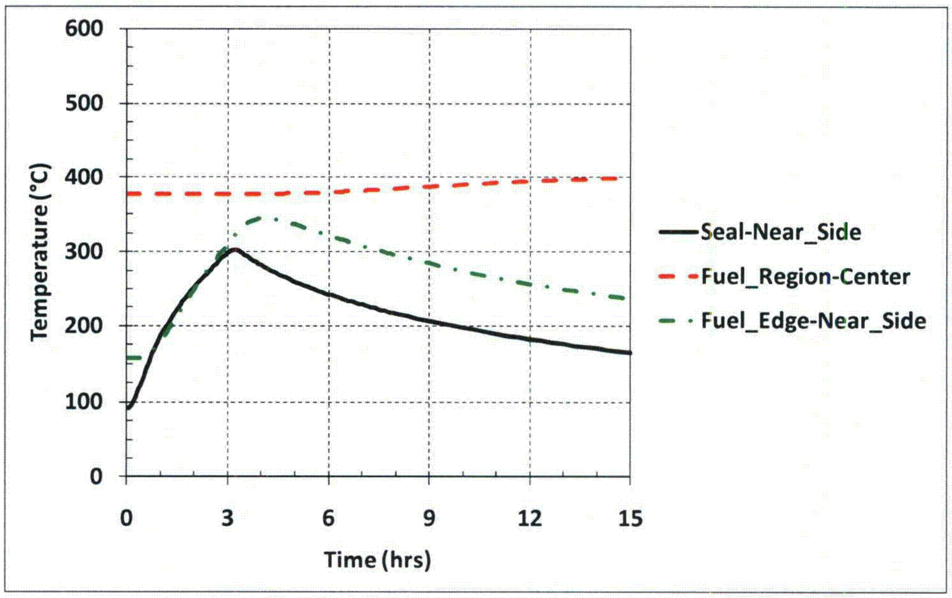


Figure 4-18. Temperature of key cask regions, Rail-Steel cask – Cask on ground, 3m offset fire

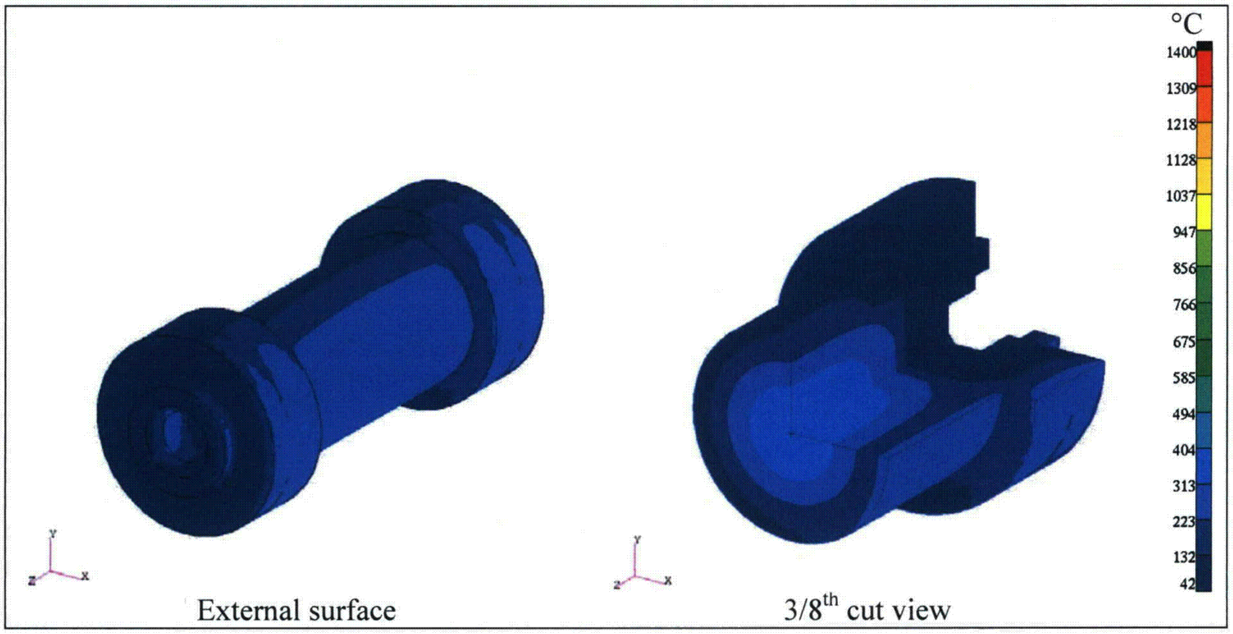


Figure 4-19. Temperature distribution of the Rail-Steel cask at the end of the 3-hour 18m offset CAFE fire - cask on ground

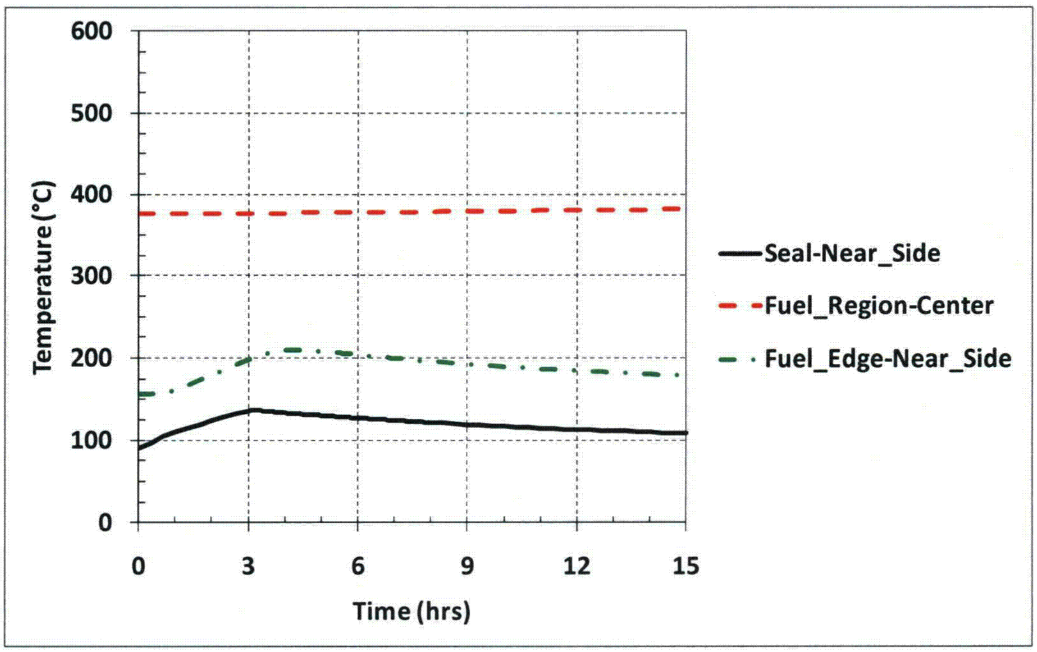


Figure 4-20. Temperature of key cask regions, Rail-Steel cask – Cask on ground, 18m offset fire

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, the fuel region stayed below 750°C and the seal region stayed under 649°C 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-Pb cask results

The thermal response of the Rail-Pb 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 4-21 through Figure 4-25.

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 4-25. As with the Rail-Steel cask, this figure illustrates that the uniform heating thermal environment described in 10 CFR 71.73 heats up the seal region of the Rail-Pb 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.

The results of the analyses of cask on the ground heated by the concentric and offset fires are summarized in Figure 4-26 through Figure 4-31. 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-Pb cask. Lead melts at 328°C 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 fuel region) in Figure 4-27 and Figure 4-29 show a change in slope at about 328°C. This effect is more clearly seen in the slower heating case shown in Figure 4-29. Once the lead melting process is complete, the cask resume 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 4-29.

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.

Appendix IV contains additional plots with more information about temperature distributions at more locations in the cask.

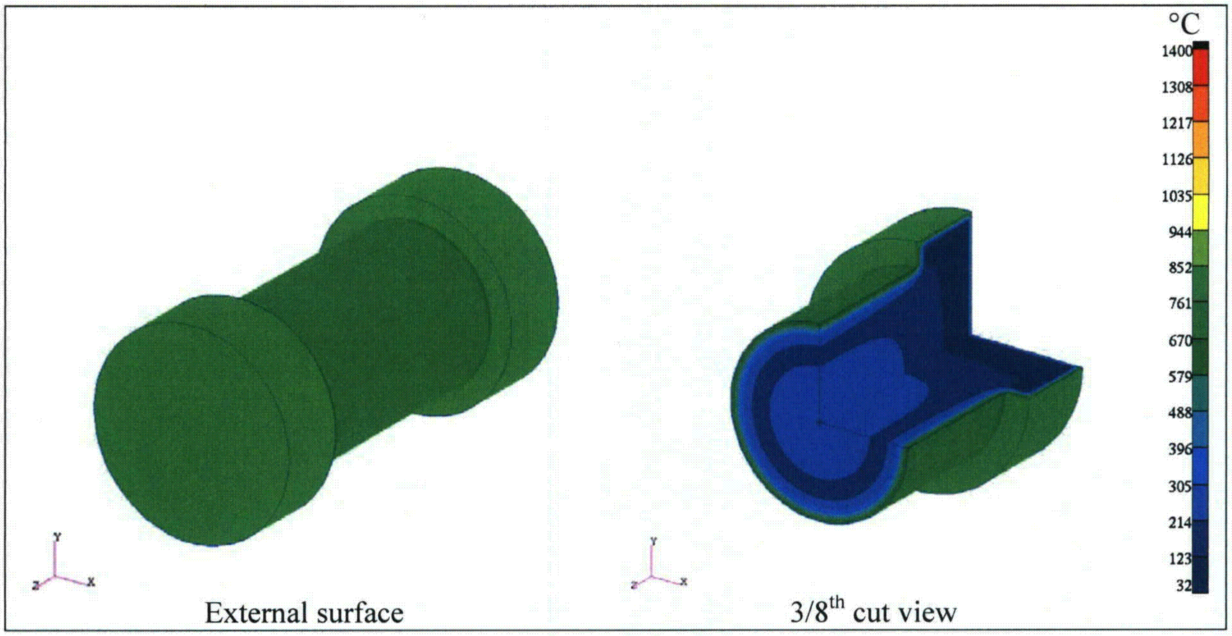


Figure 4-21. Temperature distribution of the Rail-Pb cask at the end of the 30-minute 800°C regulatory uniform heating

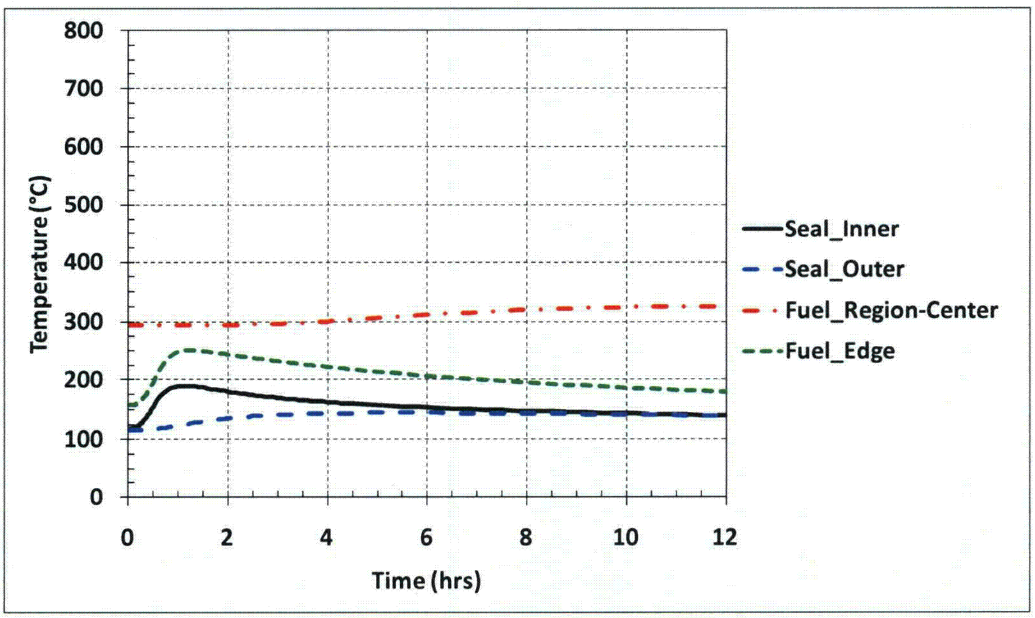


Figure 4-22. Temperature of key cask regions, Rail-Pb cask – Regulatory uniform heating

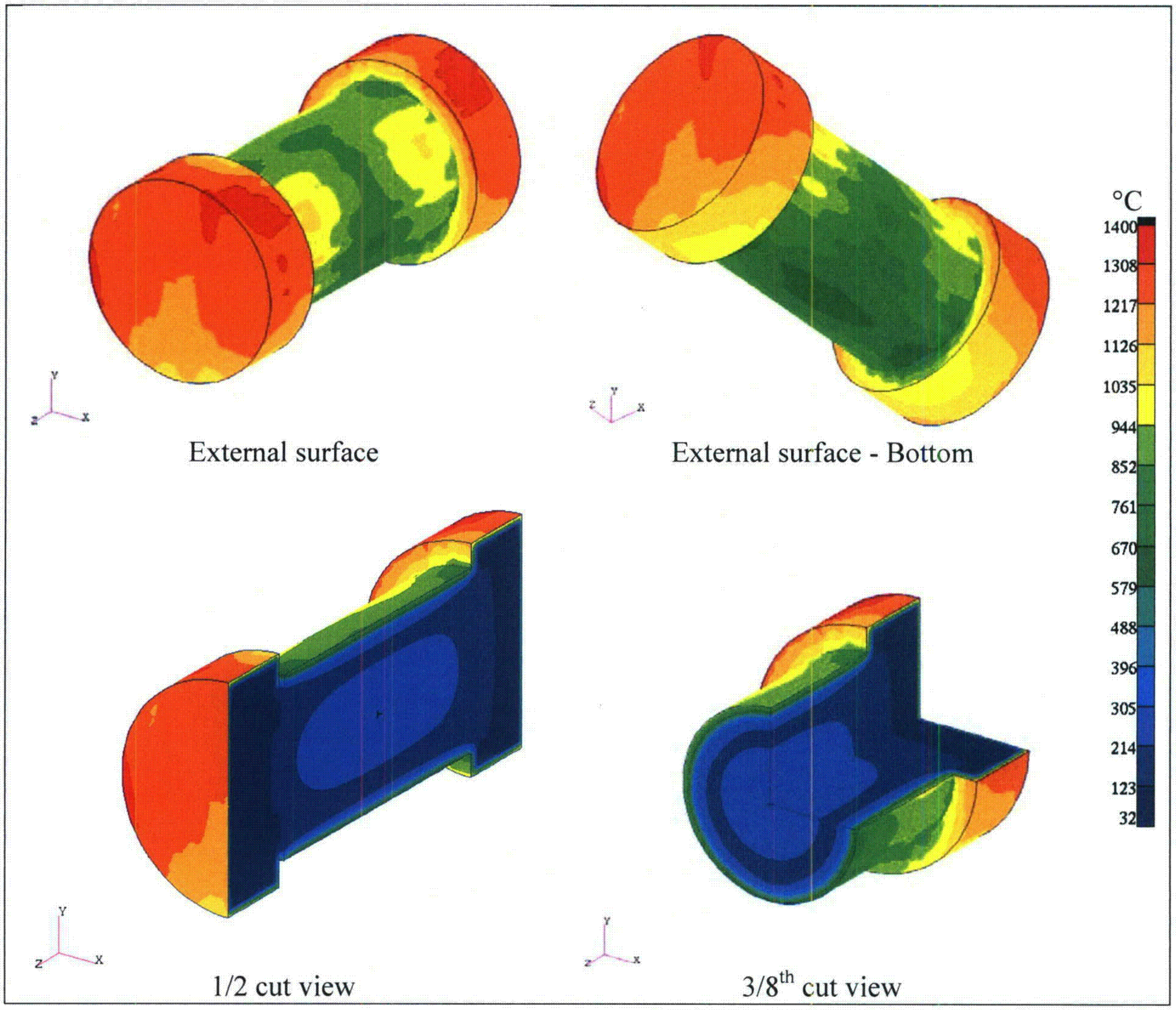


Figure 4-23. Temperature distribution of the Rail-Pb cask at the end of the 30-minute regulatory CAFE fire

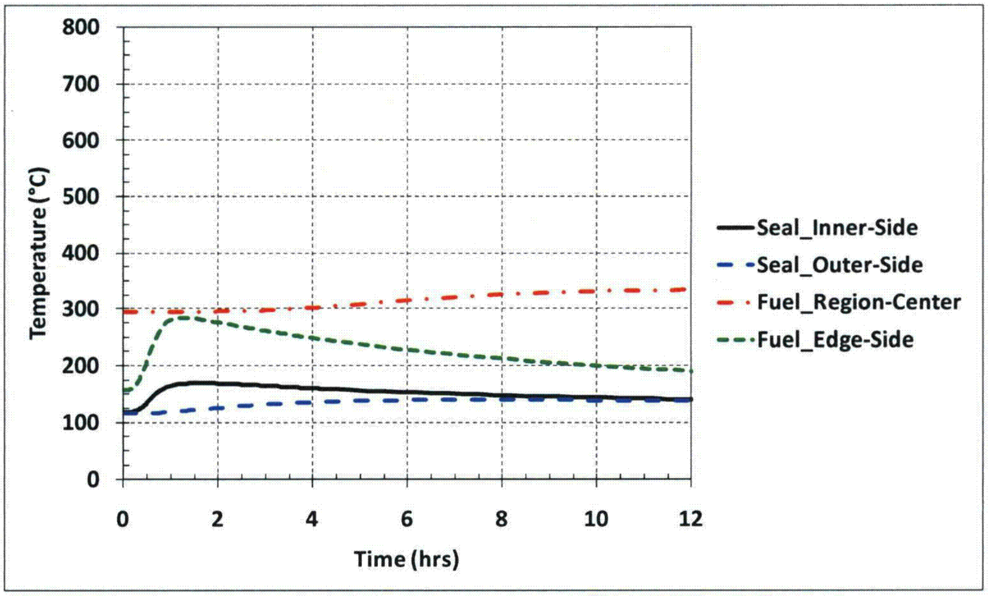


Figure 4-24. Temperature of key cask regions, Rail-Pb cask – Regulatory CAFE fire

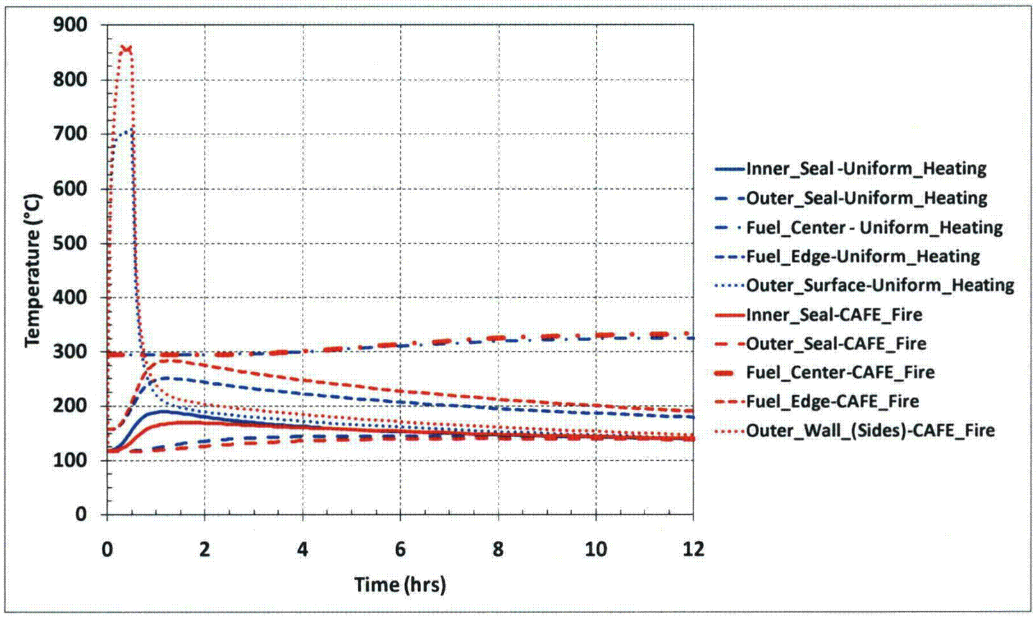


Figure 4-25. Comparison of regulatory fire analysis – Rail-Pb cask: Uniform heating vs. CAFE fire

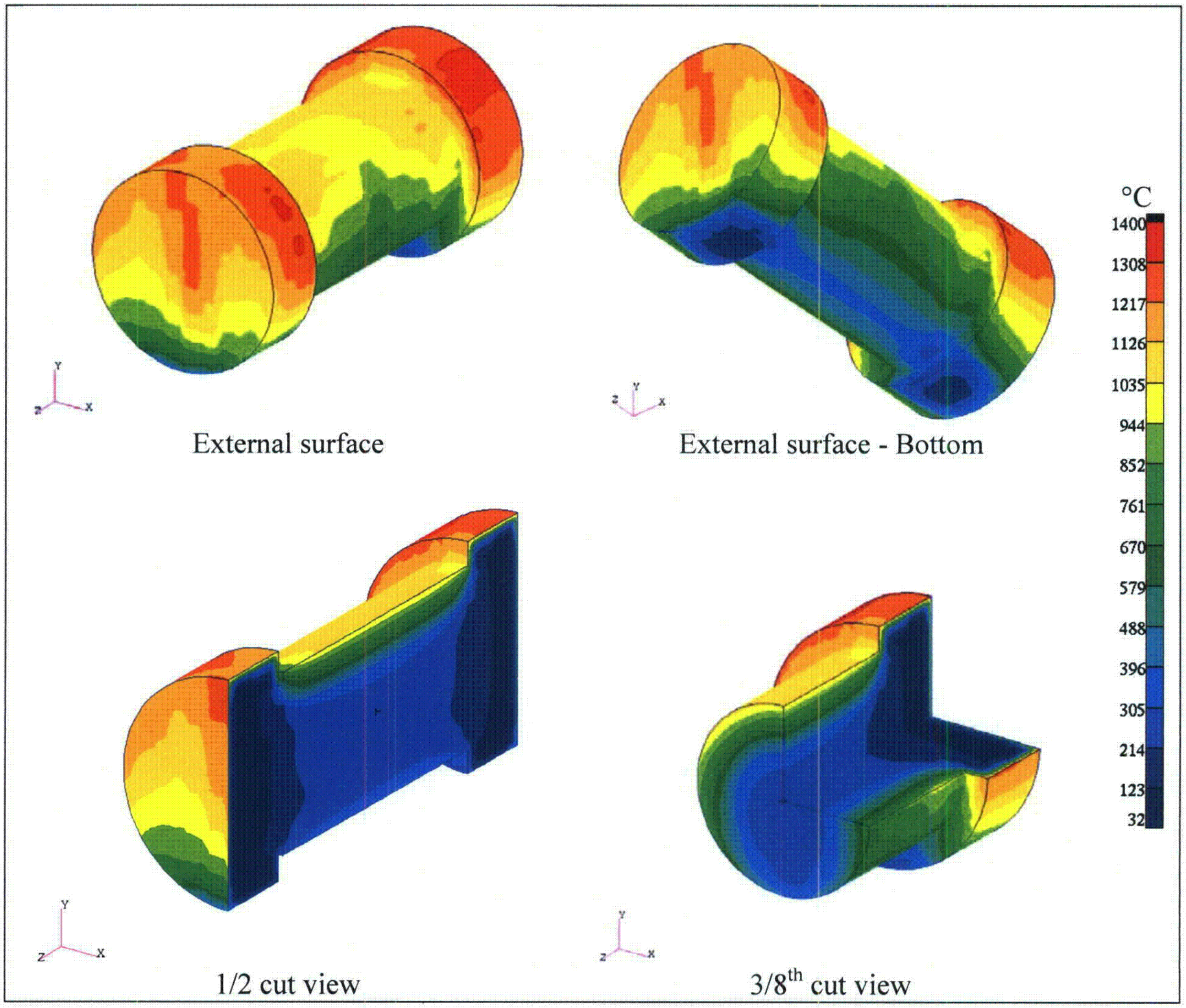


Figure 4-26. Temperature distribution of the Rail-Pb cask at the end of the 3-hour concentric CAFE fire - cask on ground

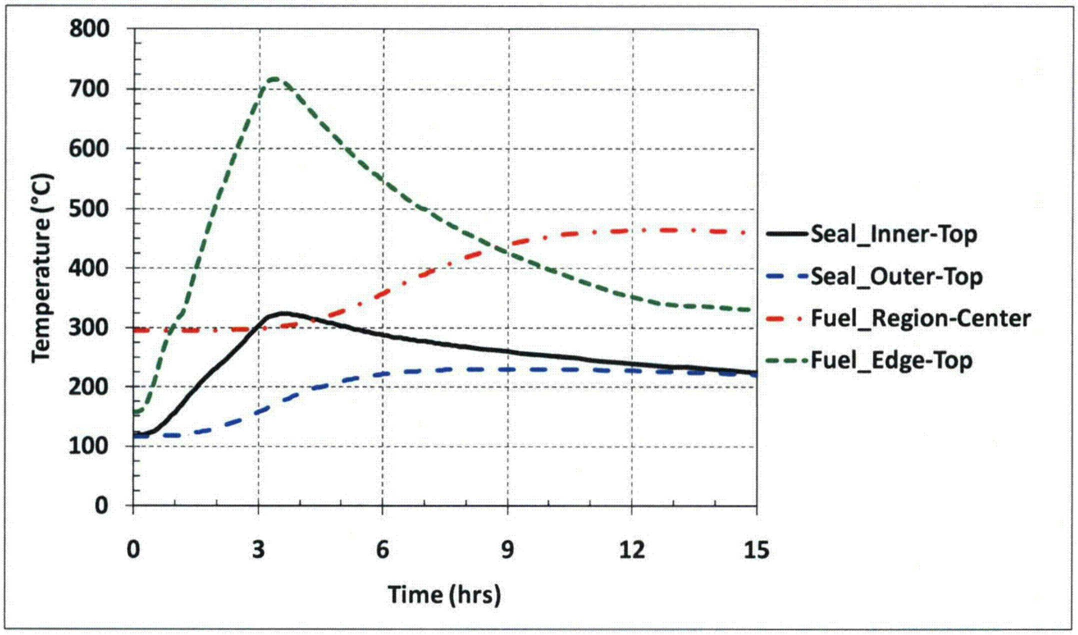


Figure 4-27. Temperature of key cask regions, Rail-Pb cask – Cask on ground, concentric fire

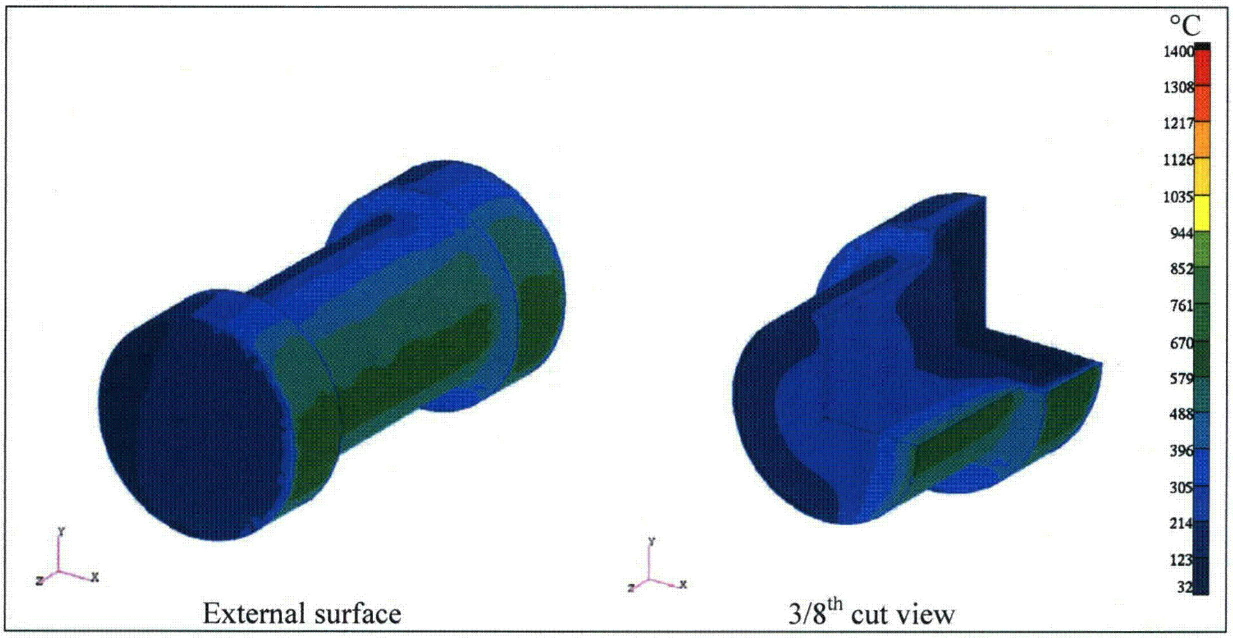


Figure 4-28. Temperature distribution of the Rail-Pb cask at the end of the 3-hour 3m offset CAFE fire - cask on ground

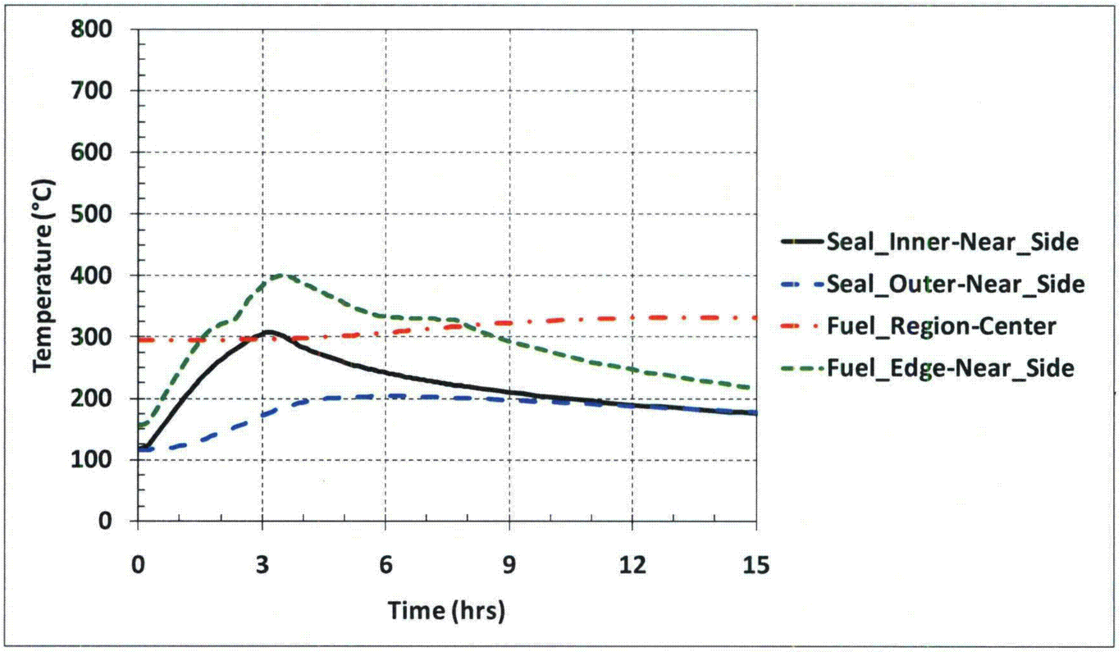


Figure 4-29. Temperature of key cask regions, Rail-Pb cask – Cask on ground, 3m offset fire

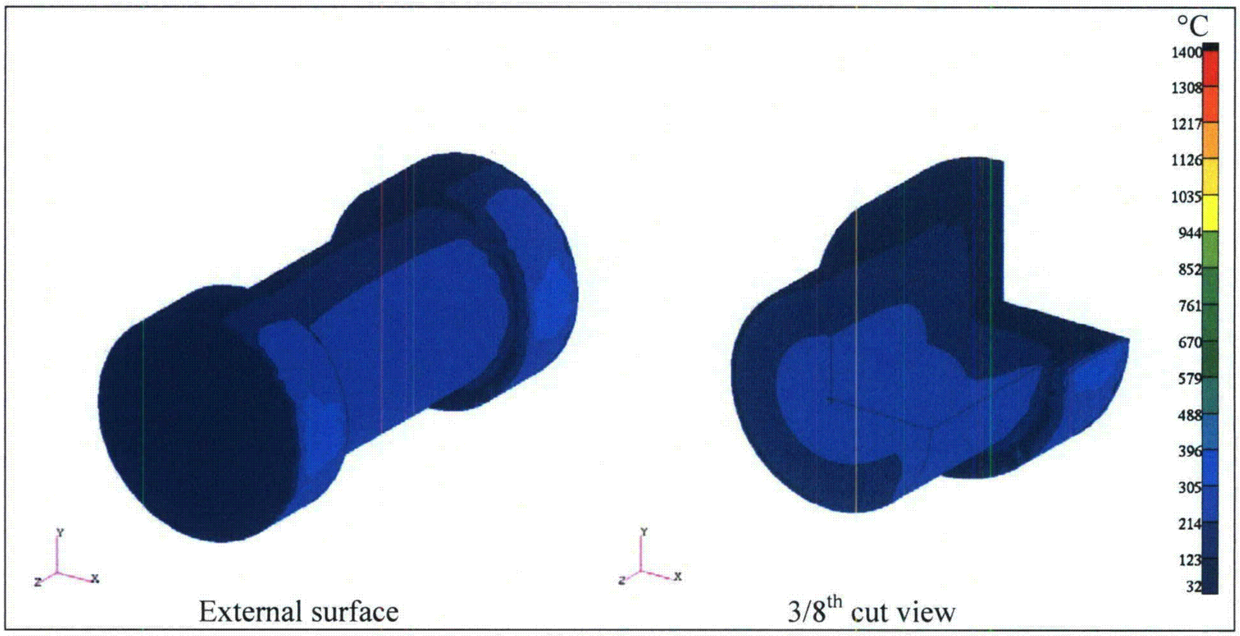


Figure 4-30. Temperature distribution of the Rail-Pb cask at the end of the 3-hour 18m offset CAFE fire - cask on ground

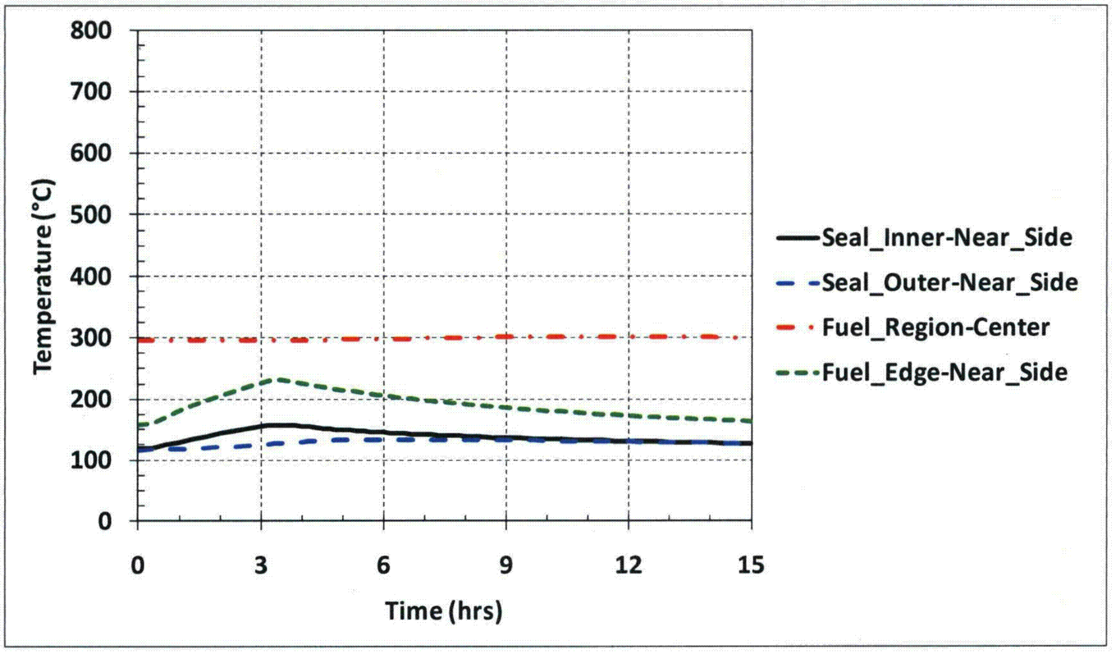


Figure 4-31. Temperature of key cask regions, Rail-Pb cask – Cask on ground, 18m offset fire

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 4-32 and Figure 4-33. Note that these two figures only show the portion of the cask wall that has lead. 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 buckled 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 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. This gap is estimated to be about 0.5m (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 4-33. For this case, the gap is estimated to be about 0.127m (5 inches), which translates to a 2% loss of shielding. These loss-of-shielding fractions are used as part of the work presented in Chapter 5 to estimate the consequences.

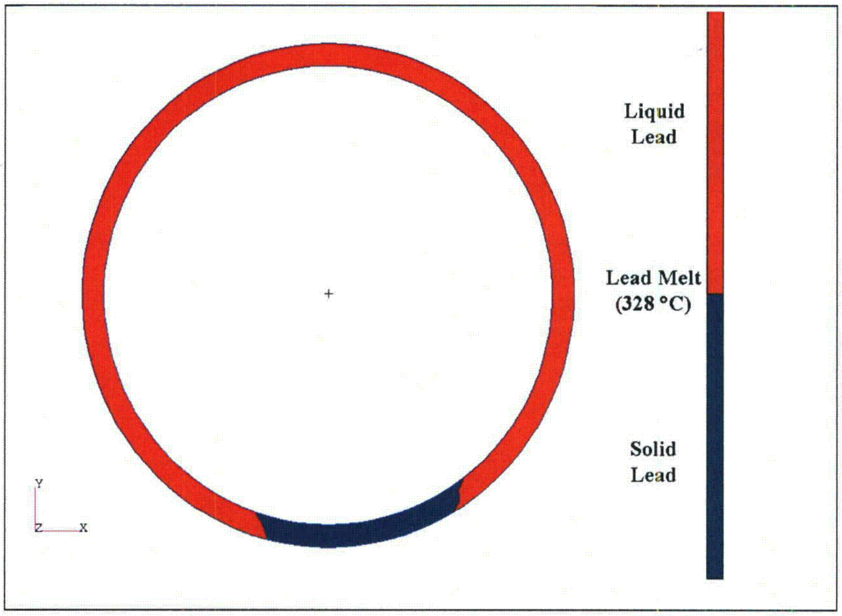


Figure 4-32. Rail-Pb cask lead gamma shield region - maximum lead melt at the middle of the cask – Scenario: Cask on ground, 3-hour concentric pool fire

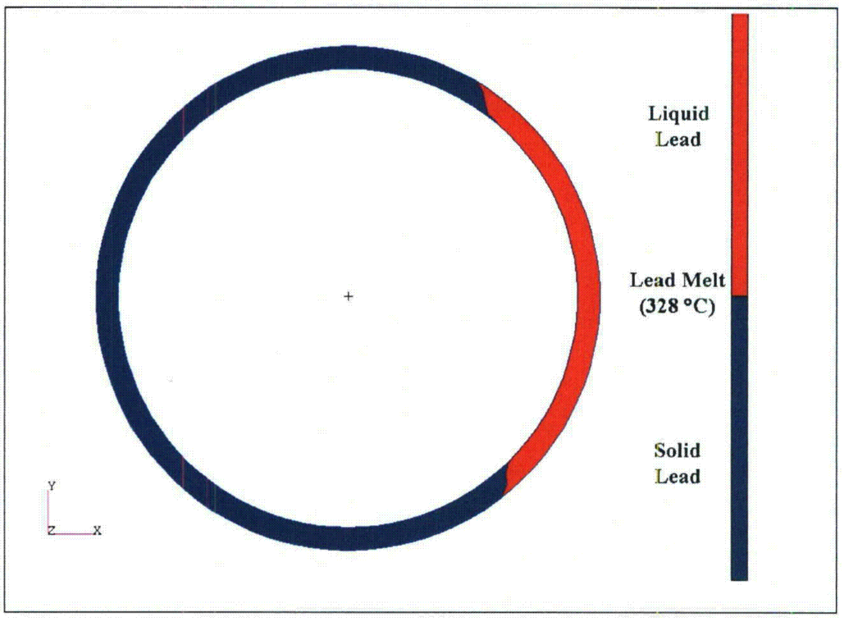


Figure 4-33. Rail-Pb cask lead gamma shield region - maximum lead melt at the middle of the cask – Scenario: Cask on ground, 3-hour 3-meter offset pool fire

Summary of Rail-Pb cask analysis results

The results presented here show that the Rail-Pb 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. 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 under any of the fire environments that are analyzed.

4.4 Truck Cask Analysis

Unlike for the rail casks, detailed three-dimensional FE analyses of the Truck-DU cask are not performed for this risk study. However, NUREG/CR-6672 provides the information necessary to determine the capacity of this cask to withstand the conservative one hour fire duration calculated in Section 4.2.2 of this document.

A comparison of the results obtained from the analyses performed for this study with those found in NUREG/CR-6672 show that the analyses performed for NUREG/CR-6672 were conservative. For example, in NUREG/CR-6672 the seal temperature of the all steel rail cask reached 350°C in about 2.37 hours and in this study the seal temperature of the Rail-Steel cask is about 275°C at the same time. In addition, after the package is exposed to the fire for one hour, the inner surface temperature of the NUREG/CR-6672 cask was 265°C compared to 229°C in this study. Another example, in NUREG/CR-6672 the seal temperature of the steel-lead-steel cask reached 350°C in about 1.69 hours and in this study the inner and outer seal temperatures of the Rail-Pb cask are about 233°C and about 129°C, respectively, at the same time. In addition, after the package was exposed to the fire for one hour, the inner surface temperature of the NUREG/CR-6672 steel-lead-steel cask was 314°C compared to 310°C for the Rail-Pb cask in this study. These results show that the analyses conducted for NUREG/CR-6672 predicted shorter times to seal failure temperature than the more accurate three-dimensional analyses performed for this study. They also show that the analysis in NUREG/CR-6672 was conservative in estimating the inner wall temperature. Therefore, utilizing the results from NUREG/CR-6672 to estimate the response of the Truck-DU cask leads to a conservative estimate of time to seal failure.

From Table 6.7 in NUREG/CR-6672, the duration of fire to cause seal failure is 1.06 hours. The conservative nature of the results in 6672 imply that a longer than a 1.06-hour fire would be required to cause seal failure of the Truck-DU cask. From Section 4.2.2 of this document, the maximum duration of an engulfing fire resulting from a typical petroleum tank truck is estimated to be about one hour. Therefore, the Truck-DU cask is expected to maintain containment in highway fire accidents.

4.5 Conclusions

This chapter presents the realistic analyses of four hypothetical fire accident scenarios. These are the regulatory 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). These analyses are performed for the Rail-Steel and the Rail-Pb casks. Results show that neither the Rail-Steel cask nor the Rail-Pb cask would lose the containment boundary seal in any of the accidents considered in this study. In addition, the fuel rods did not reach burst rupture temperature. However, some loss of gamma shielding is expected with the Rail-Pb 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 from these hypothetical fire accidents. In addition, the Truck-DU cask is able to maintain containment if it were to be exposed to a realistically maximum truck accident fire duration of about an hour. These results demonstrate the adequacy of current regulations to ensure the safe transport of radioactive material.

CHAPTER 5

TRANSPORTATION ACCIDENTS

5.1 Types of Accidents and Incidents

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

- Minor traffic accidents ("fender-benders," flat tires), resulting in minor damage to the vehicle. These are usually called "incidents."[11]
- Accidents which damage the vehicle and or trailer and cask severely 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.
- Accident in which there may be a loss of lead gamma shielding but no release of radioactive material.
- Accidents in which there is a release of radioactive material.

In this analysis the first three types are considered together, since neither type involves a release of radioactive material. In addition, the rail-canistered cask is loaded with canistered fuel, so that even in an accident there would be no release of radioactive material.

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

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. 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 unloading or otherwise opening the cask would be affected. The risk to workers opening a cask of fuel damaged in transit is not included in this study.

5.2 Accident probabilities

Risk is the product of probability and consequence of a particular accident scenario. The probability – 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

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[11] 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 result 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 and the conditional probability. 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 3000-Mile Cross-Country Trip[a]	Accident Type	Net Probability	Conditional Probability Of Accident Type[b]
0.0165	Truck collision with a gasoline tank truck	0.000041	0.82*0.003*0.0165 = 0.000041
0.00138	Rail/truck 50 mph collision at grade crossing	0.0000067	0.7355*0.985*0.0604*0.0113*0.00138 = 0.0000067
0.00087	Railcar falling off bridge at 30 mph	0.00017	0.7355*0.2665*0.9887*0.00087 = 0.00017

[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 mile (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 accident frequency varies somewhat from state to state: the U.S. average for large trucks for the period 1991 to 2007 is 0.0032 large truck accidents per thousand miles. For rail accidents, the average is 0.0018 per thousand railcar miles (DOT, 2008). The DOT has compiled and validated national accident data for truck and rail from 1971 through 2007, but the accident rates declined so sharply between 1971 and the 1990s that, for this analysis, rates from 1991 through 2008 are used. Figure 5-1 shows the accidents per truck mile and per railcar mile for this period. The logarithmic scale is used on the vertical axis in order to show the entire range.

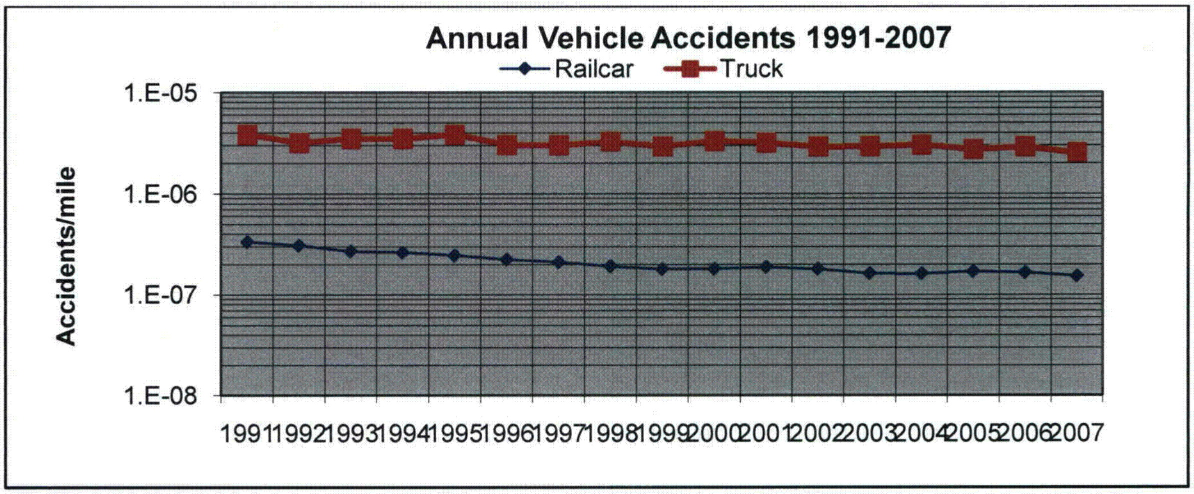


Figure 5-1. Accident frequencies in the U.S. from 1991 until 2007

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-Pb cask. These are:

- Some 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.
- Fires of long enough duration to compromise the seals.

Whether or not these accidents happen depends on the likelihood (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-5, which shows the sequence of events needed for a pool fire that can burn long enough to compromise the seals and the lead shielding.

The sum of all conditional accident probabilities is one (100 percent). Table 5-2 shows the conditional probabilities of accidents that could result in a radiation dose to a member of the public and the conditional probability of an accident in which there is neither loss of lead shielding nor a release of radioactive material; that is, there is no radiation dose to anyone from the accident. The analysis that results in these conditional probabilities may be found in Appendix V, Sections V.3 to V.5.

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

Accident Scenario for the Rail-Pb Cask	Conditional probability of gamma shield loss or release exceeding 10 CFR 71.51 quantities
Loss of lead shielding from impact	5.1×10^{-6}
Loss of lead shielding from fire	10^{-14} to 10^{-10}
Radioactive materials release from impact	3.6×10^{-6}
Radioactive materials release from fire	10^{-14} to 10^{-10}
No radiation dose attributable to the accident: no loss of lead shielding and no release of	0.999991

5.3 Accidents With No Attributable Radiation Dose

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

The doses to the public and to emergency responders from an accident in which no material is released and there is no loss of lead gamma shield are shown in Tables 5-3 and 5-4. These doses depend only on the external dose rate from the cask in the accident. The radiation dose depends on:

- The external dose rate from each 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.
- 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-All Steel accidents, and 99.999% of the Rail-Pb accidents.

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

Cask	Dose in Sv	Ten-hour allowed dose in Sv (mrem) from 10 CFR 71.51
Truck-DU	1.0 E-03	0.10
Rail-Pb	9.2E-04	0.10
Rail-All	6.9E-04	0.10

Table 5-4 and Table 5-5 show collective doses in sieverts (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 a spot on that particular segment of the route.

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

FROM	TO	Rail-Pb			Rail-All Steel		
		Rural	Suburban	Urban	Rural	Suburban	Urban
MAINE YANKEE	ORNL	3.1E-06	5.3E-05	6.6E-	2.3E-	4.0E-05	5.0E-06
	DEAF SMITH	2.3E-06	5.7E-05	6.8E-	1.7E-	4.3E-05	5.2E-06
	HANFORD	3.7E-06	5.3E-05	6.4E-	2.8E-	4.0E-05	4.8E-06
	SKULL VALLEY	2.8E-06	5.1E-05	5.3E-	2.1E-	3.9E-05	4.0E-06
KEWAUNEE	ORNL	3.1E-06	5.7E-05	7.2E-	2.3E-	4.3E-05	5.4E-06
	DEAF SMITH	1.5E-06	6.1E-05	7.2E-	1.2E-	4.6E-05	5.4E-06
	HANFORD	1.5E-06	5.3E-05	6.6E-	1.2E-	4.0E-05	5.0E-06
	SKULL VALLEY	2.0E-06	6.2E-05	6.0E-	1.5E-	4.7E-05	4.5E-06
INDIAN POINT	ORNL	2.6E-06	7.2E-05	8.7E-	2.0E-	5.4E-05	6.6E-06
	DEAF SMITH	1.9E-06	5.9E-05	7.5E-	1.4E-	4.5E-05	5.7E-06
	HANFORD	1.9E-06	5.6E-05	7.2E-	1.4E-	4.3E-05	5.5E-06
	SKULL VALLEY	2.2E-06	6.0E-05	6.6E-	1.7E-	4.6E-05	5.0E-06
IDAHO NATIONAL LAB	ORNL	1.9E-06	6.0E-05	5.8E-	1.4E-	4.6E-05	4.4E-06
	DEAF SMITH	8.0E-07	6.0E-05	5.3E-	6.0E-	4.6E-05	4.0E-06
	HANFORD	1.0E-06	6.0E-05	6.7E-	7.5E-	4.6E-05	5.1E-06
	SKULL VALLEY	2.0E-06	5.9E-05	7.1E-	1.5E-	4.4E-05	5.4E-06
AVERAGE		2.1E-06	5.8E-05	6.7E-	1.6E-	4.4E-05	5.1E-06

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

FROM	TO	Truck-DU		
		Rural	Suburban	Urban
MAINE YANKEE	ORNL	3.8E-06	6.6E-05	8.1E-06
	DEAF SMITH	2.8E-06	7.0E-05	8.4E-06
	HANFORD	4.5E-06	6.5E-05	7.9E-06
	SKULL VALLEY	3.5E-06	6.3E-05	6.6E-06
KEWAUNEE	ORNL	3.8E-06	7.1E-05	8.9E-06
	DEAF SMITH	1.9E-06	7.4E-05	8.9E-06
	HANFORD	1.9E-06	6.5E-05	8.2E-06
	SKULL VALLEY	2.4E-06	7.6E-05	7.4E-06
INDIAN POINT	ORNL	3.2E-06	8.8E-05	1.1E-05
	DEAF SMITH	2.3E-06	7.3E-05	9.2E-06
	HANFORD	2.3E-06	6.9E-05	8.9E-06
	SKULL VALLEY	2.7E-06	7.4E-05	8.2E-06
IDAHO NATIONAL LAB	ORNL	2.4E-06	7.4E-05	7.2E-06
	DEAF SMITH	9.8E-07	7.4E-05	6.6E-06
	HANFORD	1.2E-06	7.4E-05	8.3E-06
	SKULL VALLEY	2.4E-06	7.2E-05	8.8E-06
AVERAGE		2.6E-06	7.2E-05	8.3E-06

The average individual U.S. background dose for ten hours is 4.1×10^{-6} Sv (0.41 mrem). Average background doses for the 16 routes analyzed are:

- Rural: 6.9×10^{-5} person-Sv
- Suburban: 1.9×10^{-3} person-Sv
- Urban: 0.011 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: 7.2×10^{-5} person-Sv
- Suburban: 2.0×10^{-3} person-Sv
- Urban: 0.011 person-Sv

The suburban and urban collective doses would be indistinguishable from the collective background dose. Any dose to an individual is well below the doses allowed by 10 CFR 71.51, as one would expect. The total collective doses may be slightly less from Rail-Pb and Rail-All Steel casks than from a Truck-DU cask.

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 carry very radioactive material and need shielding in additional to that provided by the steel shell of the package. 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 (10 mrem) per hour at two meters from the cask, by regulation (10 CFR 71.4). The three cask types tested in this assessment meet this criterion.

Each spent fuel transportation cask analyzed each use a different gamma shield. The Rail-All Steel has a stainless steel wall thick enough to attenuate gamma radiation to acceptable levels. The Truck-DU uses metallic depleted uranium (DU). Neither of these shields would be damaged or even affected by, an accident. The Rail-Pb has a lead gamma shield which can be damaged 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, or fractions of shield lost. In the figures, the largest gaps are at the left end of the graph, and the gap size decreases from left to right. Figure 5-2 and Figure 5-3 show that doses larger than the external dose in 10 CFR 71.51 occur when the lead shielding gap is more than two percent of the shield.

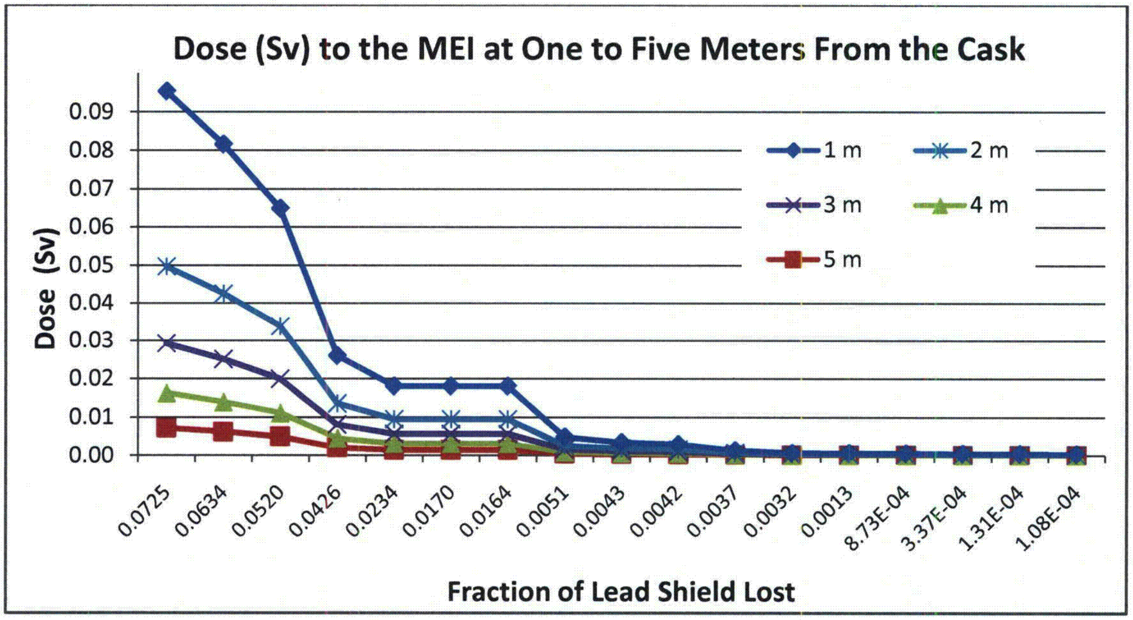


Figure 5-2. Radiation dose to the maximally exposed individual (MEI) from loss of lead gamma shielding at distances from one to five meters from the cask carrying spent fuel. The horizontal axis represents the fraction of shielding lost–the shielding gap—and is not to scale.

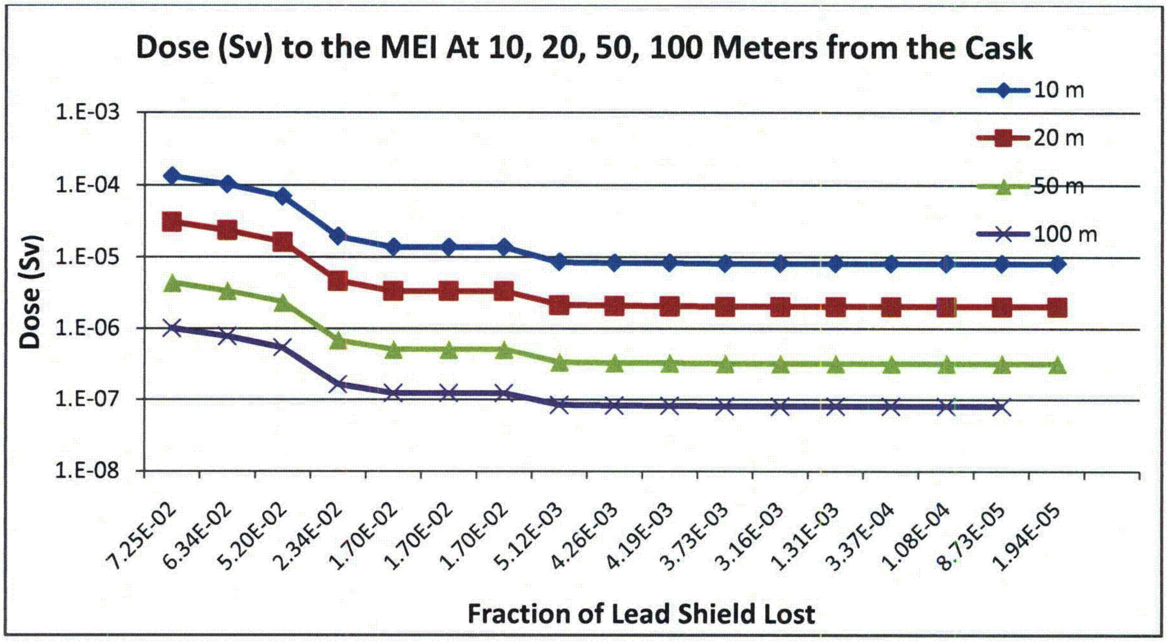


Figure 5-3. Radiation dose to the maximally exposed individual from loss of lead gamma shielding at distances from 20 to 100 meters from the cask carrying spent fuel. The vertical axis is logarithmic, so that all of the doses can be shown on the same graph. The horizontal axis represents the fraction of shielding lost–the shielding gap—and is not to scale.

Doses that are larger than the 10 CFR 71.51 accident doses pose a greater risk than the doses from routine transportation (Chapter 2) or those from an accident in which there is neither a release nor loss of shielding (Section 5.3 of this chapter). The probability of an accident resulting in these doses is a significant component of the risk. Table 5-2 shows that the probability of an impact accident causing loss of lead shielding is five per million (5×10^{-6}), or one in 200,000. The probability that the dose from that accident will be larger than allowed by 10 CFR 71.51 is less: about three per hundred million (3×10^{-8}) because the dose resulting from most accidents is less than the limit in 10 CFR 51.71.

One of every 200,000 accidents could be an impact accident that causes loss of lead shielding; the "one in 200,000 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 will happen on the route from Indian Point Nuclear Plant to Hanford is:

$$(5 \times 10^{-6})*(0.00178) = 8.9 \times 10^{-9}$$

or about one in 100 million. The probability that the lead shielding loss is large enough to deliver an acute dose is:

$$(3 \times 10^{-8})(0.00178) = 5.34 \times 10^{-11}$$

or about one in 10 billion.

These very small probabilities reflect the conclusion that such severe accidents, which are more traumatic to the cask than the tests shown in Figure 1-1, are not likely to happen. The conditions which can cause enough loss of lead shielding to challenge human health are extreme conditions.

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

ORIGIN	DESTINATION	AVERAGE ACCIDENTS PER KM	AVERAGE ACCIDENTS
MAINE YANKEE	ORNL	6.5×10^{-7}	0.00328
	DEAF SMITH	5.8×10^{-7}	0.00195
	HANFORD	4.2×10^{-7}	0.00178
	SKULL VALLEY	5.1×10^{-7}	0.00108
KEWAUNEE	ORNL	4.3×10^{-7}	0.00328
	DEAF SMITH	3.3×10^{-7}	0.00130
	HANFORD	2.4×10^{-7}	0.00062
	SKULL VALLEY	3.7×10^{-7}	0.00066
INDIAN POINT	ORNL	8.8×10^{-6}	0.00052
	DEAF SMITH	6.2×10^{-7}	0.04206
	HANFORD	5.1×10^{-7}	0.00190
	SKULL VALLEY	5.5×10^{-7}	0.00203
INL	ORNL	3.6×10^{-7}	0.00069
	DEAF SMITH	3.5×10^{-7}	0.00038
	HANFORD	3.2×10^{-7}	0.00067
	SKULL VALLEY	2.8×10^{-7}	0.00015

The overall collective dose risks to the resident population from a lead shielding loss accident on the sixteen routes studied, are shown in Table 5-7. These include some accidents that are within regulatory limits. 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 in person-Sv for a loss of lead shielding accident

SHIPMENT ORIGIN	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	4.4E-10	2.7E-10	2.4E-10	1.4E-10
KEWAUNEE	1.9E-10	9.1E-11	8.6E-11	7.7E-11
INDIAN POINT	7.4E-09	2.8E-10	2.8E-10	1.0E-10
IDAHO NATIONAL LAB	5.6E-11	9.5E-11	2.1E-11	1.3E-10

The conditional probability that lead shielding will be lost in 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 hot enough, and burns long enough, to do any damage to the lead shield.

- The train must be in an accident that results in a major derailment
- The train carrying the spent fuel cask must also be carrying at least one tank car of flammable material.
- The derailment must result in a pileup. Railcars carrying spent fuel casks are always located between buffer cars and never located next 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.

If there is no pileup and if the cask is more that a railcar length from the fire, although still close enough that the lead shield could be damaged, the probability is increased to about 10^{-10}—about one in ten billion.

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

5.4.2 Neutron shielding

The type of fuel which can be transported in the three casks considered has relatively low neutron emission but does require neutron shielding. This usually a hydrocarbon or carbohydrate polymer of some type that often contains a boron compound (borax is a good neutron absorber). Water is an excellent neutron absorber, but is no longer used. 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.

Impacts, 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 destroyed in a fire.

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

Cask	Dose in Sv	Ten-hour allowed dose in Sv from 10 CFR 71.51
Truck-DU	0.0073	1.00
Rail-Pb	0.0076	1.00
Rail-All	0.0076	1.00

The neutron doses do not exceed the dose cited in the regulation following an accident, so the loss of neutron shield is not included in the overall risk assessment. Essentially, these are not extra-regulatory accidents.. The conditional probability of this neutron dose is 0.0063 for a truck fire accident and 0.0000001 for a rail accident. The rail fire is less probable because of the series of events needed to produce a rail fire. 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; mass and energy are conserved.

The fuel studied in this analysis is pressurized water reactor (PWR) fuel that has "burned" 45,000 megawatt-days per metric ton of uranium (MWD/MTU) and has been cooled for nine years. The Rail-Pb cask, the only cask studied that could release radioactive material in an accident, is certified to carry more than 20 PWR assemblies. In this study, the Rail-Pb cask was loaded with 26 PWR assemblies.

The spent fuel inventory for accident analysis was selected by normalizing the radionuclide concentrations in the spent fuel by radiotoxicity. The resulting inventory is shown in Table 5-9.

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

Radionuclide	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
^{233}U	8.99E+02
^{241}Pu	5.75E-01
^{113m}Cd	6.13E-01

The ^{60}Co inventory listed is not part of the nuclear fuel. It is the main constituent of a corrosion product that 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, described in Chapter 3, can 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 r fuel rods to the cask interior and from the cask interior to the environment is found in Appendix V Sections V.5.4.1 and V.5.4.2.

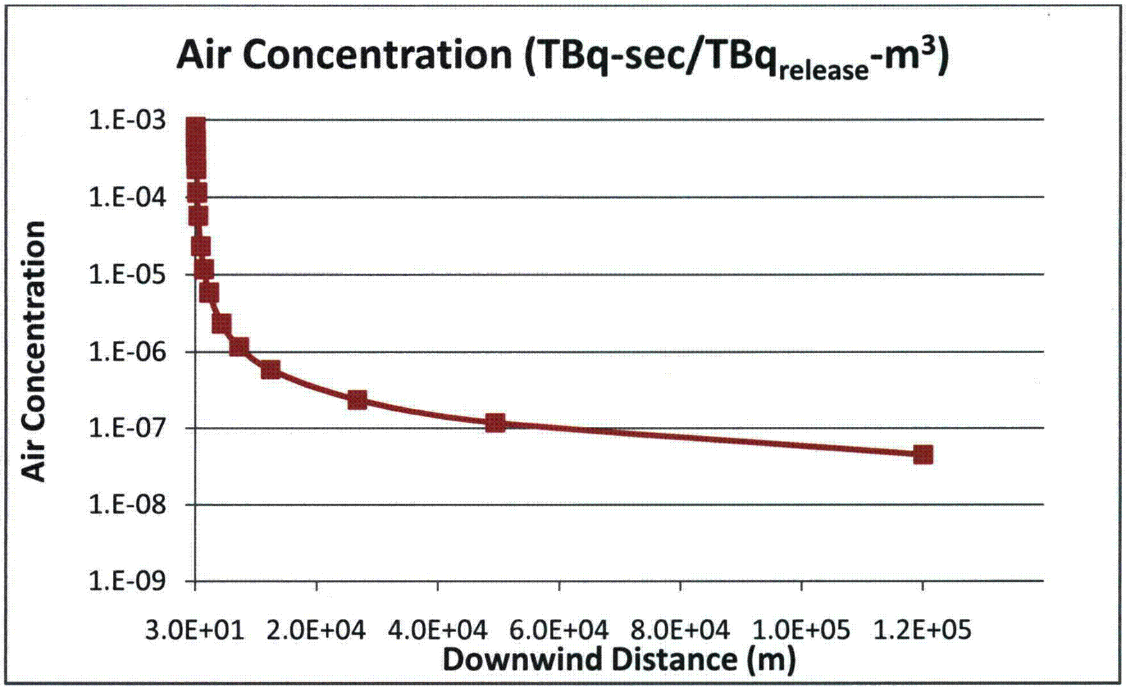
Table 5-10. Parameters for determining release functions for the accidents that would result in release of radioactive material

	Cask Orientation	End	Corner	Side	Side	Side	Side	Corner
	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**	0.005	0.005	0.005	0.005	0.005	0.005	0.005
	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	2.68E-08	1.61E-07	8.02E-08	8.02E-08	8.02E-08	8.02E-08	3.06E-06

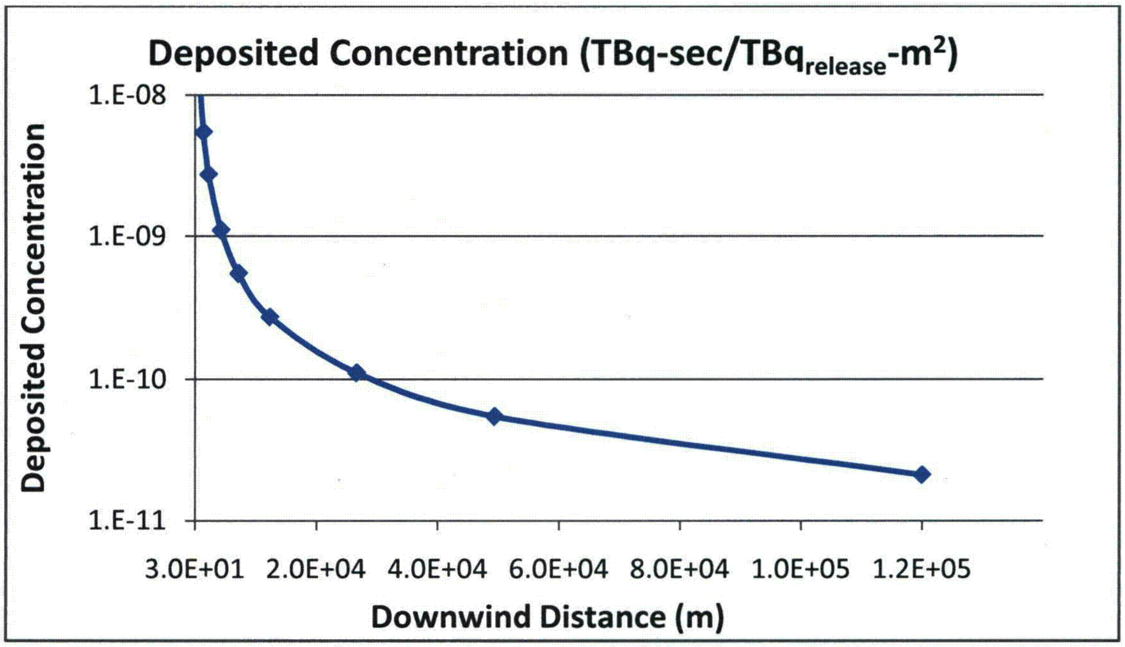
5.5.3 Dispersion

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 release is elevated, and the maximum air concentration and ground 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. These concentrations 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 an 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.

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

Cask Orientation	Impact Speed (kph)	Dose from No-release, No-shielding-loss Accident	Inhalation	Resus-pension	Cloud-shine	Ground-shine	Total
End	193	0.001	1.59	0.0137	0.0001	0.0009	1.60
Corner	193	0.001	1.59	0.0137	0.0001	0.0009	1.60
Side	193	0.001	1.59	0.0137	0.0001	0.0009	1.60
Side	193	0.001	1.59	0.0137	0.0001	0.0009	1.60
Side	145	0.001	1.58	0.0137	4.53E-06	3.61E-05	1.59
Side	145	0.001	1.59	0.0137	8.78E-05	9.42E-04	1.60
Corner	145	0.001	0.7270	0.0063	0.0001	0.0009	0.7340

The doses listed in Table 5-11 are consequences, not risks. The dose to the maximally exposed individual is not the sum of the doses. Each cask orientation is a different accident scenario and results in a different set of inhalation and external doses. These are significant doses, but they are not acute, and none would result in either acute illness or death (Shleien et al., 1998, p. 15-3). The inhalation and groundshine doses are listed separately because they have different physiological effects. External doses are exactly that, and the receptor would be suffer 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.

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. The NRC uses the total effective dose equivalent, the sum of the inhalation and external doses, as a measure.

A pool fire co-located with the cask and burning for a long enough time, could damage the seals severely. However, as has already been mentioned, and is discussed in detail in Appendix V Section V.3.1.2, the condiotional probability of the series of events required to produce such a fire scenario is about 10^{-19} 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 dose risk from the universe of release accidents is shown in Table 5-12.

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

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	2.3E-10	1.4E-10	1.2E-10	6.1E-11
KEWAUNEE	9.8E-11	4.7E-11	4.6E-11	3.3E-11
INDIAN POINT	3.9E-09	1.5E-10	1.5E-10	4.9E-11
IDAHO NATIONAL LAB	1.9E-05	7.6E-07	8.6E-10	2.6E-08

These dose risks and cancer risks are negligible by any standard.

The total dose risks from loss-of-lead shielding accidents is shown in Table 5-13, 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 shielding accident

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	4.4E-10	2.7E-10	2.4E-10	1.4E-10
KEWAUNEE	1.9E-10	9.1E-11	8.6E-11	7.7E-11
INDIAN POINT	7.4E-09	2.8E-10	2.8E-10	1.0E-10
IDAHO NATIONAL LAB	5.6E-11	9.5E-11	2.1E-11	1.3E-10

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

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	6.7E-10	4.1E-10	3.6E-10	2.0E-10
KEWAUNEE	2.9E-10	1.4E-10	1.3E-10	1.1E-10
INDIAN POINT	1.1E-08	4.3E-10	4.4E-10	1.5E-10
IDAHO NATIONAL LAB	1.9E-05	7.6E-07	8.8E-10	2.6E-08

Table 5-15 shows the total collective dose risk for an accident where there is neither loss of lead shielding nor a release;

Table 5-15. Total collective dose risk (person-Sv) from no-release, no-loss of shielding accidents

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	2.07E-07	1.29E-07	1.12E-07	6.42E-08
KEWAUNEE	2.22E-07	9.00E-08	3.80E-08	4.62E-08
INDIAN POINT	4.31E-08	2.88E-06	1.24E-07	1.40E-07
IDAHO NATIONAL LAB	4.71E-08	2.52E-08	4.56E-08	1.02E-08

Table 5-16 shows the collective accident risk for the 16 routes from loss of neutron shielding

Table 5-16. Total collective dose risk (person-Sv) from loss of neutron shielding

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	5.2E-09	3.5E-09	3.6E-09	1.5E-09
KEWAUNEE	3.3E-09	1.9E-09	2.2E-09	1.1E-09
INDIAN POINT	4.5E-09	2.9E-09	3.2E-09	1.1E-09
IDAHO NATIONAL LAB	7.6E-10	1.9E-09	2.4E-10	2.9E-09

Table 5-17 shows the collective dose risk for the 16 routes for all accidents, for the Rail-Pb cask. Of the three casks in this study, only the Rail-Pb is damaged in each kind of accident considered.

Table 5-17. Total collective dose risk (person-Sv) from all accidents

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	2.1E-07	1.3E-07	1.2E-07	6.6E-08
KEWAUNEE	2.3E-07	9.2E-08	4.0E-08	4.7E-08
INDIAN POINT	5.9E-08	2.9E-06	1.3E-07	1.4E-07
IDAHO NATIONAL LAB	1.9E-05	7.9E-07	4.7E-08	3.9E-08

5.6 Conclusions

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

- The sixteen routes selected for study are an adequate representation of U.S. routes for spent nuclear fuel, 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.
- 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 are larger than cited in the regulation of 10 CFR 71.51, and are significant, but are neither acute nor lethal.
- These results are not unexpected.

CHAPTER 6

OBSERVATIONS AND CONCLUSIONS

The present document is an assessment (or evaluation) of the risks of transporting spent nuclear fuel. It is also an update to NUREG-0170, the 1977 Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes. Both NUREG-0170 and this document provide a technical basis for the regulations of 10 CFR Part 71.

Regulation is different from risk assessment, and the technical basis for a regulation is not the regulation itself. A regulation must be conservative, because its purpose is to ensure safety, and 10 CFR Part 71, which regulates transportation, requires a conservative estimate of the damage to a cask in an accident and the radiation emitted from the cask during routine transportation. The original technical basis for 10 CFR Part 71, NUREG-0170, was also conservative, but for a different reason: only limited data were available to perform the required assessment, so NUREG-0170 deliberately used conservative parameter estimates. The NRC's conclusion was that NUREG-0170 showed transportation of radioactive materials to be safe enough, even with conservative assumptions, to support the regulation.

However, assessment and evaluation are not regulation and serve a different purpose. An assessment should be as realistic as possible. 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 present study is closer to a "real world" analysis than the previous analyses. Certified spent fuel cask types 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 the risks of spent nuclear fuel transportation, and an addition to the technical basis for 10 CFR Part 71. The results of this study are compared with preceding risk assessments in the figures that follow.

6.1 Routine Transportation

Figure 6-1and Figure 6-2 show results of routine truck and rail transportation of a single shipment of spent nuclear fuel; Figure 6-1 plots average collective radiation dose (person-Sv) from truck transportation and Figure 6-2 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.

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. As Figure 6-1 shows, the doses in this study from routine transportation are about 35 percent of the analogous collective doses in NUREG/CR-6672 and about 10% of the analogous NUREG-

0170 results. When the doses in the truck analysis are normalized to the population, vehicle density, and vehicle speed used in the NUREG/CR-6672 analysis, the collective doses are about seven percent of the NUREG/CR-6672 doses.

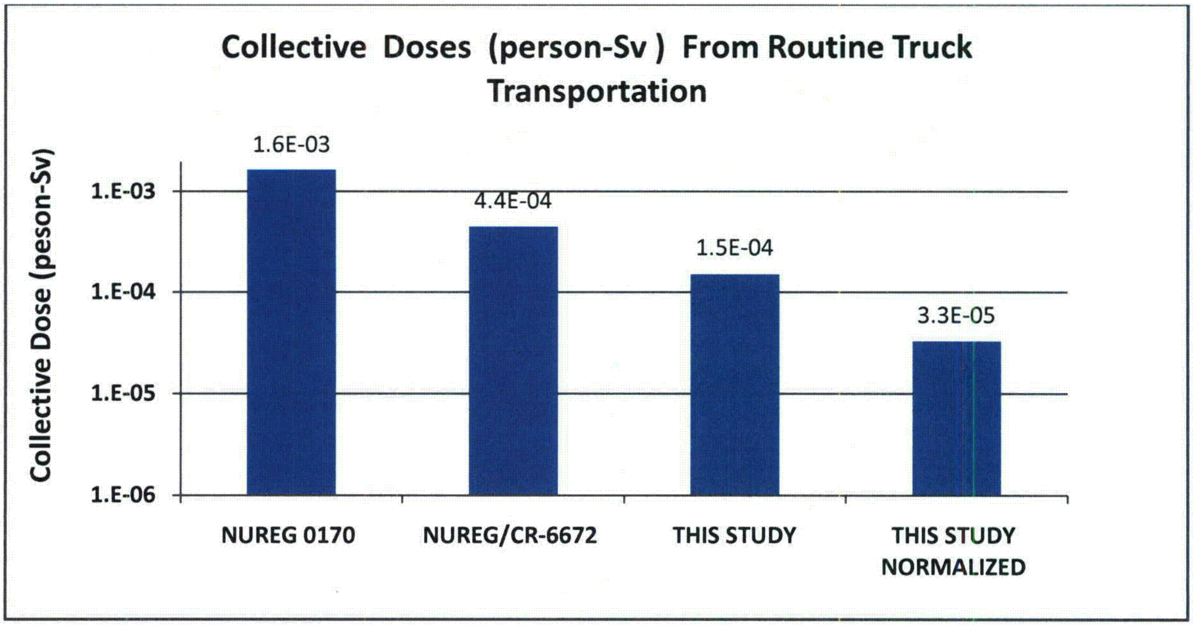


Figure 6-1. Collective doses (person-Sv) from routine truck transportation

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 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 the figure shows, the conservatism was decreased by over a factor of three.

The collective average dose in the present study might have been larger than the NUREG/CR-6672 result, because the populations are generally larger, particularly along rural routes, and the vehicle densities are much larger (see Chapter 2). These increases were offset by the considerably larger vehicle speeds used in the present study. The difference made by normalizing to the NUREG/CR-6672 input parameter values demonstrates that the collective dose depends on the number of people exposed, not on the dose to which they are exposed. The population exposed to the transportation cask is also exposed to background radiation. Thus, even in comparisons, collective dose is an artificial construct with limited relevance to an assessment of radiological effects.

Figure 6-2 shows a more predictable difference between the present study's results and NUREG/CR-6672. This difference reflects primarily the increase in population density along the rail routes. Doses to rail crew are considerably larger because crew are exposed during travel over the entire route (although this involves different individuals) Crew doses, including railyard worker doses, are shown in Figure 6-3.

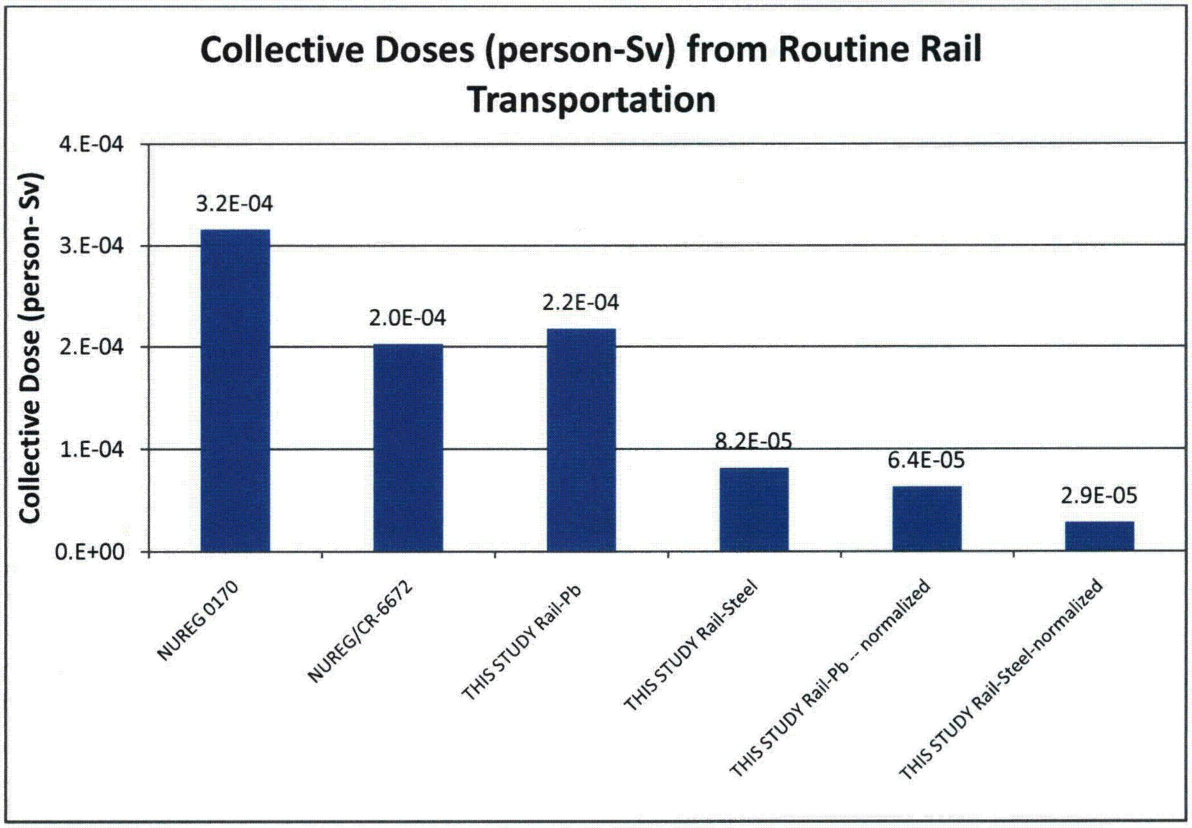


Figure 6-2. Collective doses (person-Sv) from routine rail transportation

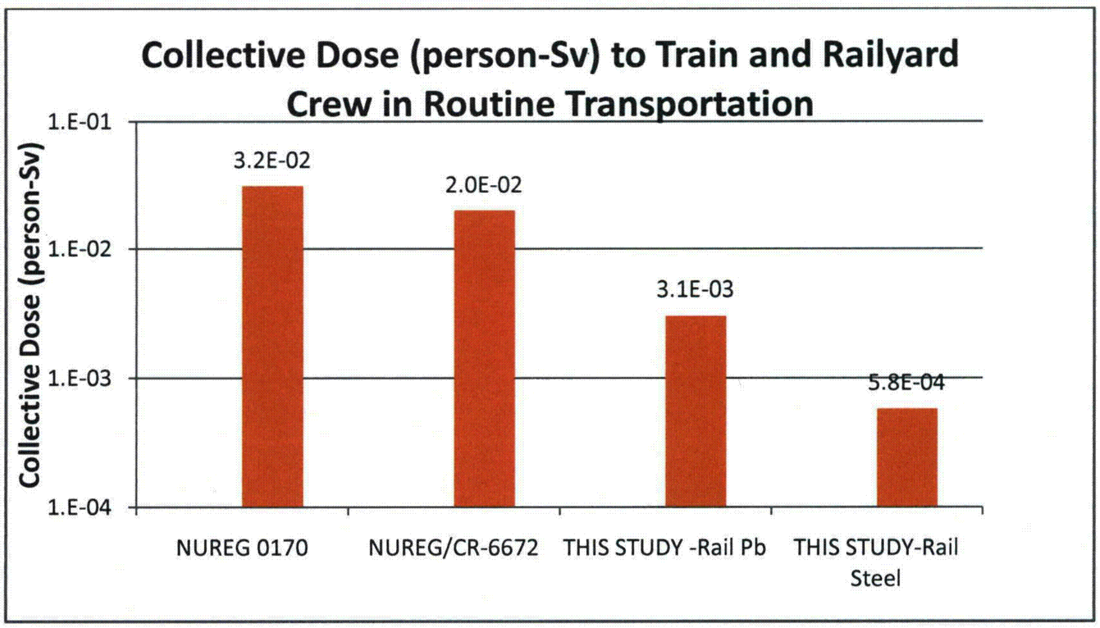


Figure 6-3. Collective dose (person-sv) to train and railyard crew in routine transportation

The difference in dose between the Rail-Pb cask and the Rail-Steel cask occurs because the latter cask has a smaller TI (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 [12] defines the same limit on external radiation (the transport index) as Part 71 defines today. The differences in results are due primarily to the following factors:

- 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 for all truck routes and 64 kph on rural rail routes, 40 kph on suburban rail routes, and 24 kph on urban rail routes. The truck speeds used in this study are 108 kph on rural routes, 102 kph on suburban routes, and 97 kph on urban routes and the rail speed is 40 kph on all routes .
- 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 for urban routes. NUREG/CR-6672 used 1990 census data provided by the code WebTRAGIS and used the mean values of Gaussian distributions of population densities on 200 routes in the United States. This study uses 2000 census data, updated to 2009, for the rural, suburban, and urban truck and rail route segments in each state traversed in each of the sixteen routes studies. 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.
- 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.

Dose to the maximally exposed individual is a better indication of the radiological effect of routine transportation than collective dose. The dose to the maximally exposed individual is shown in Figure 6-4 for NUREG-0170 and for the three cask types of this study. NUREG/CR-6672 did not calculate this dose for routine transportation.

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

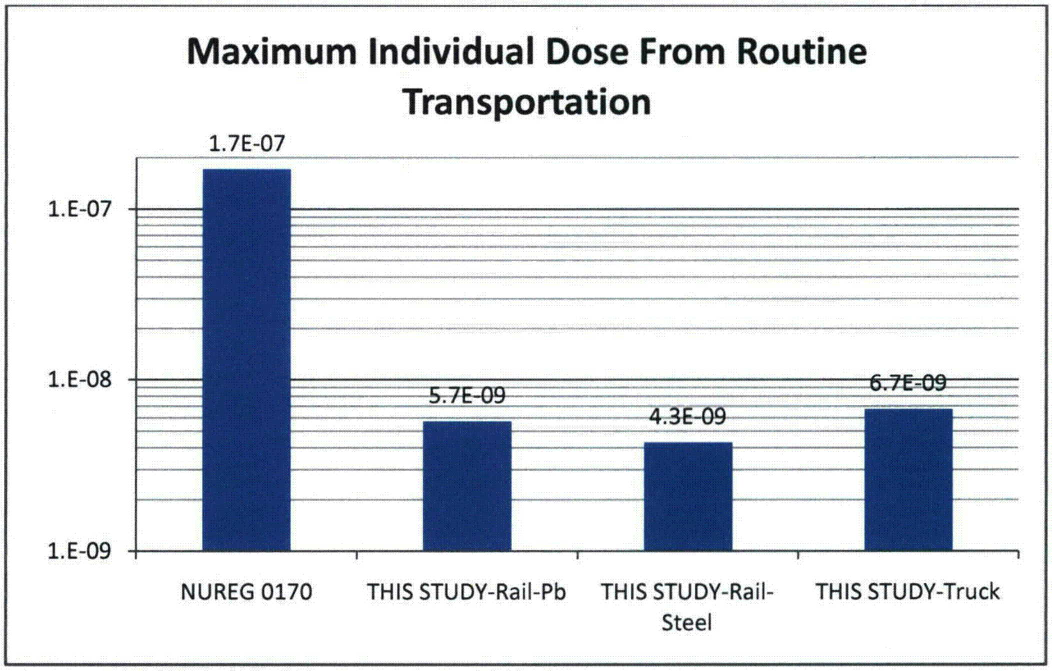

Figure 6-4. Maximum individual dose (Sv) from routine transportation.

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 RADTRAN in a currently available version to calculate dose risk, but the input parameters differed widely. 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 6-5 for this study are averages over the 16 routes studied. As was discussed in Chapters 3, 4, and 5, a lead-shielded rail cask, the Rail-Pb cask in this study, is the only cask type of the three studied that can either release radioactive material in a traffic accident or can lose lead gamma shielding.

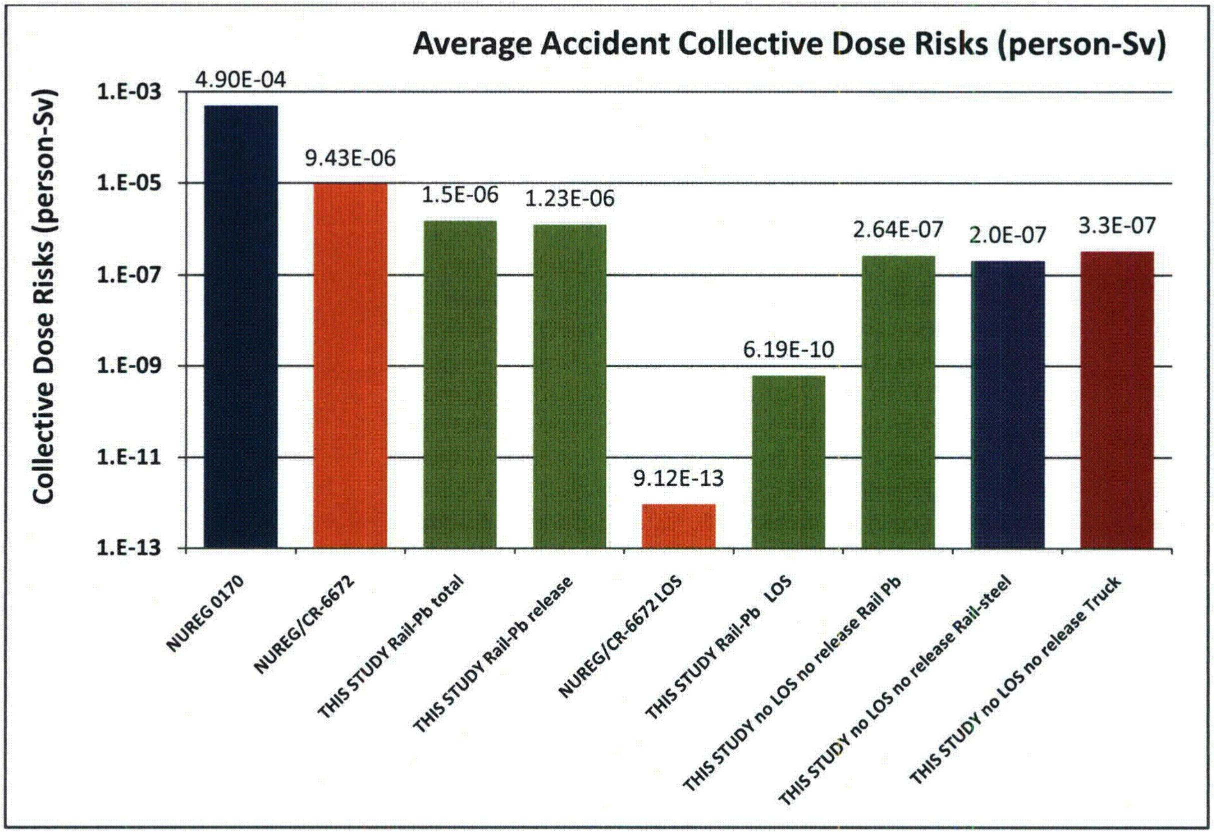

Figure 6-5. Accident collective dose risks

Unlike the results for routine transportation, the results shown in Figure 6-5 depend on different amounts of radioactive material released and different amounts of lead shielding lost. 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, 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 Chalk River Unidentified Deposits (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-Pb cask design met criteria necessary for 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 quantity of material potentially released and the overall conditional probability of release decreases with each successive study.

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 6-5.

Frequently, people who are concerned about the transportation of spent fuel want to know about consequences instead of risk. The average consequences (collective doses) from potential accidents involving release, about 110 person-Sv in NUREG-0170 and about 9000 person-Sv estimated from Figure 8.27 in NUREG/CR-6672 are orders of magnitude larger than the 2 person-Sv estimated from release in the current study.

NUREG-0170 did not consider loss of lead gamma shielding, which can increase the dose from gamma radiation significantly. 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 damaged shield and with 18 potential accident scenarios instead of 10. Much of the difference between the NUREG/CR-6672 dose risks from shield loss and this study is the inclusion of accident scenarios that have a higher conditional probability than any such scenarios in NUREG/CR-6672. The consequence of 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. Lead shield loss clearly affects only casks that have a lead gamma shield; casks using DU or thicker steel shielding would not be affected.

More than 99.999 percent of potential accident scenarios do not affect the cask at all and would not result in either release of radioactive material nor increased dose from loss of lead shielding. However, these accidents would result in an increased dose from the cask external radiation to the population near the accident, because the cask remains at the location of the accident until it can be moved. A nominal ten hours was assumed for this study. The resulting collective dose risk from this accident is shown in Figure 6-5 for all three cask types studied. Even including this additional consequence type, the collective dose risk from this study is less than that reported in either NUREG-0170 or NUREG/CR-6672.

In conclusion, the three studies reviewed here 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 loss of shielding or release of radioactive material are improbable and the consequences of such unlikely accidents are serous but not dire. Moreover, these consequences depend on the population exposed rather than on the radiation or radioactive material released. The consequences (doses) to the maximally exposed individual, 1.6 Sv from a release and 1.1 Sv from loss of lead shielding, are chronic rather than acute doses.

The most significant consequence of an accident, in addition to any non-radiological consequence of the accident itself, is the external dose from a cask immobilized at the accident location. 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.

Public perception of radiological risk of transportation does not appear to recognize that such risk depends much more on artifacts of calculation, parameter selection, and assumption than on the amount of radiation emitted. The conservative estimates of NUREG-0170 may have inadvertently contributed to this misperception. The more realistic the analysis, the greater the likelihood of redirecting public perception.

APPENDIX I

CASK DETAILS AND CERTIFICATES OF COMPLIANCE

Table of Contents

I.1 Cask Descriptions

I.1.1 Truck Casks

GA-4

Steel-DU-steel design – stiffer than lead casks, smaller deformations
The 4 PWR assembly capacity of this cask makes it the likely workhorse truck cask for any large transportation campaign.
Elastomeric seals (ethylene propylene) – allows larger closure deformations before leakage
Polymer neutron shielding
Larger capacity means larger radioactive material inventory and larger possible consequence from an accident that produces the same size of leak
Design is from the late 80s – General Atomics used finite element analyses in certification
DU shielding is made from 5 segments – possible segment-to-segment problems
Cask body has a square cross-section – this provides more possible orientations
Aluminum honeycomb impact limiter

NAC-LWT

Steel-lead-steel design – relatively flexible – should have plastic deformation of body before seal failure
Contains either a single PWR assembly or two BWR assemblies
Both elastomeric and metallic seals – low compression of elastomeric seal (metallic is primary) – allows little closure movement before leakage but may have better performance in a fire.
Lead shielding – could melt during severe fires (leads to loss of shielding)
Liquid neutron shielding – tank is likely to fail in extra-regulatory impacts
Bottom end impact limiter is attached to neutron shielding tank – makes side drop analysis more difficult
Aluminum honeycomb impact limiter
Cask is very similar to generic steel-lead-steel cask from 6672
Cask is being used for FRR shipments

I.1.2 Rail Casks

NAC-STC

Steel-lead-steel design – relatively flexible – should have plastic deformation of body before seal failure
Certified for both direct loaded fuel and for fuel in a welded canister
Contains either 26 directly loaded PWR assemblies or 1 Transportable Storage Container (3 configurations, all for PWR fuel)

Can have either elastomeric or metallic seals – must choose a configuration for analysis
Lead shielding – could melt during severe fires (leads to loss of shielding)
Polymer neutron shielding
Wood impact limiter (redwood and balsa)
Cask is similar to the steel-lead-steel rail cask from 6672
Two casks have been built and are being used outside of the US

NAC-UMS

Steel-lead-steel design – relatively flexible – should have plastic deformation of body before seal failure
Fuel in welded canister
24 PWR assemblies or 56 BWR assemblies
Elastomeric seals (EPDM) – allows larger closure deformations before leakage
Lead shielding – could melt during severe fires (leads to loss of shielding)
Polymer neutron shielding
Wood impact limiter (redwood and balsa)
Cask is similar to the steel-lead-steel rail cask from 6672
Cask has never been built

HI-STAR 100

Layered all-steel design
Fuel in welded canister
24 PWR assemblies or 68 BWR assemblies
Metallic seals – allows little closure deformations before leakage
Polymer neutron shielding
Aluminum honeycomb impact limiters
At least 7 have been built and are being used for dry storage, no impact limiters have been built
Is proposed as the transportation cask for the Private Fuel Storage facility (PFS)

TN-68

Layered all-steel design
Directly loaded fuel
68 BWR assemblies
Metallic seals – allows little closure deformations before leakage
Polymer neutron shielding
Wood impact limiter (redwood and balsa)
At least 24 have been built and are being used for dry storage, no impact limiters have been built

MP-187

Steel-lead-steel design – relatively flexible – should have plastic deformation of body before seal failure
Fuel in welded canister
24 PWR assemblies
Metallic seals – allows little closure deformations before leakage
Hydrogenous neutron shielding
Aluminum honeycomb/polyurethane foam impact limiters (chamfered rectangular parallelepiped)
Cask has never been built

MP-197

Steel-lead-steel design – relatively flexible – should have plastic deformation of body before seal failure
Fuel in welded canister
61 BWR assemblies
Elastomeric seals – allows larger closure deformations before leakage
Hydrogenous neutron shielding
Wood impact limiter (redwood and balsa)
Cask has never been built

TS125

Steel-lead-steel design – relatively flexible – should have plastic deformation of body before seal failure
Fuel in welded canister
21 PWR assemblies or 64 BWR assemblies
Metallic seals – allows little closure deformations before leakage
Polymer neutron shielding
Aluminum honeycomb impact limiters
Cask has never been built

I.2 Certificates of Compliance

NRC FORM 618 — U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9261	7	71-9261	USA/9261/B(U)F-96	1	OF	7

PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10 Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

ISSUED TO (*Name and Address*)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)* Revision 12, dated October 9, 2006, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: HI-STAR 100 System

(2) Description

The HI-STAR 100 System is a canister system comprising a Multi-Purpose Canister (MPC) inside of an overpack designed for both storage and transportation (with impact limiters) of irradiated nuclear fuel. The HI-STAR 100 System consists of interchangeable MPCs that house the spent nuclear fuel and an overpack that provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the overpack of the HI-STAR 100 is approximately 96 inches without impact limiters and approximately 128 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is 282,000 pounds. Specific tolerances germane to the safety analyses are called out in the drawings listed below. The HI-STAR 100 System includes the HI-STAR 100 Version HB (also referred to as the HI-STAR HB).

Multi-Purpose Canister

There are seven Multi-Purpose Canister (MPC) models designated as the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, MPC-68F, and the MPC-HB. All MPCs are designed to have identical exterior dimensions, except 1) MPC-24E/EFs custom-designed for the Trojan plant, which are approximately nine inches shorter than the generic Holtec MPC design; and 2) MPC-HBs custom-designed for the Humboldt Bay plant, which are approximately 6.3 feet shorter than the generic Holtec MPC designs The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 series is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies; the MPC-32 is designed to contain up to 32 intact PWR assemblies; and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. The MPC-HB is designed to contain up to 80 Humboldt Bay BWR fuel assemblies

The HI-STAR 100 MPC is a welded cylindrical structure with flat ends Each MPC is an assembly consisting of a honeycombed fuel basket, baseplate, canister shell, lid, and closure ring The outer diameter and cylindrical height of each generic MPC is fixed. The outer diameter of the Trojan MPCs is the same as the generic MPC, but the height is approximately nine inches shorter than the generic MPC design. A steel spacer is used with the Trojan plant MPCs to ensure the MPC-overpack interface is bounded by the generic design The outer diameter of the Humboldt Bay MPCs is the same as the generic MPC, but the height is approximately 6.3 feet shorter than the generic MPC design. The Humboldt Bay MPCs are transported in a shorter version of the HI-STAR overpack, designated as the HI-STAR HB. The fuel basket designs vary based on the MPC model.

Overpack

The HI-STAR 100 overpack is a multi-layer steel cylinder with a welded baseplate and bolted lid (closure plate). The inner shell of the overpack forms an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate steel shells for radiation shielding. The overpack closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the overpack inner shell, bottom plate, top flange, top closure plate, top closure inner O-ring seal, vent port plug and seal, and drain port plug and seal.

Impact Limiters

The HI-STAR 100 overpack is fitted with two impact limiters fabricated of aluminum honeycomb completely enclosed by an all-welded austenitic stainless steel skin. The two impact limiters are attached to the overpack with 20 and 16 bolts at the top and bottom, respectively.

(3) Drawings

The package shall be constructed and assembled in accordance with the following drawings or figures in Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12, as supplemented:

5 (a)(3) Drawings (continued)

(a) HI-STAR 100 Overpack	Drawing 3913, Sheets 1-9, Rev. 9
(b) MPC Enclosure Vessel	Drawing 3923, Sheets 1-5, Rev 16
(c) MPC-24E/EF Fuel Basket	Drawing 3925, Sheets 1-4, Rev. 5
(d) MPC-24 Fuel Basket Assembly	Drawing 3926, Sheets 1-4, Rev 5
(e) MPC-68/68F/68FF Fuel Basket	Drawing 3928, Sheets 1-4, Rev 5
(f) HI-STAR 100 Impact Limiter	Drawing C1765, Sheet 1, Rev. 4; Sheet 2, Rev. 3; Sheet 3, Rev. 4, Sheet 4, Rev. 4; Sheet 5, Rev. 2; Sheet 6, Rev. 3; and Sheet 7, Rev 1.
(g) HI-STAR 100 Assembly for Transport	Drawing 3930, Sheets 1-3, Rev. 2
(h) Trojan MPC-24E/EF Spacer Ring	Drawing 4111, Sheets 1-2, Rev. 0
(i) Damaged Fuel Container for Trojan Plant SNF	Drawing 4119, Sheet 1-4, Rev. 1
(j) Spacer for Trojan Failed Fuel Can	Drawing 4122, Sheets 1-2, Rev. 0
(k) Failed Fuel Can for Trojan	SNC Drawings PFFC-001, Rev. 8 and PFFC-002 Sheets 1 and 2, Rev. 7
(l) MPC-32 Fuel Basket Assembly	Drawing 3927, Sheets 1-4, Rev. 6
(m) HI-STAR HB Overpack	Drawing 4082, Sheets 1-7, Rev. 3
(n) MPC-HB Enclosure Vessel	Drawing 4102, Sheets 1-4, Rev. 1
(o) MPC-HB Fuel Basket	Drawing 4103, Sheets 1-3, Rev. 5
(p) Damaged Fuel Container HB	Drawing 4113, Sheets 1-2, Rev. 1

5.(b) Contents

(1) Type, Form, and Quantity of Material

(a) Fuel assemblies meeting the specifications and quantities provided in Appendix A to this Certificate of Compliance and meeting the requirements provided in Conditions 5.b(1)(b) through 5.b(1)(i) below are authorized for transportation.

5.(b)(1) Type, Form, and Quantity of Material (continued)

(b) The following definitions apply:

Damaged Fuel Assemblies are fuel assemblies with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered FUEL DEBRIS.

Damaged Fuel Containers (or Canisters) (DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris that permit gaseous and liquid media to escape while minimizing dispersal of gross particulates.

The DFC designs authorized for use in the HI-STAR 100 are shown in Figures 1.2.10, 1.2.11, and 1.I.1 of the HI-STAR 100 System SAR, Rev. 12, as supplemented.

Fuel Debris is ruptured fuel rods, severed rods, loose fuel pellets, and fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage, including containers and structures supporting these parts. Fuel debris also includes certain Trojan plant-specific fuel material contained in Trojan Failed Fuel Cans.

Incore Grid Spacers are fuel assembly grid spacers located within the active fuel region (i.e., not including top and bottom spacers).

Intact Fuel Assemblies are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s). Trojan fuel assemblies not loaded into DFCs or FFCs are classified as intact assemblies.

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRA), Thimble Plug Devices (TPDs), and Rod Cluster Control Assemblies (RCCAs).

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

5.(b)(1)(b) Definitions (continued)

Trojan Damaged Fuel Containers (or Canisters) are Holtec damaged fuel containers custom-designed for Trojan plant damaged fuel and fuel debris as depicted in Drawing 4119, Rev. 1

Trojan Failed Fuel Cans are non-Holtec designed Trojan plant-specific damaged fuel containers that may be loaded with Trojan plant damaged fuel assemblies, Trojan fuel assembly metal fragments (e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.), a Trojan fuel rod storage container, a Trojan Fuel Debris Process Can Capsule, or a Trojan Fuel Debris Process Can. The Trojan Failed Fuel Can is depicted in Drawings PFFC-001, Rev. 8 and PFFC-002, Rev. 7.

Trojan Fuel Debris Process Cans are Trojan plant-specific canisters containing fuel debris (metal fragments) and were used to process organic media removed from the Trojan plant spent fuel pool during cleanup operations in preparation for spent fuel pool decommissioning. Trojan Fuel Debris Process Cans are loaded into Trojan Fuel Debris Process Can Capsules or directly into Trojan Failed Fuel Cans. The Trojan Fuel Debris Process Can is depicted in Figure 1.2.10B of the HI-STAR100 System SAR, Rev. 12, as supplemented.

Trojan Fuel Debris Process Can Capsules are Trojan plant-specific canisters that contain up to five Trojan Fuel Debris Process Cans and are vacuumed, purged, backfilled with helium and then seal-welded closed. The Trojan Fuel Debris Process Can Capsule is depicted in Figure 1.2.10C of the HI-STAR 100 System SAR, Rev. 12, as supplemented.

Undamaged Fuel Assemblies are fuel assemblies where all the exterior rods in the assembly are visually inspected and shown to be intact. The interior rods of the assembly are in place; however, the cladding of these rods is of unknown condition. This definition only applies to Humboldt Bay fuel assembly array/class 6x6D and 7x7C.

ZR means any zirconium-based fuel cladding materials authorized for use in a commercial nuclear power plant reactor.

(c) For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the stainless steel clad fuel assemblies or the applicable ZR clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more

5.(b)(1)(b) Definitions (continued)

restrictive of the decay heat limits for the damaged fuel assemblies or the intact fuel assemblies

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the 6x6A, 6x6B, 6x6C, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies

(f) PWR non-fuel hardware and neutron sources are not authorized for transportation except as specifically provided for in Appendix A to this CoC.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

(h) For spent fuel assemblies to be loaded into MPC-32s, core average soluble boron, assembly average specific power, and assembly average moderator temperature in which the fuel assemblies were irradiated, shall be determined according to Section 1.2.3.7.1 of the SAR, and the values shall be compared against the limits specified in Part VI of Table A.1 in Appendix A of this Certificate of Compliance.

(i) For spent fuel assemblies to be loaded into MPC-32s, the reactor records on spent fuel assemblies average burnup shall be confirmed through physical burnup measurements as described in Section 1.2.3.7.2 of the SAR.

5.(c) Criticality Safety Index (CSI)= 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed. At a minimum, those procedures shall include the provisions provided in Chapter 7 of the HI-STAR SAR.

(b) All acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for acceptance testing and maintenance shall be developed and shall include the provisions provided in Chapter 8 of the HI-STAR SAR.

7. The maximum gross weight of the package as presented for shipment shall not exceed 282,000 pounds, except for the HI-STAR HB, where the gross weight shall not exceed 187,200 pounds.

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 9 feet (along the axis of the overpack) from the edge of the vehicle.

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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9 The personnel barrier shall be installed at all times while transporting a loaded overpack.

10 The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11 Transport by air of fissile material is not authorized.

12 Revision No. 6 of this certificate may be used until May 31, 2010.

13 Expiration Date: March 31, 2014

Attachment: Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12, dated October 9, 2006.

Holtec International supplements dated June 29, July 27, August 3, September 27, October 5, and December 18, 2007; January 9, March 19 and September 30, 2008; and February 27, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: May 8, 2009

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 7

MODEL NO. HI-STAR 100 SYSTEM

INDEX TO APPENDIX A

Page	Table.	Description:
Page A-1 to A-23	Table A.1	Fuel Assembly Limits
Page A-1		MPC-24: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-2		MPC-68: Uranium oxide, BWR intact fuel assemblies listed in Table A.3 with or without Zircaloy channels
A-3		MPC-68: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6X6C, 7x7A, or 8x8A
A-4		MPC-68: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-5		MPC-68: Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-6		MPC-68: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters
A-7		MPC-68F: Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-8		MPC-68F: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-9		MPC-68F: Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.

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Page:	Table:	Description.
A-10	Table A. 1 (Cont'd)	MPC-68F: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-11		MPC-68F. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-12		MPC-68F: Mixed Oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-13		MPC-68F: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters.
A-15		MPC-24E: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-16		MPC-24E: Trojan plant damaged fuel assemblies.
A-17		MPC-24EF: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-18		MPC-24EF: Trojan plant damaged fuel assemblies.
A-19		MPC-24EF: Trojan plant Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris.
A-20 to A-21		MPC-32: Uranium oxide, PWR intact fuel assemblies in array classes 15X15D, E, F, and H and 17X17A, B, and C as listed in Table A.2.
A-22 to A-23		MPC-HB: Uranium oxide, intact and/or undamaged fuel assemblies and damaged fuel assemblies, with or without channels, meeting the criteria specified in Table A.3 for fuel assembly array/class 6x6D or 7x7C.
A-24 to A-27	Table A.2	PWR Fuel Assembly Characteristics
A-28 to A-33	Table A.3	BWR Fuel Assembly Characteristics

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Table A.1 (Page 1 of 23)
Fuel Assembly Limits

MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment.	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
d. Decay heat per assembly:	
i. ZR Clad:	≤833 Watts
ii. SS Clad:	≤488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware or neutron sources.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-24.

E. Trojan plant fuel is not permitted to be transported in the MPC-24.

II MPC MODEL: MPC-68

A Allowable Contents

1 Uranium oxide, BWR intact fuel assemblies listed in Table A.3, except assembly classes 6x6D and 7x7C, with or without Zircaloy channels, and meeting the following specifications:

a Cladding type	ZR or stainless steel (SS) as specified in Table A.3 for the applicable fuel assembly array/class
b Maximum planar-average initial enrichment.	As specified in Table A.3 for the applicable fuel assembly array/class.
c Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies, which shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.45 wt% ^{235}U, and (2) array/class 8x8F fuel assemblies, which shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a minimum initial enrichment ≥ 2.4 wt% ^{235}U.
ii. SS clad:	An assembly cooling time after discharge ≥ 16 years, an average burnup ≤ 22,500 MWD/MTU, and a minimum initial enrichment ≥ 3.5 wt% ^{235}U.
e. Decay heat per assembly:	
i. ZR Clad:	≤272 Watts, except for array/class 8X8F fuel assemblies, which shall have a decay heat ≤183.5 Watts.
a. SS Clad:	≤83 Watts
f. Fuel assembly length:	≤ 176.2 inches (nominal design)
g. Fuel assembly width:	≤ 5.85 inches (nominal design)
h Fuel assembly weight:	≤ 700 lbs, including channels

II MPC MODEL MPC-68 (continued)

A Allowable Contents (continued)

2 Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications.

a Cladding type	ZR
b Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.45 wt% ^{235}U.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 550 lbs, including channels and damaged fuel containers

Table A.1 (Page 4 of 23)
Fuel Assembly Limits

II MPC MODEL: MPC-68 (continued)

A Allowable Contents (continued)

3 Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications.

a Cladding type	ZR
b Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A 1 (Page 5 of 23)
Fuel Assembly Limits

II MPC MODEL: MPC-68 (continued)

A Allowable Contents (continued)

4 Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a Cladding type	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B
c. Initial maximum rod enrichment.	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers.

II MPC MODEL MPC-68 (continued)

A Allowable Contents (continued)

5 Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1 2 11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications

a Cladding type	ZR
b Composition:	98.2 wt % ThO_2. 1.8 wt. % UO_2 with an enrichment of 93.5 wt % ^{235}U.
c Number of rods per Thoria Rod Canister:	≤ 18
d. Decay heat per Thoria Rod Canister:	≤ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time ≥ 18 years and an average burnup ≤ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	≤ 27 kg/canister
g. Fuel cladding O.D.:	≥ 0.412 inches
h. Fuel cladding I.D.:	≤ 0.362 inches
i. Fuel pellet O.D.:	≤ 0.358 inches
j. Active fuel length:	≤ 111 inches
k. Canister weight:	≤ 550 lbs, including fuel

B. Quantity per MPC: Up to one (1) Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

III MPC MODEL MPC-68F

A Allowable Contents

1 Uranium oxide. BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A and meet the following specifications:

a Cladding type	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.45 wt% ^{235}U.
e. Fuel assembly length:	≤ 176.2 inches (nominal design)
f. Fuel assembly width:	≤ 5.85 inches (nominal design)
g. Fuel assembly weight:	≤ 400 lbs, including channels

Table A 1 (Page 8 of 23)
Fuel Assembly Limits

III MPC MODEL: MPC-68F (continued)

A Allowable Contents (continued)

2 Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type.	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

III MPC MODEL MPC-68F (continued)

A Allowable Contents (continued)

3 Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.45 wt% ^{235}U for the original fuel assembly.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 550 lbs, including channels and damaged fuel containers

III MPC MODEL MPC-68F (continued)

A Allowable Contents (continued)

4 Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a Cladding type:	ZR
b Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A 1 (Page 11 of 23)
Fuel Assembly Limits

III MPC MODEL: MPC-68F (continued)

A Allowable Contents (continued)

5. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a Cladding type	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

Table A.1 (Page 12 of 23)
Fuel Assembly Limits

III MPC MODEL: MPC-68F (continued)

A Allowable Contents (continued)

6 Mixed oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications

a Cladding type	ZR
b Maximum planar-average initial enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods in the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

Table A 1 (Page 13 of 23)
Fuel Assembly Limits

III MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7 Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2 11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications

a Cladding Type	ZR
b Composition.	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

III MPC MODEL MPC-68F (continued)

B Quantity per MPC.

Up to four (4) damaged fuel containers containing uranium oxide or MOX BWR fuel debris The remaining MPC-68F fuel storage locations may be filled with array/class 6x6A. 6x6B. 6x6C, 7x7A. and 8x8A fuel assemblies of the following type, as applicable

1 Uranium oxide BWR intact fuel assemblies
2 MOX BWR intact fuel assemblies;
3 Uranium oxide BWR damaged fuel assemblies placed in damaged fuel containers:
4 MOX BWR damaged fuel assemblies placed in damaged fuel containers: or
5 Up to one (1) Dresden Unit 1 Thoria Rod Canister

C Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium neutron source material shall be in a water rod location.

IV MPC MODEL: MPC-24E

A Allowable Contents

1 Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications

a Cladding type	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b Maximum initial enrichment	As specified in Table A.2 for the applicable fuel assembly array/class
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii. Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup as specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources

IV MPC MODEL: MPC-24E

A Allowable Contents (continued)

2 Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications

a Cladding type	ZR
b Maximum initial enrichment	3.7% ^{235}U
c Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8 Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24E fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source per fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Fuel debris is not authorized for transport in the MPC-24E.

H. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

V MPC MODEL: MPC-24EF

A Allowable Contents

1 Uranium oxide. PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications

a. Cladding type	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b Maximum initial enrichment	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii Trojan plant fuel:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources:	Post-irradiation cooling time, and average burnup as specified in Table A.9.
d. Decay heat per assembly:	
a. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts.
b. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources.

Table A.1 (Page 18 of 23)
Fuel Assembly Limits

V MPC MODEL: MPC-24EF

A Allowable Contents (continued)

2 Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a Cladding type	ZR
b Maximum initial enrichment	3.7% ^{235}U
c Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

V MPC MODEL MPC-24EF

A Allowable Contents (continued)

3 Trojan Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris, for which the original fuel assemblies meet the applicable criteria listed in Table A.2 and meet the following specifications

a Cladding type	ZR
b Maximum initial enrichment.	3.7% ^{235}U
c. Fuel debris post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	Post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies, fuel assemblies classified as fuel debris, and/or Trojan Fuel Debris Process Can Capsules may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24EF fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies, fuel assemblies classified as fuel debris, and Fuel Debris Process Can Capsules must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source per fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 20 of 23)
Fuel Assembly Limits

VI MPC MODEL MPC-32

A Allowable Contents

1 Uranium oxide, PWR intact fuel assemblies in array/classes 15x15D, E, F, and H and 17x17A, B, and C listed in Table A.2, and meeting the following specifications

a Cladding type.	ZR
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, maximum average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.10 or A.11, as applicable.
d. Minimum average burnup per assembly (Assembly Burnup shall be confirmed per Subsection 1.2.3.7.2 of the SAR, which is hereby included by reference)	Calculated value as a function of initial enrichment. See Table A.12.
e. Decay heat per assembly:	≤ 625 Watts
f. Fuel assembly length:	≤ 176.8 inches (nominal design)
g. Fuel assembly width:	≤ 8.54 inches (nominal design)
h. Fuel assembly weight:	≤ 1,680 lbs

i. Operating parameters during irradiation of the assembly (Assembly operating parameters shall be determined per Subsection 1.2.3.7.1 of the SAR, which is hereby included by reference)

Core ave. soluble boron concentration:	≤ 1,000 ppmb
Assembly ave. moderator temperature:	≤ 601 K for array/classes 15x15D, E, F, and H ≤ 610 K for array/classes 17x17A, B, and C
Assembly ave. specific power:	≤ 47.36 kW/kg-U for array/classes 15x15D, E, F, and H ≤ 61.61 kW/kg-U for array/classes 17x17A, B, and C

Table A 1 (Page 21 of 23)
Fuel Assembly Limits

VI MP C MODEL. MPC-32 (continued)

B Quantity per MPC Up to 32 PWR intact fuel assemblies

C Fuel assemblies shall not contain non-fuel hardware

D Damaged fuel assemblies and fuel debris are not authorized for transport in MPC-32

E Trojan plant fuel is not permitted to be transported in the MPC-32.

Table A.1 (Page 22 of 23)
Fuel Assembly Limits

VII MPC MODEL MPC-HB

A Allowable Contents

1 Uranium oxide, INTACT and/or UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS, with or without channels, meeting the criteria specified in Table A.3 for fuel assembly array/class 6x6D or 7x7C and the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 29 years, an average burnup ≤ 23,000 MWD/MTU, and a minimum initial enrichment ≥ 2.09 wt% ^{235}U.
e. Fuel assembly length:	≤ 96.91 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 400 lbs, including channels and DFC
h. Decay heat per assembly:	≤ 50 W
h. Decay heat per MPC:	≤ 2000 W

Table A 1 (Page 23 of 23)
Fuel Assembly Limits

VII MPC MODEL. MPC-HB (continued)

B Quantity per MPC-HB Up to 80 fuel assemblies

C Damaged fuel assemblies and fuel debris must be stored in a damaged fuel container Allowable Loading Configurations: Up to 28 damaged fuel assemblies/fuel debris in damaged fuel containers, may be placed into the peripheral fuel storage locations as shown in SAR Figure 6.I.3, or up to 40 damaged fuel assemblies/fuel debris, in damaged fuel containers, can be placed in a checkerboard pattern as shown in SAR Figure 6.I.4. The remaining fuel locations may be filled with intact and/or undamaged fuel assemblies meeting the above applicable specifications, or with intact and/or undamaged fuel assemblies placed in damaged fuel containers.

NOTE 1: The total quantity of damaged fuel or fuel debris permitted in a single damaged fuel container is limited to the equivalent weight and special nuclear material quantity of one intact assembly.

NOTE 2: Fuel debris includes material in the form of loose debris consisting of zirconium clad pellets, stainless steel clad pellets, unclad pellets, or rod segments up to a maximum of one equivalent fuel assembly. A maximum of 1.5 kg of stainless steel clad is allowed per cask.

Table A.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**14x14A**	**14x14B**	**14x14C**	**14x14D**	**14x14E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 5.0 (24E/EF)	≤ 5.0
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide Tubes	17	17	5 (Note 4)	16	0
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15A**	**15x15B**	**15x15C**	**15x15D**	**15x15E**	**15x15F**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	< 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	N/A	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.015	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table A.2 (Page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	15x15G	15x15H	16x16A	17x17A	17x17B	17x17C
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475	≤ 443	≤ 467	≤ 467	≤ 474
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.0 (24) ≤ 4.5 (24E/EF)	≤ 3.8 (24) ≤ 4.2 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF) (Note 7)	≤ 4.0 (24) ≤ 4.4 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	(Note 5)	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table A.2 (Page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class
2. ZR Designates cladding material made of Zirconium or Zirconium alloys
3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.
4. Each guide tube replaces four fuel rods.
5. Minimum burnup and maximum initial enrichment as specified in Table A.12.
6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.
7. Trojan plant-specific fuel is governed by the limits specified for array/class 17x17B and will be transported in the custom-designed Trojan MPC-24E/EF canisters. The Trojan MPC-24E/EF design is authorized to transport only Trojan plant fuel with a maximum initial enrichment of 3.7 wt.% ^{235}U.

Table A.3 (Page 1 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	6x6A	6x6B	6x6C	7x7A	7x7B	8x8A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195	≤ 120
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	≥ 0	≥ 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table A.3 (Page 2 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	8x8F	9x9A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177
Maximum planar-average initial enrichment (wt % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	< 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

Table A.3 (Page 3 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	9x9B	9x9C	9x9D	9x9E (Note 13)	9x9F (Note 13)	9x9G
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table A.3 (Page 4 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**10x10A**	**10x10B**	**10x10C**	**10x10D**	**10x10E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 186	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table A.3 (Page 5 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**6x6D**	**7x7C**
Clad Material (Note 2)	Zr	Zr
Design Initial U (kg/assy.)(Note 3)	≤ 78	≤ 78
Maximum planar-average initial enrichment (wt % ^{235}U)	≤ 2.6	≤ 2.6
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0 (Note 14)	≤ 4.0
No. of Fuel Rod Locations	36	49
Fuel Clad O.D. (in.)	≥ 0.5585	≥ 0.486
Fuel Clad I.D. (in.)	≤ 0.505	≤ 0.426
Fuel Pellet Dia. (in.)	≤ 0.488	≤ 0.411
Fuel Rod Pitch (in.)	≤ 0.740	≤ 0.631
Active Fuel Length (in.)	≤ 80	≤ 80
No. of Water Rods (Note 11)	0	0
Water Rod Thickness (in.)	N/A	N/A
Channel Thickness (in.)	≤ 0.060	≤ 0.060

Table A.3 (Page 6 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes

1. All dimensions are design nominal values Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class

2. ZR designates cladding material made from Zirconium or Zirconium alloys

3. Design initial uranium weight is the uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5% for comparison with users' fuel records to account for manufacturer's tolerances.

4. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).

5. This assembly class contains 74 total fuel rods; 66 full length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable

8. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.

9. This assembly class contains 91 total fuel rods, 83 full length rods and 8 partial length rods.

10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.

11. These rods may be sealed at both ends and contain Zr material in lieu of water.

12. This assembly is known as "QUAD+" and has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

14. Only two assemblies may contain one rod each with an initial maximum enrichment up to 5.5 wt%.

Table A 4

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-24/24E/24/EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 9	≤ 24,500	≥ 2.3
≥ 11	≤ 29,500	≥ 2.6
≥ 13	≤ 34,500	≥ 2.9
≥ 15	≤ 39,500	≥ 3.2
≥ 18	≤ 44,500	≥ 3.4

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 6	≤ 24,500	≥ 2.3
≥ 7	≤ 29,500	≥ 2.6
≥ 9	≤ 34,500	≥ 2.9
≥ 11	≤ 39,500	≥ 3.2
≥ 14	≤ 44,500	≥ 3.4

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH STAINLESS STEEL CLAD

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 19	≤ 30,000	≥ 3 1
≥ 24	≤ 40,000	≥ 3 1

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-68

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 5	≤ 10,000	≥ 0.7
≥ 7	≤ 20,000	≥ 1.35
≥ 8	≤ 24,500	≥ 2.1
≥ 9	≤ 29,500	≥ 2.4
≥ 11	≤ 34,500	≥ 2.6
≥ 14	≤ 39,500	≥ 2.9
≥ 19	≤ 44,500	≥ 3.0

Table A.8

TROJAN PLANT FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT LIMITS (Note 1)

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% ^{235}U)
≥16	≤42,000	≥3.09
≥16	≤37,500	≥2.6
≥16	≤30,000	≥2.1

NOTES:

1. Each fuel assembly must only meet one set of limits (i.e., one row)

Table A.9

TROJAN PLANT NON-FUEL HARDWARE AND NEUTRON SOURCES COOLING AND BURNUP LIMITS

Type of Hardware or Neutron Source	Burnup (MWD/MTU)	Post-Irradiation Cooling Time (Years)
BPRAs	≤15,998	≥24
TPDs	≤118,674	≥11
RCCAs	≤125,515	≥9
Cf neutron source	≤15,998	≥24
Sb-Be neutron source with 4 source rods, 16 burnable poison rods, and 4 thimble plug rods	≤45,361	≥19
Sb-Be neutron source with 4 source rods, 20 thimble plug rods	≤88,547	≥9

Table A 10

FUEL ASSEMBLY COOLING. AVERAGE BURNUP. AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥12	≤24,500	≥2.3
≥14	≤29,500	≥2.6
≥16	≤34,500	≥2.9
≥19	≤39,500	≥3.2
≥20	≤42,500	≥3.4

Table A.11

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% U-235)
≥8	≤24,500	≥2.3
≥9	≤29,500	≥2.6
≥12	≤34,500	≥2.9
≥14	≤39,500	≥3.2
≥19	≤44,500	≥3.4

Table A.12

FUEL ASSEMBLY MAXIMUM ENRICHMENT AND MINIMUM BURNUP REQUIREMENTS FOR TRANSPORTATION IN MPC-32

Fuel Assembly Array/Class	Configuration (Note 2)	Maximum Enrichment (wt.% U-235)	Minimum Burnup (B) as a Function of Initial Enrichment (E) (Note 1) (GWD/MTU)
15x15D, E, F, H	A	4.65	$B = (1.6733)^*E^3-(18.72)^*E^2+(80.5967)^*E-88.3$
	B	4.38	$B = (2.175)^*E^3-(23.355)^*E^2+(94.77)^*E-99.95$
	C	4.48	$B = (1.9517)^*E^3-(21.45)^*E^2+(89.1783)^*E-94.6$
	D	4.45	$B = (1.93)^*E^3-(21.095)^*E^2+(87.785)^*E-93.06$
17x17A,B,C	A	4.49	$B = (1.08)^*E^3-(12.25)^*E^2+(60.13)^*E-70.86$
	B	4.04	$B = (1.1)^*E^3-(11.56)^*E^2+(56.6)^*E-62.59$
	C	4.28	$B = (1.36)^*E^3-(14.83)^*E^2+(67.27)^*E-72.93$
	D	4.16	$B = (1.4917)^*E^3-(16.26)^*E^2+(72.9883)^*E-79.7$

NOTES:

1. E = Initial enrichment (e.g., for 4.05 wt.%, E = 4.05).
2. See Table A.13.
3. Fuel Assemblies must be cooled 5 years or more.

Table A 13

LO ADING CONFIGURATIONS FOR THE MPC-32

CONFIGURATION	ASSEMBLY SPECIFICATIONS
A	• Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures): or • Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures), but where it can be demonstrated, based on operating records, that the insertion never exceeded 8 inches from the top of the active length during full power operation.
B	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. There is no limit on the duration (in terms of burnup) under this bank. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
C	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 20 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
D	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 30 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.

REFERENCES:

Holtec International Report No HI-951251. *Safety Analysis Report for the Holtec International Storage, Transport. And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12 dated October 6. 2006. as supplemented

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	11	71-9235	USA/9235/B(U)F-96	1	OF	12

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71 "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

NAC International
3930 East Jones Bridge Road, Suite 200
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc., application dated February 19, 2009

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below

5. (a) Packaging

(1) Model No.: NAC-STC

(2) Description: For descriptive purposes, all dimensions are approximate nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

A steel, lead and polymer (NS4FR) shielded shipping cask for (a) directly loaded irradiated PWR fuel assemblies, (b) intact, damaged and/or the fuel debris of Yankee Class or Connecticut Yankee irradiated PWR fuel assemblies in a canister, and (c) non-fissile, solid radioactive materials (referred to hereafter as Greater Than Class C (GTCC) as defined in 10 CFR Part 61) waste in a canister. The cask body is a right circular cylinder with an impact limiter at each end. The package has approximate dimensions as follows:

Cavity diameter	71 inches
Cavity length	165 inches
Cask body outer diameter	87 inches
Neutron shield outer diameter	99 inches
Lead shield thickness	3.7 inches
Neutron shield thickness	5.5 inches
Impact limiter diameter	124 inches
Package length:	
without impact limiters	193 inches
with impact limiters	257 inches

The maximum gross weight of the package is about 260,000 lbs.

The cask body is made of two concentric stainless steel shells. The inner shell is 1.5 inches thick and has an inside diameter of 71 inches. The outer shell is 2.65 inches thick and has

5.(a)(2) Description (Continued)

an outside diameter of 86.7 inches. The annulus between the inner and outer shells is filled with lead

The inner and outer shells are welded to steel forgings at the top and bottom ends of the cask. The bottom end of the cask consists of two stainless steel circular plates which are welded to the bottom end forging. The inner bottom plate is 6.2 inches thick and the outer bottom plate is 5.45 inches thick. The space between the two bottom plates is filled with a 2-inch thick disk of a synthetic polymer (NS4FR) neutron shielding material

The cask is closed by two steel lids which are bolted to the upper end forging The inner lid (containment boundary) is 9 inches thick and is made of Type 304 stainless steel. The outer lid is 5.25 inches thick and is made of SA-705 Type 630, H1150 or 17-4PH stainless steel. The inner lid is fastened by 42, 1-1/2-inch diameter bolts and the outer lid is fastened by 36, 1-inch diameter bolts. The inner lid is sealed by two O-ring seals. The outer lid is equipped with a single O-ring seal. The inner lid is fitted with a vent and drain port which are sealed by O-rings and cover plates. The containment system seals may be metallic or Viton. Viton seals are used only for directly-loaded fuel that is to be shipped without long-term interim storage.

The cask body is surrounded by a 1/4-inch thick jacket shell constructed of 24 stainless steel plates. The jacket shell is 99 inches in diameter and is supported by 24 longitudinal stainless steel fins which are connected to the outer shell of the cask body. Copper plates are bonded to the fins. The space between the fins is filled with NS4FR shielding material.

Four lifting trunnions are welded to the top end forging. The package is shipped in a horizontal orientation and is supported by a cradle under the top forging and by two trunnion sockets located near the bottom end of the cask.

The package is equipped at each end with an impact limiter made of redwood and balsa. Two impact limiter designs consisting of a combination of redwood and balsa wood, encased in Type 304 stainless steel are provided to limit the g-loads acting on the cask during an accident. The predominantly balsa wood impact limiter is designed for use with all the proposed contents. The predominately redwood impact limiters may only be used with directly loaded fuel or the Yankee-MPC configuration.

The contents are transported either directly loaded (uncanistered) into a stainless steel fuel basket or within a stainless steel transportable storage canister (TSC).

The directly loaded fuel basket within the cask cavity can accommodate up to 26 PWR fuel assemblies. The fuel assemblies are positioned within square sleeves made of stainless steel. Boral or TalBor sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 31, ½-inch thick, 71-inch diameter stainless steel disks. The basket also has 20 heat transfer disks made of Type 6061-T651 aluminum alloy. The support disks

5.(a)(2) Description (Continued)

and heat transfer disks are connected by six, 1-5/8-inch diameter by 161-inch long threaded rods made of Type 17-4 PH stainless steel.

The TSC shell, bottom plate, and welded shield and structural lids are fabricated from stainless steel. The bottom is a 1-inch thick steel plate for the Yankee-MPC and 1.75-inch thick steel plate for the CY-MPC The shell is constructed of 5/8-inch thick rolled steel plate and is 70 inches in diameter The shield lid is a 5-inch thick steel plate and contains drain and fill penetrations for the canister The structural lid is a 3-inch thick steel plate The canister contains a stainless steel fuel basket that can accommodate up to 36 intact Yankee Class fuel assemblies and Reconfigured Fuel Assemblies (RFAs), or up to 26 intact Connecticut Yankee fuel assemblies with RFAs, with a maximum weight limit of 35,100 lbs. Alternatively, a stainless steel GTCC waste basket is used for up to 24 containers of waste.

One TSC fuel basket configuration can store up to 36 intact Yankee Class fuel assemblies or up to 36 RFAs within square sleeves made of stainless steel. Boral sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 22 ½-inch thick , 69-inch diameter stainless steel disks, which are spaced about 4 inches apart. The support disks are retained by split spacers on eight 1.125-inch diameter stainless steel tie rods. The basket also has 14 heat transfer disks made of Type 6061-T651 aluminum alloy.

The second fuel basket is designed to store up to 26 Connecticut Yankee Zirc-clad assemblies enriched to 3.93 wt. percent, stainless steel clad assemblies enriched up to 4.03 wt. percent, RFAs, or damaged fuel in CY-MPC damaged fuel cans (DFCs). Zirc-clad fuel enriched to between 3.93 and 4.61 wt. percent, such as Westinghouse Vantage 5H fuel, must be stored in the 24-assembly basket. Assemblies approved for transport in the 26-assembly configuration may also be shipped in the 24-assembly configuration. The construction of the two basket configurations is identical except that two fuel loading positions of the 26-assembly basket are blocked to form the 24-assembly basket.

RFAs can accommodate up to 64 Yankee Class fuel rods or up to 100 Connecticut Yankee fuel rods, as intact or damaged fuel or fuel debris, in an 8x8 or 10x10 array of stainless steel tubes, respectively. Intact and damaged Yankee Class or Connecticut Yankee fuel rods, as well as fuel debris, are held in the fuel tubes. The RFAs have the same external dimensions as a standard intact Yankee Class, or Connecticut Yankee fuel assembly.

The TSC GTCC basket positions up to 24 Yankee Class or Connecticut Yankee waste containers within square stainless steel sleeves. The Yankee Class basket is supported laterally by eight 1-inch thick, 69-inch diameter stainless steel disks. The Yankee Class basket sleeves are supported full-length by 2.5-inch thick stainless steel support walls. The support disks are welded into position at the support walls. The Connecticut Yankee GTCC basket is a right-circular cylinder formed by a series of 1.75-inch thick Type 304 stainless steel plates, laterally supported by 12 equally spaced welded 1.25-inch thick Type 304 stainless steel outer ribs. The GTCC waste containers accommodate radiation activated and surface contaminated steel, cutting debris (dross) or filter media, and have the same external dimensions of Yankee Class or Connecticut Yankee fuel assemblies.

5.(a)(2) Description (Continued)

The Yankee Class TSC is axially positioned in the cask cavity by two aluminum honeycomb spacers. The spacers, which are enclosed in a Type 6061-T651 aluminum alloy shell, position the canister within the cask during normal conditions of transport. The bottom spacer is 14-inches high and 70-inches in diameter, and the top spacer is 28-inches high and also 70-inches in diameter.

The Connecticut Yankee TSC is axially positioned in the cask cavity by one stainless steel spacer located in the bottom of the cask cavity.

5.(a)(3) Drawings

(i) The cask is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-800, sheets 1-3, Rev. 14
423-802, sheets 1-7, Rev. 20
423-803, sheets 1-2, Rev. 8
423-804, sheets 1-3, Rev. 8
423-805, sheets 1-2, Rev. 6
423-806, Rev. 7
423-807, sheets 1-3, Rev. 3
423-811, sheets 1-2, Rev. 11
423-812, Rev. 6
423-900, Rev. 6
423-209, Rev. 0
423-210, Rev. 0
423-901, Rev. 2

(ii) For the directly loaded configuration, the basket is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-870, Rev. 5
423-871, Rev. 5
423-872, Rev. 6
423-873, Rev. 2
423-874, Rev. 2
423-875, sheets 1-2, Rev. 7

(iii) For the Yankee Class TSC configuration, the canister, and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

455-800, sheets 1-2, Rev. 2
455-801, sheets 1-2, Rev. 3
455-820, sheets 1-2, Rev. 2
455-870, Rev. 5
455-871, sheets 1-2, Rev. 8
455-871, sheets 1-3, Rev. 7P2[1]
455-872, sheets 1-2, Rev. 12
455-872, sheets 1-2, Rev. 11P1[1]
455-873, Rev. 4
455-881, sheets 1-3, Rev. 8
455-887, sheets 1-3, Rev. 4
455-888, sheets 1-2, Rev. 8
455-891, sheets 1-2, Rev. 1
455-891, sheets 1-3, Rev. 2PO[1]
455-892, sheets 1-2, Rev. 3
455-892, sheets 1-3, Rev. 3P0[1]
455-893, Rev. 3
455-894, Rev. 2
455-895, sheets 1-2, Rev. 5
455-895, sheets 1-2, Rev. 5P0[1]
455-901, Rev. 0P0[1]
455-902, sheets 1-5, Rev. 0P4[1]
455-919, Rev. 2

[1]Drawing defines the alternate configuration that accommodates the Yankee-MPC damaged fuel can.

5.(a)(3) Drawings (Continued)

(iv) For the Yankee Class TSC configuration, RFAs are constructed and assembled in accordance with the following Yankee Atomic Electric Company Drawing Nos.:

YR-00-060, Rev. D3
YR-00-061, Rev. D4
YR-00-062, sheet 1, Rev D4
YR-00-062, sheet 2, Rev D2
YR-00-062, sheet 3, Rev. D1
YR-00-063, Rev. D4
YR-00-064, Rev. D4
YR-00-065, Rev. D2
YR-00-066, sheet 1, Rev. D5
YR-00-066, sheet 2, Rev. D3

(v) The Balsa Impact Limiters are constructed and assembled in accordance with the following NAC International Drawing Nos.:

423-257, Rev. 2
423-258, Rev. 2
423-843, Rev. 2
423-859, Rev. 0

(vi) For the Connecticut Yankee TSC configuration, the canister and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-801, sheets, 1-2 Rev. 1
414-820, Rev. 0
414-870, Rev. 3
414-871, sheets 1-2, Rev. 6
414-872, sheets 1-3, Rev. 6
414-873, Rev. 2
414-874, Rev. 0
414-875, Rev. 0
414-881, sheets 1-2, Rev. 4
414-882, sheets 1-2, Rev. 4
414-887, sheets 1-4, Rev. 4
414-888, sheets 1-2, Rev. 4
414-889, sheets 1-3, Rev. 7
414-891, Rev. 3
414-892, sheets 1-3, Rev. 3
414-893, sheets 1-2, Rev. 2
414-894, Rev. 0
414-895, sheets 1-2, Rev. 4

(vii) For the Connecticut Yankee TSC configuration, DFCs and RFAs are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-901, Rev. 1
414-902, sheets 1-3, Rev. 3
414-903, sheets 1-2, Rev. 1
414-904, sheets 1-3, Rev. 0

5.(b) Contents

(1) Type and form of material

(i) Irradiated PWR fuel assemblies with uranium oxide pellets. Each fuel assembly may have a maximum burnup of 45 GWD/MTU. The minimum fuel cool time is defined in the Fuel Cool Time Table, below. The maximum heat load per assembly is 850 watts Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications:

Assembly Type	14x14	15x15	16x16	17x17	17x17 (OFA)	Framatome-Cogema 17x17
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirconium Alloy
Maximum Initial Uranium Content (kg/assembly)	407	469	402.5	464	426	464
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.2	4.2	4.2	4.5
Minimum Initial Enrichment (wt% ^{235}U)	1.7	1.7	1.7	1.7	1.7	1.7
Assembly Cross-Section (inches)	7.76 to 8.11	8.20 to 8.54	8.10 to 8.14	8.43 to 8.54	8.43	8.425 to 8.518
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264	264	264[1]
Fuel Rod OD (inch)	0.422 to 0.440	0.418 to 0.430	0.382	0.374 to 0.379	0.360	0.3714 to 0.3740
Minimum Cladding Thickness (inch)	0.023	0.024	0.025	0.023	0.023	0.0204
Pellet Diameter (inch)	0.344 to 0.377	0.358 to 0.390	0.325	0.3225 to 0.3232	0.3088	0.3224 to 0.3230
Maximum Active Fuel Length (inches)	146	144	137	144	144	144.25

Notes:
(1) - Fuel rod positions may also be occupied by solid poison shim rods or solid zirconium alloy or stainless steel fill rods.

5.(b)(1)(i) Contents - Type and Form of Material - Irradiated PWR fuel assemblies (Continued)

FUEL COOL TIME TABLE
Minimum Fuel Cool Time in Years

Uranium Enrichment (wt% U-235)	Fuel Assembly Burnup (BU): BU ≤ 30 GWD/MTU				30 < BU ≤ 35 GWD/MTU				35 < BU ≤ 40 GWD/MTU				40 < BU ≤ 45 GWD/MTU			
Fuel Type	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.7≤E<1.9	8	7	6	7	10	10	7	9	--	--	--	--	--	--	--	--
1.9≤E<2.1	7	7	5	7	9	9	7	8	12	13	9	11	--	--	--	--
2.1≤E<2.3	7	7	5	6	9	8	6	8	11	11	8	10	--	--	--	--
2.3≤E<2.5	6	6	5	6	8	8	6	7	10	10	8	9	14	15	12	14
2.5≤E<2.7	6	6	5	6	8	7	6	7	10	9	7	9	13	14	10	12
2.7≤E<2.9	6	6	5	5	7	7	5	6	9	9	7	8	12	12	9	11
2.9≤E<3.1	6	5	5	5	7	7	5	6	9	8	6	8	11	11	8	10
3.1≤E<3.3	5	5	5	5	7	6	5	6	8	8	6	7	10	10	8	9
3.3≤E<3.5	5	5	5	5	6	6	5	6	8	7	6	7	10	10	7	9
3.5≤E<3.7	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.7≤E<3.9	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.9≤E<4.1	5	5	5	5	6	6	5	6	7	7	6	7	8	9	7	9
4.1≤E≤4.2	5	5	5	5	5	6	5	6	6	7	6	7	8	8	7	9
4.2<E<4.3	--	--	--	5[1]	--	--	--	6[1]	--	--	--	7[1]	--	--	--	9[1]
4.3≤E≤4.5	--	--	--	5[1]	--	--	--	6[1]	--	--	--	7[1]	--	--	--	8[1]

Notes:
[1] - Framatome-Cogema 17x17 fuel only.

5.(b)(1) Contents - Type and Form of Material (Continued)

(ii) Irradiated intact Yankee Class PWR fuel assemblies or RFAs within the TSC. The maximum initial fuel pin pressure is 315 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	UN 16x16	CE [1] 16x16	West. 18x18	Exxon [2] 16x16	Yankee RFA	Yankee DFC
Cladding Material	Zircaloy	Zircaloy	SS	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Rods per Assembly	237	231	305	231	64	305
Maximum Initial Uranium Content (kg/assembly)	246	240	287	240	70	287
Maximum Initial Enrichment (wt% ^{235}U)	4.0	3.9	4.94	4.0	4.94	4.97 [3]
Minimum Initial Enrichment (wt% ^{235}U)	4.0	3.7	4.94	3.5	3.5	3.5 [3]
Maximum Assembly Weight (lbs)	≤ 950	≤ 950	≤ 950	≤ 950	≤ 950	≤ 950
Maximum Burnup (MWD/MTU)	32,000	36,000	32,000	36,000	36,000	36,000
Maximum Decay Heat per Assembly (kW)	0.28	0.347	0.28	0.34	0.11	0.347
Minimum Cool Time (yrs)	11.0	8.1	22.0	10.0	8.0	8.0
Maximum Active Fuel Length (in)	91	91	92	91	92	N/A

Notes:

[1] Combustion Engineering (CE) fuel with a maximum burnup of 32,000 MWD/MTU, a minimum enrichment of 3.5 wt. percent ^{235}U, a minimum cool time of 8.0 years, and a maximum decay heat per assembly of 0.304 kW is authorized.

[2] Exxon assemblies with stainless steel in-core hardware shall be cooled a minimum of 16.0 years with a maximum decay heat per assembly of 0.269 kW.

[3] Stated enrichments are nominal values (fabrication tolerances are not included).

5.(b)(1) Contents - Type and Form of Material (Continued)

(iii) Solid, irradiated, and contaminated hardware and solid, particulate debris (dross) or filter media placed in a GTCC waste container, provided the quantity of fissile material does not exceed a Type A quantity, and does not exceed the mass limits of 10 CFR 71.15

(iv) Irradiated intact and damaged Connecticut Yankee (CY) Class PWR fuel assemblies (including optional stainless steel rods inserted into the CY intact and damaged fuel assembly reactor control cluster assembly (RCCA) guide tubes that do not contain RCCAs), RFAs, or DFCs within the TSC. The maximum initial fuel pin pressure is 475 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	PWR [1] 15x15	PWR [2] 15x15	PWR [3]	CY-MPC RFA[4]	CY-MPC DFC[5]
Cladding Material	SS	Zircaloy	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Assemblies	26	26	24	4	4
Maximum Initial Uranium Content (kg/assembly)	433.7	397.1	390	212	433.7
Maximum Initial Enrichment (wt% ^{235}U)	4.03	3.93	4.61	4.61[6]	4.61[6]
Minimum Initial Enrichment (wt% ^{235}U)	3.0	2.95	2.95	2.95	2.95
Maximum Assembly Weight (lbs)	≤ 1,500	≤ 1,500	≤ 1,500	≤ 1,600	≤ 1,600
Maximum Burnup (MWD/MTU)	38,000	43,000	43,000	43,000	43,000
Maximum Decay Heat per Assembly (kW)	0.654	0.654	0.654	0.321	0.654
Minimum Cool Time (yrs)	10.0	10.0	10.0	10.0	10.0
Maximum Active Fuel Length (in)	121.8	121.35	120.6	121.8	121.8

Notes:

1. Stainless steel assemblies manufactured by Westinghouse Electric Co., Babcock & Wilcox Fuel Co., Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co.
2. Zircaloy spent fuel assemblies manufactured by Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co., and Babcock & Wilcox Fuel Co.
3. Westinghouse Vantage 5H zircaloy clad spent fuel assemblies have an initial uranium enrichment > 3.93 % wt. U^{235}.
4. Reconfigured Fuel Assemblies (RFA) must be loaded in one of the 4 oversize fuel loading positions.
5. Damaged Fuel Cans (DFC) must be loaded in one of the 4 oversize fuel loading positions.
6. Enrichment of the fuel within each DFC or RFA is limited to that of the basket configuration in which it is loaded.

5.(b) Contents (Continued)

(2) Maximum quantity of material per package

(i) For the contents described in Item 5.(b)(1)(i) 26 PWR fuel assemblies with a maximum total weight of 39,650 lbs. and a maximum decay heat not to exceed 22.1 kW per package.

(ii) For the contents described in Item 5.(b)(1)(ii) Up to 36 intact fuel assemblies to the maximum content weight limit of 30,600 lbs with a maximum decay heat of 12.5 kW per package. Intact fuel assemblies shall not contain empty fuel rod positions and any missing rods shall be replaced by a solid Zircaloy or stainless steel rod that displaces an equal amount of water as the original fuel rod Mixing of intact fuel assembly types is authorized.

(iii) For intact fuel rods, damaged fuel rods and fuel debris of the type described in Item 5.(b)(1)(ii): up to 36 RFAs, each with a maximum equivalent of 64 full length Yankee Class fuel rods and within fuel tubes. Mixing of directly loaded intact assemblies and damaged fuel (within RFAs) is authorized. The total weight of damaged fuel within RFAs or mixed damaged RFA and intact assemblies shall not exceed 30,600 lbs. with a maximum decay heat of 12.5 kW per package.

(iv) For the contents described in Item 5.(b)(1)(iii): for Connecticut Yankee GTCC waste up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 196,000 curies. The total weight of the waste containers shall not exceed 18,743 lbs. with a maximum decay heat of 5.0 kW. For all others, up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 125,000 curies. The total weight of the waste and containers shall not exceed 12,340 lbs. with a maximum decay heat of 2.9 kW.

(i) For the contents described in Item 5.(b)(1)(iv): up to 26 Connecticut Yankee fuel assemblies, RFAs or damaged fuel in CY-MPC DFCs for stainless steel clad assemblies enriched up to 4.03 wt. percent and Zirc-clad assemblies enriched up to 3.93 wt. percent. Westinghouse Vantage 5H fuel and other Zirc-clad assemblies enriched up to 4.61 wt. percent must be installed in the 24-assembly basket, which may also hold other Connecticut Yankee fuel types. The construction of the two basket configurations is identical except that two fuel loading positions of the 26 assembly basket are blocked to form the 24 assembly basket. The total weight of damaged fuel within RFAs or mixed damaged RFAs and intact assemblies shall not exceed 35,100 lbs. with a maximum decay heat of 0.654 kW per assembly for a canister of 26 assemblies. A maximum decay heat of 0.321 kW per assembly for Connecticut Yankee RFAs and of 0.654 kW per canister for the Connecticut Yankee DFCs is authorized.

5.(c) Criticality Safety Index: 0.0

6 Known or suspected damaged fuel assemblies or rods (fuel with cladding defects greater than pin holes and hairline cracks) are not authorized, except as described in Item 5.(b)(2)(iii)

7. For contents placed in a GTCC waste container and described in Item 5.(b)(1)(iii): and which contain organic substances which could radiolytically generate combustible gases, a determination must be made by tests and measurements or by analysis that the following criteria are met over a period of time that is twice the expected shipment time:

The hydrogen generated must be limited to a molar quantity that would be no more than 4% by volume (or equivalent limits for other inflammable gases) of the TSC gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F). For determinations performed by analysis, the amount of hydrogen generated since the time that the TSC was sealed shall be considered.

8. For damaged fuel rods and fuel debris of the quantity described in Item 5.(b)(2)(iii) and 5.(b)(2)(v): if the total damaged fuel plutonium content of a package is greater than 20 Ci, all damaged fuel shall be enclosed in a TSC which has been leak tested at the time of closure. For the Yankee Class TSC the leak test shall have a test sensitivity of at least 4.0×10^{-8} cm^3/sec (helium) and shown to have a leak rate no greater than 8.0×10^{-8} cm^3/sec (helium). For the Connecticut Class TSC the leak test shall have a test sensitivity of at least 1.0×10^{-7} cm^3/sec (helium) and shown to have a leak rate no greater than 2.0×10^{-7} cm^3/sec (helium).

9. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented, except that the thermal testing of the package (including the thermal acceptance test and periodic thermal tests) must be performed as described in NAC-STC Safety Analysis Report.

(c) For packaging Serial Numbers STC-1 and STC-2, only one of these two packagings must be subjected to the thermal acceptance test as described in Section 8.1.6 of the NAC-STC Safety Analysis Report.

10. Prior to transport by rail, the Association of American Railroads must have evaluated and approved the railcar and the system used to support and secure the package during transport.

11. Prior to marine or barge transport, the National Cargo Bureau, Inc., must have evaluated and approved the system used to support and secure the package to the barge or vessel, and must have certified that package stowage is in accordance with the regulations of the Commandant, United States Coast Guard.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a CERTIFICATE NUMBER	b REVISION NUMBER	c DOCKET NUMBER	d PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	11	71-9235	USA/9235/B(U)F-96	12	OF	12

12 Transport by air is not authorized.

13 Packagings must be marked with Package Identification Number USA/9235/B(U)F-96.

14 The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

15 Revision No. 9 of this certificate may be used until May 31. 2010.

16 Expiration date: May 31, 2014

REFERENCES

NAC International, Inc., application dated: February 19, 2009.

As supplemented June 3, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 12, 2009

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a CERTIFICATE NUMBER	b REVISION NUMBER	c DOCKET NUMBER	d PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9226	3	71-9226	USA/9226/B(U)F-85	1 OF	9

2 PREAMBLE

a This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies including the government of any country through or into which the package will be transported

3 THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a ISSUED TO *(Name and Address)*

General Atomics
3550 General Atomics Court
San Diego, California 92121-1122

b TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics application dated
January 6, 2009

4 CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5

a. Packaging

(1) Model No.: GA-4

(2) Description

The GA-4 Legal Weight Truck Spent Fuel Shipping Cask consists of the packaging (cask and impact limiters) and the radioactive contents. The packaging is designed to transport up to four intact pressurized-water reactor (PWR) irradiated spent fuel assemblies as authorized contents. The packaging includes the cask assembly and two impact limiters, each of which is attached to the cask with eight bolts. The overall dimensions of the packaging are approximately 90 inches in diameter and 234 inches long

The containment system includes the cask body (cask body wall, flange, and bottom plate); cask closure; closure bolts; gas sample valve body; drain valve; and primary O-ring seals for the closure, gas sample valve, and drain valve.

Cask Assembly
The cask assembly includes the cask, the closure, and the closure bolts. Fuel spacers are also provided when shipping specified short fuel assemblies to limit the movement of the fuel. The cask is constructed of stainless steel, depleted uranium, and a hydrogenous neutron shield. The cask external dimensions are approximately 188 inches long and 40 inches in diameter. A fixed fuel support structure divides the cask cavity into four spent fuel compartments, each approximately 8.8 inches square and 167 inches long. The closure is recessed into the cask body and is attached to the cask flange with 12 1-inch diameter bolts. The closure is approximately 26 inches square, 11 inches thick, and weighs about 1510 lbs.

5.a. (2) (continued)

The cask has two ports allowing access to the cask cavity. The closure lid has an integral half-inch diameter port (hereafter referred to as the gas sample valve) for gas sampling, venting, pressurizing, vacuum drying, leakage testing, or inerting. A 1-inch diameter port in the bottom plate allows draining, leakage testing, or filling the cavity with water. A separate drain valve opens and closes the port The primary seals for the gas sample valve and drain valve are recessed from the outside cask surface as protection from punctures The gas sample valve and the drain valve also have covers to protect them during transport

Cask

The cask includes the containment (flange, cask body, bottom plate and drain valve seals), the cavity liner and fuel support structure; the impact limiter support structure; the trunnions and redundant lift sockets; the depleted uranium gamma shield; and the neutron shield and its outer shell. The cask body is square, with rounded corners and a transition to a round outer shell for the neutron shield. The cask has approximately a 1.5 inch thick stainless steel body wall, 2.6 inch thick depleted uranium shield (reduced at the corners), and 0.4 inch thick stainless steel fuel cavity liner.

The cruciform fuel support structure consists of stainless steel panels with boron-carbide (B_4C) pellets for criticality control. A continuous series of holes in each panel, at right angles with the fuel support structure axis, provides cavities for the B_4C pellets. The fuel support structure is welded to the cavity liner and is approximately 18 inches square by 166 inches long and weighs about 750 lbs.

The flange connects the cask body wall and fuel cavity liner at the top of the cask, and the bottom plate connects them at the bottom. The gamma shield is made up of five rings, which are assembled with zero axial tolerance clearance within the depleted uranium cavity, to minimize gaps. The impact limiter support structure is a slightly tapered 0.4 inch thick shell on each end of the cask. The shell mates with the impact limiter's cavity and is connected to the cask body by 36 ribs

The neutron shield is located between the cask body and the outer shell. The neutron shield design maintains continuous shielding immediately adjacent to the cask body under normal conditions of transport The details of the design are proprietary The design, in conjunction with the operating procedures, ensures the availability of the neutron shield to perform its function under normal conditions of transport.

Two lifting and tie-down trunnions are located about 34 inches from the top of the cask body, and another pair is located about the same distance from the bottom. The trunnion outside diameter is 10 inches, increasing to 11.5 inches at the cask interface. Two redundant lift sockets are located about 26 inches from the top of the cask body and are flush with the outer skin.

5.a. (2) (continued)

Materials
All major cask components are stainless steel, except the neutron shield, the depleted uranium gamma shield, and the B_4C pellets contained in the fuel support structure. All O-ring seals are fabricated of ethylene propylene

Impact Limiters
The impact limiters are fabricated of aluminum honeycomb, completely enclosed by an all-welded austenitic stainless steel skin. Each of the two identical impact limiters is attached to the cask with eight bolts. Each impact limiter weighs approximately 2,000 lbs

(3) Drawings
The packaging is constructed and assembled in accordance with the following GA Drawing Number:

Drawing No. 031348,
sheets 1 through 19, Revision D (Proprietary Version)
GA-4 Spent Fuel Shipping Cask Packaging Assembly

5.(b) Contents

(1) Type and Form of Material

(a) Intact fuel assemblies. Fuel with known or suspected cladding defects greater than hairline cracks or pinhole leaks is not authorized for shipment.

(b) The fuel authorized for shipment in the GA-4 package is irradiated 14x14 and 15x15 PWR fuel assemblies with uranium oxide fuel pellets. Before irradiation, the maximum enrichment of any assembly to be transported is 3.15 percent by weight of uranium-235 (^{235}U). The total initial uranium content is not to exceed 407 Kg per assembly for 14x14 arrays and 469 Kg per assembly for 15x15 arrays.

c) Fuel assemblies are authorized to be transported with or without control rods or other non-fuel assembly hardware (NFAH) Spacers shall be used for the specific fuel types, as shown on sheet 17 of the Drawings

(d) The maximum burnup for each fuel assembly is 35,000 MWd/MTU with a minimum cooling time of 10 years and a minimum enrichment of 3.0 percent by weight of ^{235}U or 45,000 MWd/MTU with a minimum cooling time of 15 years (no minimum enrichment).

(e) The maximum assembly decay heat of an individual assembly is 0.617 kW. The maximum total allowable cask heat load is 2.468 kW (including control components and other NFAH when present).

5.b. (1) (continued)

(f) The PWR fuel assembly types authorized for transport are listed in Table 1. All parameters are design nominal values.

(2) Maximum Quantity of Material per Package

(a) For material described in 5.b.(1): four (4) PWR fuel assemblies

(b) For material described in 5.b.(1): the maximum assembly weight (including control components or other NFAH when present) is 1,662 lbs The maximum weight of the cask contents (including control components or other NFAH when present) is 6,648 lbs., and the maximum gross weight of the package is 55,000 lbs.

Table 1 - PWR Fuel Assembly Characteristics

Fuel Type Mfr.-Array (Versions)	Design Initial U (kg/assy.)	No. of Fuel Rods	Fuel Rod Pitch (in.)	Pellet Diameter (in.)	Zr Clad Thickness (in.)	Active Fuel Length (in.)
W-15x15 (Std/ZC)	469	204	0.563	0.3659	0.0242	144
W-15x15 (OFA)	463	204	0.563	0.3659	0.0242	144
BW-15x15 (Mk.B,BZ,BGD)	464	208	0.568	0.3686	0.0265	142
Exx/A-15x15 (WE)	432	204	0.563	0.3565	0.030	144
CE-15x15 (Palisades)	413	204	0.550	0.358	0.026	144
CE-14x14 (Ft.Calhoun)	376	176	0.580	0.3765	0.028	128
W-14x14 (Model C)	397	176	0.580	0.3805	0.026	137
CE-14x14 (Std/Gen.)	386	176	0.580	0.3765	0.028	137
Exx/A-14x14 (CE)	381	176	0.580	0.370	0.031	137

5.b(2)(b)(continued)

Fuel Type Mfr.-Array (Versions)	Design Initial U (kg/assy.)	No. of Fuel Rods	Fuel Rod Pitch (in.)	Pellet Diameter (in.)	Zr Clad Thickness (in.)	Active Fuel Length (in.)
W-14x14 (OFA)	358	179	0.556	0.3444	0.0243	144
W-14x14 (Std/ZCA,/ZCB)	407	179	0.556	0.3674	0.0225	145.5
Exx/A-14x14 (WE)	379	179	0.556	0.3505	0.030	142

5.c. Criticality Safety Index (CSI): 100

6. Fuel assemblies with missing fuel pins shall not be shipped unless dummy fuel pins that displace an equal amount of water have been installed in the fuel assembly.

7. In addition to the requirements of Subpart G of 10 CFR 71:

   a. Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed using the specifications contained within the application. At a minimum, those procedures shall require the following provisions:

   (1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b of the CoC.

   (2) That before shipment the licensee shall:

   (a) Perform a measured radiation survey to assure compliance with 49 CFR 173.441 and 10 CFR 71.47 and assure that the neutron and gamma measurement instruments are calibrated for the energy spectrums being emitted from the package.
   (b) Verify that measured dose rates meet the following correlation to demonstrate compliance with the design bases calculated hypothetical accident dose rates: 3.4 x (peak neutron dose rate at any point on cask surface at its midlength) + 1.0 x (gamma dose rate at that location) $\leq$ 1000 mR/hr.
   (c) Verify that the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87.

7.a.(2) (continued)

(d) Inspect all containment seals and closure sealing surfaces for damage. Leak test all containment seals with a gas pressure rise test after final closure of the package. The leak test shall have a test sensitivity of at least 1 $\times 10^{-3}$ standard cubic centimeters per second of air (std-cm^3/sec) and there shall be no detectable pressure rise. A higher sensitivity acceptance and maintenance test may be required as discussed in Condition 7.b.(5), below

(3) Before leak testing, the following closure bolt and valve torque specifications:

(a) The cask lid bolts shall be torqued to 235 ± 15 ft-lbs.
(b) The gas sample valve and drain valve shall be torqued to 20 ± 2 ft-lbs.

(4) During wet loading operations and prior to leak testing, the removal of water and residual moisture from the containment vessel in accordance with the following specifications:

(a) Cask evacuation to a pressure of 0.2 psia (10 mm Hg) or less for a minimum of 1 hour.
(b) Verifying that the cask pressure rise is less than 0.1 psi in 10 minutes.

(5) Before shipment, independent verification of the material condition of the neutron shield as described in SAR Section 7.1.1.4 or 7.1.2.4.

b. All fabrication acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed using the specifications contained within the application and shall include the following provisions:

(1) All containment boundary welds, except the final fabrication weld joint connecting the cask body wall to the bottom plate, shall be radiographed and liquid-penetrant examined in accordance with ASME Code Section III, Division 1, Subsection NB. Examination of the final fabrication weld joint connecting the cask body wall to the bottom plate may be ultrasonic and progressive liquid penetrant examined in lieu of radiographic and liquid penetrant examination.

(2) The upper lifting trunnions and redundant lifting sockets shall be load tested, in the cask axial direction, to 300 percent of their maximum working load (79,500 lbs. minimum) per trunnion and per lifting socket, in accordance with the requirements of ANSI N14.6. The upper and lower lifting trunnions shall be load tested, in the cask transverse direction, to 150 percent of their maximum working load (20,625 lbs. minimum) per trunnion, in accordance with the requirements of ANSI N14.6.

7.b.(continued)

(3) The cask containment boundary shall be pressure tested to 1.5 times the Maximum Normal Operating Pressure of 80 psig. The minimum test pressure shall be 120 psig.

(4) All containment seals shall be replaced within the 12-month period prior to each shipment.

(5) A fabrication leakage test shall be performed on all containment components including the O-ring seals prior to first use. Additionally, all containment seals shall be leak tested after the third use of each package and within the 12-month period prior to each shipment. Any replaced or repaired containment system component shall be leak tested. The leakage tests shall verify that the containment boundary leakage rate does not exceed the design leakage rate of 1×10^{-7} std-cm^3/sec. The leak tests shall have a test sensitivity of at least 5×10^{-8} std-cm^3/sec.

(6) The depleted uranium shield shall be gamma scanned with 100 percent inspection coverage during fabrication to ensure that there are no shielding discontinuities. The neutron shield supplier shall certify that the shield material meets the minimum specified requirements (proprietary) used in the applicant's shielding analysis.

(7) Qualification and verification tests to demonstrate the crush strength of each aluminum honeycomb type and lot to be utilized in the impact limiters shall be performed.

(8) The boron carbide pellets, fuel support structure and fuel cavity dimensions, and ^{235}U content in the depleted uranium shall be fabricated and verified to be within the specifications of Table 2 to ensure criticality safety.

Table 2

Specified Parameter	Minimum	Maximum
B_4C boron enrichment	96 wt% ^{10}B	N/A
Diameter of each B_4C pellet	0.426 in	0.430 in
Height of each B_4C pellet stack	7.986 in	8.046 in
Mass of ^{10}B in each B_4C pellet stack	31.5 g	N/A
Mass of each B_4C pellet stack	43.0 g	45.0 g
Diameter of each fuel support structure hole	0.432 in	0.44 in
Fuel support structure nominal hole pitch	N/A	0.55 in
Fuel support structure hole depth minus B_4C pellet-stack height (at room temperature)	0.009 in	0.129 in
Thickness of each fuel support structure panel	0.600 in	0.620 in
Fuel cavity width	N/A	9.135 in
^{235}U content in depleted uranium shielding material	N/A	0.2 wt%

8. Transport of fissile material by air is not authorized.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Expiration Date: October 31, 2013.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9226	3	71-9226	USA/9226/B(U)F-85	9	OF	9

REFERENCES

General Atomics Application for the GA-4 Legal Weight Truck Spent Fuel Shipping Cask, January 6, 2009.

FOR THE U.S NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date 2/5/09

APPENDIX II

DETAILS OF RISK ANALYSIS OF ROUTINE, INCIDENT-FREE TRANSPORTATION

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APPENDIX II

DETAILS OF RISK ANALYSIS OF ROUTINE, INCIDENT-FREE TRANSPORTATION

II.1 Introduction

NUREG-0170 (NRC, 1977) 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. The estimates were calculated using Version 1 of the RADTRAN code (Taylor and Daniel, 1977), which was developed for the NRC by Sandia National Laboratories specifically to support the conduct of the NUREG-0170 study. RADTRAN Version 6, integrated with the input file generator RADCAT, (Neuhauser, et al[1], 2000; Weiner, et al, 2009) is the computational tool used in this study.

The basic risk assessment method employed in the RADTRAN code is widely accepted. Changes to the code are tracked by a software quality assurance plan that is consistent with American National Standards Institute guidelines. The incident-free module of an earlier version of RADTRAN, RADTRAN 5.25, was validated by measurement (Steinman, et al, 2002); this module is the same in RADTRAN 6.0, the version used in the current study. Verification and validation of RADTRAN 6.0 are documented in Dennis, et al (2008).

II.2 The RADTRAN Model of Routine Transportation

II.2.1 Description of the RADTRAN program

RADTRAN calculates the radiological consequences and risks associated with the shipment of a specific radioactive material in a specific package along a specific route. Shipments that take place without the occurrence of accidents are routine, incident free shipments, and the radiation doses to various receptors are called "incident-free doses." Since the probability of routine, incident-free shipment is essentially equal to one, RADTRAN calculates a dose rather than a risk for such shipments. The dose from a routine shipment is based on the external dose from the part of the vehicle carrying the radioactive cargo, referred to as the "vehicle" in this discussion of RADTRAN. Doses to receptors from the external radiation from the vehicle depend on the distance between the receptor and the radioactive cargo being transported and the exposure time. Exposure time is the length of time the receptor is exposed to external emissions from the radioactive cargo. The doses in routine transportation depend only on the external dose rate from the cargo and not on the radioactive inventory of the cargo.

RADTRAN 6.0, integrated with the input file generator RADCAT, (Neuhauser, et al, 2000; 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).

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[1] 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 and the technical manual for RADTRAN 6 is not yet available.

RADTRAN models the vehicle as a spherical radiation source traveling along the route. The source strength is the transport index (TI), one percent of the dose rate in mSv/hour[2] at 1 m from the cask. which is treated as an isotropically radiating virtual source at the center of the sphere, as shown in Figure II-1.

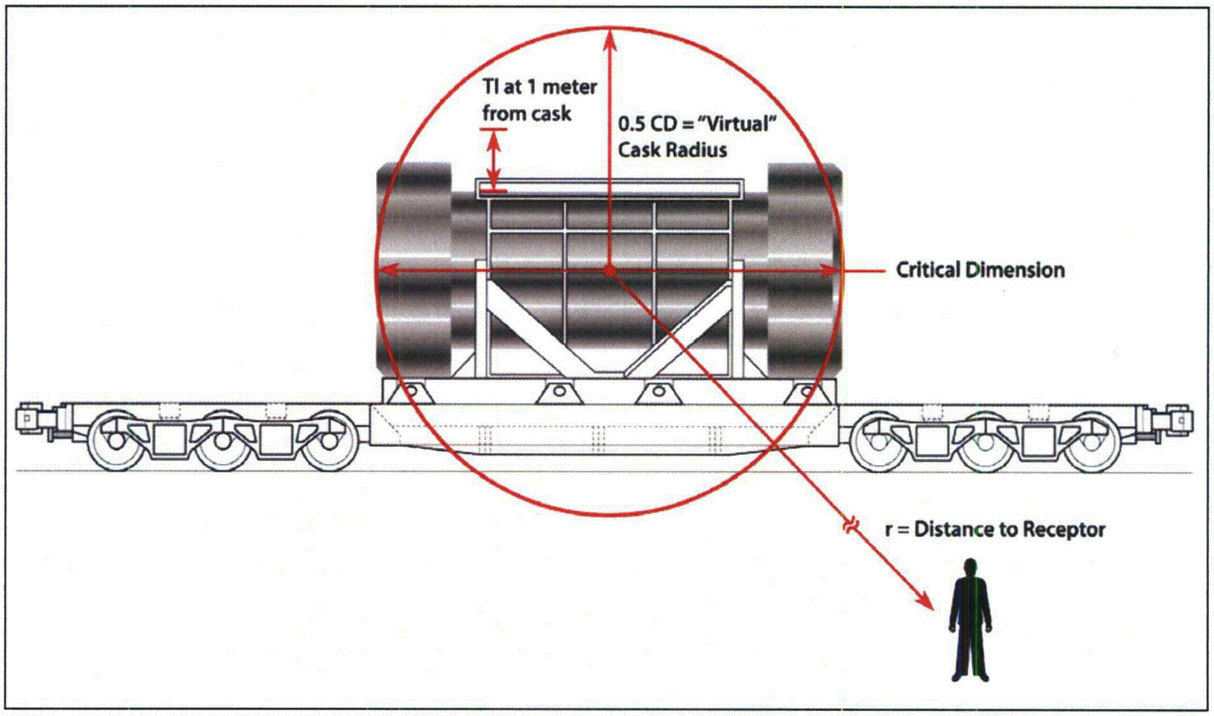

Figure II- 1. RADTRAN model of the vehicle in routine, incident-free transportation

When the distance to the receptor r 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 TIs for the Rail-Pb and the Rail-Steel casks, were calculated from the dose rates at 2 meters as reported in the Safety Analysis Reports of these casks (Holtec International, 2004, NAC international, 2004) and are shown in Table II.3-1.

The basic equation for calculating incident-free doses to a population along a transportation route is Equation II-1:

(II-1) $$D(x) = \frac{Qk_0 DR_v}{V} \int_{-\infty}^{\infty} \int_{x\min}^{x\max} \left\{ \frac{(\exp(-\mu r))(B(r))}{r\sqrt{r^2 - x^2}} \right\} dx dr$$

where x is the distance between the receptor and the source, perpendicular to the route

[2] One mSv = 100 mrem. Thus, 1% of the doe rate in mSv at one meter from the package is equivakent to the dose rate in mrem.hr.

Q includes factors that correct for unit differences
k_0 is the package shape factor
DR_v is the vehicle external dose rate: the TI
V is the vehicle speed
μ is the radiation attenuation factor
B is the radiation buildup factor
r is the distance between the receptor and the source along the route

Details of the application of this and similar equations may be found in Neuhauser et al (2000).

External radiation from casks carrying used nuclear fuel includes both gamma and neutron radiation. For calculating doses from gamma radiation, RADTRAN uses Equation II-2,

(II.2) $$(e^{-\mu r}) * B(r) = 1$$

for conservatism. For calculating doses from neutron radiation, on the other hand, RADTRAN uses Equation II-3

(II-3) $$(e^{-\mu r}) * B(r) = (e^{-\mu r}) * (1 + a_1 r + a_2 r^2 + a_3 r^3 + a_4 r^4)$$

where the coefficients are characteristics of the material. The default coefficients in RADTRAN are those for steel.

Collective (population) doses are calculated by integrating over the band along the route where the population resides (the *x* integration in Equation II-1) and then integrating along the route from minus to plus infinity ($-\infty$ to ∞) along the route (the *r* integration in Equation II-1). This is illustrated for a truck route in Figure II-2. The *x* integration limits in Figure II-2 are not to scale: *xmin* is usually 30 m. and *xmax* is usually 800 m. Integration of x to distances greater than 800 results in risks not significantly different from integration to 800 meters, since the decrease in dose with distance is exponential.

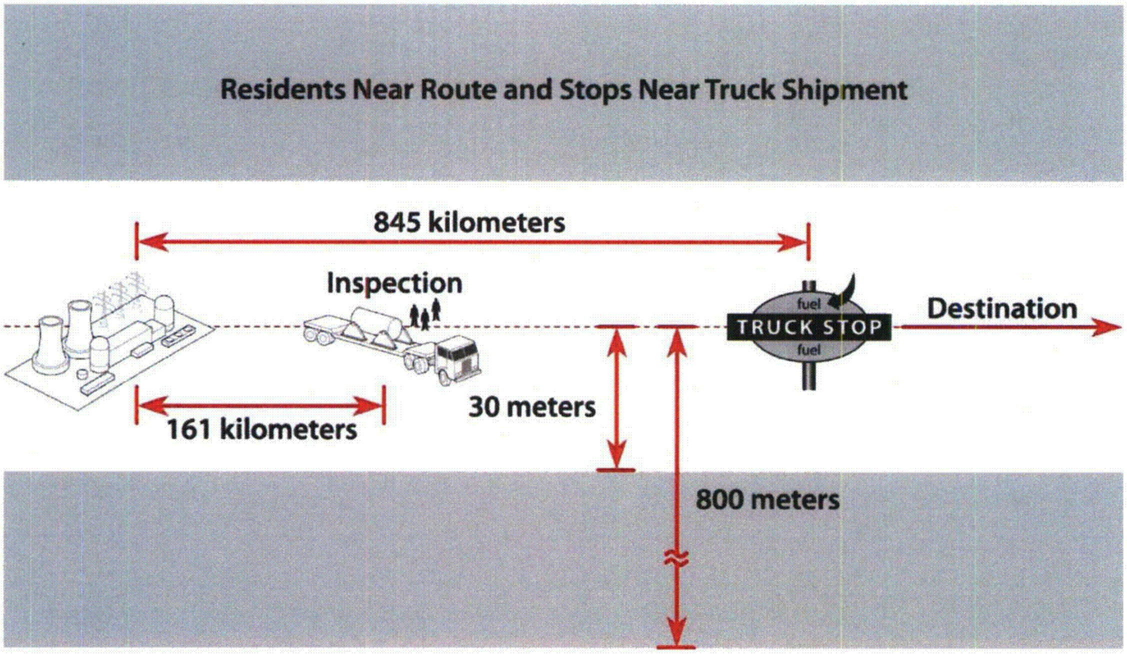

Figure II-2. Diagram of a truck route as modeled in RADTRAN; 845 km is the average distance a very large truck travels on half of its fuel capacity. The 161 km (100 miles) is the distance between spent fuel shipment inspections required by regulation (DOE, 2002).

Variants of Equation II-1 are used to calculate doses to members of the public at stops, vehicle crew members and other workers, occupants of vehicles that share the route with the vehicle carrying the radioactive cargo, and any other receptor identified. Figure II-3 is a diagram of the model used to calculate doses at truck stops. The inner circle defines the area occupied by people who are between the spent fuel truck and the building, and who are not shielded from the truck's external radiation. The dimensions of this circle and the average number of people who occupy it, along with the method used to determine these, are found in Griego et al (1996).

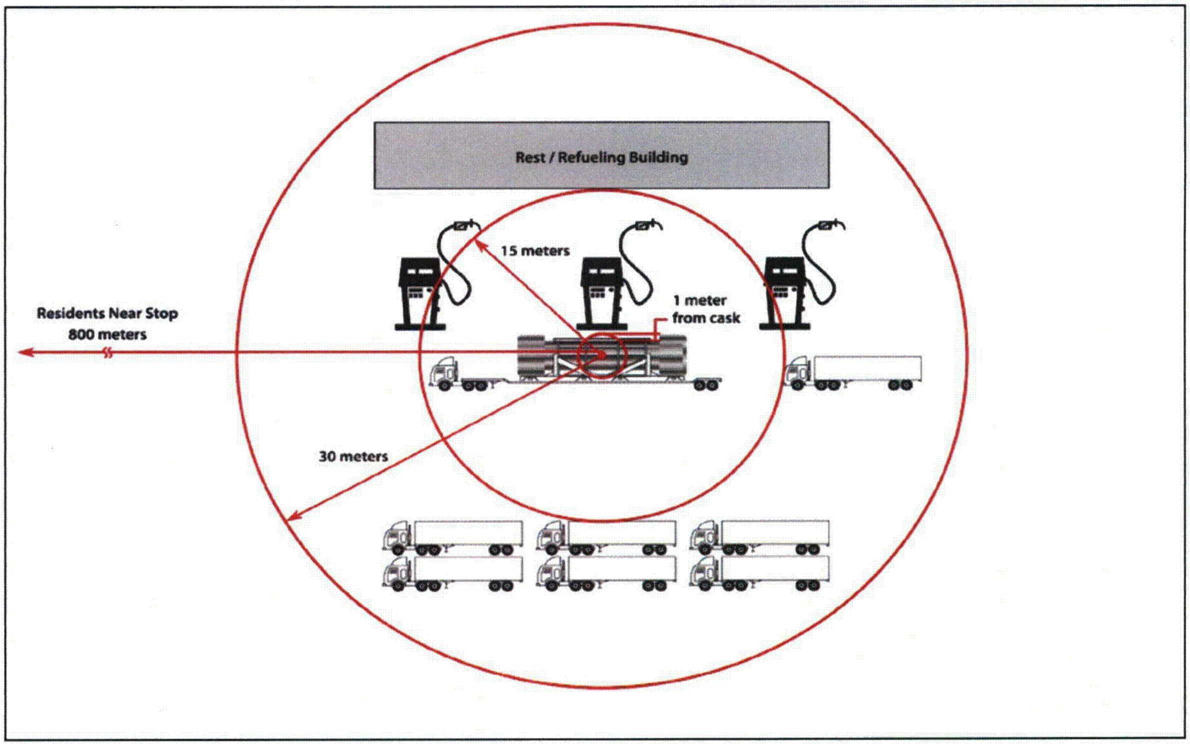

Figure II-3. Diagram of truck stop model

II.2.2 The RADTRAN Software

This section is a brief description of the RADTRAN software program. A full description of the software and how to use it may be found in the RADCAT User Guide (Weiner, et al, 2009).

The equations that RADTRAN uses, variants of Equation II-1, are programmed in FORTRAN 95. RADTRAN reads in

- an input text file that contains the input parameters as defined by the RADTRAN user,
- a text file that contains an internal library of 148 radionuclides with their associated dose conversion factors and half-lives,
- a binary file that contains the societal ingestion doses for one curie of each radionuclide in the internal radionuclide library,
- dilution factors and isopleths areas for several weather patterns.

Only the first of these is used in calculating doses from incident-free transportation; the other three are used in the accident analysis and will be discussed in Appendix V.

The input text file can be written directly using a text editor, or can be constructed using the input file generator RADCAT. RADCAT, programmed in XML and running under Java Webstart, provides a series of screens that guide the user in entering values for RADTRAN input parameters. Figure II-4 shows a RADCAT screen.

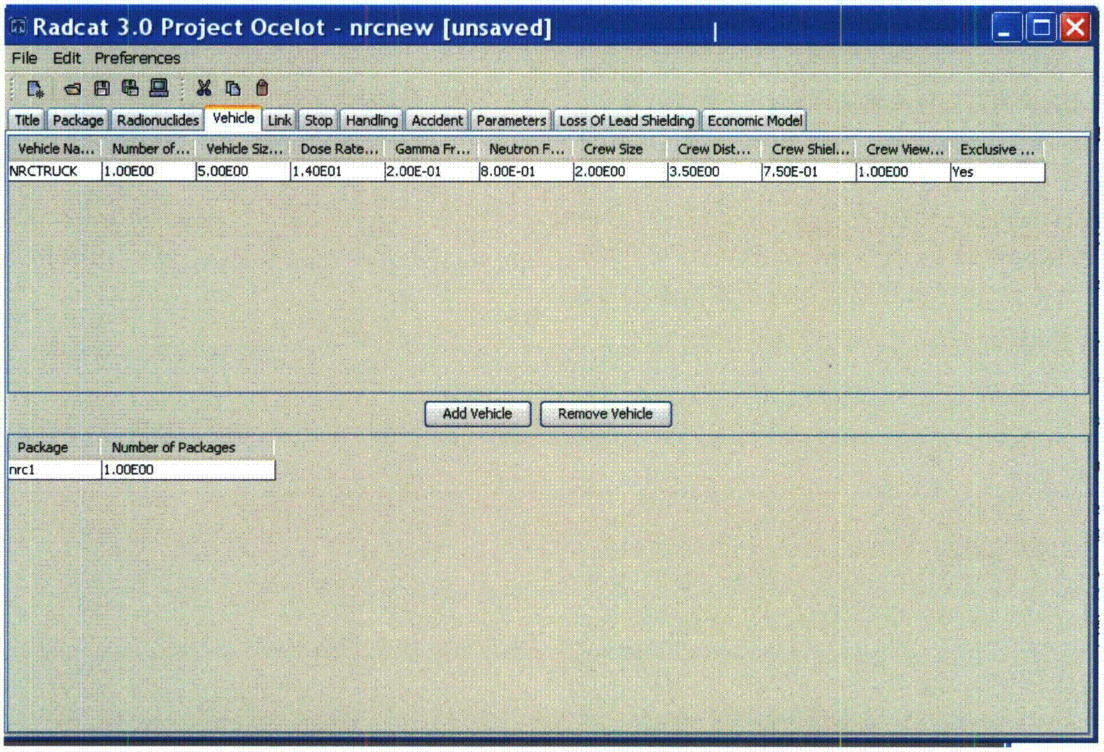

Figure II-4. RADCAT vehicle screen.

RADTRAN output is a text file that can be saved as text or as a spreadsheet.

II.3 RADTRAN Input Parameters

II.3.1 Vehicle-specific Input Parameters

RADTRAN does not allow for the offset of the package from the trailer edge, so the physical dimensions of the package are considered the physical dimensions of the vehicle. Table II-1 shows the vehicle-specific input parameters to RADTRAN and shows the parameter values used in this analysis. The Rail-Steel is modeled transporting canistered fuel; the Rail-Steel-Pb is modeled transporting uncanistered fuel. The Truck-DU is a truck cask; the other two are rail casks. In this analysis, the Truck-DU is assumed to be transported by truck; the Rail-Steel-Pb and the Rail-Steel, by rail.

Table II-1. Vehicle-specific parameters

	Truck-DU	Rail-Steel-Pb	Rail-Steel
Transportation mode	highway	rail	Rail
Length (critical dimension)	5.94 m	4.90m	5.08 m
Diameter ("crew view")	2.29 m	2.5 m	3.2 m
Distance from cargo to crew cab	3.5 m	150 m	150 m minimum
TI	14	14.02	10.34
Gamma fraction	0.77	0.89	0.90
Neutron fraction	0.23	0.11	0.10
Number of packages per vehicle	1 per truck	1 per railcar	1 per railcar
Number of crew	2	3	3
Exclusive use?	yes	NA	NA
Dedicated rail	NA	no	No
17 x 17 PWR assemblies	4	26	24

II.3.2 Route-Specific Input Parameters

Route-specific input parameters are shown in Table II-2 for a unit risk calculation. These are the common input parameters for the he sixteen specific routes analyzed, The vehicle density for rail assumes that one car of a freight train carries a spent fuel cask.

Table II-2. Route parameters for unit risk calculation (USDOT 2004a, b)

Parameter	Interstate Highway	Freight Rail
Vehicle speed (U.S. average kph)		
Rural	108	40.4
Suburban	102	40.4
Urban	97	24
Vehicle density (U.S. average vehicles/hr)		
Rural	1119	17[a]
Suburban	2464	17
Urban	5384	17
Persons per vehicle	1.5	2
Farm fraction	0.5	0.5
Stops		
Minimum distance from nearby residents (m)	30	200
Maximum distance from nearby residents (m)	800	800
Stop time for classification (hours)	NA	27
Stop time in transit for railroad change (hours)	NA	0.5
Stop time for truck inspections (hours)	0.75	NA
Stop time at truck stops (hours)	0.83	NA
Average number of people sharing the stop	6.9[b]	NA
Minimum distance to people sharing the stop (m)	1[b]	NA
Maximum distance to people sharing the stop (m)	15[b]	NA
Truck stop worker distance from cask (m)	15	NA
Truck stop worker shielding factor	0.018	NA
Truck crew shielding factor (no regulatory limit)[c]	0.377	NA
Escort distance from cask (m)	4	16

[a]Railcars per hr [b]Griego et al, 1996. [c] From crew doses with and without the regulatory limit.

II.3.3. Other parameters

RADTRAN includes a set of parameters whose values are not generally known by the user and which have been used routinely in transportation risk assessments. RADTRAN contains default values for these parameters, but all default values can be changed by the user. Table II-3 lists the parameter values used in the incident-free analysis.

Table II-3. Parameter values in the RADTRAN 6 analysis

Parameter	Value
Shielding factor for residents (fraction of energy impacting the receptor): R= rural, S=suburban, U=urban	R=1.0 S=0.87 U=0.018
Fraction of outside air in urban buildings	0.25
Fraction of urban population on sidewalk	0.48
Fraction of urban population in buildings	0.52
Ratio of non-residents to residents in urban areas	6
Distance from in-transit shipment for maximum exposure in m. (RMEI exposure)	30
Vehicle speed for maximum exposure in km/hr. (RMEI exposure)	24
Distance from in-transit shipment to nearest resident in rural and suburban areas, m	30
Distance from in-transit shipment to nearest resident in urban areas, m	27
Population bandwidth m	800
Distance between vehicles (trains), m	3.0
Minimum number of rail classification stops	1

Additional input parameters are rural, suburban, and urban route lengths and population densities, characteristics of stops along a route and the TI of the package.

II.4. Routes

This study analyzes both the per-km doses from a single shipment on rural, suburban, and urban route segments and doses to receptors from a single shipment on 16 representative routes, chosen to represent a range of route lengths and a variety of populations. The actual truck and rail routes were selected for a number of reasons. The combination of four origins and four destinations represent a variety of route lengths and population densities and both private and government facilities, a large number of states, and includes origins and destinations that were included in the analyses of NUREG/CR-6672.

Power reactor spent fuel (SNF) and high-level radioactive waste (HLW) are currently stored at 77 locations in the U.S.(67 nuclear generating plants, five storage facilities at sites of decommissioned nuclear plants, and five DOE defense facilities). The origin sites (Table II-1) include two nuclear generating plants (Indian Point and Kewaunee) a storage site (Maine Yankee) and a National Laboratory (Idaho National Laboratory). The destination sites include the two proposed repository sites not characterized (Deaf Smith County, TX and Hanford, WA) (DOE, 1986), the site of the proposed Private Fuel Storage facility (Skull Valley, UT), and a National Laboratory site (Oak Ridge, TN). The routes modeled are shown in Table II-4. Both truck and rail versions of each route are analyzed.

Table II-4. Specific routes modeled

Origin	Destination
Maine Yankee site, ME	Hanford, WA
	Deaf Smith County, TX
	Skull Valley, UT
	Oak Ridge, TN
Kewaunee NP, WI	Hanford, WA
	Deaf Smith County, TX
	Skull Valley, UT
	Oak Ridge, TN
Indian Point NP, NY	Hanford, WA
	Deaf Smith County, TX
	Skull Valley, UT
	Oak Ridge, TN
Idaho National Lab, ID	Hanford, WA
	Deaf Smith County, TX
	Skull Valley, UT
	Oak Ridge, TN

Route segments and population densities are provided by WebTRAGIS (Johnson and Michelhaugh (2003). WebTRAGIS uses census data from the 2000 census. Updated population data to 2006 were provided in the 2008 Statistical Abstract (U.S. Census Bureau 2008). Table 13 of U.S. Census Bureau (2008) shows the percent increase in population for each of the 50 states of the United States, as well as for the U. S. as a whole, and Table 21 shows the percent increase in population for the 50 largest metropolitan areas in the U.S. Data from these two tables were combined to give population multipliers for states along routes for which the collective dose and the population increase were significant enough to make a correction.

The population multipliers used are shown in Table II-5. "Significant" was taken to mean that the population difference was more than 1% (i.e., multipliers between 0.99 and 1.01 were not considered significant). The state-specific multiplier was applied to rural and suburban routes through the state, and the multiplier for the largest metropolitan area in that state was applied to the urban routes. The U.S. multiplier was applied to ingestion doses.

Table II-5. Population multipliers

State	Rural, Suburban, Urban Designation	Population Multiplier	State	Rural, Suburban, Urban Designation	Population Multiplier
Arizona	Rural, Suburban	1.202	Nevada	Rural, Suburban	1.249
	Urban	1.242		Urban	1.292
Arkansas	Rural, Suburban	1.051	New Mexico	Rural, Suburban	1.075
	Urban	1.051		Urban	1.075
California	Rural, Suburban	1.076	Oklahoma	Rural, Suburban	1.037
	Urban	1.15		Urban	1.07
Colorado	Rural, Suburban	1.105	Pennsylvania	Rural, Suburban	1.013
	Urban	1.105		Urban	1.025
Georgia	Rural, Suburban	1.144	Oregon	Rural, Suburban	1.082
	Urban	1.21		Urban	1.109
Idaho	Rural, Suburban	1.133	Tennessee	Rural, Suburban	1.061
	Urban	1.133		Urban	1.109
Illinois	Rural, Suburban	1.033	Texas	Rural, Suburban	1.127
	Urban	0.959		Urban	1.175
Maryland	Rural, Suburban	1.037	Utah	Rural, Suburban	1.142
	Urban	1.041		Urban	1.102
Missouri	Rural, Suburban	1.044	Virginia	Rural, Suburban	1.08
	Urban	1.044		Urban	1.103

Parameters like population density and route segment lengths, that are specific to each route, were developed using WebTRAGIS. Figure II-5 is a WebTRAGIS output map showing highway routes from Idaho National Laboratory (INL) to the four destinations in Table II-4.

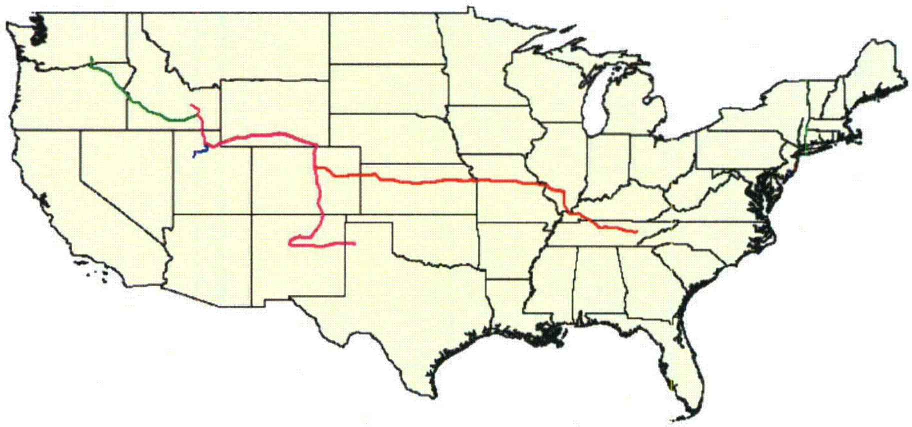

Figure II-5. WebTRAGIS map of truck routes from INL.

II.5 Results

II.5.1 Maximally exposed resident in-transit dose

The largest dose from a moving vehicle to an individual member of the public is sustained when that individual is 30 meters (a conservative estimate of the interstate right-of-way) from the moving vehicle, and the vehicle is moving at the slowest speed it would be likely to maintain. This speed is 24 kph (16 mph) for both rail and truck. Table II-6 shows the maximum individual dose, in Sv, for each package. These doses are directly proportional to the external dose rate (TI) of each package. For comparison, a single dental x-ray delivers a dose of 4×10^{-5} Sv (Stabin, 2009), about 7000 times the doses shown in Table II-6.

Table II-6 Maximum individual doses.

Package	Dose in Sv
Rail-Steel-Pb (rail)	5.7×10^{-9}
Rail-Steel (rail)	4.3×10^{-9}
Truck-DU (truck)	6.7×10^{-9}

II.5.2 Unit risk: rail routes

The doses to railyard workers along the route, to residents and others along the route, and to occupants of vehicles that share the route from a single shipment (one rail cask) traveling one km past a population density of one person/km^2 are shown in Table II-7. The dose units are person- Sv. The doses are calculated assuming one cask on a train, because railcar-km is the unit usually used to describe freight rail transport. The data in this table may be used to calculate collective doses along routes.as follows:

- Multiply the railyard crew dose by the kilometers of each type of route traveled. This is a conservatively calculated dose that assumes that the railyard crew receives 1.8 percent of the classification yard dose, per km of travel, when the train stops. The classification yard occupational collective dose (Wooden, 1980), assuming a 30-hour classification stop, is integrated into RADTRAN. The dose was adjusted to reflect the 27-hour stop (Table II-3) (DOT, 2004b).

- The area of the band occupied by the population along the route is equal to the kilometers traveled multiplied by, e.g., 1.6 for a band width of 800 m on each side of the route. Therefore, multiply the "population along route" dose by this area and the appropriate population density (obtained from a routing code like WebTRAGIS).

Table II-7. Average individual doses ("unit risks") to various receptors, rail routes. The units of the average dose to the residents near the yard, Sv- km^2/hour (mrem-km^2/hour), reflect the output of the RADTRAN stop model, which incorporates the area occupied.

Cask and route type	Resident along route, Sv	Resident near yard Sv-km^2/hour	Occupants of vehicles sharing the route person-Sv
Rail-Steel-Pb rural	7.3E-10	3.5E-07	6.5E-09
Rail-Steel-Pb suburban	6.3E-10	3.5E-07	6.5E-09
Rail-Steel-Pb urban	2.2E-10	3.5E-07	9.1E-08
Rail-Steel rural	5.6E-10	2.7E-07	4.9E-09
Rail-Steel suburban	4.8E-10	2.7E-07	4.9E-09
Rail-Steel urban	1.7E-11	2.7E-07	1.4E-08

II.5.3 Unit risk: truck routes

The doses to truck crew, residents and others along the route, and to occupants of vehicles that share the route from a single shipment (one truck cask) traveling one kilometer past a population density of one persom/km^2 are shown in Table II-8. The dose units are person-Sv. Rural, suburban, and urban doses to residents living near stops are calculated by multiplying the appropriate stop dose - truck stops are not typically located in urban areas) by the appropriate population density (obtained from a routing code like WebTRAGIS). The number of stops on each route segment is calculated by dividing the length of the route segment by 845 km (average distance between refueling stops for a large semi-detached trailer truck (DOE, 2002, Appendix J). The area of the band occupied by the population along the route is equal to the kilometers traveled multiplied by, e.g., 1.6 for a band width of 800 m on each side of the route.

Table II-8. Average individual dose ("unit risk") to various receptors, truck routes.

	Resident near stops Sv-km^2/hour	Resident along route Sv	Occupants of vehicles sharing route
Truck-DU rural	3.26E-06	3.1E-10	1.2E-07
Truck-DU suburban	2.84E-06	2.7E-10	2.7E-07
Truck-DU urban		5.2E-12	6.0E-07
Truck-DU urban rush hour		1.2E-12	5.5E-07
6.9 people sharing stop (person-rem)	2.3E-04		

II.5.4 Doses along selected routes.

Doses to receptors along the routes shown in Table II-5 are presented below.

II.5.4.1 Collective doses to receptors along the route

Using route data from Web TRAGIS, collective doses from incident-free transportation were calculated. For rural and suburban route segments, collective doses calculated were doses sustained by the resident population. Non-resident populations were included with residents as receptors along the urban segments of the routes. Tables II-9 to II-12 show collective doses along rail routes and Tables II-13 to II-116, along highway routes. Blank cells in the tables indicate that no route miles or population was present in those cells.

Table II-9. Collective doses to residents along the route (person-Sv) from rail transportation; shipment origin INL

DESTINATION AND ROUTE	Rail-Pb			Rail-Steel		
	Rural	Suburban	Urban	Rural	Suburban	Urban
ORNL						
Colorado	2.1E-07	9.3E-07		1.6E-07	7.1E-07	
Idaho	2.8E-06	1.2E-05	5.5E-06	2.1E-06	9.3E-06	4.2E-07
Illinois	2.8E-06	2.7E-05	7.2E-06	2.8E-06	2.7E-05	5.5E-07
Indiana	2.7E-06	1.3E-05	2.9E-06	2.1E-06	1.0E-05	2.2E-07
Kansas	2.0E-06	1.1E-05	2.5E-06	1.6E-06	8.1E-06	1.9E-07
Kentucky	4.2E-06	3.4E-05	1.4E-05	3.2E-06	2.6E-05	1.0E-06
Missouri	3.8E-06	3.6E-05	1.8E-05	2.9E-06	2.7E-05	1.4E-06
Nebraska	5.6E-06	2.0E-05	5.6E-06	4.3E-06	1.5E-05	4.2E-07
Tennessee	2.0E-06	1.3E-05	6.7E-07	1.5E-06	9.6E-06	5.1E-08
Wyoming	2.3E-06	1.4E-05	3.3E-06	1.7E-06	1.1E-05	2.5E-07
DEAF SMITH						
Colorado	5.2E-06	6.6E-05	3.0E-05	2.7E-05	5.1E-05	2.1E-06
Idaho	2.8E-06	1.2E-05	6.0E-06	5.5E-06	9.3E-06	4.2E-07
Oklahoma	1.7E-07	2.9E-07	0.0E+00	1.3E-07	2.2E-07	0.0E+00
Texas	6.5E-07	5.4E-06	9.4E-07	5.0E-07	4.1E-06	7.1E-08
Wyoming	1.8E-06	9.6E-06	2.4E-06	2.4E-06	7.3E-06	1.8E-07
HANFORD						
Idaho	6.0E-06	2.6E-05	9.6E-06	9.6E-06	4.0E-06	7.3E-07
Oregon	2.3E-06	1.5E-05	3.5E-06	3.5E-06	1.5E-06	2.7E-07
Washington	1.9E-07	7.0E-06	4.2E-06	4.2E-06	1.2E-07	3.2E-07
SKULL VALLEY						
Idaho	2.3E-06	1.0E-05	5.3E-06	5.3E-06	1.5E-06	4.1E-07
Utah	2.6E-06	3.0E-05	1.8E-05	1.8E-05	2.3E-05	1.4E-06

Table II-10. Collective doses to residents along the route (person-Sv), rail transportation, shipment origin Indian Point

DESTINATION	Rail-Steel-Pb			Rail-Steel		
	Rural	Suburban	Urban	Rural	Suburban	Urban
ORNL						
Delaware	2.0E-08	1.2E-05	1.3E-05	1.5E-08	8.9E-06	1.0E-06
Washington DC	5.1E-09	1.4E-06	7.3E-06	3.9E-09	1.1E-06	5.5E-07
Maryland	1.1E-06	3.6E-05	3.2E-05	8.4E-07	2.7E-05	2.4E-06
New Jersey	6.5E-07	1.9E-05	2.2E-05	4.9E-07	1.5E-05	1.7E-06
New York	4.9E-08	2.6E-06	5.4E-05	3.7E-08	2.0E-06	4.1E-06
Pennsylvania	7.9E-08	1.4E-05	5.1E-05	6.0E-08	1.0E-05	3.9E-06
Tennessee	3.6E-06	4.9E-05	1.0E-05	2.7E-06	3.7E-05	8.0E-07
Virginia	6.5E-06	9.4E-05	3.9E-05	5.0E-06	7.2E-05	2.7E-06
DEAF SMITH						
Illinois	2.4E-06	4.3E-05	3.9E-05	1.8E-06	3.3E-05	2.9E-06
Indiana	3.3E-06	1.8E-05	8.6E-06	2.5E-06	1.4E-05	6.6E-07
Iowa	4.7E-07	1.0E-06	5.0E-07	3.6E-07	7.6E-07	3.8E-08
Kansas	3.2E-06	2.9E-05	1.3E-05	2.5E-06	2.2E-05	9.6E-07
Missouri	1.9E-06	1.1E-05	3.8E-06	1.5E-06	8.6E-06	2.9E-07
New York	8.7E-06	9.8E-05	7.9E-05	6.6E-06	7.4E-05	6.0E-06
Ohio	3.9E-06	5.1E-05	3.7E-05	3.0E-06	3.9E-05	2.8E-06
Oklahoma	7.2E-07	6.4E-06	8.3E-07	5.5E-07	4.9E-06	6.3E-08
Pennsylvania	6.6E-07	1.5E-05	7.8E-06	5.0E-07	4.4E-07	4.0E-07
Texas	1.2E-06	8.2E-06	2.0E-06	8.9E-07	6.2E-06	1.5E-07
HANFORD						
Idaho	1.6E-06	1.1E-05	1.5E-06	1.2E-06	1.0E-06	2.9E-07
Illinois	2.1E-06	3.2E-05	3.7E-05	1.6E-06	1.4E-06	8.6E-07
Indiana	3.4E-06	1.8E-05	8.6E-06	2.9E-06	2.2E-06	4.8E-07
Minnesota	5.1E-06	4.7E-05	1.9E-05	3.9E-06	3.4E-06	1.3E-06
Montana	0.0E+00	2.1E-05	2.2E-06	0.0E+00	0.0E+00	5.6E-07
New York	8.7E-06	9.8E-05	7.9E-05	6.6E-06	7.4E-05	6.0E-06
North Dakota	1.6E-06	1.3E-05	4.1E-06	1.2E-06	1.1E-06	3.5E-07
Ohio	3.9E-06	5.1E-05	3.7E-05	3.0E-06	3.9E-05	2.8E-06
Pennsylvania	6.6E-07	1.5E-05	7.8E-06	5.0E-07	4.4E-07	4.0E-07
Washington	1.8E-06	2.2E-05	1.0E-05	1.4E-06	1.2E-06	5.7E-07
Wisconsin	2.7E-06	1.3E-05	6.0E-06	2.0E-06	1.8E-06	3.5E-07
SKULL VALLEY						0
Colorado	2.1E-07	9.3E-07	0.0E+00	1.6E-07	7.1E-07	0.0E+00
Illinois	2.1E-06	3.3E-05	4.3E-05	1.6E-06	2.5E-05	3.3E-06
Indiana	3.3E-06	1.8E-05	8.6E-06	2.5E-06	1.4E-05	6.6E-07
Iowa	6.4E-06	2.9E-05	5.5E-06	6.4E-06	2.9E-05	6.0E-06
Nebraska	6.7E-06	3.2E-05	9.9E-06	5.1E-06	2.4E-05	7.5E-07
New York	8.7E-06	9.8E-05	7.9E-05	6.6E-06	7.4E-05	6.0E-06
Ohio	3.9E-06	5.1E-05	3.7E-05	3.0E-06	3.9E-05	2.8E-06
Pennsylvania	6.6E-07	1.5E-05	7.8E-06	5.0E-07	4.4E-07	4.0E-07
Utah	2.0E-06	2.9E-05	1.8E-05	1.6E-06	1.4E-06	7.4E-07
Wyoming	2.2E-06	1.5E-05	3.7E-06	1.6E-06	1.4E-06	4.0E-07

Table II-11. Collective doses to residents along the route (person-Sv) rail transportation; shipment origin Kewaunee

DESTINATION AND ROUTES	Rail-Steel-Pb			Rail-Steel		
	Rural	Suburban	Urban	Rural	Suburban	Urban
ORNL						
Illinois	3.8E-07	3.3E-05	4.0E-05	2.9E-07	2.5E-05	3.0E-06
Indiana	3.3E-06	1.8E-05	8.6E-06	2.5E-06	1.4E-05	6.6E-07
Kentucky	5.1E-06	2.6E-05	1.1E-05	3.9E-06	2.0E-05	8.6E-07
Ohio	3.5E-06	4.8E-05	2.3E-05	2.6E-06	3.6E-05	1.7E-06
Tennessee	1.2E-06	7.9E-06	6.5E-07	9.0E-07	6.1E-06	5.0E-08
Wisconsin	3.1E-06	4.0E-05	2.4E-05	6.0E-08	1.0E-05	1.8E-06
DEAF SMITH						
Illinois	2.6E-06	5.6E-05	4.8E-05	2.0E-06	4.3E-05	3.7E-06
Iowa	4.7E-07	1.0E-06	5.0E-07	3.6E-07	7.6E-07	3.8E-08
Kansas	3.2E-06	2.9E-05	1.3E-05	2.5E-06	2.2E-05	9.6E-07
Missouri	1.9E-06	1.1E-05	4.5E-06	1.5E-06	8.6E-06	2.9E-07
Oklahoma	6.7E-07	6.0E-06	7.3E-07	5.1E-07	4.6E-06	5.6E-08
Texas	1.2E-06	8.2E-06	2.0E-06	8.9E-07	6.2E-06	1.5E-07
Wisconsin	3.1E-06	4.0E-05	2.4E-05	6.0E-08	1.0E-05	1.8E-06
HANFORD						
Idaho	1.4E-06	1.1E-05	1.5E-06	1.2E-06	1.0E-06	1.2E-07
Minnesota	5.3E-06	4.8E-05	1.5E-05	4.0E-06	3.5E-06	1.1E-06
Montana	0.0E+00	2.1E-05	2.2E-06	0.0E+00	1.6E-05	1.7E-07
North Dakota	1.6E-06	1.3E-05	4.1E-06	1.2E-06	1.1E-06	3.5E-07
Washington	1.8E-06	2.2E-05	1.0E-05	1.4E-06	1.2E-06	7.9E-07
Wisconsin	5.7E-06	3.5E-05	1.4E-05	4.3E-06	3.8E-06	1.1E-06
SKULL VALLEY						
Colorado	2.1E-07	9.3E-07		1.6E-07	7.1E-07	
Illinois	2.3E-06	4.3E-05	4.4E-05	1.7E-06	3.3E-05	3.4E-06
Iowa	6.4E-06	2.9E-05	5.5E-06	4.9E-06	2.2E-05	4.2E-07
Nebraska	6.7E-06	3.2E-05	9.9E-06	5.1E-06	2.4E-05	7.5E-07
Utah	2.0E-06	2.9E-05	1.8E-05	1.6E-06	1.4E-06	1.4E-06
Wisconsin	3.1E-06	4.0E-05	2.4E-05	6.0E-08	1.0E-05	1.8E-06
Wyoming	2.2E-06	1.5E-05	3.7E-06	1.6E-06	1.4E-06	2.8E-07

Table II-12. Collective doses to residents along the route (person-Sv) rail shipment origin Maine Yankee

DESTINATION AND ROUTES	Rail-Steel-Pb Rural	Rail-Steel-Pb Suburban	Rail-Steel-Pb Urban	Rail-Steel Rural	Rail-Steel Suburban	Rail-Steel Urban
ORNL						
Kentucky	5.1E-06	2.6E-05	1.1E-05	3.9E-06	2.0E-05	8.6E-07
Maine	1.5E-06	2.5E-05	9.9E-06	1.1E-06	1.9E-05	7.5E-07
Massachusetts	2.1E-06	4.6E-05	2.8E-05	1.6E-06	3.5E-05	2.2E-06
New Hampshire	6.1E-07	1.2E-05	4.0E-06	4.6E-07	9.2E-06	3.1E-07
New York	7.7E-06	8.3E-05	2.9E-05	5.9E-06	6.3E-05	2.2E-06
Ohio	5.7E-06	7.8E-05	5.2E-05	4.4E-06	6.0E-05	3.9E-06
Pennsylvania	6.7E-07	1.5E-05	8.1E-06	5.1E-07	1.1E-05	6.2E-07
Tennessee	1.2E-06	7.9E-06	6.5E-07	9.0E-07	6.1E-06	5.0E-08
Vermont	1.1E-07	8.3E-07		8.1E-08	6.3E-07	
DEAF SMITH						
Illinois	2.4E-06	4.3E-05	3.9E-05	1.8E-06	3.3E-05	2.9E-06
Indiana	3.3E-06	1.8E-05	8.6E-06	2.5E-06	1.4E-05	6.6E-07
Iowa	4.7E-07	1.0E-06	5.0E-07	3.6E-07	7.6E-07	3.8E-08
Kansas	3.2E-06	2.9E-05	1.3E-05	2.5E-06	2.2E-05	9.6E-07
Maine	1.5E-06	2.5E-05	9.9E-06	1.1E-06	1.9E-05	7.5E-07
Massachusetts	2.1E-06	4.6E-05	2.8E-05	1.6E-06	3.5E-05	2.2E-06
Missouri	1.9E-06	1.1E-05	3.8E-06	1.5E-06	8.6E-06	2.9E-07
New Hampshire	6.1E-07	1.2E-05	4.0E-06	4.6E-07	9.2E-06	3.1E-07
New York	7.7E-06	8.3E-05	2.9E-05	5.9E-06	6.3E-05	2.2E-06
Ohio	3.9E-06	5.1E-05	3.7E-05	3.0E-06	3.9E-05	2.8E-06
Oklahoma	6.9E-07	6.2E-06	7.8E-07	5.3E-07	4.7E-06	5.9E-08
Pennsylvania	6.6E-07	1.5E-05	7.8E-06	5.0E-07	1.1E-05	5.9E-07
Texas	1.2E-06	8.2E-06	2.0E-06	8.9E-07	6.2E-06	1.5E-07
Vermont	1.1E-07	8.3E-07		8.1E-08	6.3E-07	

Table II-12. Collective doses to residents along the route (person-Sv) from rail transportation; shipment origin Maine Yankee -- continued

DESTINATION	Rail-Steel-Pb			Rail-Steel		
	Rural	Suburban	Urban	Rural	Suburban	Urban
HANFORD						
Idaho	1.6E-06	1.1E-05	1.5E-06	1.2E-06	8.2E-06	1.2E-07
Illinois	2.2E-06	3.3E-05	3.5E-05	1.7E-06	2.5E-05	2.7E-06
Indiana	3.3E-06	1.8E-05	8.6E-06	2.5E-06	1.4E-05	6.6E-07
Maine	1.5E-06	2.5E-05	9.9E-06	1.1E-06	1.9E-05	7.5E-07
Massachusetts	2.1E-06	4.6E-05	2.8E-05	1.6E-06	3.5E-05	2.2E-06
Minnesota	5.1E-06	4.7E-05	1.9E-05	3.9E-06	3.6E-05	1.4E-06
Montana	3.5E-06	2.1E-05	2.2E-06	2.7E-06	1.6E-05	1.7E-07
New Hampshire	6.1E-07	1.2E-05	4.0E-06	4.6E-07	9.2E-06	3.1E-07
New York	7.7E-06	8.3E-05	3.5E-05	5.9E-06	6.3E-05	2.2E-06
North Dakota	1.6E-06	1.3E-05	4.1E-06	1.2E-06	1.0E-05	3.2E-07
Ohio	3.9E-06	5.1E-05	3.7E-05	3.0E-06	3.9E-05	2.8E-06
Pennsylvania	6.6E-07	1.5E-05	7.8E-06	5.0E-07	1.1E-05	5.9E-07
Vermont	1.1E-07	8.3E-07		8.1E-08	6.3E-07	
Washington	1.8E-06	2.2E-05	1.0E-05	1.4E-06	1.2E-06	5.7E-07
Wisconsin	2.7E-06	1.3E-05	6.0E-06	2.0E-06	1.0E-05	4.6E-07
SKULL VALLEY						
Colorado	1.0E-06	4.3E-05	2.0E-05	7.8E-07	3.2E-05	1.5E-06
Illinois	3.2E-06	2.6E-05	8.2E-06	2.4E-06	1.9E-05	6.0E-07
Indiana	3.2E-06	2.7E-05	7.8E-06	2.4E-06	2.1E-05	5.9E-07
Iowa	7.2E-06	2.6E-05		5.5E-06	2.0E-05	
Maine	1.5E-06	2.6E-05		1.2E-06	2.0E-05	
Massachusetts	1.0E-06	4.5E-05		7.9E-07	3.4E-05	
Nebraska	7.6E-06	2.1E-05	5.9E-06	5.8E-06	1.6E-05	4.5E-07
New Hampshire	1.8E-07	5.9E-06	7.8E-07	1.4E-07	4.5E-06	5.9E-08
New York	7.7E-06	8.3E-05	2.9E-05	5.9E-06	6.3E-05	2.2E-06
Ohio	3.9E-06	5.1E-05	3.7E-05	3.0E-06	3.9E-05	2.8E-06
Pennsylvania	7.8E-07	1.8E-05	9.5E-06	6.0E-07	1.3E-05	7.2E-07
Utah	2.0E-06	2.9E-05	1.8E-05	1.6E-06	1.4E-06	1.4E-06
Vermont	1.1E-07	8.3E-07		8.1E-08	6.3E-07	
Wyoming	2.2E-06	1.5E-05	3.7E-06	1.6E-06	1.4E-06	2.8E-07

Table II-13. Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin Maine Yankee

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour
ORNL	**Connecticut**	4.5E-07	2.4E-05	1.6E-08	3.6E-10
	Maine	6.3E-07	1.2E-05	1.9E-09	4.2E-11
	Maryland	5.4E-08	2.1E-06	3.9E-10	8.6E-12
	Massachusetts	4.3E-07	1.9E-05	5.8E-09	1.3E-10
	New Hampshire	7.6E-08	2.4E-06	2.1E-10	4.7E-12
	New Jersey	2.9E-07	1.0E-05	8.9E-09	2.0E-10
	New York	3.4E-09	2.6E-06	8.9E-09	2.0E-10
	Pennsylvania	1.8E-06	2.1E-05	4.6E-09	1.0E-10
	Tennessee	1.2E-06	1.5E-05	2.3E-09	5.1E-11
	Virginia	2.8E-06	3.1E-05	3.6E-09	8.0E-11
	West Virginia	1.8E-07	4.2E-06	2.1E-10	4.6E-12
DEAF SMITH	Connecticut	4.6E-07	2.4E-05	1.6E-08	3.6E-10
	Maine	6.4E-07	1.1E-05	1.1E-09	2.4E-11
	Maryland	5.4E-08	2.1E-06	3.9E-10	8.6E-12
	Massachusetts	4.3E-07	1.9E-05	5.8E-09	1.3E-10
	New Hampshire	7.6E-08	2.4E-06	2.1E-10	4.7E-12
	New Jersey	3.9E-07	1.4E-05	5.0E-09	1.1E-10
	New York	3.8E-08	6.8E-06	5.8E-09	1.3E-10
	Oklahoma	2.6E-06	1.3E-05	3.2E-09	7.0E-11
	Pennsylvania	1.5E-06	1.7E-05	3.8E-09	8.5E-11
	Tennessee	4.7E-06	4.0E-05	1.1E-08	2.5E-10
	Texas	6.3E-07	3.6E-06	2.7E-09	6.1E-11
	Virginia	2.7E-06	3.0E-05	3.5E-09	7.8E-11
	West Virginia	1.8E-07	4.2E-06	2.1E-10	4.6E-12

Table II-13. Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin Maine Yankee -- continued

	ROUTES	Rural	Suburban	Urban	Urban Rush Hour
HANFORD	Connecticut	4.3E-07	1.8E-05	8.6E-09	1.9E-10
	Idaho	2.2E-06	1.1E-05	2.8E-09	6.1E-11
	Illinois	1.3E-06	1.0E-05	3.5E-09	7.8E-11
	Indiana	1.3E-06	1.1E-05	3.3E-09	7.4E-11
	iowa	3.0E-06	1.1E-05	1.7E-09	3.7E-11
	Maine	6.4E-07	1.1E-05	1.1E-09	2.4E-11
	Massachusetts	4.5E-07	2.0E-05	6.0E-09	1.3E-10
	Nebraska	3.2E-06	8.6E-06	2.5E-09	5.6E-11
	New Hampshire	7.6E-08	2.4E-06	2.1E-10	4.7E-12
	New York	4.4E-07	9.1E-06	1.4E-09	3.2E-11
	Ohio	2.0E-06	2.0E-05	4.9E-09	1.1E-10
	Oregon	1.3E-06	4.6E-06	7.6E-10	1.7E-11
	Pennsylvania	3.2E-06	1.8E-05	2.4E-09	5.3E-11
	Utah	1.0E-06	6.4E-06	5.2E-10	1.2E-11
	Washington	1.4E-07	1.3E-06	1.4E-09	3.2E-11
	Wyoming	1.5E-06	5.7E-06	1.0E-09	2.2E-11
SKULL VALLEY	Connecticut	4.3E-07	1.8E-05	8.6E-09	1.9E-10
	Illinois	1.3E-06	1.0E-05	3.5E-09	7.8E-11
	Indiana	1.3E-06	1.1E-05	3.3E-09	7.4E-11
	Iowa	3.0E-06	1.1E-05	1.7E-09	3.7E-11
	Maine	6.4E-07	1.1E-05	1.1E-09	2.4E-11
	Massachusetss	4.3E-07	1.9E-05	5.8E-09	1.3E-10
	Nebraska	3.2E-06	8.6E-06	2.5E-09	5.6E-11
	New Hampshire	7.6E-08	2.4E-06	2.1E-10	4.7E-12
	New York	4.4E-07	9.1E-06	1.4E-09	3.2E-11
	Ohio	2.0E-06	2.0E-05	4.9E-09	1.1E-10
	Pennsylvamia	3.2E-06	1.8E-05	2.4E-09	5.3E-11
	Utah	8.2E-07	7.4E-06	5.9E-09	1.3E-10
	Wyoming	1.5E-06	5.7E-06	1.0E-09	2.2E-11

Table II-14 Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin Indian Point.

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour
ORNL	Maryland	5.4E-08	2.1E-06	3.9E-10	8.3E-12
	New Jersey	3.9E-07	1.4E-05	5.0E-09	1.1E-10
	New York	7.5E-08	7.0E-06	6.3E-09	1.4E-10
	Pennsylvania	1.5E-06	1.7E-05	3.8E-09	8.3E-11
	Tennessee	1.3E-06	1.6E-05	1.9E-09	3.8E-11
	Virginia	2.7E-06	3.0E-05	3.5E-09	7.8E-11
	West virginia	1.8E-07	4.2E-06	2.1E-10	4.6E-12
DEAF SMITH	Arkansas	2.3E-06	1.6E-05	2.8E-09	6.2E-11
	Maryland	5.4E-08	2.1E-06	3.9E-10	8.6E-12
	New Jersey	3.9E-07	1.4E-05	5.0E-09	1.1E-10
	New York	7.5E-08	7.0E-06	6.3E-09	1.4E-10
	Oklahoma	2.7E-06	1.4E-05	3.3E-09	6.6E-11
	Pennsylvania	1.5E-06	1.7E-05	3.8E-09	8.3E-11
	Texas	6.3E-07	3.6E-06	2.7E-09	5.2E-11
	Virginia	2.7E-06	3.0E-05	3.5E-09	7.8E-11
	West Virginia	1.8E-07	4.2E-06	2.1E-10	4.6E-12
HANFORD	Idaho	2.2E-06	1.1E-05	2.8E-09	5.4E-11
	Illinois	1.3E-06	1.0E-05	3.6E-09	8.1E-11
	Indiana	1.4E-06	1.1E-05	3.3E-09	7.4E-11
	Iowa	3.0E-06	1.1E-05	1.7E-09	3.7E-11
	Nebraska	3.2E-06	8.6E-06	2.5E-09	5.6E-11
	New Jersey	4.2E-07	1.1E-05	4.1E-09	9.1E-11
	New York	7.5E-08	7.0E-06	6.3E-09	1.4E-10
	Ohio	2.0E-06	2.0E-05	4.9E-09	1.1E-10
	Oregon	1.4E-06	5.1E-06	7.6E-10	1.7E-11
	Pennsylvania	2.9E-06	1.4E-05	1.1E-09	2.3E-11
	Utah	1.0E-06	6.4E-06	5.2E-10	1.2E-11
	Washington	1.4E-07	1.3E-06	1.4E-09	3.2E-11
	Wyoming	1.5E-06	5.7E-06	1.0E-09	2.2E-11
SKULL VALLEY	Illinois	1.3E-06	1.0E-05	3.5E-09	7.8E-11
	Indiana	1.3E-06	1.1E-05	3.3E-09	7.4E-11
	Iowa	3.0E-06	1.1E-05	1.7E-09	3.7E-11
	Nebraska	3.2E-06	8.6E-06	2.5E-09	5.6E-11
	New Jersey	4.2E-07	1.1E-05	4.1E-09	9.1E-11
	New York	7.5E-08	7.0E-06	6.3E-09	1.4E-10
	Ohio	2.0E-06	2.0E-05	4.9E-09	1.1E-10
	Pennsylvania	2.9E-06	1.4E-05	1.1E-09	2.3E-11
	Utah	8.2E-07	7.4E-06	5.9E-09	1.3E-10
	Wyoming	1.5E-06	5.7E-06	1.0E-09	2.2E-11

Table II-15. Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin INL

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour
ORNL	Colorado	1.7E-06	7.4E-06	4.0E-09	8.8E-11
	Idaho	1.0E-06	4.3E-06	7.1E-10	1.4E-11
	Illinois	1.4E-06	9.7E-06	5.6E-10	1.3E-11
	Kansas	2.6E-06	1.1E-05	3.3E-09	7.3E-11
	Kentucky	9.2E-07	3.7E-06	5.3E-11	1.2E-12
	Missouri	1.9E-06	2.4E-05	9.0E-09	2.0E-10
	Tennessee	2.2E-06	1.4E-05	3.3E-09	7.3E-11
	Utah	1.1E-06	6.7E-06	5.2E-10	1.2E-11
	Wyoming	1.3E-06	4.0E-06	7.1E-10	1.6E-11
DEAF SMITH	Colorado	2.0E-06	2.5E-05	1.3E-08	2.8E-10
	Idaho	1.0E-06	4.3E-06	7.1E-10	1.6E-11
	New Mexico	1.9E-06	8.9E-06	5.3E-09	1.2E-10
	Texas	8.4E-08	1.5E-07	0.0E+00	0.0E+00
	Utah	9.6E-07	6.1E-06	5.0E-10	1.1E-11
	Wyoming	1.3E-06	4.1E-06	7.4E-10	1.6E-11
HANFORD	Idaho	2.7E-06	1.4E-05	3.2E-09	7.1E-11
	Oregon	1.3E-06	4.7E-06	7.3E-10	1.6E-11
	Washington	1.4E-07	1.3E-06	1.4E-09	3.2E-11
SKULL VALLEY	Idaho	1.0E-06	4.3E-06	7.1E-10	1.6E-11
	Utah	9.6E-07	1.2E-05	7.0E-09	1.5E-10

Table II-16. Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin Kewaunee.

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour
ORNL	Illinois	3.4E-07	1.6E-05	1.0E-08	2.4E-10
	Indiana	2.1E-06	1.9E-05	5.8E-09	1.3E-10
	Kentucky	1.9E-06	1.7E-05	3.7E-09	8.3E-11
	Ohio	9.5E-08	1.3E-06	3.1E-10	7.0E-12
	Tennessee	5.8E-07	9.9E-06	2.3E-09	4.5E-11
	Wisconsin	1.6E-06	1.2E-05	1.0E-08	2.2E-10
DEAF SMITH	Illinois	1.2E-06	5.2E-06	3.5E-10	8.2E-12
	Iowa	2.3E-06	1.2E-05	1.8E-09	4.0E-11
	Kansas	1.7E-06	1.1E-05	4.6E-09	1.0E-10
	Missouri	1.0E-06	9.7E-06	1.8E-09	3.2E-11
	Oklahoma	1.7E-06	9.6E-06	2.7E-09	6.0E-11
	Texas	6.3E-07	3.6E-06	1.8E-09	5.2E-11
	Wisconsin	2.0E-06	1.2E-05	8.0E-09	1.8E-10
HANFORD	Idaho	3.3E-07	6.6E-06	1.7E-09	3.4E-11
	Minnesota	2.7E-06	4.4E-06	3.8E-10	8.3E-12
	Montana	3.3E-06	1.3E-05	2.9E-09	6.4E-11
	South Dakota	2.3E-06	6.2E-06	8.9E-10	2.0E-11
	Washington	1.6E-06	1.5E-05	5.9E-09	1.3E-10
	Wisconsin	3.3E-06	1.8E-05	7.8E-09	1.7E-10
	Wyoming	8.9E-07	2.6E-06	6.3E-10	1.4E-11
SKULL VALLEY	Illinois	1.2E-06	5.2E-06	3.5E-10	8.2E-12
	Iowa	3.0E-06	1.1E-05	1.7E-09	3.7E-11
	Nebraska	3.2E-06	8.6E-06	2.5E-09	5.6E-11
	Utah	8.2E-07	7.4E-06	5.9E-09	1.2E-10
	Wisconsin	2.0E-06	1.2E-05	8.0E-09	1.8E-10
	Wyoming	1.5E-06	5.7E-06	1.0E-09	2.2E-11

Collective dose is best used in making comparisons; e.g., in comparing the risks of routine transportation along different routes. All collective doses modeled are of the order of 10^{-5} person-Sv or less. The tables show that, in general, urban residents sustain a slightly larger dose from rail transportation than from truck transportation on the same state route, even though urban population densities are similar; e.g., for the Maine urban segment of the Maine Yankee-to-ORNL route,

- the truck route urban population density is 2706 persons/km^2 and the collective dose is 1×10^{-7} person-Sv
- the rail route urban population density is 2527 persons/km^2, but the collective dose is 9.9×10^{-4} person-Sv from the Rail-Steel-Pb cask is almost 100 times larger than the dose from the Truck-DU cask, even though the external dose rates from the two casks are nearly the same.

Doses from rail transportation through urban areas are larger than those from truck transportation because train transportation was designed, and train tracks were laid, to go from city center to city center. Trucks carrying spent fuel, on the other hand, are required to use the interstate highway system, and to use bypasses around cities where such bypasses exist. In the example presented, the truck traverses 5 km of urban route while the train traverses 13 urban km. In addition, the average urban train speed is 24 km/hour (15mph) while the average urban truck speed is 102 km/hour (63.4 mph). A truck carrying a cask through an urban area at about four times the speed of a train carrying a similar cask will deliver ¼ the dose of the trail cask.

II.5.4.2 Doses to occupants of vehicles sharing the route

Rail

The dose to occupants of trains other than the train carrying the radioactive cargo is provided in Table II-17. The vehicle occupancies used to calculate the table, one person on rural and suburban segments, and five people on urban segments, have been used historically in RADTRAN since 1988. The occupancy is consistent with the following considerations:

- Freight trains carry a crew of three, but all but one or two of the 60 to 120 cars on a freight train are unoccupied.
- Urban track carries almost all passenger rail traffic.
- Dose is calculated for one cask on a train, and rail statistics are per railcar, not per train.

The net dose to occupants of other trains depends on train speed and the external dose rate from the spent fuel cask. Train speeds are available only for the entire U.S., not for each state. Therefore the doses to occupants of trains that share the route with either a loaded Rail-Steel-Pb cask or a loaded Rail-Steel cask are shown in Table II-17 for rural, suburban, and urban segments of each entire route. The rural and about half of the suburban collective doses may be unrealistically large.

Table II-17. Collective doses (person-Sv) to occupants of trains sharing the route.

SHIPMENT ORIGIN	SHIPMENT DESTINATION	Rail-Steel-Pb CASK			Rail-Steel CASK		
		Rural	Suburban	Urban	Rural	Suburban	Urban
MAINE YANKEE	ORNL	5.3E-06	1.6E-05	1.1E-04	4.0E-06	1.2E-05	7.6E-06
	DEAF SMITH	1.0E-05	1.8E-05	1.4E-04	7.7E-06	1.4E-05	9.9E-06
	HANFORD	1.5E-05	2.2E-05	1.5E-04	1.2E-05	1.7E-05	1.1E-05
	SKULL VALLEY	1.3E-05	2.4E-05	1.2E-04	9.9E-06	1.9E-05	8.5E-06
KEWAUNEE	ORNL	3.7E-06	9.4E-06	8.5E-05	2.8E-06	7.1E-06	5.9E-06
	DEAF SMITH	6.4E-06	7.0E-06	7.4E-05	4.8E-06	5.3E-06	5.2E-06
	HANFORD	6.7E-06	9.0E-06	4.1E-05	5.0E-06	6.9E-06	2.8E-06
	SKULL VALLEY	9.4E-06	1.0E-05	8.5E-05	7.2E-06	7.9E-06	5.9E-06
INDIAN POINT	ORNL	2.5E-06	1.1E-05	1.4E-04	1.9E-06	8.2E-06	9.7E-06
	DEAF SMITH	9.8E-06	1.4E-05	1.4E-04	7.4E-06	1.1E-05	9.6E-06
	HANFORD	1.2E-05	1.9E-05	1.5E-04	8.8E-06	1.5E-05	1.1E-05
	SKULL VALLEY	5.9E-06	4.2E-05	7.1E-05	4.4E-06	3.2E-05	2.7E-05
INL	ORNL	4.0E-06	5.3E-05	5.5E-05	3.0E-06	4.0E-05	3.8E-06
	DEAF SMITH	7.3E-06	4.4E-06	2.7E-05	5.6E-06	3.3E-06	1.9E-06
	HANFORD	4.1E-06	2.3E-06	1.4E-05	3.1E-06	1.8E-06	9.4E-07
	SKULL VALLEY	1.5E-06	2.0E-06	1.7E-05	1.1E-06	1.5E-06	1.2E-06

Truck

Vehicle density data for large semi-detached trailer trucks traveling U.S. interstates and primary highways is available and well qualified. Every state records traffic counts on major (and most minor) highways, and publishes these routinely. Average vehicle density data from each of the 10 EPA regions was used (Weiner,et al. 2009, Appendix D). The EPA regions were used because they include all of the "lower 48" U.S. states (Alaska and Hawaii are included in EPA Region 10 but are not considered in this risk assessment). Table II-18 shows the 10 EPA regions.

Table II-18. States comprising the ten EPA regions

Region	States Included in Region
1	Connecticut, Massachusetts, Maine, New Hampshire, Rhode Island, Vermont
2	New Jersey, New York
3	Delaware. Maryland, Pennsylvania, Virginia, West Virginia
4	Alabama, Florida, Georgia, Kentucky, Mississippi, North Carolina, South Carolina, Tennessee
5	Illinois, Indiana, Michigan, Minnesota, Ohio, Wisconsin
6	Arkansas, Louisiana, New Mexico, Oklahoma, Texas
7	Iowa, Kansas, Missouri, Nebraska
8	Colorado, Montana, North Dakota, South Dakota, Utah, Wyoming
9	Arizona, California, Nevada
10	Idaho, Oregon , Washington

The calculation of doses to occupants sharing the highway route with the radioactive materials truck is complex in that vehicles traveling in both directions are considered (the equations that describe this calculation are Equations 28 and 34 of Neuhauser et al, 2000). Figure II-6 is the diagram accompanying these equations and shows the parameters used in the calculation. Parameter values are in Table II-1.

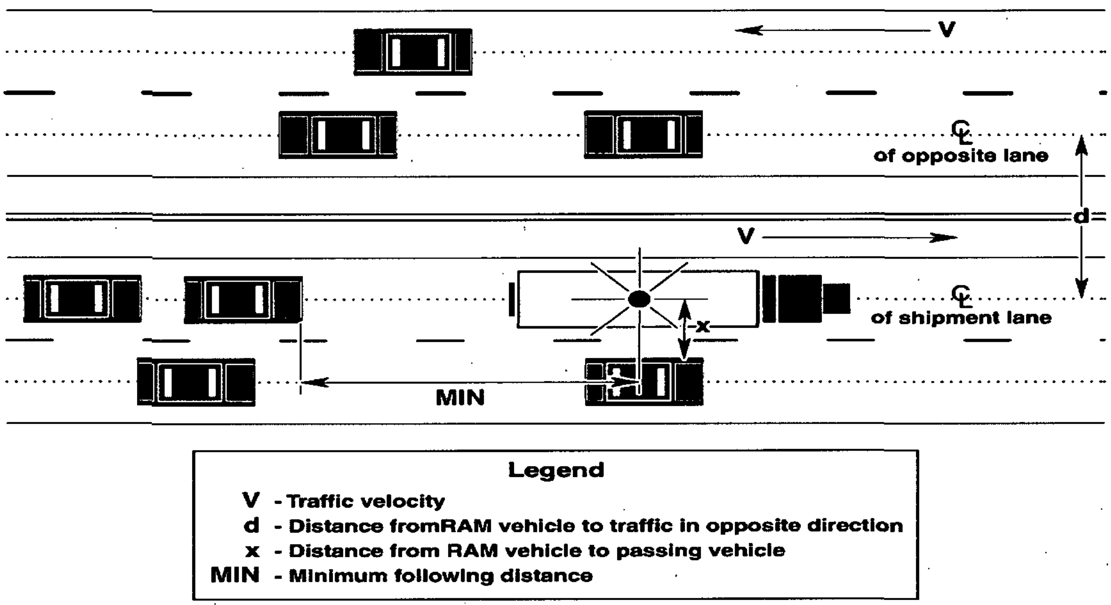

Figure II-6. Parameters for calculating doses to occupants of highway vehicles sharing the route with the radioactive shipment (From Figure 3-2 of Neuhauser, et al, 2000).

Tables II-19 to II-22 show the doses to individuals in vehicles sharing the highway route with the truck carrying a loaded Truck-DU cask.

Table II-19. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin Maine Yankee

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour
ORNL	Connecticut	1.9E-06	9.1E-06	9.1E-06	8.5E-07
	Maine	2.9E-06	6.7E-06	1.1E-06	1.0E-07
	Maryland	1.3E-06	4.9E-06	9.0E-07	8.3E-08
	Massachusetts	1.7E-06	8.7E-06	3.4E-06	3.2E-07
	New Hampshire	3.7E-07	1.4E-06	1.9E-07	1.8E-08
	New Jersey	5.1E-06	1.2E-05	9.2E-06	8.5E-07
	New York	1.8E-07	2.1E-06	1.3E-05	1.2E-06
	Pennsylvania	3.0E-05	4.8E-05	7.0E-06	6.5E-07
	Tennessee	1.7E-05	3.2E-05	4.2E-06	3.9E-07
	Virginia	6.4E-05	9.3E-05	6.2E-06	5.7E-07
	West Virginia	2.8E-06	1.2E-05	4.5E-07	4.1E-08
DEAF SMITH	Connecticut	3.1E-05	2.1E-05	2.8E-06	2.6E-07
	Maine	2.0E-06	9.2E-06	9.2E-06	8.5E-07
	Maryland	2.9E-06	6.8E-06	7.3E-07	6.8E-08
	Massachusetts	1.3E-06	4.9E-06	9.0E-07	8.3E-08
	New Hampshire	4.2E-06	2.9E-05	8.7E-06	8.0E-06
	New Jersey	9.5E-07	4.8E-06	4.8E-07	4.4E-07
	New York	4.5E-06	1.6E-05	6.6E-06	6.1E-07
	Oklahoma	7.5E-07	6.8E-06	6.9E-06	6.4E-07
	Pennsylvania	4.2E-05	1.6E-05	2.8E-06	2.6E-07
	Tennessee	3.0E-05	4.8E-05	7.0E-06	6.5E-07
	Texas	7.8E-05	8.6E-05	2.0E-05	1.8E-06
	Virginia	2.2E-05	3.1E-06	2.4E-06	2.2E-07
	West Virginia	6.4E-05	9.3E-05	6.2E-06	5.7E-07

Table II-19. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin Maine Yankee -- continued

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour
HANFORD	Connecticut	1.7E-06	8.0E-06	5.1E-06	4.7E-07
	Idaho	4.4E-05	2.3E-05	4.6E-06	4.2E-07
	Illinois	2.4E-05	2.0E-05	5.0E-06	4.6E-07
	Indiana	1.8E-05	2.6E-05	4.6E-06	4.3E-07
	iowa	4.0E-05	1.7E-05	1.4E-06	1.3E-07
	Maine	2.9E-06	6.8E-06	7.3E-07	6.8E-08
	Massachusetts	1.7E-06	8.7E-06	3.4E-06	3.2E-07
	Nebraska	6.7E-05	1.3E-05	1.9E-06	1.8E-07
	New Hampshire	3.7E-07	1.4E-06	1.9E-07	1.8E-08
	New York	2.5E-06	4.6E-06	1.1E-06	9.9E-08
	Ohio	8.7E-05	6.9E-05	4.0E-06	3.7E-07
	Oregon	3.7E-05	9.5E-06	1.4E-06	1.3E-07
	Pennsylvania	8.7E-05	6.9E-05	4.0E-06	3.7E-07
	Utah	1.6E-05	1.1E-05	6.2E-07	5.7E-08
	Washington	7.6E-06	2.1E-06	2.6E-06	2.4E-07
	Wyoming	7.5E-05	1.0E-05	2.1E-06	2.0E-07
SKULL VALLEY	Connecticut	1.7E-06	8.0E-06	5.1E-06	4.7E-07
	Illinois	2.4E-05	2.0E-05	5.0E-06	4.6E-07
	Indiana	1.8E-05	2.6E-05	4.6E-06	4.3E-07
	Iowa	4.0E-05	1.7E-05	1.4E-06	1.3E-07
	Maine	2.9E-06	6.8E-06	7.3E-07	6.8E-08
	Massachusetss	1.7E-06	8.7E-06	3.4E-06	3.2E-07
	Nebraska	6.7E-05	1.3E-05	1.9E-06	1.8E-07
	New Hampshire	9.5E-07	4.8E-06	4.8E-07	4.4E-07
	New York	5.8E-06	1.3E-05	2.1E-06	1.9E-07
	Ohio	8.7E-05	6.9E-05	4.0E-06	3.7E-07
	Pennsylvamia	8.7E-05	6.9E-05	4.0E-06	3.7E-07
	Utah	1.8E-05	8.1E-06	6.1E-06	5.6E-07
	Wyoming	7.5E-05	1.0E-05	2.1E-06	2.0E-07

Table II-20. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin Indian Point.

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour
ORNL	Maryland	1.3E-06	4.9E-06	9.0E-07	8.3E-08
	New Jersey	4.5E-06	1.6E-05	6.6E-06	6.1E-07
	New York	1.3E-06	6.5E-06	7.6E-06	7.0E-07
	Pennsylvania	3.0E-05	4.8E-05	7.0E-06	6.5E-07
	Tennessee	1.7E-05	3.4E-05	3.8E-06	3.5E-07
	Virginia	6.4E-05	9.3E-05	6.2E-06	5.7E-07
	West Virginia	6.4E-05	1.2E-05	4.5E-07	4.1E-08
DEAF SMITH	Arkansas	3.1E-05	2.1E-05	2.8E-06	2.6E-07
	Maryland	1.3E-06	4.9E-06	9.0E-07	8.3E-08
	New Jersey	4.5E-06	1.6E-05	6.6E-06	6.1E-07
	New York	1.3E-06	6.5E-06	7.6E-06	7.0E-07
	Oklahoma	4.2E-05	1.6E-05	2.8E-06	2.6E-07
	Pennsylvania	3.0E-05	4.8E-05	7.0E-06	6.5E-07
	Texas	7.8E-05	8.6E-05	2.0E-05	1.8E-06
	Virginia	2.2E-05	3.1E-06	2.4E-06	2.2E-07
	West Virginia	6.4E-05	9.3E-05	6.2E-06	5.7E-07
HANFORD	Idaho	2.8E-06	1.2E-05	4.5E-07	4.1E-08
	Illinois	4.4E-05	2.3E-05	4.6E-06	4.2E-07
	Indiana	2.4E-05	2.0E-05	5.0E-06	4.6E-07
	Iowa	1.8E-05	2.6E-05	4.6E-06	4.3E-07
	Nebraska	4.0E-05	1.7E-05	1.4E-06	1.3E-07
	New Jersey	6.7E-05	1.3E-05	1.9E-06	1.8E-07
	New York	4.8E-06	1.3E-05	5.6E-06	5.2E-07
	Ohio	1.3E-06	6.5E-06	7.6E-06	7.0E-07
	Oregon	1.5E-06	7.6E-06	8.1E-06	7.4E-07
	Pennsylvania	3.7E-05	9.5E-06	1.4E-06	1.3E-07
	Utah	8.0E-05	5.7E-05	2.2E-06	2.0E-07
	Washington	1.6E-05	1.1E-05	6.2E-07	5.7E-08
	Wyoming	7.6E-06	2.1E-06	2.6E-06	2.4E-07
SKULL VALLEY	Illinois	7.5E-05	1.0E-05	2.1E-06	2.0E-07
	Indiana	2.4E-05	2.0E-05	5.0E-06	4.6E-07
	Iowa	1.8E-05	2.6E-05	4.6E-06	4.3E-07
	Nebraska	4.0E-05	1.7E-05	1.4E-06	1.3E-07
	New Jersey	6.7E-05	1.3E-05	1.9E-06	1.8E-07
	New York	5.6E-06	1.5E-05	5.9E-06	5.5E-07
	Ohio	1.5E-06	7.6E-06	8.1E-06	7.4E-07
	Pennsylvania	2.8E-05	4.1E-05	7.3E-06	6.7E-07
	Utah	8.0E-05	5.7E-05	2.2E-06	2.0E-07
	Wyoming	1.7E-05	8.1E-06	6.1E-06	5.6E-07

Table II-21. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin INL

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour
ORNL	Colorado	3.1E-05	1.1E-05	4.0E-06	3.7E-07
	Idaho	2.2E-05	8.0E-06	1.3E-06	1.2E-07
	Illinois	2.5E-05	2.4E-05	1.1E-06	1.0E-07
	Kansas	6.2E-05	1.4E-05	2.7E-06	2.5E-07
	Kentucky	1.8E-05	1.1E-05	1.2E-07	1.2E-08
	Missouri	2.5E-05	2.3E-05	7.2E-06	6.7E-07
	Tennessee	3.3E-05	3.5E-05	5.2E-06	4.8E-07
	Utah	1.3E-05	1.1E-05	6.2E-07	5.7E-08
	Wyoming	7.0E-05	7.6E-06	1.5E-06	1.4E-07
DEAF SMITH	Colorado	3.9E-05	3.6E-05	1.9E-05	1.8E-05
	Idaho	2.2E-05	8.0E-06	1.3E-06	1.2E-07
	New Mexico	6.4E-05	9.8E-06	4.8E-06	4.4E-07
	Texas	7.7E-06	1.7E-07	0.0E+00	0.0E+00
	Utah	1.3E-05	1.1E-05	6.2E-07	5.7E-08
	Wyoming	7.0E-05	7.6E-06	1.5E-06	1.4E-07
HANFORD	Idaho	5.5E-05	6.3E-05	5.4E-06	5.0E-07
	Oregon	3.7E-05	2.0E-05	1.4E-06	1.31E-07
	Washington	7.6E-06	2.1E-06	2.6E-06	2.4E-07
SKULL VALLEY	Idaho	2.2E-05	8.0E-06	1.3E-06	1.2E-07
	Utah	1.5E-05	1.5E-05	7.2E-06	6.6E-07

Table II-22. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin Kewaunee.

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban
ORNL	Illinois	3.7E-06	2.0E-05	1.4E-05	1.3E-06
	Indiana	3.3E-05	3.8E-05	8.3E-06	7.7E-07
	Kentucky	2.7E-05	4.3E-05	7.2E-06	6.7E-07
	Ohio	1.4E-06	2.5E-06	5.4E-07	5.0E-08
	Tennessee	1.1E-05	1.8E-05	4.4E-06	4.1E-07
	Wisconsin	2.0E-05	2.1E-05	1.3E-05	1.2E-06
DEAF SMITH	Illinois	2.0E-05	1.2E-05	5.9E-07	5.4E-08
	Iowa	3.2E-05	1.6E-05	1.6E-06	1.4E-07
	Kansas	2.9E-05	1.2E-05	3.5E-06	3.2E-07
	Missouri	1.4E-05	1.1E-05	1.3E-06	1.2E-07
	Oklahoma	3.4E-05	1.1E-05	2.8E-06	2.6E-07
	Texas	2.2E-05	3.1E-06	2.4E-06	2.2E-07
	Wisconsin	2.5E-05	2.3E-05	9.8E-06	9.0E-07
HANFORD	Idaho	9.3E-06	1.1E-05	3.0E-06	2.8E-07
	Minnesota	5.2E-05	1.3E-05	5.4E-07	5.0E-08
	Montana	9.6E-05	3.0E-05	5.4E-06	5.0E-07
	South Dakota	5.3E-05	1.2E-05	1.0E-06	9.5E-08
	Washington	4.6E-05	3.0E-05	1.1E-05	1.0E-06
	Wisconsin	4.6E-05	4.0E-05	9.9E-06	9.2E-07
	Wyoming	4.0E-05	4.1E-06	1.4E-06	1.3E-07
SKULL VALLEY	Illinois	2.0E-05	1.2E-05	5.9E-07	5.4E-08
	Iowa	4.0E-05	1.7E-05	1.4E-06	1.3E-07
	Nebraska	6.7E-05	1.3E-05	1.9E-06	1.8E-07
	Utah	2.4E-05	1.0E-05	8.8E-06	8.1E-06
	Wisconsin	2.5E-05	2.3E-05	9.8E-06	9.0E-07
	Wyoming	7.5E-05	1.0E-05	2.1E-06	2.0E-07

II.5.4.3 Doses from stopped vehicles

Rail

Trains are stopped in classification yards at the origin and destination of the trip. The usual length of these classification stops is 27 hours. The collective dose to the railyard workers at these classification stops from the radioactive cargo is calculated internally by RADTRAN and is based on calculations of Wooden (1986) which authors of this document have verified. This "classification yard dose" for the two rail casks studied is:

- For the Rail-Steel-Pb: 1.5 x 10^{-5} person-Sv
- For the Rail-Steel: 1.1 x 10^{-5} person-Sv

- These collective doses include doses to the train crew while the train is in the yard.

The collective dose to people living near a classification yard is calculated by multiplying the average dose from the rail cask to an individual living near a classification yard, as shown in Table II-7, by the population density between 200 and 800 meters from the rail yard. The population density is obtained from WebTRAGIS, and the integration from 200 to 800 meters (Table II-2) is performed by RADTRAN.

Most train stops along any route are shown in the WebTRAGIS output for that route. The stops on the rail route from Maine Yankee to Hanford are shown in Table II-23 as an example.

Table II-23. Example of rail stops on the Maine Yankee-to-Hanford rail route

Stop	Reason	Route type (R, S, U)[a] and State	Time (hours)
1	Railroad transfer (short line to ST)	S, ME	4.0
2	Railroad transfer (ST to CSXT)	R, NY	4.0
3	Railroad transfer (CSXT to IMB)	S, IL	4.0
4	Railroad transfer (IMB to BNSF)	S, IL	0
5	Railroad transfer (BNSF to UP)	S, WA	0

[a]Determined by the user from the WebTRAGIS output

Railyard worker collective doses can then be calculated for Stops 1, 2, and 3 in Table II-23. Parametr values are from Table II-23 and page 36.:

Dose: $(4/27)*(1.5 \times 10^{-5}) = 2.2 \times 10^{-6}$ person- Sv for the Rail-Pb cask.

Dose: $(4/27)*(1.1 \times 10^{-5}) = 1.6 \times 10^{-6}$ person-Sv for the Rail-Steel cask.

The factor of 4/27 is in the equation because the classification stop doses are calculated by RADTRAN for activities lasting a total of 27 hours, and the in-transit stops are for only four hours.

The average dose to an individual living 200 to 800 meters from a classification yard, as calculated by RADTRAN, is

- 3.5×10^{-7} Sv from the Rail-Pb cask.
- 2.7×10^{-7} Sv from the Rail-Steel cask.

Collective doses to residents near a yard (a classification yard or railroad stop) are then calculated from the general expression:

(II-4) Dose (person-Sv) = (Population density)*(Dose/hr to resident near yard)*(Stop time)

Thus, for a rural population density of 13.2 persons/km^2 (the average along the Maine Yankee-to-Hanford route) living near Stop 1 in Table II-23,

Dose = (13.2 persons/km^2)*(3.5×10^{-7} Sv-km^2/hour)*(4 hours) = 1.9×10^{-5} person-Sv.

Results for the stops are in Table II-24.

Table II-24. Doses at rail stops on the Maine Yankee-to-Hanford rail route

Stop	Route type (R, S, U)[a]	Time (hours)	Railyard worker dose (person-Sv)		Residents near stop (person-Sv)	
			Rail-Pb	Rail-Steel	Rail- Pb	Rail-Steel
1	S, ME	4.0	2.16E-06	1.61E-06	3.42E-06	2.59E-06
2	R, NY	4.0	2.16E-06	1.61E-06	9.15E-07	6.94E-07
3	S, IL	2.0	1.08E-06	8.05E-07	1.24E-05	9.37E-06

[a]Determined by the user from the WebTRAGIS output

Truck

Doses at truck stops are calculated differently. There are two types of receptors at a truck stop, in addition to the truck crew: residents who live near the stop and people who share the stop with the refueling truck. Griego, et al (1996) conducted some time and motion studies at a number of truck stops. They found that the average number of people at a stop between the gas pumps and the nearest building was 6.9, the average distance from the fuel pump to the nearest building was 15 meters, and the longest refueling time for a large semi-detached trailer truck was 0.83 hour (50 minutes). With these parameters, the collective dose to the people sharing the stop would be 2.3×10^{-4} person-Sv (Table II-8). The relationship between the collective dose and the number of receptors is not linear in this case. If there are more people sharing the stop, the analysis should be repeated using RADTRAN.

The collective dose to residents near the stop is calculated in the same way as for rail transportation, using data in Table II-8, the population density of the region around the stop, and the stop time.

(II-5) Dose (person-Sv) = (Population density)*(Dose/hr to resident near stop)*(Stop time)

Thus, for a rural population density of 15.1 persons/km^2 (the average along the Maine Yankee-to-Hanford route)

Dose = (15.1 persons/km^2)*(3.3 x 10^{-6} Sv-km^2/hour)*(0.83 hours) = 4.1 x 10^{-5} person-Sv.

The population density used in the calculation is the density around the truck stop; appropriate residential shielding factors are used in the calculation. Unlike a train, the truck will stop several times on any truck route to fill the fuel tanks. Very large trucks generally carry two 80-gallon tanks each and stop for fuel when the tanks are half empty. A semi carrying a Truck-DU cask can travel an average of 845 km (DOE, 2002) before needing to refuel. The number of refueling (and rest) stops depends on the length of each type of route segment. The following equations are used in this calculation

(II-6) Route segment length (km)/(845 km/stop) = stops/route segment

(II-7) Dose (person-Sv) = (population/km^2)*(dose to resident near stop(Sv-km^2/hr))* (stops/route segment)*(hours/stop)

Table II-25 shows the collective doses to residents near stops for the rural and suburban segments of the 16 routes in Table II.4.1. Trucks carrying Truck-DU casks of spent fuel are unlikely to stop in urban areas.

Table II-25. Collective doses to residents near truck stops

Origin	Route	Type	Persons/km²	Average number of stops	Person-Sv
Maine Yankee	ORNL	Rural	19.9	1.73	1.1E-06
		Suburban	395	2.09	2.3E-05
	Deaf Smith	Rural	18.6	2.47	1.5E-06
		Suburban	371	1.6	1.7E-05
	Hanford	Rural	15.4	4.33	2.2E-06
		Suburban	325	1.5	1.4E-05
	Skull Valley	Rural	16.9	3.5	1.9E-06
		Suburban	332.5	1.3	1.2E-05
Kewaunee	ORNL	Rural	19.8	0.81	5.2E-07
		Suburban	361 360.0	0.59	0.0E+00
	Deaf Smith	Rural	13.5	2.0	6.0E-06
		Suburban	339	0.52	8.6E-07
	Hanford	Rural	10.5	3.4	5.0E-06
		Suburban	316	0.60	1.2E-06
	Skull Valley	Rural	12.5	2.6	5.4E-06
		Suburban	324.5	0.44	1.1E-06
Indian Point	ORNL	Rural	20.5	0.71	4.1E-06
		Suburban	388	0.71	4.7E-07
	Deaf Smith	Rural	17.1	2.3	7.8E-06
		Suburban	370	1.2	1.3E-06
	Hanford	Of stops	13.0	4.1	1.3E-06
		Suburban	338	1.1	1.8E-06
	Skull Valley	Rural	14.2	3.3	1.1E-05
		Suburban	351	0.93	1.5E-06
INL	ORNL	Rural	12.4	3.1	9.3E-06
		Suburban	304	0.72	1.3E-06
	Deaf Smith	Rural	7.8	2.3	6.3E-06
		Suburban	339	0.35	5.8E-07
	Hanford	Rural	6.5	0.43	3.4E-06
		Suburban	200	0.57	9.0E-08
	Skull Valley	Rural	10.1	0.42	3.2E-06
		Suburban	343	0.11	1.4E-07

[a]The number of stops is the kilometers of the route segment divided by 845 km, the distance between stops, so that it may be a fraction. Retaining the fraction allows the calculation to be repeated.

The rural and suburban population densities in Table II-25 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. The route from Indian Point to ORNL illustrates another situation. This route is 1028 km long, and would thus include one truck stop, which could be in either a rural or a suburban area.

II.5.4.4 Occupational Doses

Occupational doses from routine, incident-free radioactive materials transportation include doses to truck and train crew, railyard workers, inspectors, and escorts. Workers who handle spent fuel containers in storage, loading and unloading casks from vehicles or during intermodal transfer are not addressed in this analysis. Truck refueling stops in the U.S. no longer have attendants who refuel trucks.[3] Gas station and truck stop workers are in concrete or brick buildings and would be shielded from the radiation with the same shielding as in urban housing (83% shielded).

Table II-26 summarizes the occupational doses.

Table II-26. Occupational doses per shipment from routine incident-free transportation

Cask and route type	Train crew: 3 people; person-Sv	Truck crew 2 people; person-Sv	Escort: Sv/hour	Inspector: Sv/hour per inspection	Truck stop worker: Sv per shipment	Rail classification yard workers: person-Sv
Rail-Pb rural/suburban	5.4E-09		5.8 E-06			1.5E-05
Rail-Pb urban	9.1E-08		5.8 E-06			
Rail-All Steel rural/suburban	4.1E-09		4.4 E-06			1.1E-05
Rail-All Steel urban	6.8E-09		4.4 E-06			
TRUCK - DU rural/suburban		3.8E-09	3.2E-09	3.2E-09	2.0E-09	
TRUCK - DU urban		3.6E-09	3.2E-09			

II.6 Interpretation of Collective Dose

Collective dose is essentially the product of an average radiation dose and the number of people who receive that average dose. Together with the linear non-threshold theory (BEIR VII, 2006, p.16), collective dose provides a method to estimate the number of "health effects," cancer in particular, that will occur in a group of people. The following example – a state suburban segment on a particular route – is typical of all routes in all states; only the specific numbers change.

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[3] The State of Oregon still requires gas station attendants to refuel cars and light duty vehicles, but heavy truck crew do their own refueling.

The following parameters characterize a particular segment of the Maine Yankee-to-Hanford truck route: the suburban segment through Illinois:

- Route segment length: 73 km
- Suburban population density: 324 persons/km^2
- Area occupied by that population: 0.800 km x 2 x 73 = 116.8 km^2
- Total suburban population exposed to the shipment = 37,800 people
- From Table II-13, the collective radiation dose to that population, from routine, incident-free transportation, is 1.0×10^{-5} person-Sv.
- U.S. background is 0.0036 Sv per year or 4.1×10^{-7} Sv per hour. At an average speed of 108 kph, the population is exposed for 0.675 hour.

The background dose sustained by each member of this population is 2.8×10^{-7} Sv for a total collective dose of 0.11 person-Sv. The total collective dose is thus 0.11001 person-Sv with the shipment , and 0.11 person-Sv without the shipment. Estimates of the collective radiation risk from shipments of spent fuel are only valid when compared to the collective risk to the particular population when there is no shipment.