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
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COMMENTS ON DRAFT REPORT

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For any questions about the material in this report, please contact:

Mr. John Cook
Mail Stop E3-D2M
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Phone: 301-492-3318
Fax: 301-492-xxxx
E-mail: **John.Cook@nrc.gov**

ABSTRACT

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

- No realistic accident will lead to release of radioactive material for transportation in a rail cask with an inner welded canister.
- No realistic accident will lead to release of radioactive material for truck transportation in a truck cask.
- None of the extreme fire scenarios studied in this report led to a release from any of the casks studied.
- For a rail cask without an inner canister an accident that leads to any release of radioactive material occurs less than once in 25 million shipments.
- In the worst-case accident, the maximum individual dose is less than two sieverts.

These results strongly demonstrate that the regulations of the NRC ensure the safety of the transportation of spent nuclear fuel.

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ACRONYMS AND ABBREVIATIONS

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

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

CHEMICAL SYMBOLS

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

PUBLIC SUMMARY

Nuclear fission produces a large amount of energy which has been harnessed for the production of electricity. Fission also creates radioactive products which are contained in nuclear fuel. Therefore, 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 United States Nuclear Regulatory Commission (NRC) responded to this concern by estimating what the radiological 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 (EIS) for transportation of all types of radioactive material by road, rail, air, and water, and concluded:

- The average radiation dose to members of the public from routine transportation of radioactive materials is a fraction of the background dose[1].

- 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, NRC regulations in 1981 were considered "adequate to protect the public against unreasonable risk from the transport of radioactive materials." However, the adequacy of these regulations continued to be questioned in part because the EIS was based on estimates of radiation dose and accident rates, for which not much data or information had been available. Questions about "reasonable" risk and about accident consequences ("what if the accident does happen?") have also been raised. The present work uses advanced models, risk assessment methods, and updated data to provide a current assessment of the risks and consequences of transporting spent nuclear fuel.

All commodities that are transported by truck or rail can be involved in accidents. Trucks and railcars carrying spent nuclear fuel transportation casks are no exception. The NRC recognizes this, and requires that spent fuel casks be designed and built to withstand severe transportation accidents. NUREG 0170 and later studies of casks have considered accident conditions more severe than the regulations require for cask certification. A 1987 study applied actual accident statistics to projected spent fuel transportation (Fischer et al., 1987). 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 the limited number of accidents that had occurred involving spent fuel shipments was not sufficient to support projections or predictions. The Modal Study's refinement of modeling techniques and use of accident frequency data resulted in smaller assessed risks than had been projected by NUREG 0170.

A 2000 study of two generic truck casks and two generic rail casks analyzed the cask structures and response to accidents using computer modeling techniques (Sprung et al., 2000). Semi-trailer

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[1] The background dose is the average dose any individual will receive over the period of a year while conducting routine, everyday activities (3.6 millisieverts)

truck and rail accident statistics for general freight shipments were used because even by 2000 there had been too few accidents involving spent fuel shipments to provide statistically valid data.

The release of radioactive material from a cask in an accident and its subsequent dispersion has also been modeled with increasing refinement in the series of risk assessments. NUREG-0170 assumed that most very severe accidents would result in release of all of the releasable cask contents to the environment; this engineering judgment overstated the release but was nevertheless used because analytical capabilities at the time did not permit a more accurate assessment. 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 (NRC, 1977). 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 where the lid is attached. In other words, an accident could cause seal failure, but would not breach the cask body (Sprung et al., 2000).

The present study models real casks (rather than generic casks) and the commercial spent nuclear fuel that these casks are certified to transport. Two rail casks and a truck cask are evaluated.

Almost all spent fuel casks are shipped without incident. However, even this routine, incident-free transportation causes radiation exposures because all loaded spent fuel casks emit some external radiation. The radiation dose rates for spent fuel shipments are measured before each shipment and must be maintained within regulatory limits. 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.

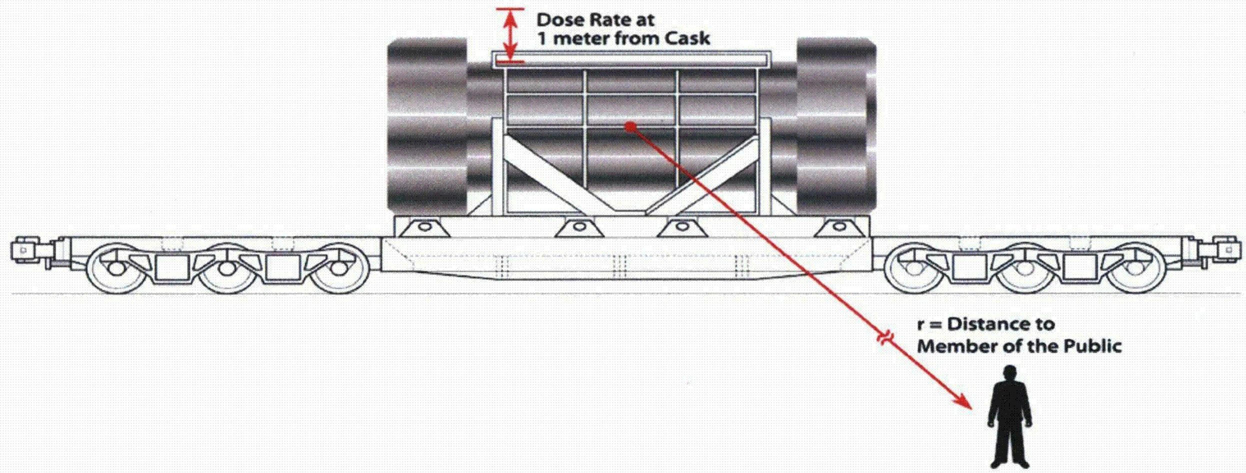

Figure PS-1. Model of a spent fuel cask in routine, incident-free transportation and radiation dose to a member of the public. Relative sizes of the cask and receptor are approximately to scale.

The external radiation from the spent fuel cask results in a very small dose to each member of the public 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 and Figure PS-2 show the total dose in person-sieverts (person-Sv) to all of the exposed workers and members of the public for one of these routes, the truck shipment from the Maine Yankee Nuclear Power Plant to Oak Ridge National Laboratory (ORNL). The background radiation dose to exposed workers and members of the public during the time of the shipment is included in Table PS-1 and Figure PS-2.

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

Exposed Population	Rural	Suburban	Urban	Urban Rush Hour	Total
Residents near route	5.0×10^{-6}	8.9×10^{-5}	2.0×10^{-6}	4.5×10^{-8}	9.6×10^{-5}
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.

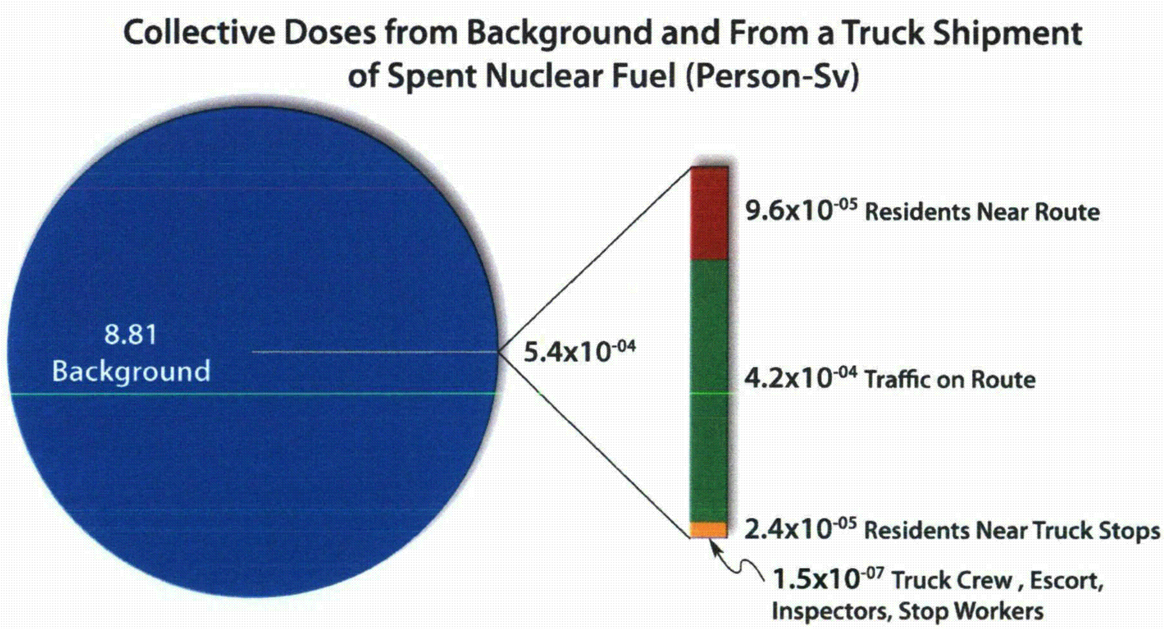

Figure PS-2. Collective doses from background and from a truck shipment of spent nuclear fuel (person-Sv).

The collective doses calculated for routine transportation are approximately the same for NUREG/CR-6672 and for this study, and are about 40 percent of the doses reported in NUREG

0170 (Sprung et al., 2000). Figure PS-3 shows a comparison of the collective doses from truck transportation from the three studies. In NUREG 0170, the analysis was for a single route, in NUREG/CR-6672, the analysis was for 200 representative routes, and in this study the analysis is for 16 actual routes (Sprung et al., 2000). The collective average dose in the present study is larger than the NUREG/CR-6672 result because present populations are generally larger, particularly along rural routes, and the vehicle densities are much larger (see Chapter 2). These increases were somewhat offset by the greater vehicle speeds used in the present study.

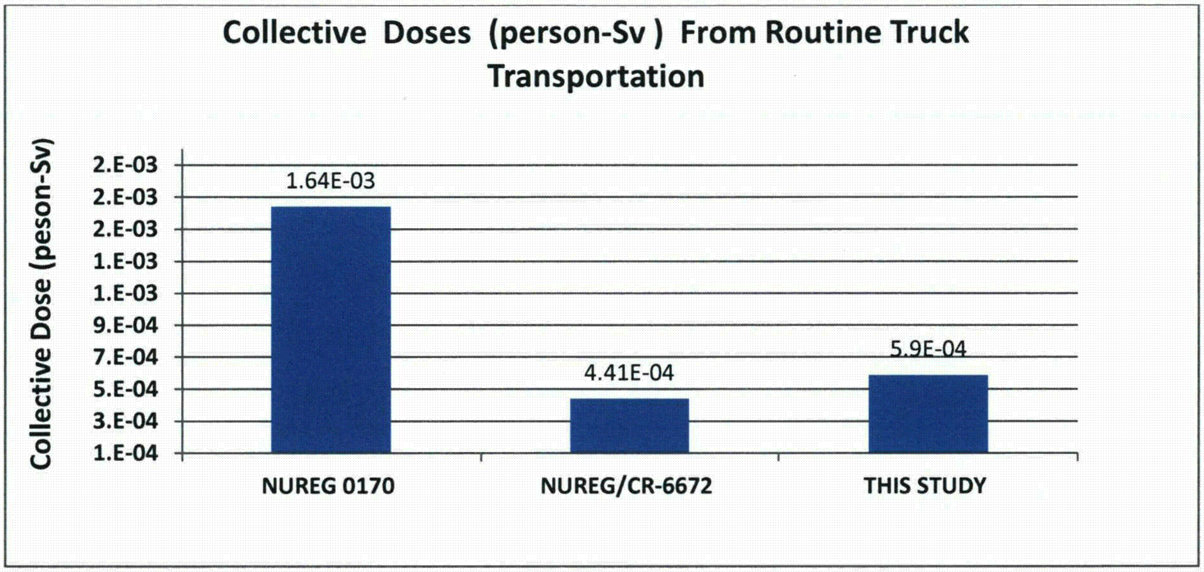

Figure PS-3. Collective doses (person-Sv) from routine truck transportation.

This 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 casks would respond to the accident scenarios. Figure PS-4 shows one impact scenario, a 97 kilometer per hour (kph), or 60 mile per hour (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 kph—equal to 30, 60, 90, and 120 mph—in end-on (lid down), corner, and side-on orientations for two cask designs. These impact speeds encompass all accidents for truck and rail transportation.

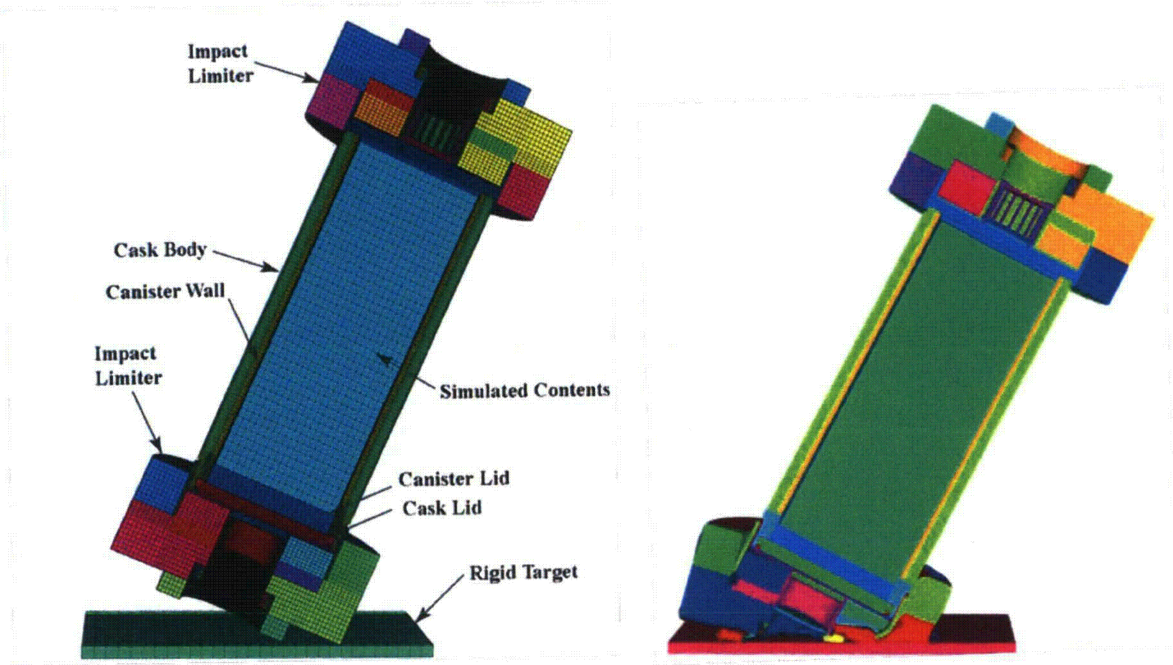

Figure PS-4. Corner impact onto a rigid target at 97 kph (60 mph) accident scenario for a spent fuel cask and the deformations produced by the impact.

Figure PS-5 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 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-5. Engulfing fire scenario and the temperature contours in the rail cask following a three-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 no accident involving the truck transportation cask would result in release of radioactive material or reduction in the effectiveness of the gamma shielding. The only radiological consequence of an accident would be exposure to external radiation from the cask because of the long duration stop associated with the accident. The stop needs to be long enough for responders to clear the accident scene and to arrange for shipment resumption. 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 would be a small fraction of the allowed occupational dose.

For rail transport of spent fuel that is in an inner welded canister, this study shows that there would be no release of radioactive material. For casks using lead gamma shielding, the most severe accidents evaluated led to reduction in the effectiveness of that shielding, which resulting in an elevated external radiation level. In addition, for rail transport of spent fuel that is not in an inner welded canister, some radioactive material is released following exceptionally severe and improbable accidents.

The calculated collective dose risk from accidents has decreased with each successive risk assessment. Figure PS-6 shows a comparison of average collective doses from releases and loss of lead shielding from the three studies (NUREG 0170 did not calculate loss of lead shielding (LOS)). This study also considered accident doses from a source that was not analyzed in the prior studies, the dose that results from accidents in which there is neither release nor loss of lead shielding, but there is increased exposure to a cask that is stopped for an extended period of time.

Average collective doses for this scenario for the three casks studies are shown in Figure PS-7. This scenario is important because more than 99.999 percent of all accident scenarios do not lead to either release of radioactive material or loss of shielding.

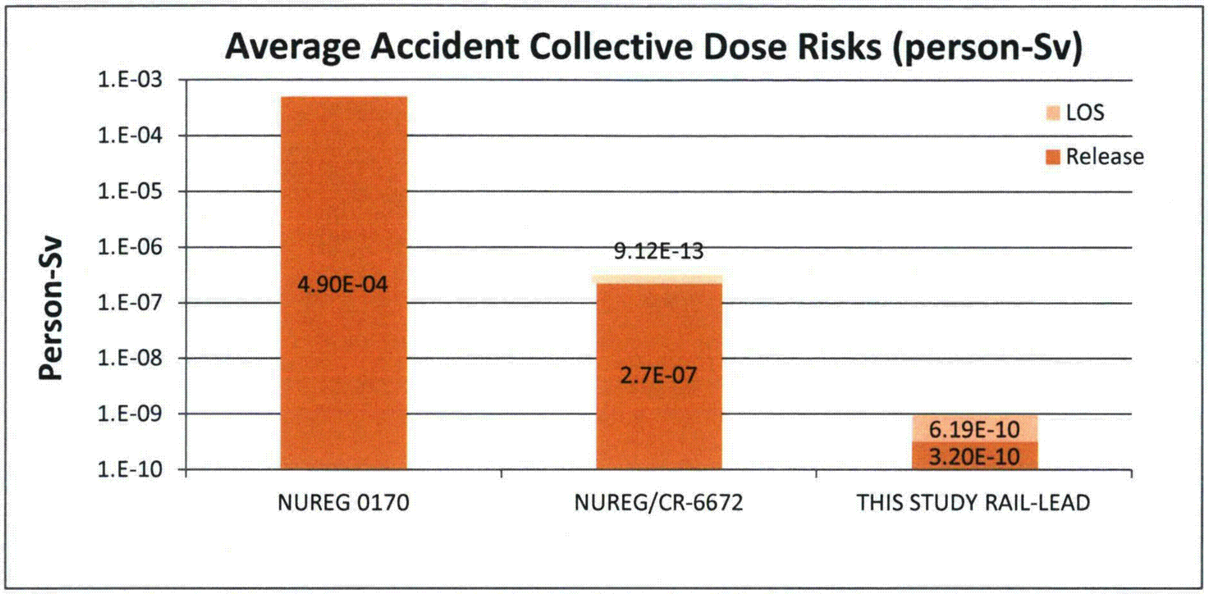

Figure PS-6. Accident collective dose risks from release and LOS accidents. The LOS bar representing the NUREG/CR-6672 collective dose is not to scale.

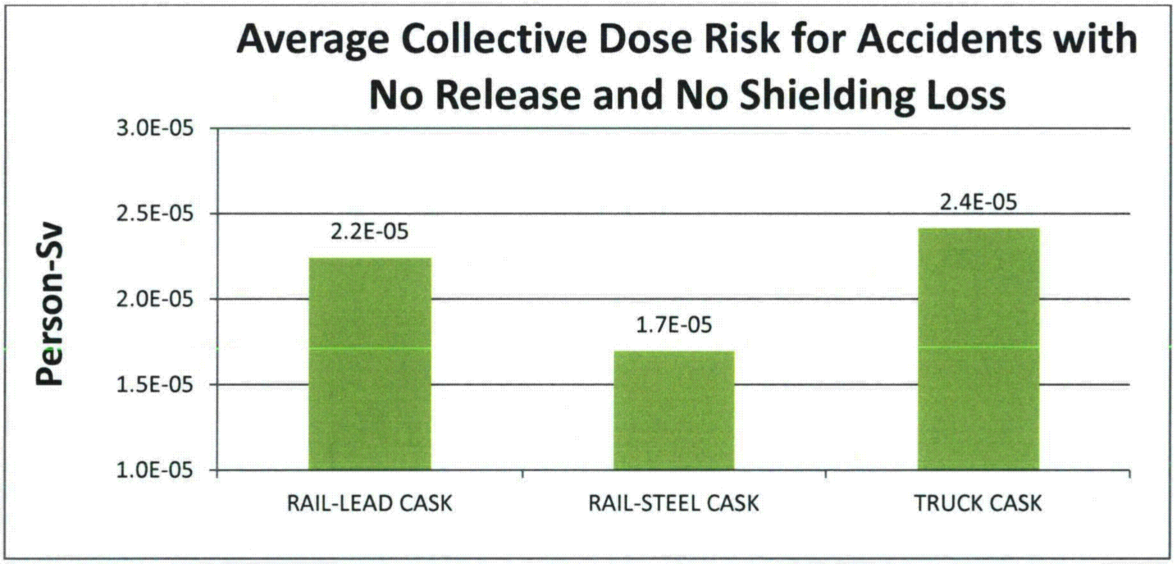

Figure PS-7. Average collective dose from accidents that have no impact on the cargo.

A final 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 5×10^{-7} person-Sv, or about 2 Sv to the maximally exposed individual.

Not only is the estimated risk of spent fuel transportation exceedingly small, but the estimated maximum consequence is also very small.

As noted above, the purpose of this analysis was to reproduce (and, in some cases, extend) risk analyses previously considered in NUREG 0170, the Modal Study, and NUREG/CR-6672, using updated models and methods. The following conclusions are reached in the completed analysis.

- The 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 collective dose risks from routine transportation are vanishingly small. Theses doses are about four to five orders of magnitude less than collective background dose.
- Radioactive material would not be released in an accident if the fuel is in a canister in the cask.
- Only rail casks without inner welded canisters would release radioactive material in extremely severe accidents; 99.999 percent of potential accidents would not result in a release of radioactive material.
- The collective dose risks for the two types of extra-regulatory accidents, accidents involving a release of radioactive material and loss of lead shielding accidents, are negligible compared to the risk from a no-release, no-loss of shielding accident.
- The risk of either a release or loss of shielding from a fire is negligible.
- The maximum consequence (dose) to an individual from the most severe accident is 1.6 Sv. The maximum collective consequence (dose) on the routes studied is 0.28 person-Sv.
- These results are in agreement with previous studies

The analyses and results described in this report confirm and extend the assurance provided by NUREG 0170 that spent fuel shipments can be completed safely, and NRC regulations governing transportation of spent nuclear fuel are effective.

CHAPTER 1

INTRODUCTION

1.1 Organization of this Report

The body of the report consists of a public summary and six chapters. The chapters describe the risk analysis qualitatively. Each chapter in this study except for Chapter 6, Observations and Conclusions, has an associated appendix that describes the analytical methods and calculations used to arrive at the results discussed in the chapters. Descriptions of programs, calculations, and codes used are in the relevant appendices.

1.1.1 Chapter 1 and Appendix I

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

1.1.2 Chapter 2 and Appendix II

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

1.1.3 Chapter 3 and Appendix III

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

1.1.4 Chapter 4 and Appendix IV

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

1.1.5 Chapter 5 and Appendix V

Chapter 5 and Appendix V address RADTRAN analysis of transportation accidents, development of accident event trees and conditional probabilities, development of the radionuclide inventory and radioactive materials releases and dispersion of released material in the environment. The chapter also discusses accidents in which there are no releases – the most likely accidents – in which the radioactive cargo is not affected at all, but the vehicle can sit for many hours at the accident location.

1.1.6 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.

1.1.7 Bibliography

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

1.2 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 primarily analyzes cask behavior rather than the behavior of the spent fuel being transported. It is the latest in a series of assessments of this type and analyzes the behavior of casks certified by NRC carrying fuel of known isotopic composition and burnup. The studies that preceded by this one were based on conservative and generic assumptions.

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

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

- The average radiation dose to members of the public from routine transportation of radioactive materials is a fraction of the 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 of the environmental impact statement (1981) were "adequate to protect the public against unreasonable risk from the transport of radioactive materials" (46 FR 21629, April 13, 1981).

The risk assessment of NUREG-0170 was based on very conservative estimates of risk parameters and on 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 implementing this policy, two comprehensive contractor reports dealing with spent fuel transportation have been issued since 1977: the Modal Study (Fischer et al., 1987) and NUREG/CR-6672 (Sprung et al., 2000).[2] Both were advances in transportation risk assessment. The Modal Study was the first intensive examination of vehicle accident statistics, and the first to categorize the frequency of severe accidents by structural and thermal response of the cask. The Modal Study concluded that the frequency of accidents severe enough to produce significant cask damage was considerably less than NUREG-0170 had estimated. Although the Modal Study was not a risk analysis, since it did not consider the radiological consequence of accidents, risks less than those estimated in NUREG-0170 could be inferred.

NUREG/CR-6672 built on the Modal Study by refining the mechanical stress/thermal stress combinations and recasting them as a matrix of accident related impact speeds and fire temperatures. In addition, NUREG/CR-6672 developed expressions for the behavior of spent fuel in accidents and potential release of this material, and analyzed the potential releases. The enhanced modeling capabilities available for NUREG/CR-6672 allowed analyses of the detailed structural and thermal response of transportation casks to accidents. NUREG/CR-6672 also used results of experiments by Lorenz (1980), Sandoval, et al. (1988), and Sanders, et al. (1992) to estimate releases of radioactive material from the fuel rods to the cask interior and from the cask interior to the environment following very severe accidents. The radionuclides available for release in the accidents studied in NUREG/CR-6672 are from relatively low burnup (30 gigawatt-days per metric ton uranium, or GWD/MTU) and relatively high burnup (60 GWD/MTU) pressurized water reactor (PWR) and boiling water reactor (BWR) fuel, although the transportability of the high burnup fuel was not considered. [The particular characteristics of high-burnup fuel shown by Einziger (2007) and Einziger and Beyer (2007) were investigated after the publication of NUREG/CR-6672 and therefore were not included in the NUREG/CR-6672 analysis.] NUREG/CR-6672 studied the behavior of two generic truck casks and two generic rail casks—each generic cask encompassed design features of several NRC certified casks.

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

[2] "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.

The present study analyzes the behavior of three currently certified casks carrying Westinghouse 17×17 PWR fuel assemblies with 45,000 megawatt-days per metric ton of uranium (MWD/MTU) burnup, the highest burnup that any of the three casks are currently certified to carry as of 2008. In the future these casks may be certified to carry higher burnup fuel that has been cooled for a longer time, and with a similar source term. For routine transportation, the risks are slightly larger than those estimated in NUREG/CR-6672 because although the actual external dose rates are less than the regulatory maximum used in the other studies, populations along the routes have increased significantly. For accidents, the radiological risks calculated in the current study are at least an order of magnitude less. The reduction in the estimates of risk from those in NUREG-0170 and NUREG/CR-6672 is the result of new data and observations, and improved modeling techniques.

1.3 Risk

Risk assessment provides understanding of events that might happen in the future. Because risks are projections of potential future events, calculations of risk are estimates based on historical data, experimental observations, and analyses using realistic assumptions about future events.

Understanding transportation risk is integral to understanding the environmental and related human health impact of radioactive materials transportation. A large amount of data exists for deaths, injuries, and damage from traffic accidents, but there are no data on health effects caused by radioactive materials transportation because no such effects have been observed. Therefore, both regulators and the public rely on risk estimates to gauge the impact of radioactive materials transportation. The risk estimates project 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 answering the questions posed 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 (probability = 1) of occurrence is an event that is certain to happen. In reality, very few events are certain to happen or certain not to happen (zero probability). The probability of most events is between these two extremes. Transportation accidents involving large trucks, for example, have a very low probability. The probability of a traffic accident for all vehicles is about 0.0001 per mile (or one in 100,000 miles) according to the Department of Transportation Bureau of Transportation Statistics (DOT, 2007), and the probability of a particular traffic accident scenario is much smaller still, as shown in the event trees in Appendix V (Figures V-1 and V-2) of this document.

1.3.1 Accident Data

The only data available to estimate the future probability of a scenario are how often that scenario has occurred in the past. The frequency of the scenario can be considered the same as its probability. In the case of transportation accidents, enough accidents must have occurred that future accidents per kilometer can be predicted with reasonable accuracy. That is, the sample must be large enough to be sampled randomly. The most applicable frequency would be the frequency of accidents involving vehicles carrying spent nuclear fuel, but there have been too few of these for a statistically valid prediction.[3] The sample size could have been increased by using international data, but regulations and practices in other countries are not consistent with those in the U.S. In any case, there have not been enough accidents worldwide involving spent fuel transportation to provide an adequate statistical data base. Even accidents involving all hazardous materials transportation do not provide a large enough data base from which to generate statistics on a state-by-state basis. The database used in this study is the frequency of highway accidents involving large semi-trailer trucks and the frequency of freight rail accidents (DOT, 2007). Freight rail accident frequency is based on accidents per railcar-mile.

1.3.2 Spent Nuclear Fuel Transportation Scenarios

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

- What can happen? The scenario is routine incident-free transportation.
- How likely is it? The probability is 99.999% (see Chapter 5).
- What if it happens? The consequence is a radiation dose less than one percent of background to individuals near the cask or along the route.

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

The accident scenarios discussed in this study are:

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

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

- Accidents involving loss of shielding but no release of radioactive material.
- Accidents in which there is a release of radioactive material.

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

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

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

1.4 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 regulations, under 49 CFR Parts 173 to 178. The regulations of 10 CFR Part 20 are also relevant. NRC transportation regulations apply primarily to the packages being transported, and DOT regulations include labeling, occupational and vehicle standards, registration requirements, and reporting requirements, as well as packaging regulations. In general, the DOT packaging regulations apply to industrial and Type A packaging, and the NRC regulations apply to Type A(F) fissile materials packaging and Type B packaging. Industrial and Type A non-fissile packages are designed to resist the stresses of routine transportation and are not certified to maintain their integrity in accidents, though many do. Type B packages are used to transport very hazardous quantities of radioactive materials. They are designed to maintain their integrity in severe accidents because the NRC recognizes that any transport package and vehicle may be in traffic accidents. This study addresses the transportation of spent nuclear fuel, and thus concerns itself only with Type B packaging. [The cask is the packaging; the cask plus contents is the package.]

Nuclear fuel that has undergone fission in a reactor is both extremely hot and extremely radioactive when it is removed from the reactor. In order to cool the fuel thermally and to allow the very radioactive and short-lived fission products in the fuel to decay, the fuel is discharged from the reactor into a large pool of water. The fuel usually remains in the pool as long as there is space for it. After the fuel has cooled sufficiently, it can be removed from the pool to dry surface storage at the reactor, or it can be transported to a storage site or other destination. In the U.S., fuel is almost never transported before it has cooled for five years. The transportation casks used are rated for heat load, and this rating often determines the cooling time needed for the fuel to be transported.

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[4]. Containers that can meet this requirement are Type B containers and include the casks considered in this analysis, the NAC-STC and Holtec HI-STAR 100 rail casks and the GA-4 legal weight truck casks.

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

1. A 30-foot drop onto an essentially unyielding horizontal surface. "Essentially unyielding" in this context means that the target is hard enough and heavy enough that the package absorbs nearly all of the impact energy, and the target absorbs very little energy. This test condition is more severe than most transportation accidents.
2. A 40-inch drop onto a fixed 6-inch-diameter steel cylinder, to test the package's resistance to punctures.
3. A 1475°F fire that fully engulfs the package for 30 minutes.
4. Immersion under three feet of water. Casks carrying spent fuel are also required to withstand a non-sequential immersion in 670 feet of water for one hour.

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

Figure 1-1 illustrates this sequence of tests.

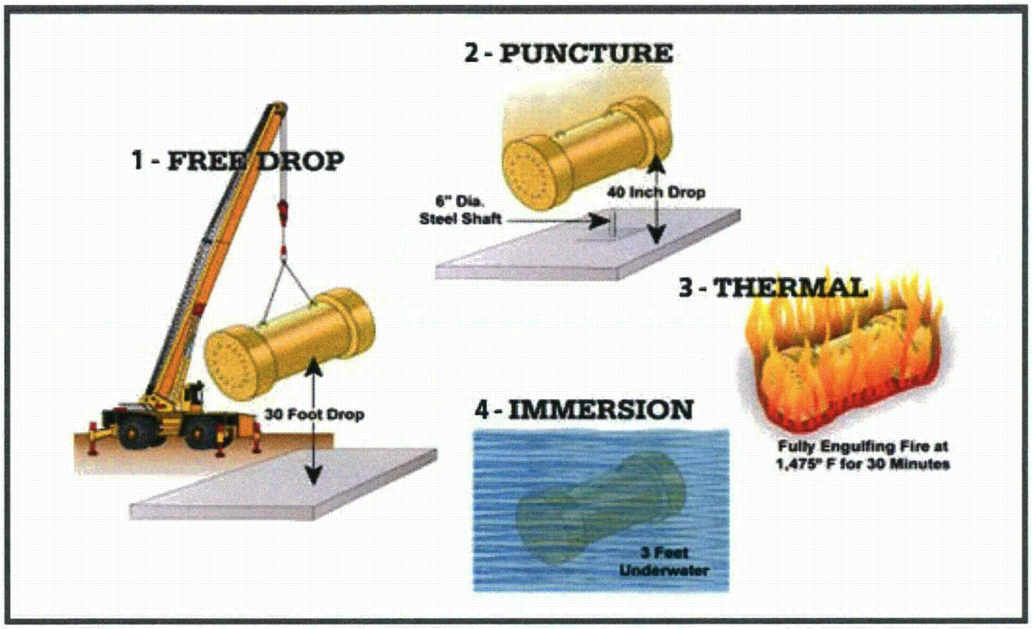

Figure 1-1. The four tests for Type B packages.

The package tests in 10 CFR 71.73 are generally representative of a hypothetical accident. These tests are not intended to represent any specific transportation route, any specific historical transportation accident, or a "worst-case" accident. These tests are intended to simulate the damaging effects of a severe transportation accident in a manner that provides international acceptability, uniformity, and repeatability. This test sequence has been adopted by all International Atomic Energy Agency (IAEA) member states.

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

NRC regulations specify that release of material from the package can be no more than that allowed to be shipped in a non-accident resistant Type A package. The regulation also specifies a maximum post-test external radiation dose rate of 0.01 Sv per hour at one meter (40 inches) from the package surface.

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 regulations. These doses are:

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

Although the regulations state clearly that these dose limits do not include background, background can provide a useful comparison to other sources of radiation exposure, since it affects everyone. The average background radiation dose in the United States is 3.6 mSv per year. This Part also regulates occupational doses to:

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

1.5 Selection of Casks

Past risk assessments for the transportation of spent fuel have used generic cask designs with features similar to real casks, but generally without all of the conservatisms that are part of real cask designs. In this effort, the risk assessment was performed using actual cask designs with all of the features that contribute to their robustness. Because it is too costly and time consuming to examine all casks, a subset of casks was chosen for the risk assessment. Appendix I lists the various spent fuel casks that were certified by the NRC at the time the study began, gives options for the method of choosing the casks to be used, gives some of the important features of the various cask designs, and finally concludes with the 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 were no longer used, but still had valid certificates. Those below the heavy line were 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) was used to evaluate truck shipments, but detailed impact analyses of this cask were not performed because prior analyses of both truck and rail casks have shown that truck casks have significantly lower probability of release of radioactive material in impact accidents (Sprung, et al., 2000). The impact analyses from Sprung et al. were used to assess the response of the GA-4 cask. The complete Certificates of Compliance (COC) for each of these casks (as of April 12, 2010) is included in Appendix I. The NAC-STC cask was chosen because it is certified for transport of spent fuel either with or without an internal welded canister and, for transport or spent fuel without an internal canister, its certificate of compliance allows use of either elastomeric o-rings or metallic o-rings. Even though there were five casks in the group that use lead as their gamma shielding, only the NAC-STC cask of this group can transport fuel that is not contained within an inner welded canister. This ensured, as noted in the analyses of Chapters 3 to 5, that the maximum potential for radioactive material released into the environment was considered. The HI-STAR 100 rail cask was chosen because it was the only all-steel cask in the group that was certified for transport of fuel in an inner welded canister. The GA-4 truck cask was chosen because it has a larger capacity than the NAC-LWT, and therefore was more likely to be used in any large transportation campaign. The chosen casks included all three of the most common shielding options: lead, depleted uranium (DU), and steel.

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

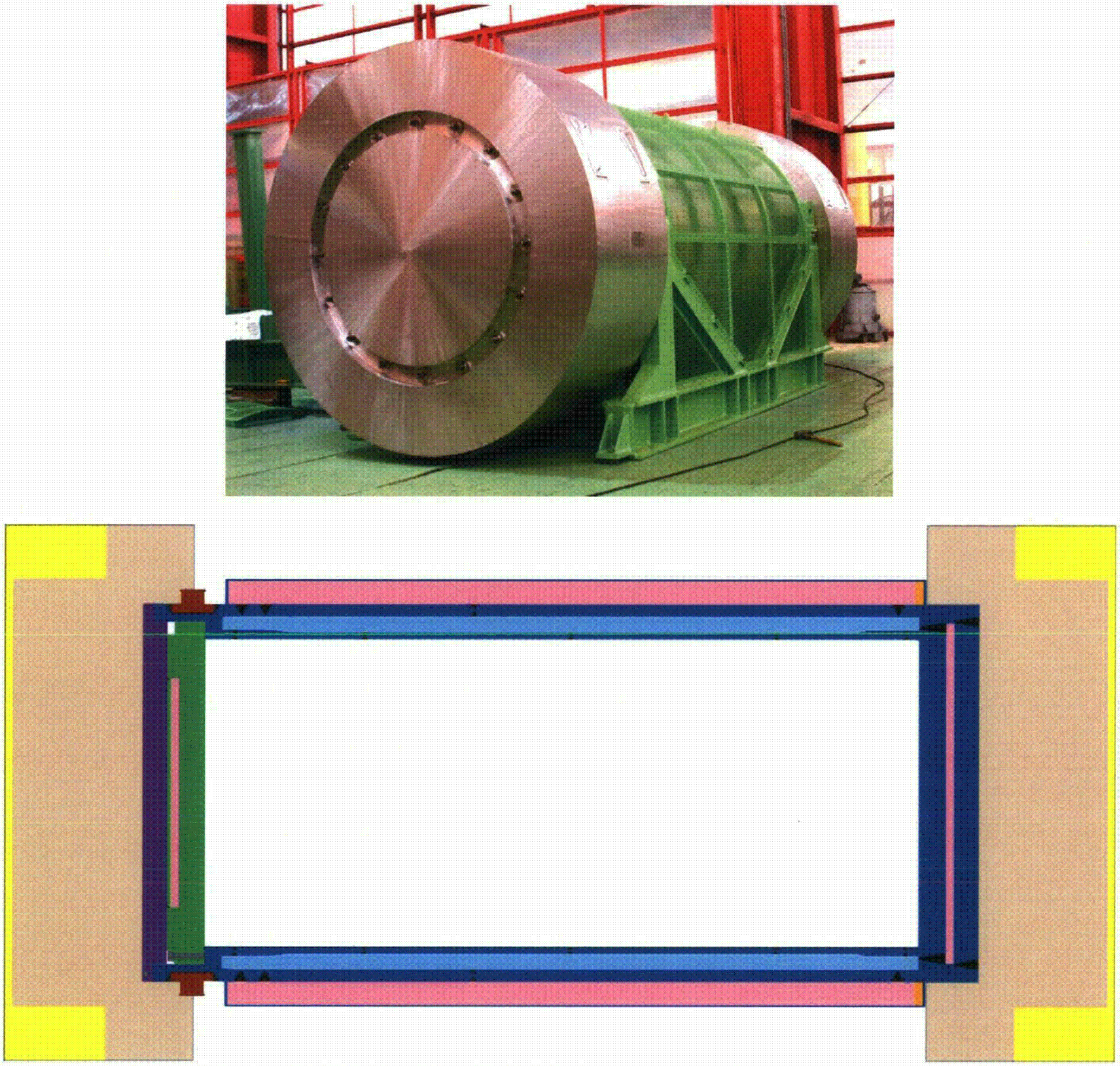

Figure 1-2. Photograph and cross-section of the NAC-STC cask (courtesy of NAC International).

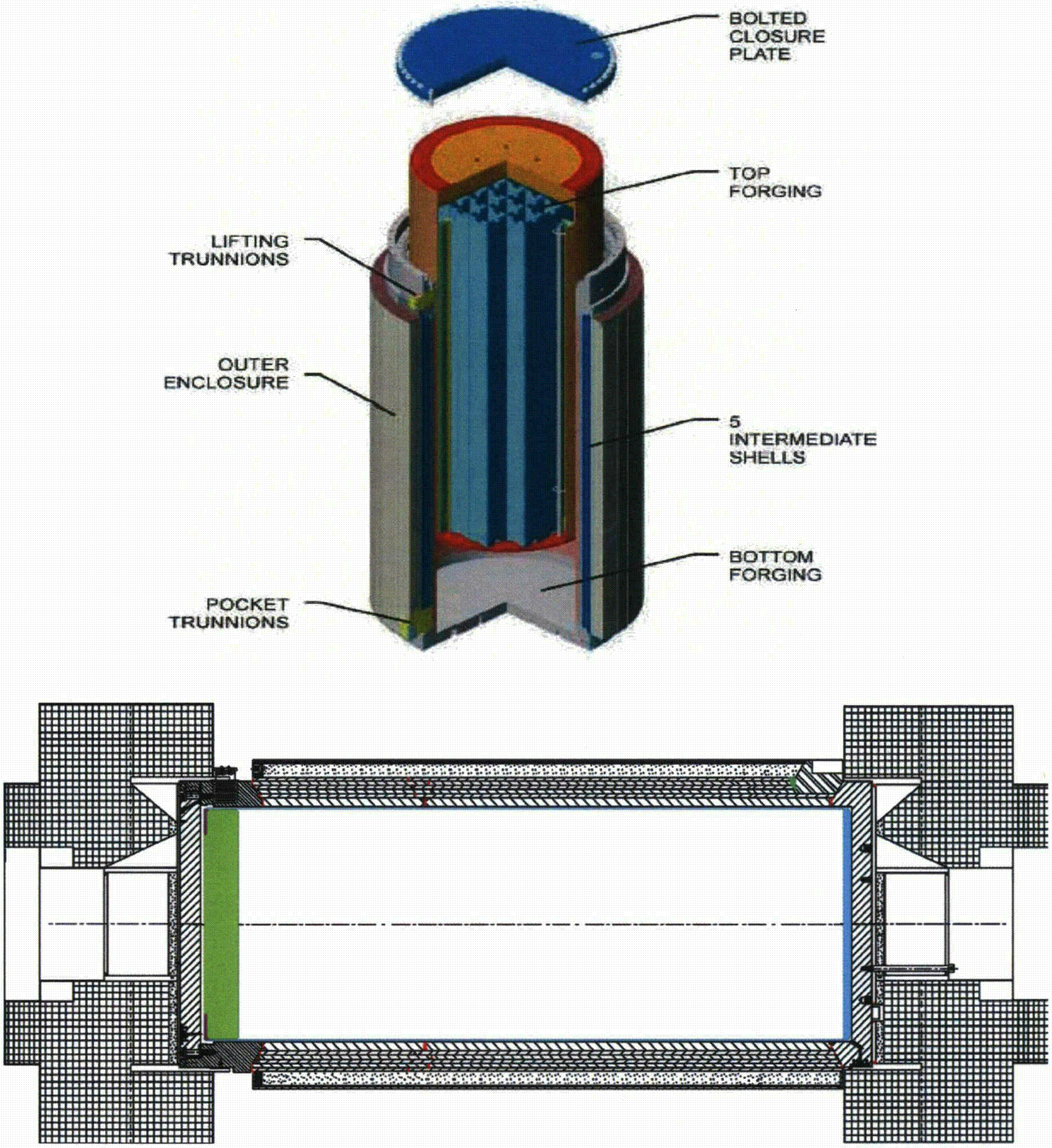

Figure 1-3. Basic layout and cross-section of the HI-STAR 100 rail transport cask (from Haire and Swaney, 2005, and Holtec International, 2004).

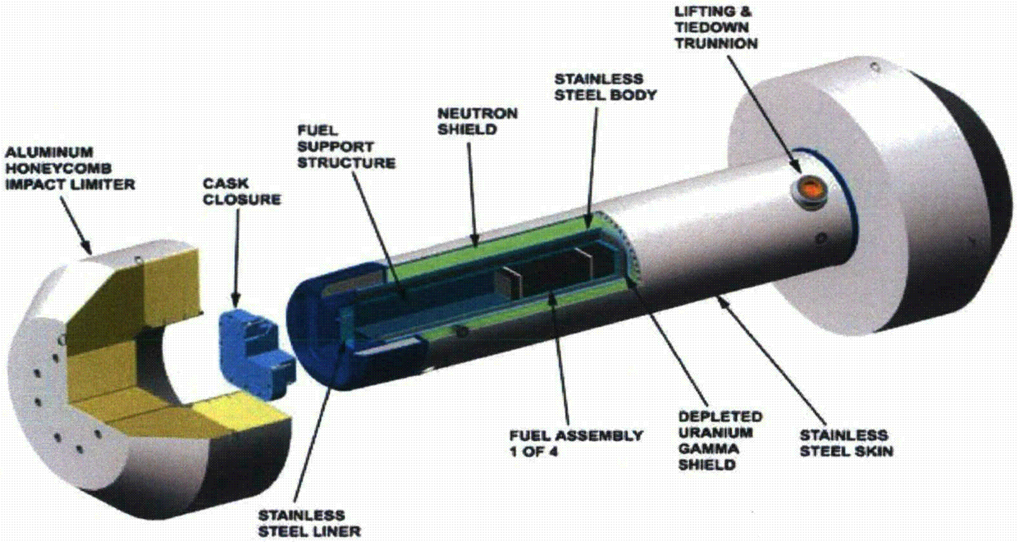

Figure 1-4. GA-4 cask (courtesy of General Atomics).

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

Table 1-2. Casks chosen and reasons for selection

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

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

CHAPTER 2

RISK ANALYSIS OF ROUTINE TRANSPORTATION

2.1 Introduction

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

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

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

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

This chapter includes the following:

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

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

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

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

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

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

2.2 Radiation Emitted during Routine Transportation

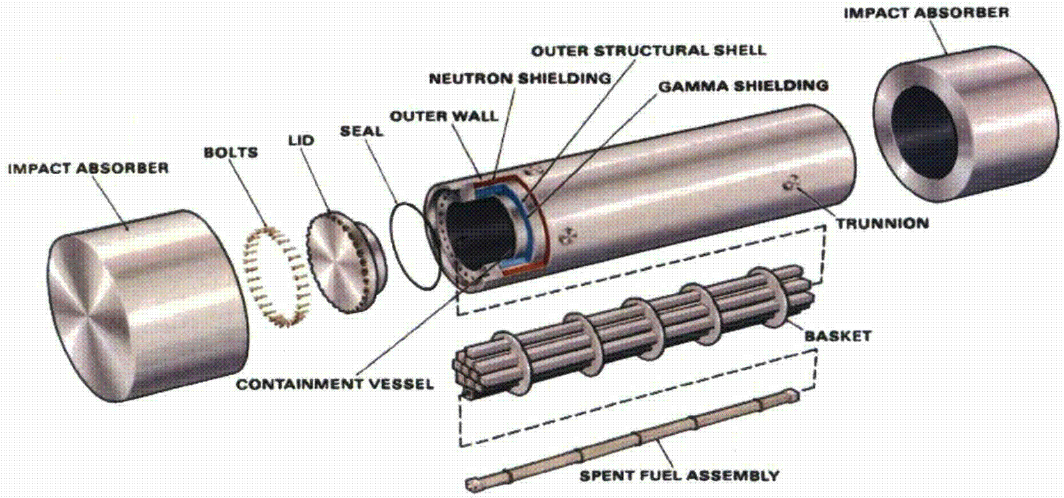
The RADTRAN model for calculating radiation doses is based on the well-understood behavior of ionizing radiation. Like all radiation, ionizing radiation moves in straight lines. It can be absorbed by various materials, including air. Absorption of ionizing radiation depends on the energy and type of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

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),[9] 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 Part 71)), or about 0.00014 Sv/hour (14 mrem per hour) at one meter from a cask four to five meters long.

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



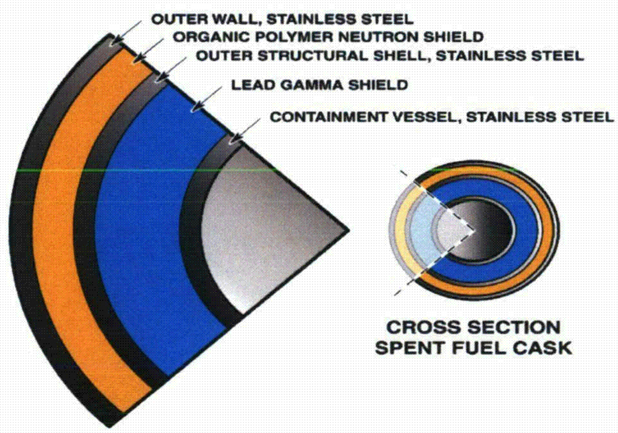


Figure 2-1. The upper figure is an exploded view of a generic spent fuel cask. The lower figure is a cross-section of the layers of the cask wall (Sandia National Laboratories archive).

The external radiation doses from the casks in this study (Figures 1-3 to 1-5), measured at two meters from the cask and reported in the cask Safety Analysis Reports, 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-Lead	Rail-Steel
Transportation mode	Highway	Rail	Rail
Dose rate Sv/hr (mrem/hr) at 1 m	0.00014 (14)	0.00014(14)	0.000103 (10.3)
Gamma fraction	0.77	0.89	0.90
Neutron fraction	0.23	0.11	0.10

The calculated radiation dose to workers and members of the public from a routine shipment is based on the external dose rate at one meter from the spent fuel cask, as shown in Figure 2-2. This dose rate, when expressed in Sv per hour, is called the transport index, or TI. Although the radioactive content of the spent fuel in the cask determines the shielding needed to meet the regulated external dose rate, it does not directly enter into the calculation of the doses from routine transportation. Doses from the external radiation from the cask depend on the external dose rate (which is a function of the contents), the distance of the receptor from the cask, and on the exposure time.

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

2.3.1 The Basic RADTRAN Model

For analysis of routine transportation, RADTRAN models the cask as a sphere with a radiation source at its center, and assumes that the dimensions of the trailer or railcar carrying the cask are the same as the cask dimensions. The emission rate of the radiation source is the dose rate in Sv/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 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.

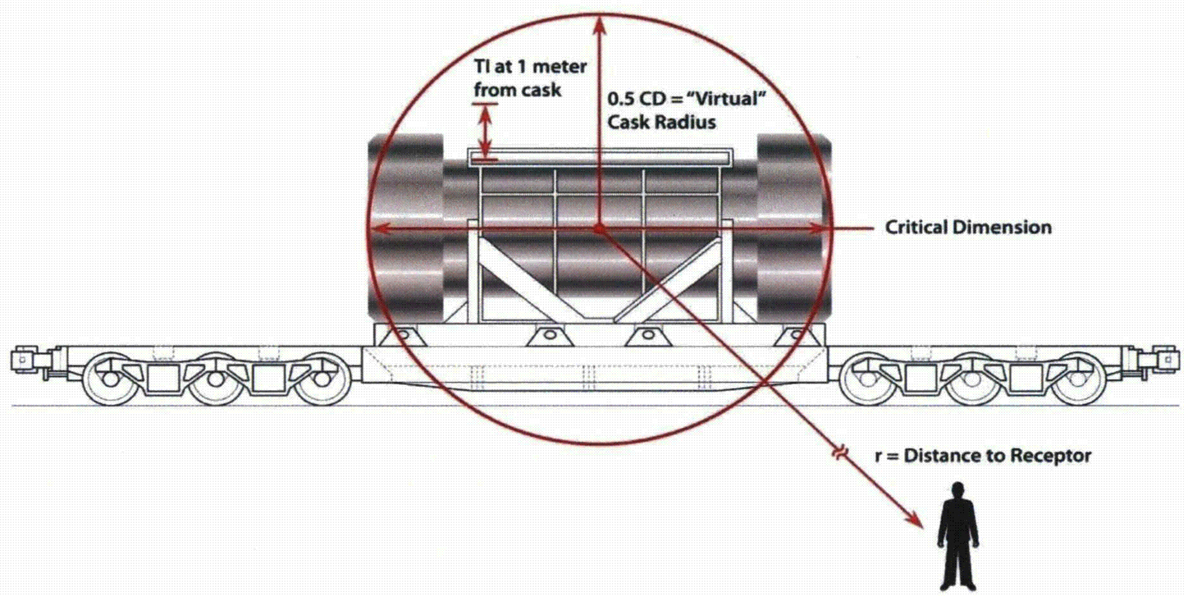


Figure 2-2. RADTRAN model of the vehicle in routine, incident-free transportation. The cask in this diagram is positioned horizontally, and the critical dimension is the cask length.

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, the distance from the cask, and the cask's external radiation. When the vehicle carrying the cask is traveling along the route, the faster the vehicle goes, the less exposure to anyone along the vehicle's route. Therefore, an individual member of the public receives the largest dose from a moving vehicle when he or she is as close as possible to the vehicle, and the vehicle is traveling as slowly as possible. For trucks and trains carrying spent fuel, a speed of 24 km per hour (15 mph) and distance of 30 meters (approximately 100 feet) are assumed for maximum exposure.[10] 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).

Table 2-2. Maximum individual in-transit doses

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

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.[11] RADTRAN calculates collective doses along transportation routes by integrating over the width of a band along the route where the population resides (the *r* in Figure 2-2) and then integrating along the route. Collective doses to people on both sides of the route are included. The exposed population is in a band 770 meters (approximately 0.5 mile) on either side of the route: from 30 meters (10 feet) from the center of the route to 800 meters. Figure 2-3 shows how these bands are defined with examples of distances within the bands.

[10] 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.

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

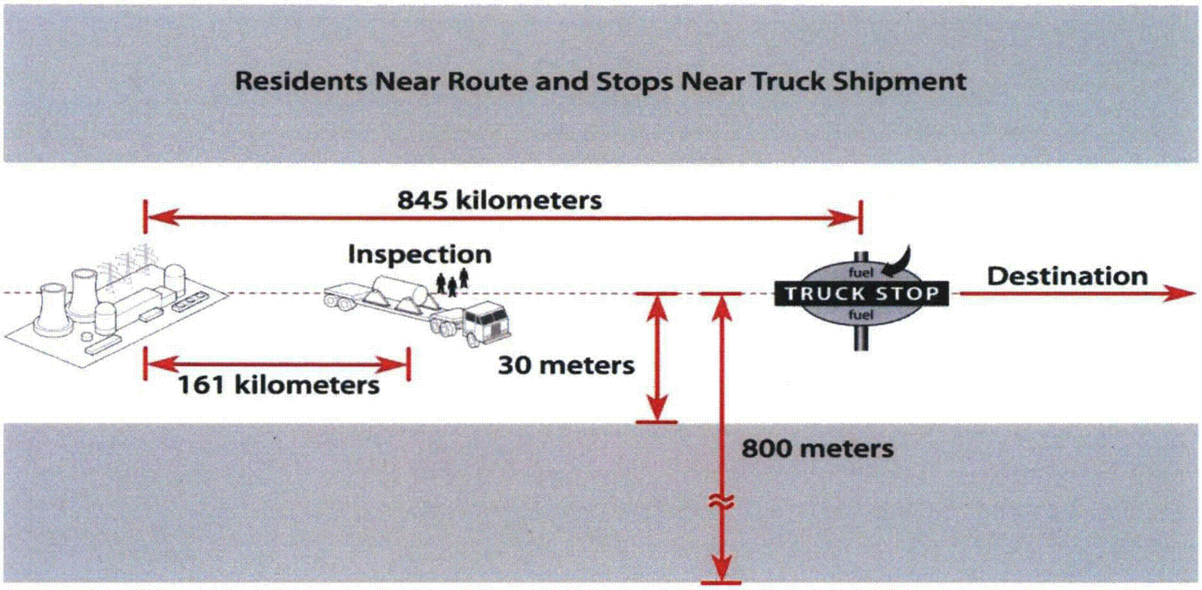


Figure 2-3. Diagram of a truck route as modeled in RADTRAN (not to scale).

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

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

Table 2-2. 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[b,e]	1119	2464	5384	17	17	17
Occupants of other vehicles[b,f]	1.5	1.5	1.5	1	1	5

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

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

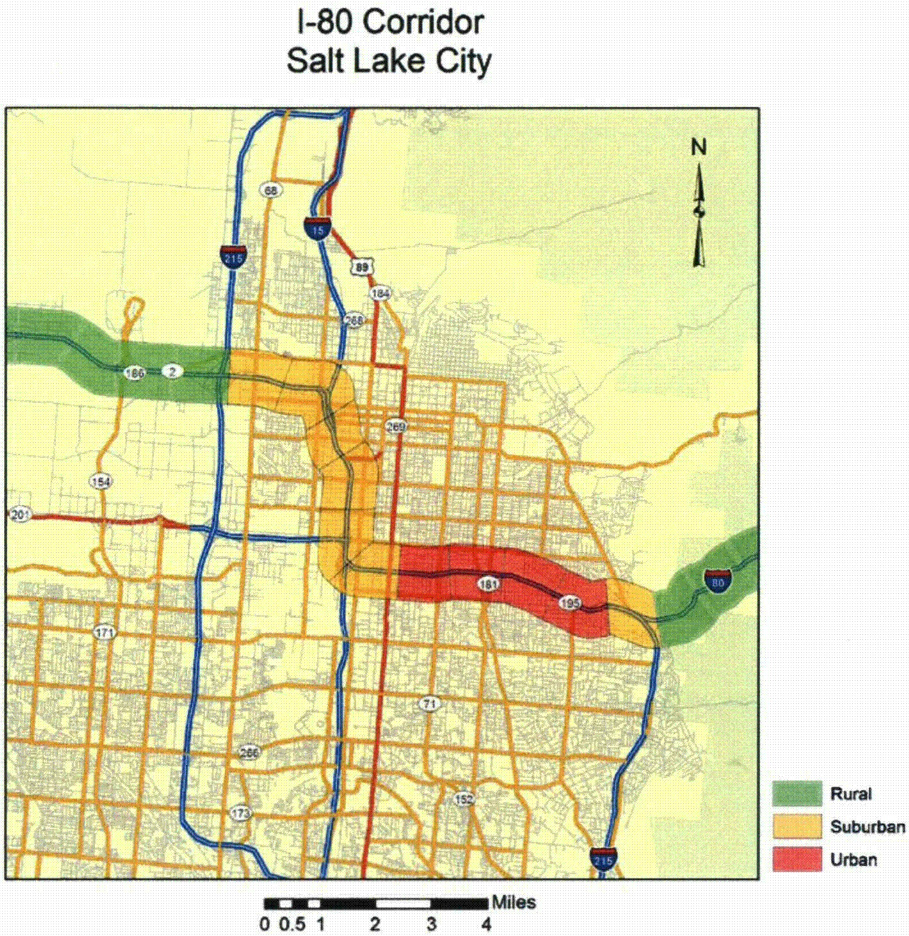


Figure 2-4. A segment of I-80 through Salt Lake City, Utah.

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

Table 2-3. Rail 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 maps of Figures 2-5 through 2-8 show the sixteen truck and sixteen rail routes analyzed in this report. The maps are adapted from the output of the routing code WebTRAGIS (Johnson and Michelhaugh, 2003).

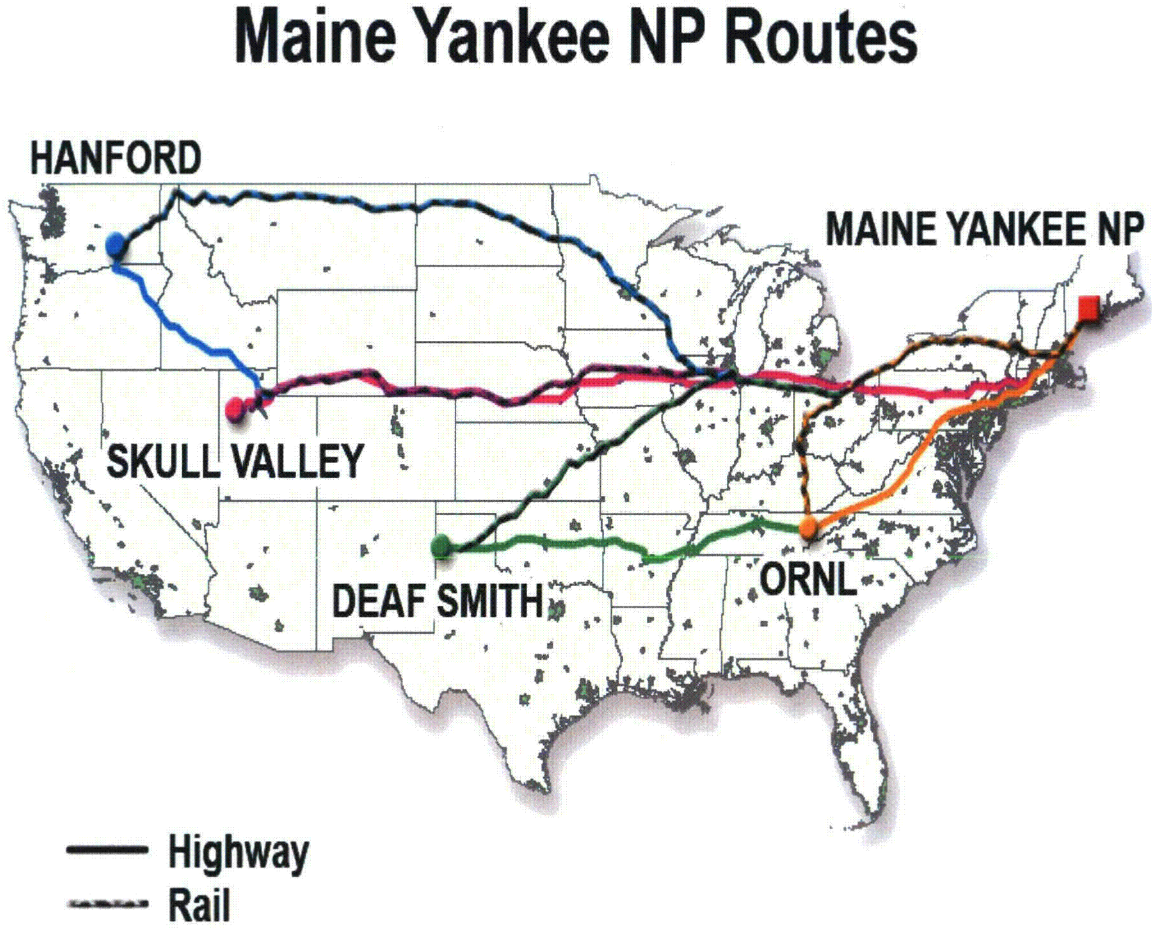


Figure 2-5. Highway and rail routes from Maine Yankee Nuclear Plant site.

(NP stands for Nuclear Plant and ORNL stands for Oak Ridge National Laboratory.)

Kewanee NP Routes

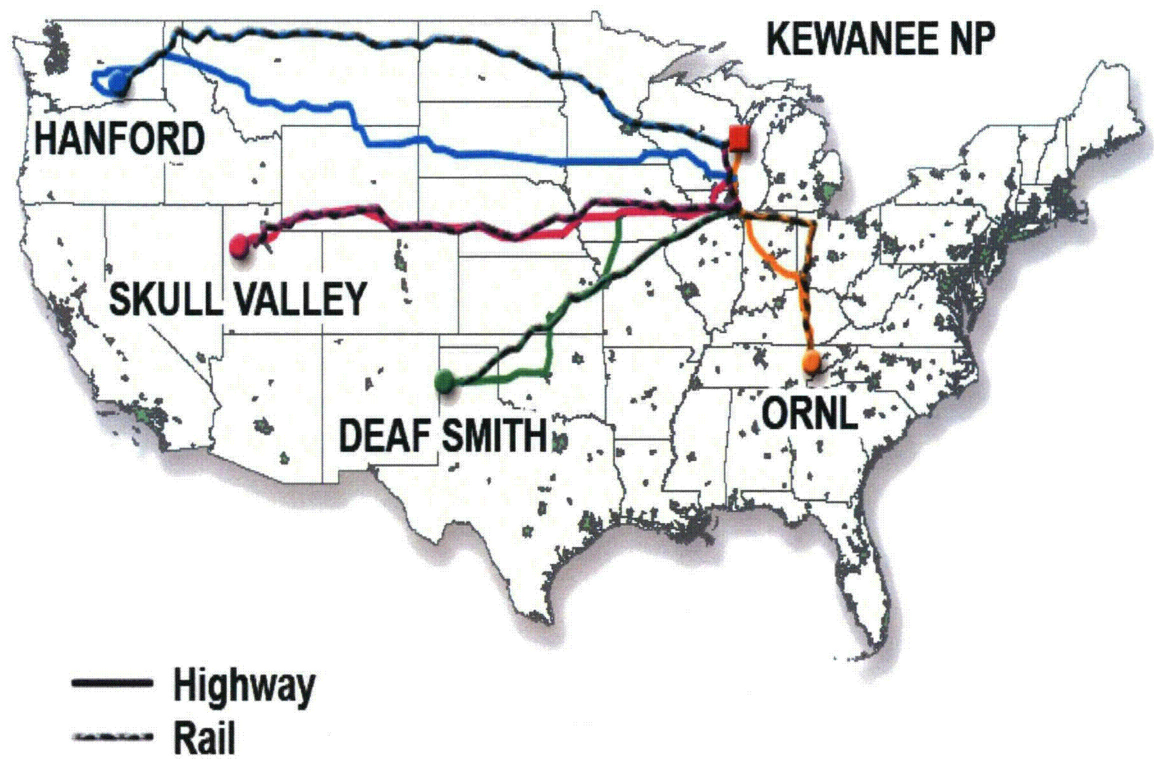


Figure 2-6. Highway and rail routes from Kewaunee Nuclear Plant.

(NP stands for Nuclear Plant and ORNL stands for Oak Ridge National Laboratory.)

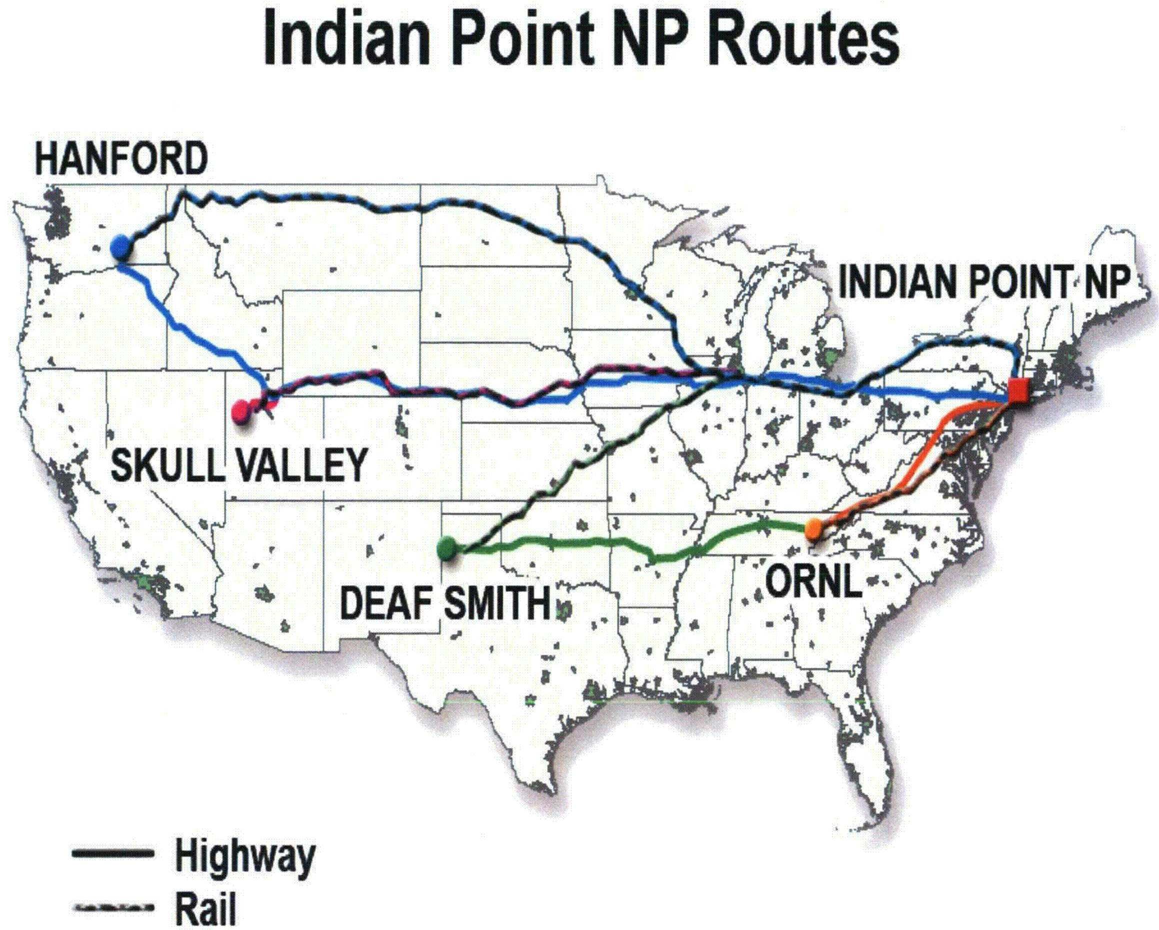


Figure 2-7. Highway and rail routes from Indian Point Nuclear Plant.

(NP stands for Nuclear Plant and ORNL stands for Oak Ridge National Laboratory.)

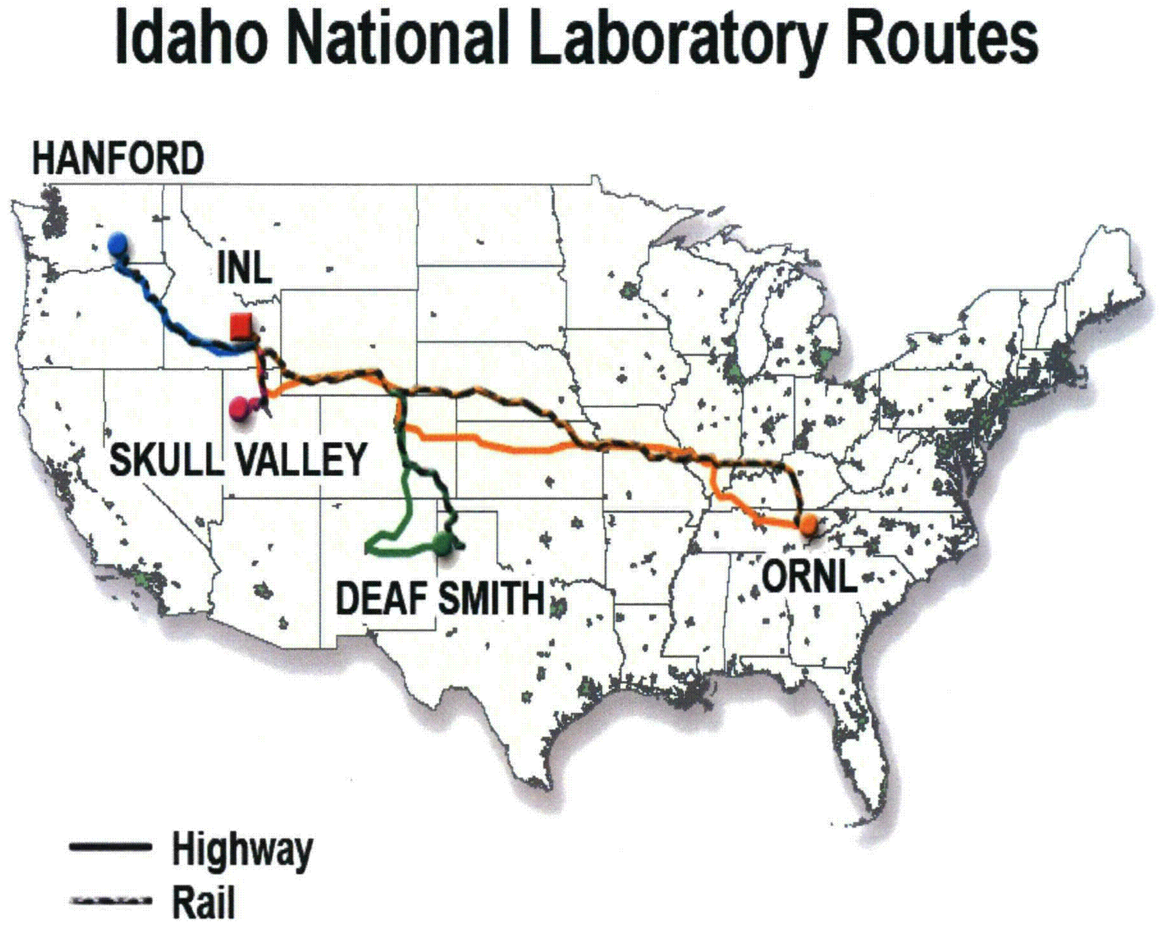


Figure 2-8. Highway and rail routes from Idaho National Laboratory.

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

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

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

Table 2-4. 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

These routes represent a variety of route lengths and populations. The routes include eastern U.S., western U.S., and cross country routes, are of varying lengths, and include a variety of urban areas. Two of the three nuclear plants chosen as origin sites: Kewaunee, WI, and Maine Yankee, ME, and two of the destination sites, Hanford, WA, and Skull Valley, TX, are origins and destinations used in NUREG/CR-6672 (Sprung et al., 2000). Indian Point Nuclear Plant, NY, involves a somewhat different set of cross-country and east coast routes than Maine Yankee, and is an operating nuclear plant, while Maine Yankee has been decommissioned and is now only a surface storage facility. Because this study deals with both commercial and non-commercial spent fuel shipments, INL was included as an origin site. The destination sites include two proposed repository sites (Deaf Smith County, TX, and Hanford, WA) (DOE, 1986), the site of the proposed Private Fuel Storage facility (Skull Valley, UT), and ORNL.

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

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

FROM	TO	Rail-Lead			Rail-Steel		
		Rural	Suburban	Urban	Rural	Suburban	Urban
MAINE YANKEE	ORNL	1.5E-05	1.8E-04	9.0E-06	1.2E-05	1.4E-04	6.8E-06
	DEAF SMITH	1.9E-05	2.5E-04	1.1E-05	1.6E-05	1.9E-04	9.5E-06
	HANFORD	2.4E-05	2.6E-04	1.3E-05	1.8E-05	2.0E-04	9.9E-06
	SKULL VALLEY	2.6E-05	2.7E-04	8.8E-06	2.0E-05	2.0E-04	6.7E-06
KEWAUNEE	ORNL	1.0E-05	1.1E-04	6.7E-06	7.9E-06	8.3E-05	5.1E-06
	DEAF SMITH	8.2E-06	9.5E-05	5.8E-06	6.3E-06	7.2E-05	4.4E-06
	HANFORD	9.9E-06	9.4E-05	3.0E-06	7.6E-06	6.6E-05	2.3E-06
	SKULL VALLEY	1.4E-05	1.2E-04	6.6E-06	1.1E-05	9.0E-05	5.0E-06
INDIAN POINT	ORNL	7.5E-06	2.0E-04	3.6E-05	8.8E-06	1.6E-04	1.6E-05
	DEAF SMITH	1.7E-05	1.4E-04	1.2E-05	1.3E-05	1.3E-04	9.0E-06
	HANFORD	2.2E-05	2.2E-04	1.3E-05	1.7E-05	1.5E-05	5.8E-06
	SKULL VALLEY	2.3E-05	2.1E-04	1.3E-05	1.8E-05	1.6E-04	1.0E-05
IDAHO NATIONAL LAB	ORNL	1.8E-05	1.2E-04	3.7E-06	1.4E-05	9.3E-05	2.8E-06
	DEAF SMITH	6.6E-06	5.6E-05	5.6E-06	4.8E-06	4.2E-05	4.2E-06
	HANFORD	5.3E-06	3.0E-05	1.1E-06	4.0E-06	2.3E-05	8.2E-07
	SKULL VALLEY	3.0E-06	2.5E-05	1.5E-06	2.3E-06	1.9E-05	1.1E-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	5.0E-06	8.9E-05	2.0E-06	4.5E-08
	DEAF SMITH	1.0E-05	1.2E-04	2.1E-06	4.8E-08
	HANFORD	1.4E-05	1.0E-04	1.5E-06	3.2E-08
	SKULL VALLEY	1.1E-05	9.5E-05	1.5E-06	3.3E-08
KEWAUNEE	ORNL	4.1E-06	4.6E-05	1.1E-06	2.5E-08
	DEAF SMITH	6.6E-06	3.9E-05	1.2E-02	1.6E-08
	HANFORD	9.1E-06	4.1E-05	7.0E-07	1.5E-08
	SKULL VALLEY	7.3E-06	3.1E-05	6.7E-07	1.5E-08
INDIAN POINT	ORNL	4.0E-06	6.4E-05	9.8E-07	2.1E-08
	DEAF SMITH	1.0E-05	1.0E-04	1.5E-06	3.3E-08
	HANFORD	1.3E-05	7.6E-05	1.2E-02	2.6E-08
	SKULL VALLEY	1.0E-05	6.6E-05	1.2E-06	2.6E-08
IDAHO NATIONAL LAB	ORNL	8.8E-06	5.3E-05	7.7E-07	1.7E-08
	DEAF SMITH	4.6E-06	3.0E-05	6.9E-07	1.5E-08
	HANFORD	5.5E-06	8.8E-06	1.1E-07	2.5E-09
	SKULL VALLEY	1.2E-06	1.0E-05	2.7E-07	5.9E-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 Table 2-6 and Table 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. In several cases shown in Table 2-5, the rail route had twice as many urban miles as the corresponding truck route.
- 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.
- The collective doses shown in Table 2-6 and Table 2-7 are all very small. However, they are not the only doses the people along the route receive. Background radiation is 0.0036 Sv per year in the U.S., or 4.1×10^{-7} Sv/ hour. The contribution of a single shipment to the population's collective dose is illustrated by the following example of the Maine Yankee to ORNL truck route:

- From Table 2-7 the total collective dose for this segment is 9.6×10^{-5} person-Sv.
- From Table 2-5, there are 1.34 million people within a half mile of the route.
- Background is 4.1×10^{-7} Sv/hour, which everyone is exposed all the time, whether a shipment occurs or not.
- A truck traveling at an average of 108 km per hour 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 100,000 times the collective dose from the shipment.
- The total collective dose during a shipment to these 1.34 million people is not 9.6×10^{-5} person-Sv), but 8.810096 person-Sv.
- The NRC recommends that collective dose be used only for comparative purposes (NRC, 2008).
- The appropriate comparison between the collective dose from this shipment of spent fuel is not a comparison between 9.6×10^{-5} person-Sv from the shipment and zero dose if there is no shipment, but between 8.810096 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 trains are represented by one person in each passing railcar in rural and suburban areas, and five people in urban areas.[12] The rural and suburban collective doses are probably unrealistically large, since most freight rail going through rural and many suburban areas never encounters a passenger train. Data were not available to account for the occupancy of actual passenger trains, including light rail, that share rail routes with freight trains.

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

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

SHIPMENT ORIGIN	SHIPMENT DESTINATION	Rail-Lead			Rail-Steel		
		Rural	Suburban	Urban	Rural	Suburban	Urban
MAINE YANKEE	ORNL	2.8E-05	1.7E-05	2.1E-05	2.1E-05	1.3E-05	1.7E-05
	DEAF SMITH	5.4E-05	1.9E-05	2.7E-05	4.1E-05	1.4E-05	2.2E-05
	HANFORD	8.3E-05	2.6E-05	4.6E-05	6.3E-05	2.0E-05	3.8E-05
	SKULL VALLEY	6.9E-05	2.6E-05	2.3E-05	5.2E-05	2.0E-05	1.9E-05
KEWAUNEE	ORNL	1.9E-05	9.9E-06	1.6E-05	1.5E-05	7.5E-06	1.3E-05
	DEAF SMITH	3.4E-05	7.4E-06	1.4E-05	2.5E-05	5.6E-06	1.2E-05
	HANFORD	3.5E-05	9.5E-06	7.8E-06	2.7E-05	7.2E-06	6.4E-06
	SKULL VALLEY	5.0E-05	1.1E-05	1.6E-05	3.8E-05	8.4E-06	1.3E-05
INDIAN POINT	ORNL	1.3E-05	1.1E-05	2.7E-05	9.8E-06	8.7E-06	2.2E-05
	DEAF SMITH	5.1E-05	1.5E-05	2.6E-05	3.9E-05	1.2E-05	2.2E-05
	HANFORD	8.5E-05	2.0E-05	2.9E-05	6.4E-05	1.5E-05	2.4E-05
	SKULL VALLEY	6.8E-05	1.9E-05	5.3E-06	5.1E-05	1.4E-05	4.0E-06
IDAHO NATIONAL LAB	ORNL	6.5E-05	1.0E-05	9.6E-06	4.9E-05	7.6E-06	7.9E-06
	DEAF SMITH	3.9E-05	4.6E-06	5.2E-06	2.9E-05	3.5E-06	4.3E-06
	HANFORD	2.2E-05	2.5E-06	2.6E-06	1.6E-05	1.9E-06	2.1E-06
	SKULL VALLEY	7.8E-06	2.1E-06	3.3E-06	5.9E-06	1.6E-06	2.7E-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 used in 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. Occupants of the vehicles that share the route are closer to the cask than residents or others beside the route. Occupants of vehicles moving in the opposite direction from the cask are exposed to radiation from the cask for considerably less time because the vehicles involved are moving past each other. The exposure time for vehicles traveling in the same direction as the cask is assumed to be the time needed to travel the link at the average speed (Neuhauser et al., 2000). The number of other vehicles that share truck routes is very large; the average number of vehicles per hour on U.S. interstate and primary highways in 2004[13] (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.

---

[13] 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 are assumed, the collective doses are one-third larger.

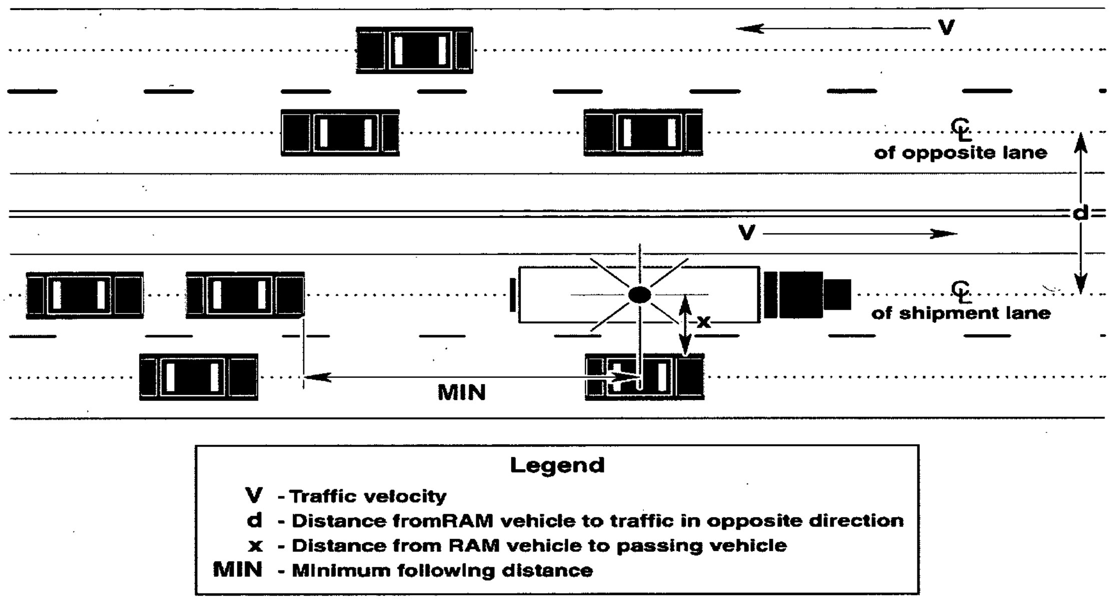


Figure 2-5. Diagram used in RADTRAN for calculating radiation doses to occupants of other vehicles (from Neuhauser et al., 2000).

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

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	5.0E-04	2.9E-04	4.3E-05	4.0E-06
	SKULL VALLEY	4.4E-04	2.8E-04	4.1E-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	3.4E-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	9.4E-06	8.7E-7
	SKULL VALLEY	3.7E-05	2.3E-05	8.5E-06	7.8E-07

2.3.4 Doses at Truck and Train Stops

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

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

Figure 2-6 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.

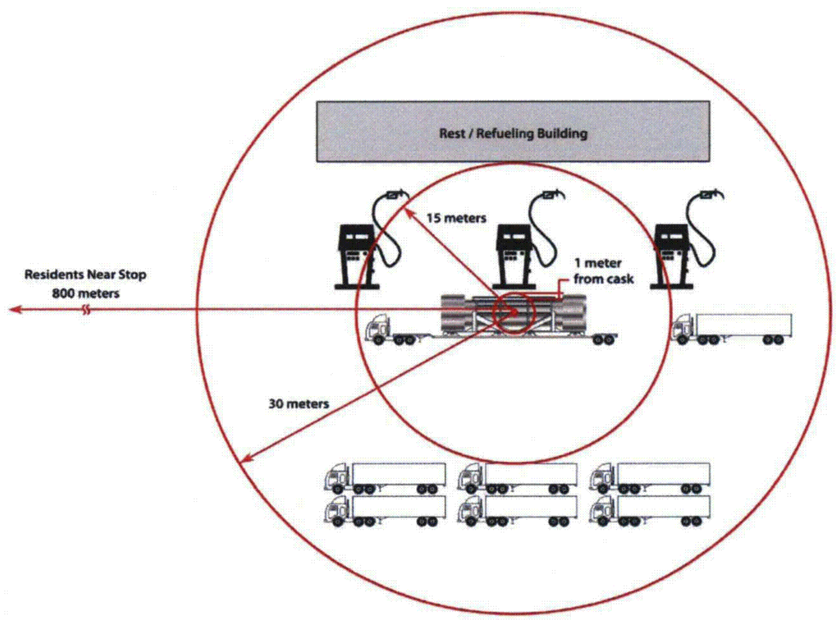


Figure 2-6. Diagram of truck stop model (not to scale).

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

Table 2-9. Some sample data for calculating doses at truck and train stops

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

[a]From Griego et al., 1996

Rail

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

- For the Rail-Lead: 1.5×10^{-5} person-Sv (1.5 person-mrem)
- For the Rail-Steel: 1.1×10^{-5} person-Sv (1.1 person-mrem)

The average dose to an individual living 200 to 800 meters from a classification yard is

- 3.5×10^{-7} Sv (0.035 mrem) from the Rail-Lead
- 2.7×10^{-7} Sv (0.027 mrem) from the Hi-STAR 100

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

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

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

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 detailed discussion of these calculations is provided 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 include one truck stop. This stop could be in either a rural or a suburban area. The results shown in Table 2-12 are general average doses at stops.

2.4 Doses to Workers

Radiation doses to workers are limited in accordance with the regulations of 10 CFR Part 20 and the practice of ALARA: maintaining the worker exposure to ionizing radiation "as low as reasonably achievable." ALARA applies to occupational doses because workers are potentially exposed to much larger doses than the general public. For example, the cab of a truck carrying a loaded Truck-DU cask is shielded so that 63% of the radiation from the end of the cask is blocked. 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. Not included are workers who handle spent fuel containers in storage, loading and unloading casks from vehicles or during intermodal transfer, and attendants who would refuel trucks, because truck refueling stops in the U.S. no longer have such attendants.[14] Table 2-13 summarizes the occupational doses. All doses are reported per hour except for the truck stop worker (reported for the maximum truck stop time) and the rail classification yard workers. All doses are individual doses (Sv) except for the railyard worker collective doses

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

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

Cask and route type	One train crew member: Sv per hour	One truck crew member; Sv per hour	One escort: Sv/hour	One Inspector: Sv per 1-hour inspection	One truck stop worker: Sv per shipment (0.83 hour)	Rail classification yard workers: person-Sv
Rail-Lead rural/suburban	2.1E-06		2.8E-06			1.5E-05
Rail-Lead urban	2.1E-06		2.8E-06			
Rail-All Steel rural/suburban	2.1E-06		2.8E-06			1.1E-05
Rail-All Steel urban	2.1E-06		2.7E-06			
Truck - DU rural/suburban		2.0E-05	7.0E-04	3.7E-04	6.7E-06	
Truck - DU urban		2.0E-05	7.0E-04			

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

Truck crew members are shielded so that they receive a maximum dose of 2.0×10^{-5} Sv per hour. This regulatory maximum was imposed in the RADTRAN calculation. Truck inspectors generally spend about an hour within one meter of the cargo (Weiner and Neuhauser, 1992), resulting in a relatively large dose. An upper bound to the duration of a truck refueling stop is about 49 minutes (0.83 hours) (Griego, et al., 1996). The truck stop worker whose dose is reflected in Table 2-13 is assumed to be outside (unshielded) at 15 meters from the truck during the stop. Truck stop workers that are in concrete or brick buildings would be shielded from any radiation.

2.5 Unit Risk

RADTRAN, the model used for the calculation of transportation risk, multiplies numbers. The only calculation that RADTRAN makes that is not a simple multiplication is calculating emissions from the spherical model shown in Figure 2-2. For routine transportation, 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 of possible effects, and a code that estimates risk can never be completely 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. Individual doses are comparable to background doses and less than doses from many medical diagnostic procedures. Collective doses are orders of magnitude less than the collective background dose, as shown in Figure 2-7 for an example shipment from Maine Yankee to ORNL. The NRC recommends that collective doses (average doses integrated over a population) be used only for comparisons (NRC, 2008). The proper comparison for collective doses is between the background collective dose plus the shipment dose and the background dose if there is no shipment. The collective dose is not zero in the absence of a shipment.

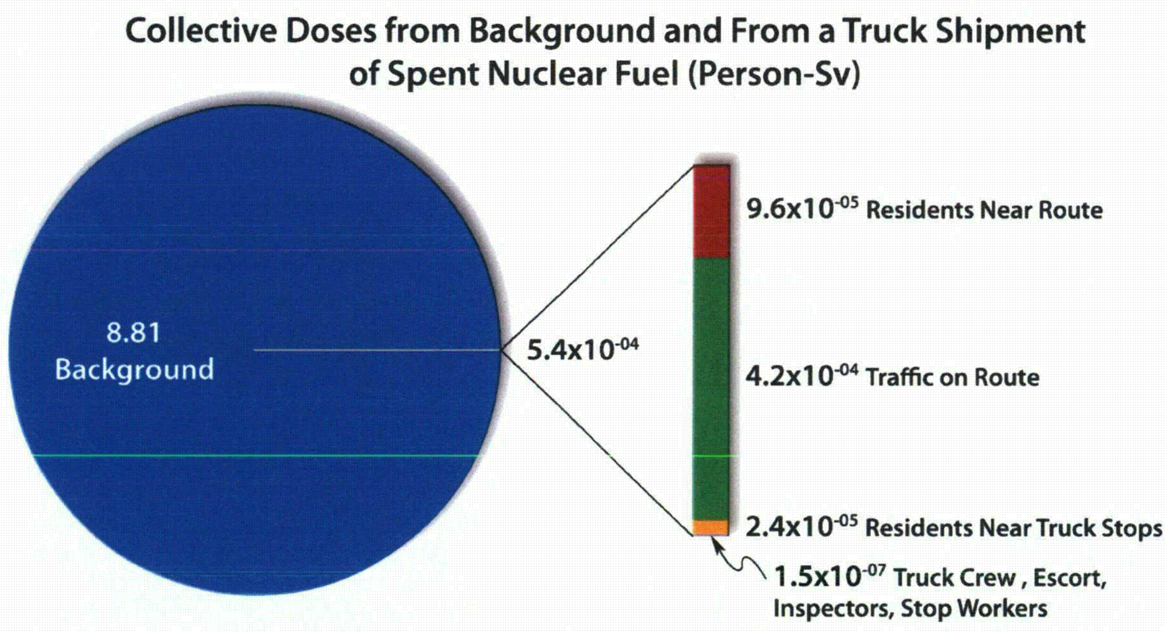


Figure 2-7. Collective doses from background and from Maine Yankee to ORNL truck shipments of spent nuclear fuel (person-Sv).

CHAPTER 3

CASK RESPONSE TO IMPACT ACCIDENTS

3.1 Introduction

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

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

3.2 Finite Element Analyses of Casks

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

The response to impacts of 48, 97,145, and 193 kilometers per hour (kph)—equal to 30, 60, 90, and 120 mph—onto an unyielding target in the end, corner, and side orientations for the Rail-Steel and Rail-Lead spent fuel transportation casks were determined using the non-linear transient dynamics explicit finite element (FE) code PRESTO (SIERRA, 2009). PRESTO is a Lagrangian code, using a mesh that follows the deformation to analyze solids subjected to large, suddenly applied loads. The code is designed for a massively parallel computing environment and for problems with large deformations, nonlinear material behavior, and contact. PRESTO has a versatile element library that incorporates both continuum (3D) elements and structural elements, such as beams and shells.

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

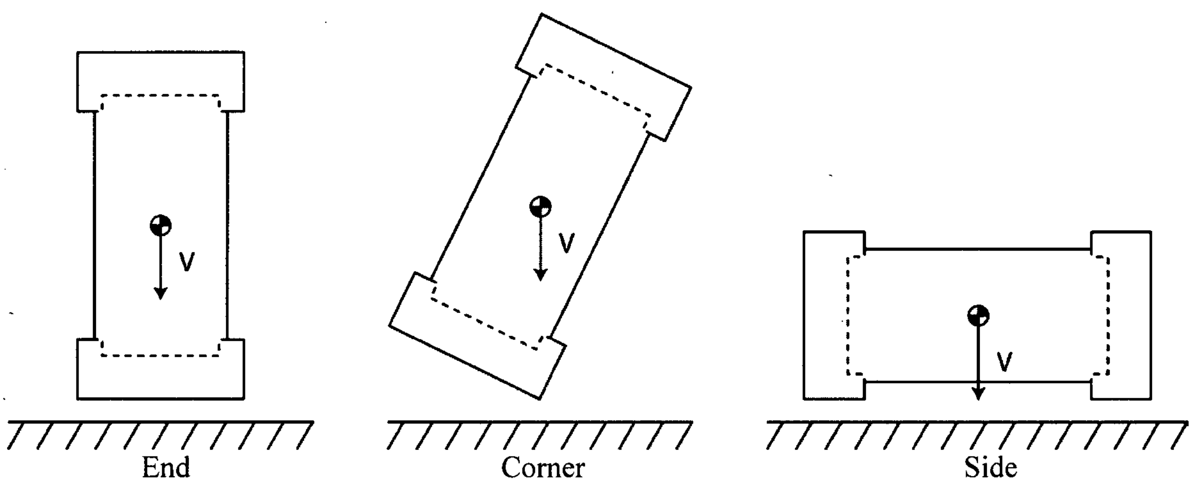


Figure 3-1. Impact orientations analyzed.

3.2.1 Rail-Steel Cask

__Finite element model__

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

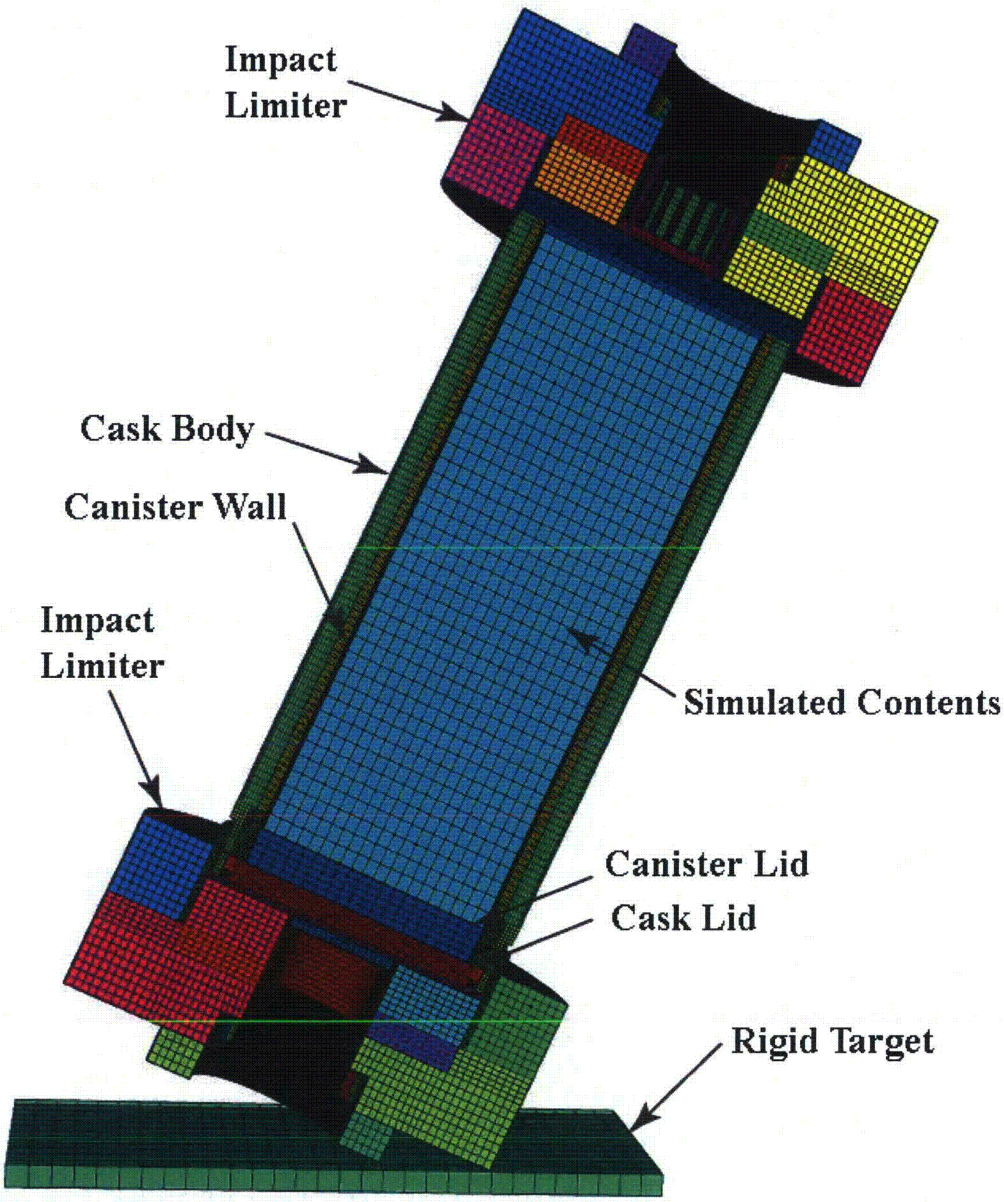
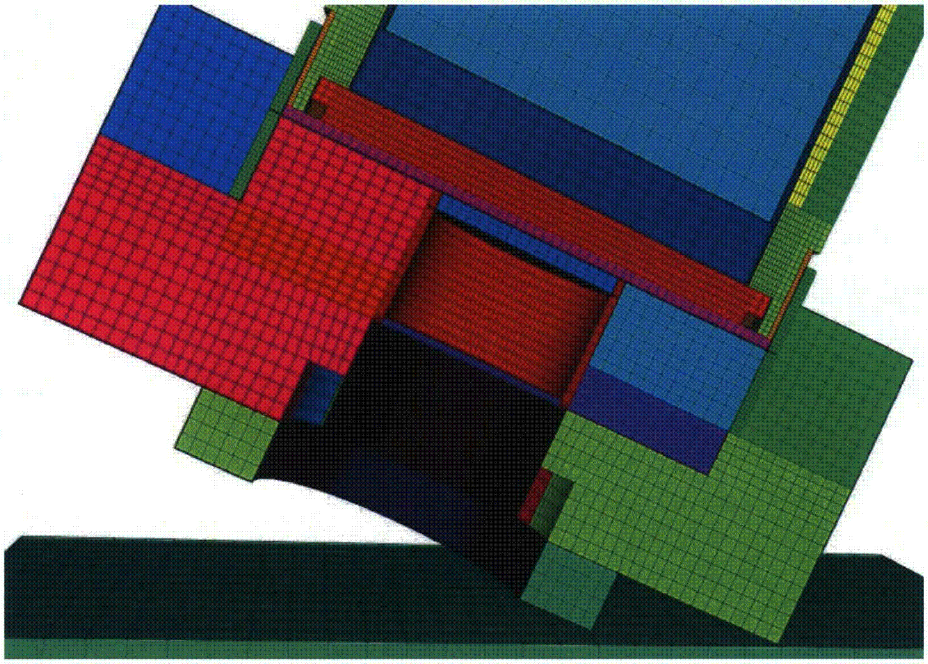
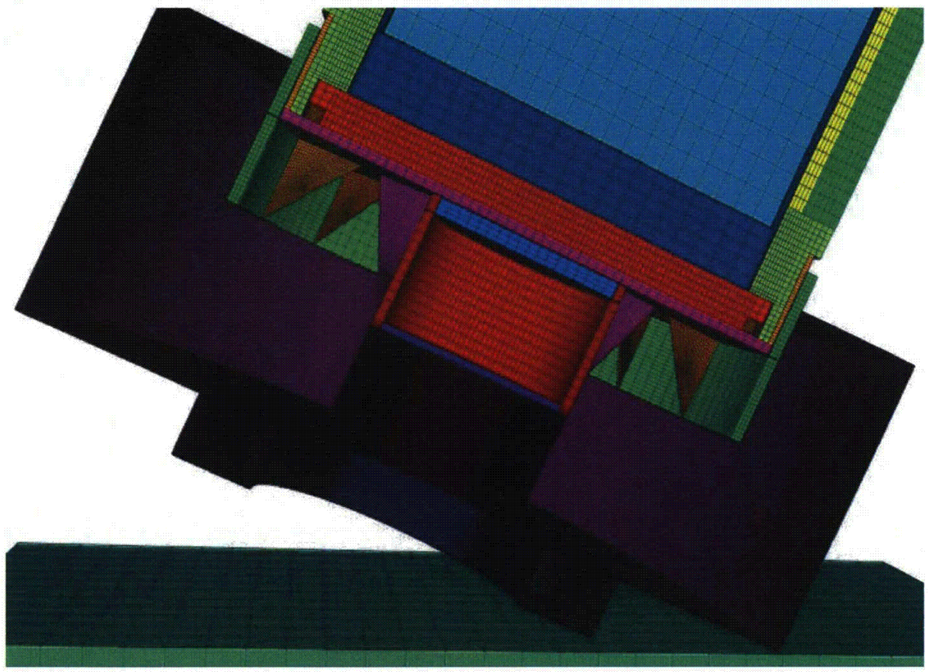


Figure 3-2. Finite element mesh of the Rail-Steel cask.



Impact limiter showing the various blocks of honeycomb



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

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

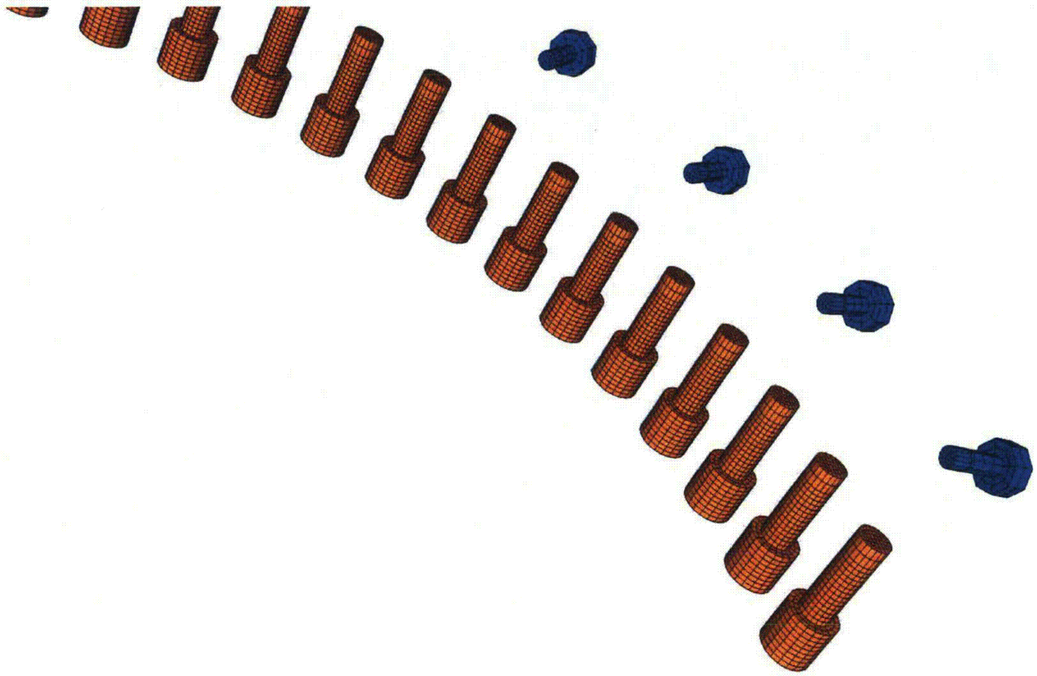


Figure 3-4. Finite element mesh of the Rail-Steel closure bolts and the closure end impact limiter attachment bolts. The highly refined mesh in these critical parts assures an accurate assessment of the closure response.

Analysis results

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

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

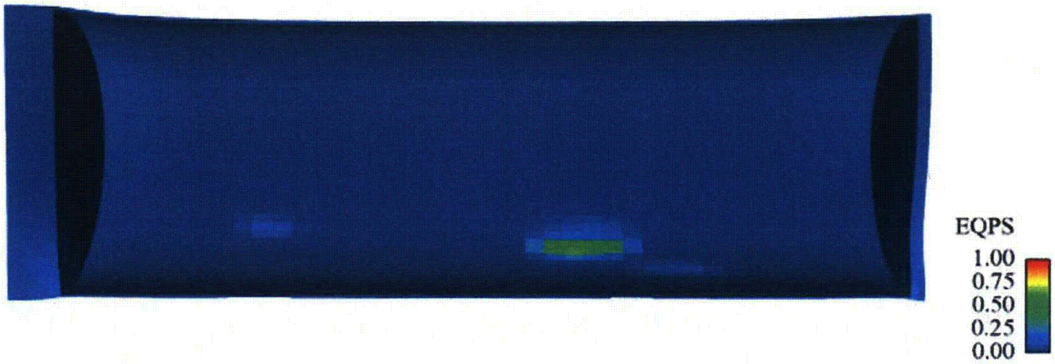


Figure 3-5. Plastic strain in the welded canister of the Rail-Steel for the 193 kph side impact case.

3.2.2 Rail-Lead Cask

Finite Element Model

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

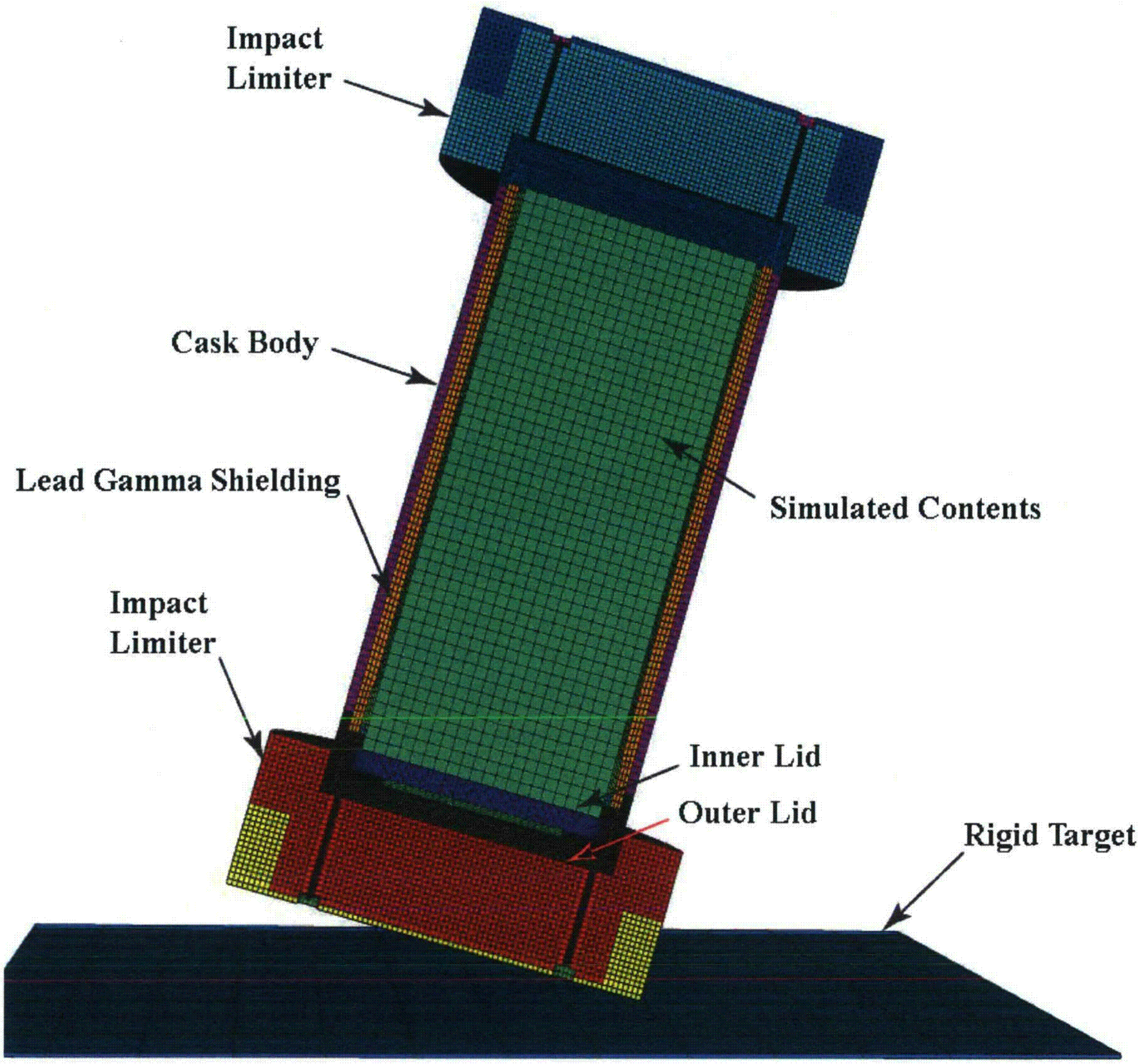
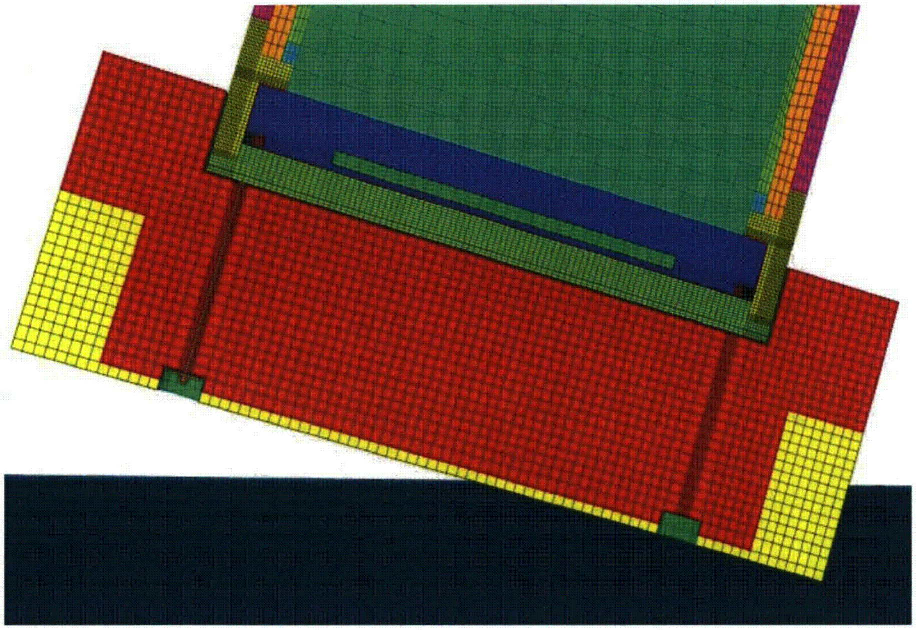
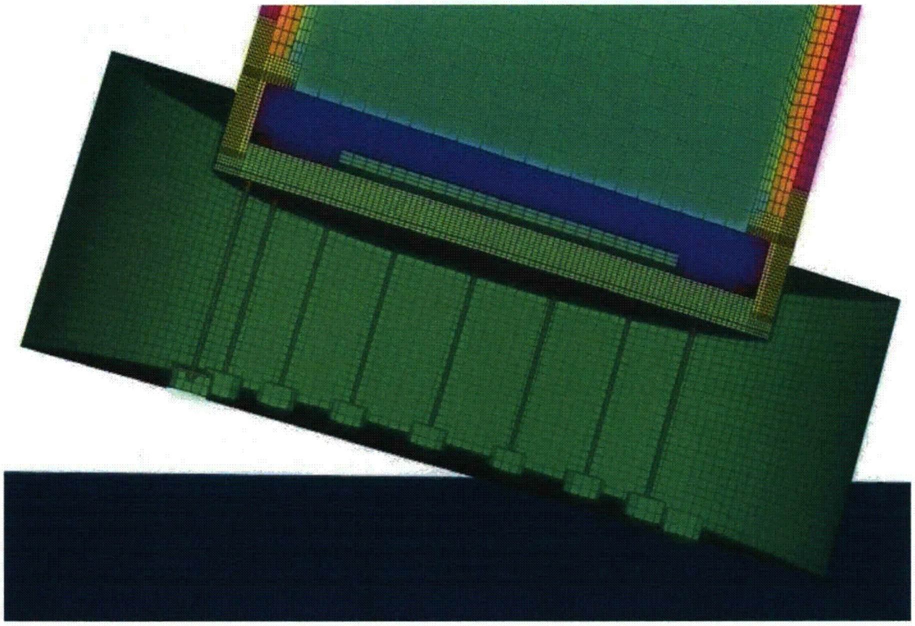


Figure 3-6. Finite element mesh of the Rail-Lead cask.



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



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

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

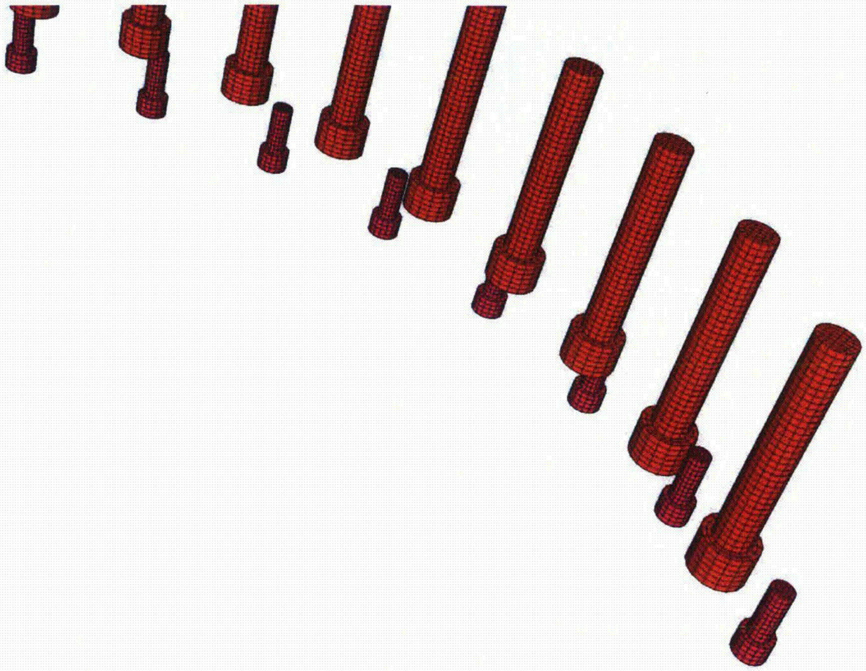


Figure 3-8. Finite element mesh of the Rail-Lead closure bolts for both the inner and outer lids. The longer bolts are for the inner lid and the shorter ones for the outer lid.

Analysis Results

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

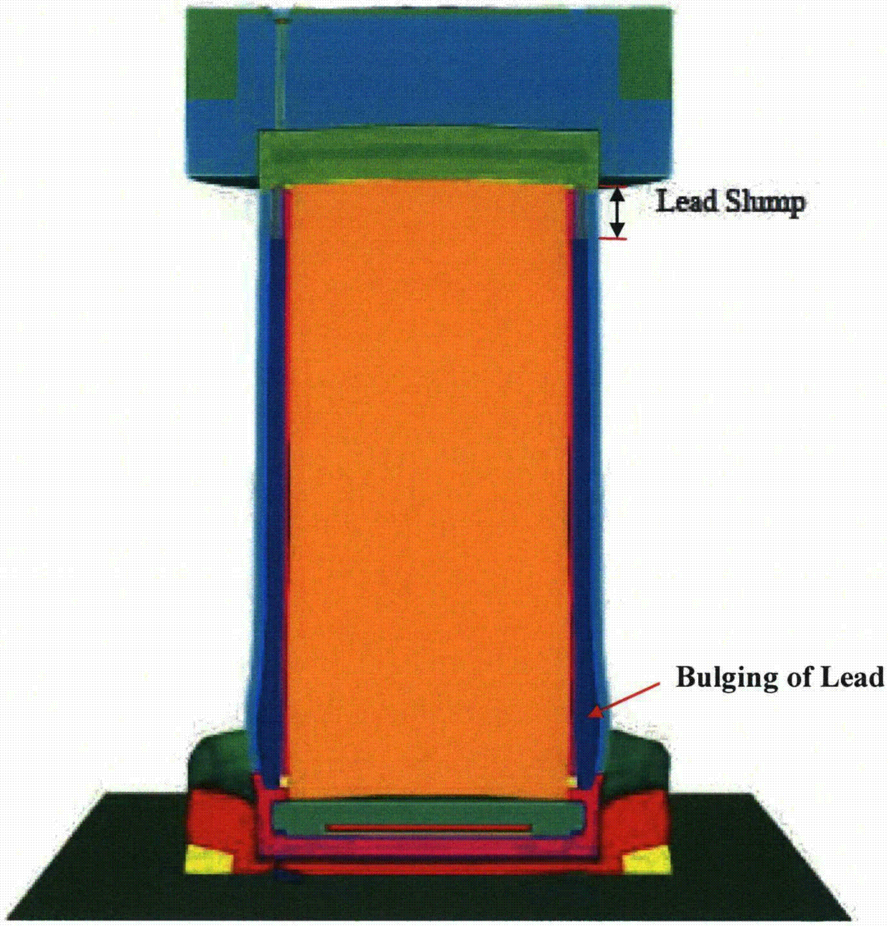


Figure 3-9. Deformed shape of the Rail-Lead cask following the 193-kph impact onto an unyielding target in the end-on orientation.

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

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

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

For the corner impacts at 97 and 145 kph, there is some damage to the cask body, in addition to deformation of the impact limiter, that results in lead slump and closure bolt deformation. The amount of deformation to the closure in these two cases is not sufficient to cause a leak if the cask is sealed with elastomeric o-rings, but is enough to cause a leak if the cask is sealed with metallic o-rings. For a corner impact at 193 kph there is more significant deformation to the cask, more lead slump, and a larger gap between the lid and the cask body. Figure 3-10 shows the deformed shape of the cask for this impact analysis. The deformation in the seal region is sufficient to cause a leak if the cask has metallic o-rings but not if it has elastomeric o-rings. The maximum amount of lead slump is 31 cm.

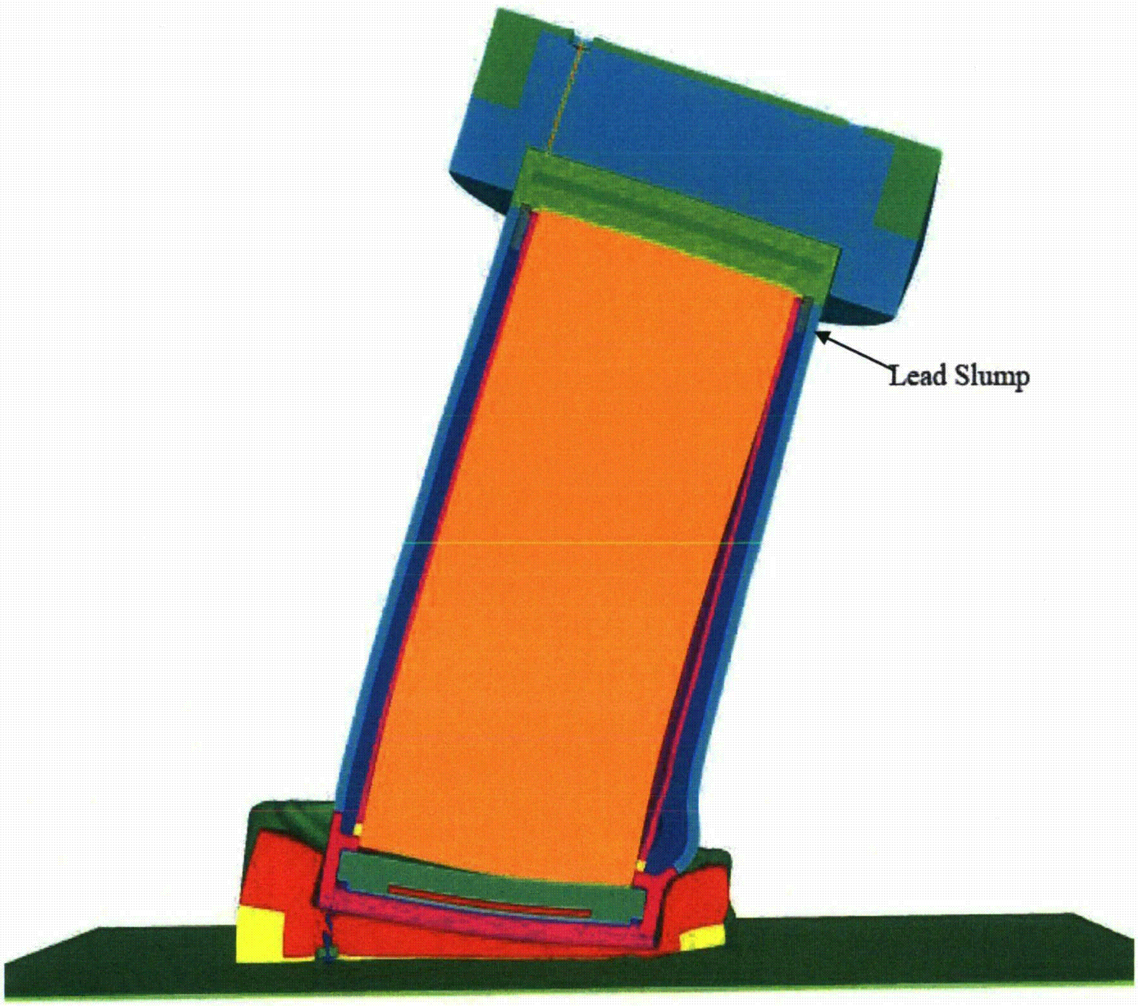


Figure 3-10. Deformed shape of the Rail-Lead following the 193 kph impact onto an unyielding target in the corner orientation.

In the side impact as the impact velocity increases from 48 kph to 97 kph, the impact limiter ceases to absorb additional energy and there is permanent deformation of the cask and closure bolts. The resulting gap in between the lids and the cask body is sufficient to allow leakage if there is a metallic seal, but not enough to leak if there is an elastomeric seal. This calculation of gap between the cask body and lid is conservative because the clamping force applied by bolt preload was neglected in the analysis (the clamping force acts to keep the lid and cask body together). When the impact speed is increased to 145 kph, the amount of damage to the cask increases significantly. In this case, many of the bolts from both the inner and outer lid fail in shear and there is a gap between each of the lids and the cask. This gap is sufficient to allow leakage if the cask is sealed with either elastomeric or metallic o-rings. Figure 3-11 shows the deformed shape of the cask following this impact. The response of the cask to the 193 kph impact is similar to that from the 145 kph impact, except that the gaps between the lids and the cask are larger. Deformed shapes for all of the analysis cases are shown in Appendix III.

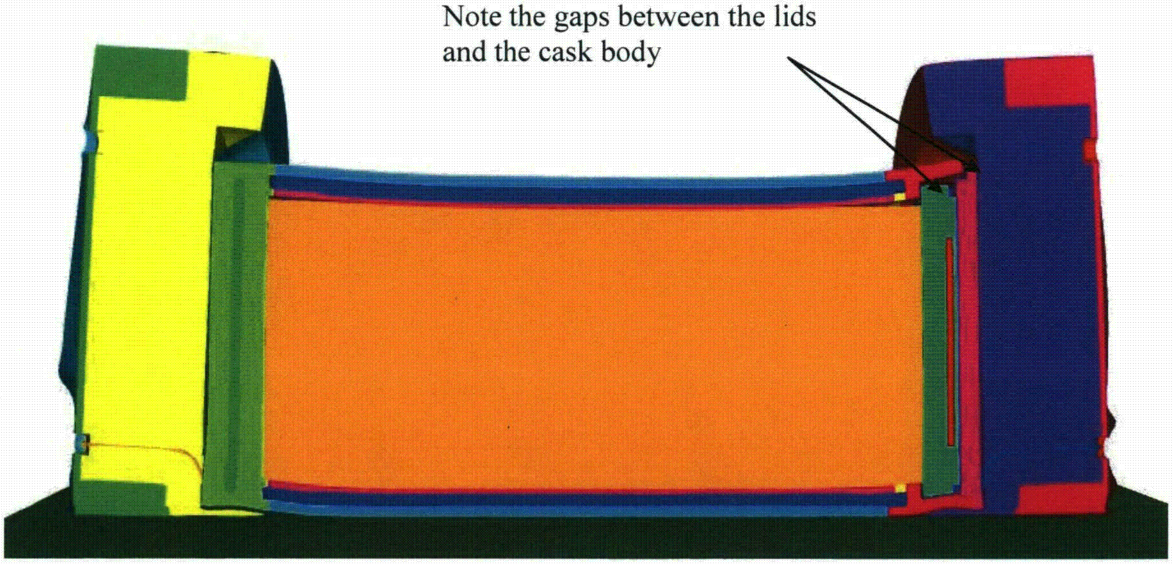


Figure 3-11. Deformed shape of the Rail-Lead cask following the 145 kph impact onto an unyielding target in the side orientation.

Leak Area

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

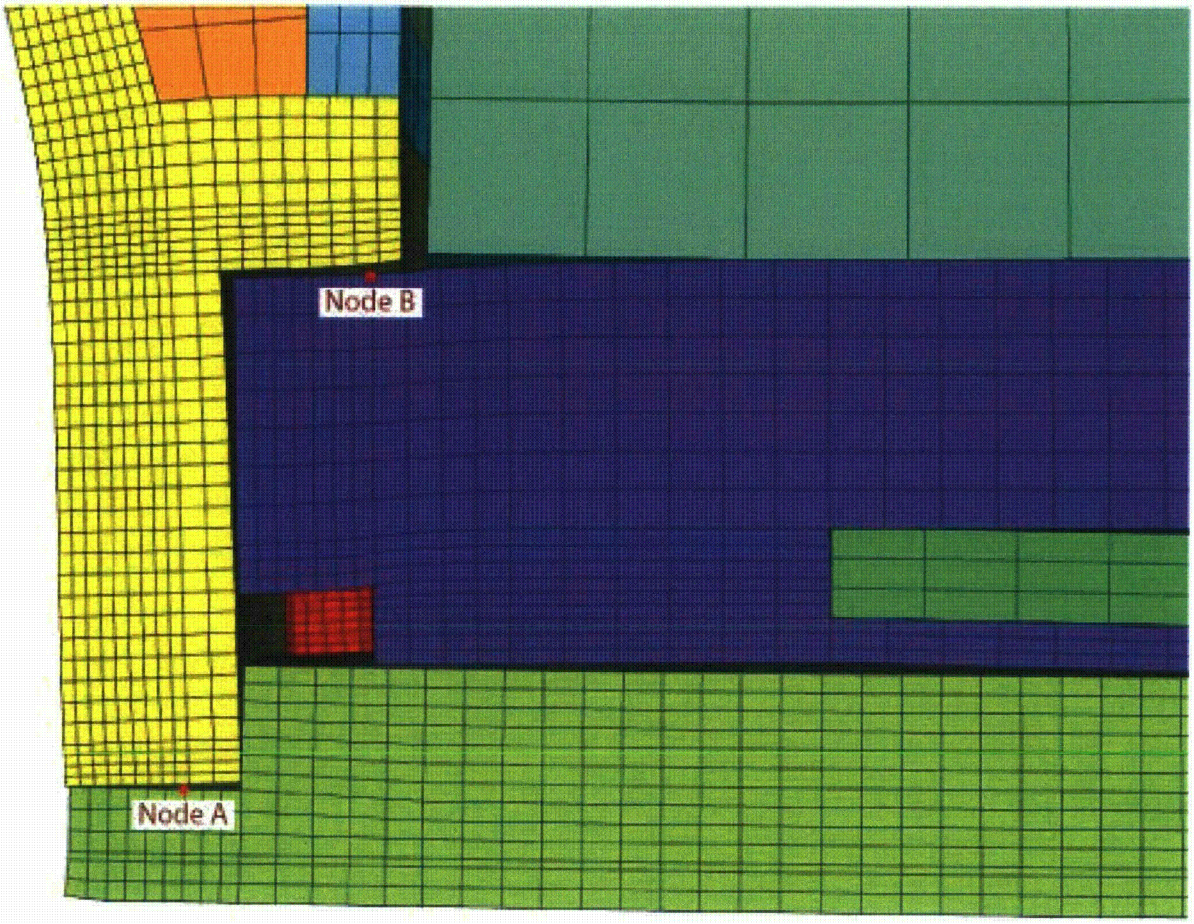


Figure 3-12. Measurement of closure gaps.

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

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

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

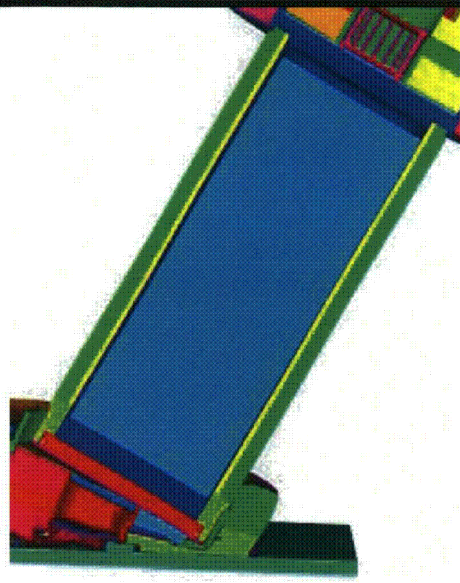
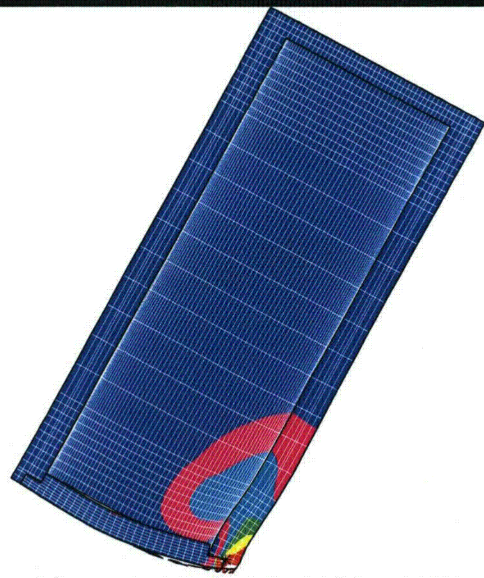
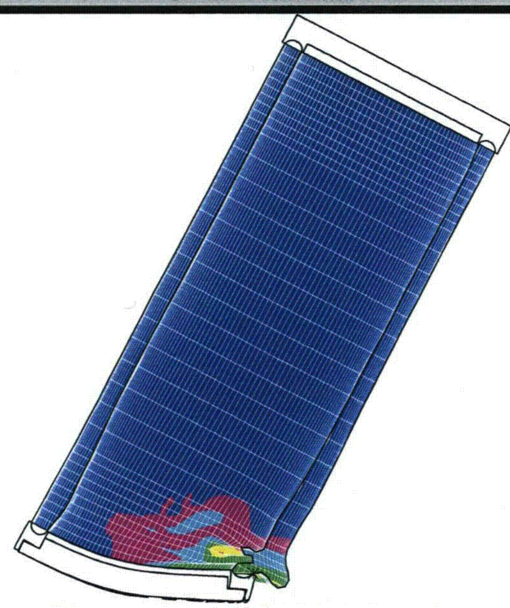
* Estimated. The method used to calculate the gaps for the other cases is explained in Appendix III. For these cases there was bolt failure and the gap was too large to measure using the standard method, but the resultant leak area is sufficiently large that any change to it would not change the cask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

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

3.2.3 Truck-DU Cask

Detailed finite element analyses of the Truck-DU cask were not performed for this study because the response of the truck casks in NUREG/CR-6672 indicated no gaps between the lid and the cask body at any impact speed. Therefore, the results discussed here are based upon the finite element analysis of the generic steel-DU-steel truck cask performed for NUREG/CR-6672. In general, the results from the analyses performed for this study have shown that the analyses performed for NUREG/CR-6672 were conservative (see Table 3-3), so the results discussed below are likely to be an overestimate of the damage to the Truck-DU cask from severe impacts. Figure 3-13 shows the deformed shape and plastic strain contours for the generic steel-DU-truck cask from Appendix A of NUREG/CR-6672 (Figures A-15, A-19, and A-22). None of the impacts caused strains that are great enough to fail the cask wall, and in all cases the deformation in the closure region was insufficient to cause seal failure. Table 3-4 (extracted from Table 5.6 of NUREG/CR-6672) provides the deformation in the seal region for each case. For all of these cases there would be no release of radioactive contents.

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

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

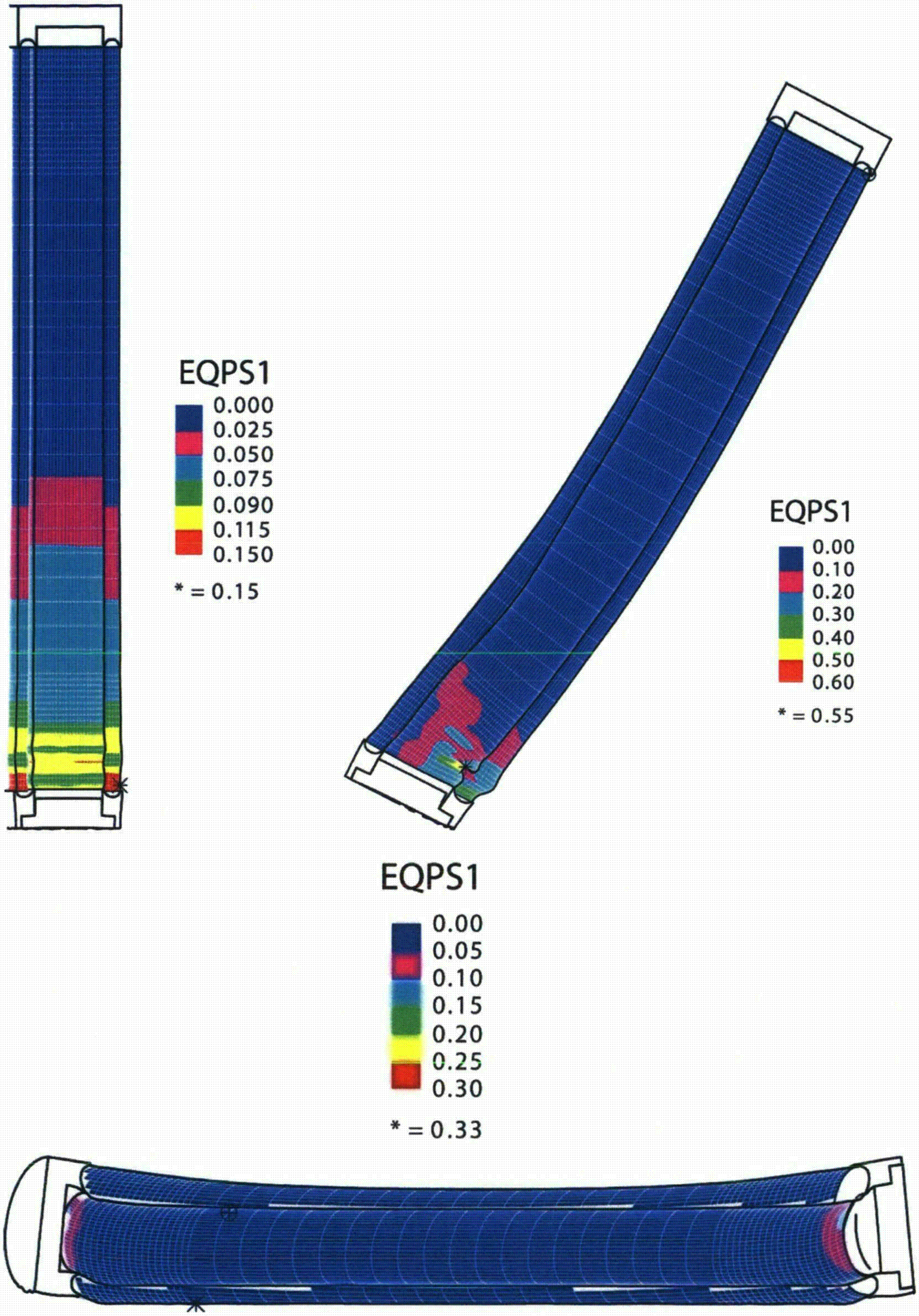


Figure 3-13. Deformed shapes and plastic strains in the generic steel-DU-steel truck cask from NUREG/CR-6672 (impact limiter removed) following 193 kph impacts in the (clockwise from top left) end-on, CG-over-corner, and side-on orientation.

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

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

3.3 Impacts onto Yielding Targets

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

For the Rail-Lead cask (the only one of the three investigated in this study that has any release), the peak force associated with each of the impact analyses performed is given in Table 3-5. In this table, the cases that have non-zero hole sizes from Table 3-4 have bold text. It can be seen that in order to produce sufficient damage for the cask to release any material, the yielding target has to be able to apply a force to the cask greater than 146 million Newtons (MN), or 33 million pounds. Very few real targets are capable of applying this amount of force. The target type that is the closest to an unyielding target is hard rock. In this study, hard rock is defined as rock that requires blasting operations to remove. While not all classes of this type of rock are equally strong, all of them are assumed to absorb negligible energy during an impact and are thus treated as rigid.

If the cask hits a flat target, such as the ground, roadway, or railway, it will penetrate into the surface. The greater the contact force between the cask and the ground, the greater the penetration depth. Figure 3-14 shows the relationship between penetration depth and force for the Rail-Lead cask impacting onto hard desert soil. As the cask penetrates the surface, some of its kinetic energy is absorbed by the surface. The amount of energy absorbed by the target is equal to the area underneath the force vs. penetration curve of Figure 3-14. As an example, the end impact at 97 kph onto an unyielding target requires a contact force of 124 MN. A penetration depth of approximately 2.2 meters will cause the soil to exert this amount of force. The soil absorbs 142 million Joules (MJ) of energy in being penetrated this distance. Adding the energy absorbed by the soil to the 41 MJ of energy absorbed by the cask gives a total absorbed energy of 183 MJ. For the cask to have this amount of kinetic energy, it would have to be traveling at 205 kph. Therefore, a 205 kph impact onto hard desert soil causes the same amount of damage as a 97 kph impact onto an unyielding target. A similar calculation can be performed for other impact speeds, orientations, and target types. Table 3-6 provides the resulting equivalent velocities. Similar to Table 3-5, the cases that result in non-zero hole sizes have bold text. Where the calculated velocity is more than 250 kph, the value in the table is listed as ">250." No accident velocities are more than this. The concrete target used is a 23-cm-thick slab on engineered fill. This is typical of many concrete roadways and concrete retaining walls adjacent to highways. Details on the calculation of equivalent velocities are included in Appendix III.

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

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

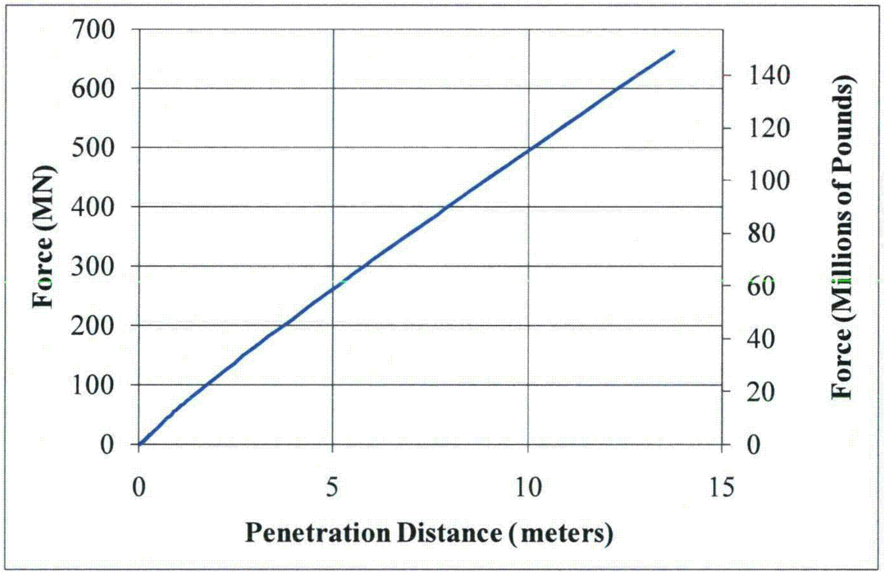


Figure 3-14. Force generated by the Rail-Lead cask penetrating hard desert soil.

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

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

3.4 Effect of Impact Angle

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

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

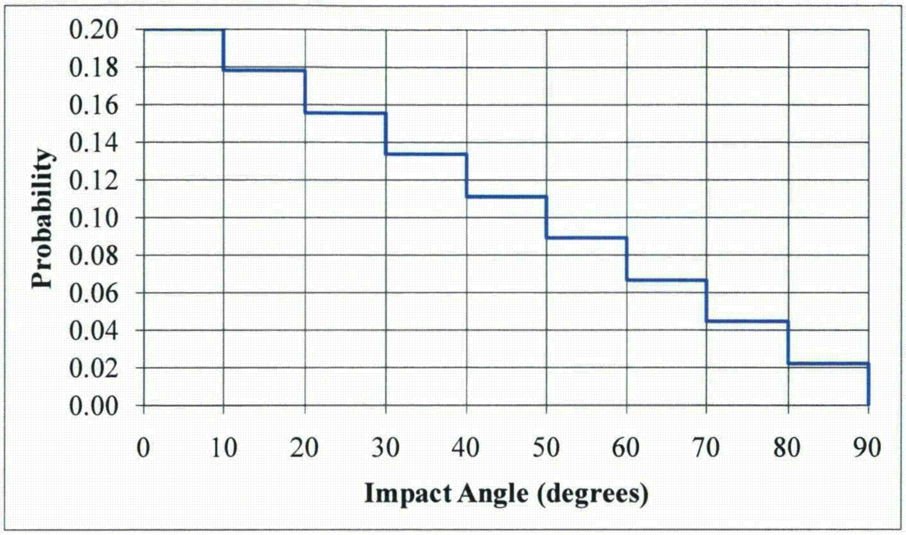


Figure 3-15. Probability distribution for impact angles.

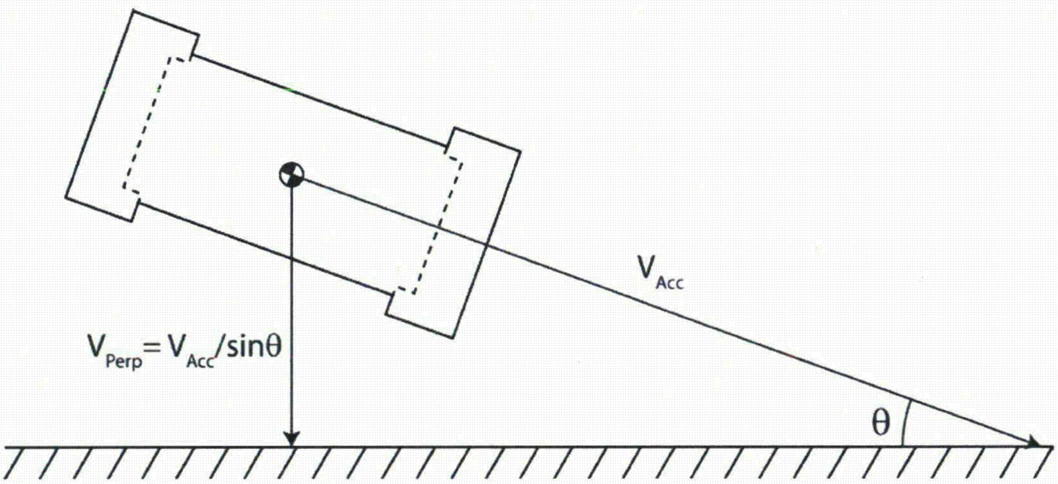


Figure 3-16. Influence of impact angle on effective velocity.

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

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

3.5 Impacts with Objects

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

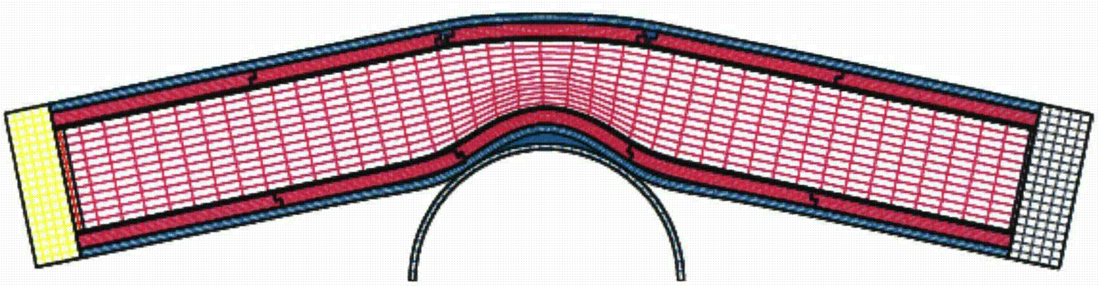


Figure 3-17. Deformations to the GA-4 truck cask after a 96 kph side impact onto a rigid semi-circular column, from (NRC, 2003b).

Another type of accident that could potentially damage a cask is the collision by a railroad locomotive. This is probably the most severe type of collision with another vehicle that is possible. Several different scenarios of this type of collision were investigated by Ammerman et al. (2005). The overall configuration of the general analysis case is shown in Figure 3-18. Most trains involve more locomotives and more trailing cars than were used in this analysis, but additional train mass has little effect on the force acting on the cask. The duration of impact is short and the coupling between the cars is flexible, so the impact is over before the inertia of more cars can have an influence on it. Variations on the general configuration included using the two most common types of locomotives, having a level crossing (such that the tires of the truck and the wheels of the locomotive are at the same elevation), having a raised crossing where the bottom of the main beams of the trailer are at the same elevation as the top of the tracks, and having a skewed crossing so the impact is at 67° instead of at 90°. For all analyses, the truck was assumed to be stopped. Train velocities of 113 kph and 129 kph were considered.

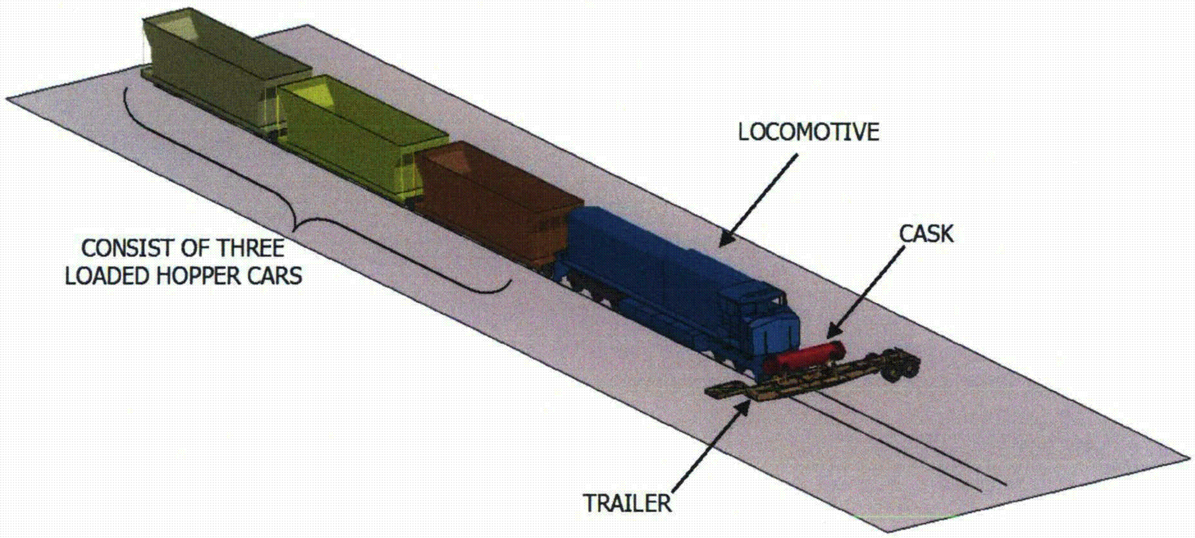


Figure 3-18. Configuration of locomotive impact analysis (Ammerman et al., 2005).

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

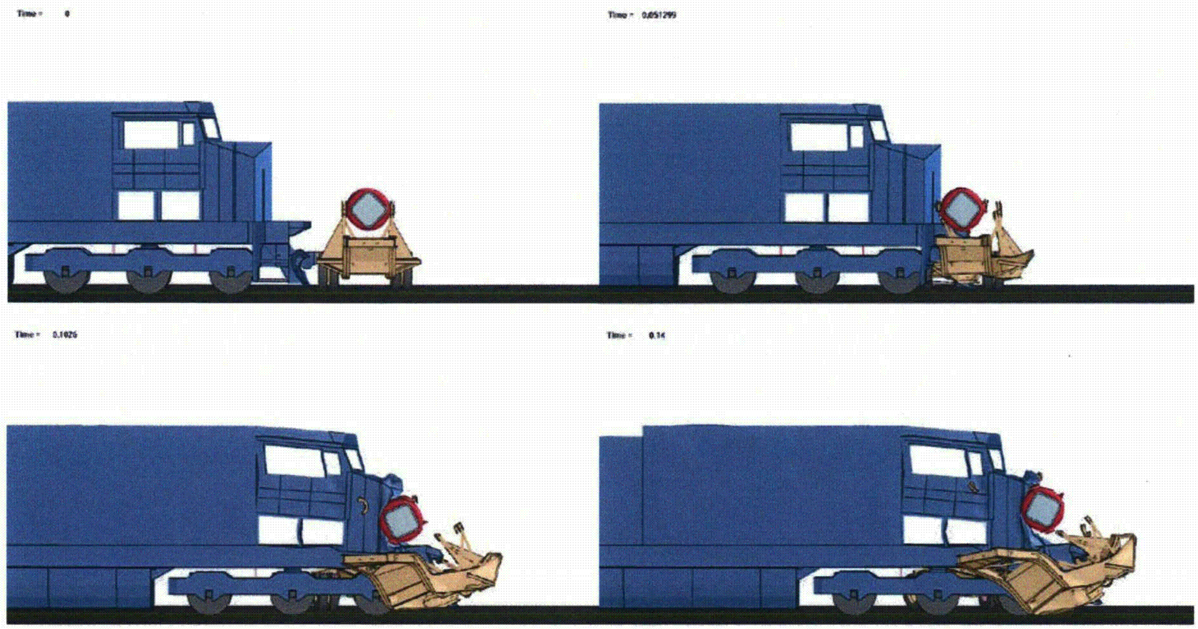


Figure 3-19. Sequential views of a 129 kph impact of a locomotive into a GA-4 truck cask (Ammerman et al., 2005).

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

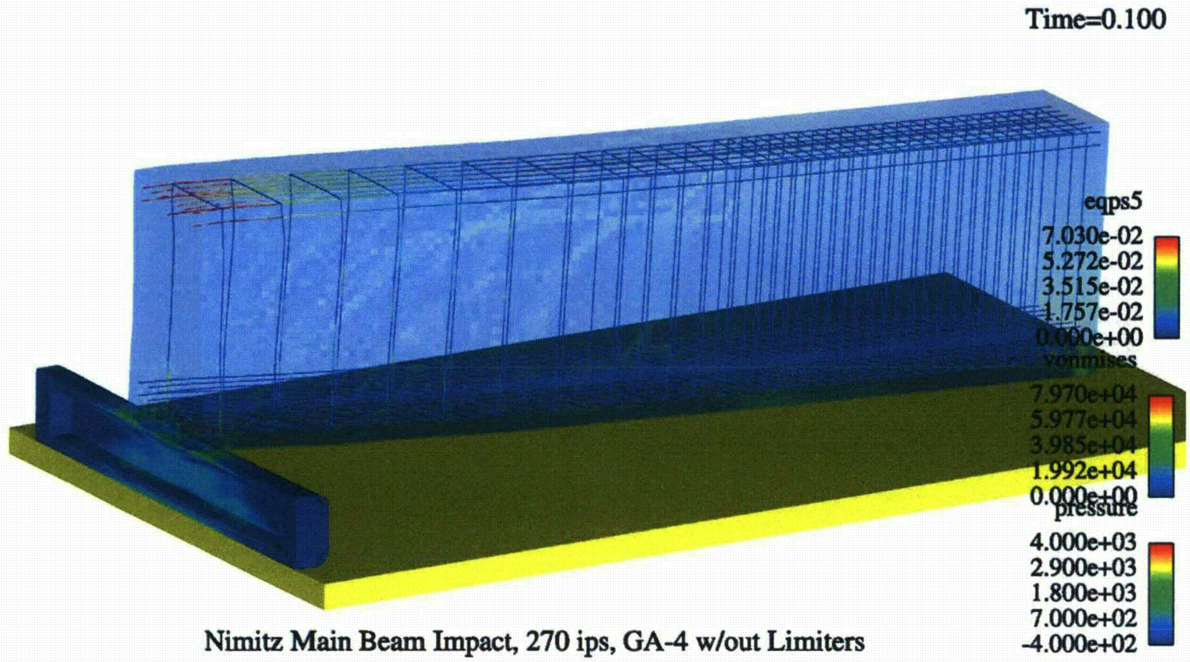


Figure 3-20. Results of a finite element simulation of an elevated freeway collapse onto a GA-4 spent fuel cask (Ammerman and Gwinn, 2004).

3.6 Response of Spent Fuel Assemblies

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

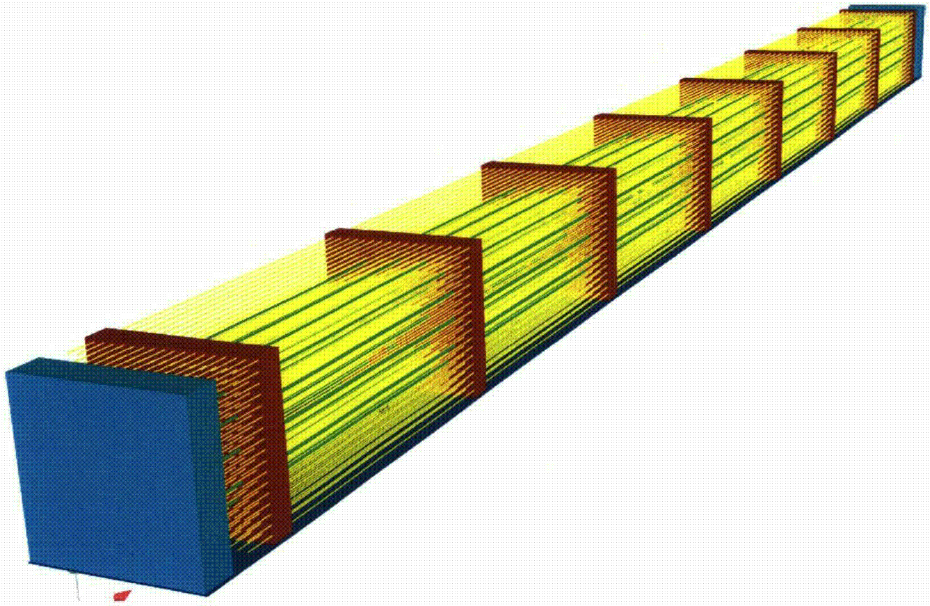


Figure 3-21. Finite element model of a PWR fuel assembly.

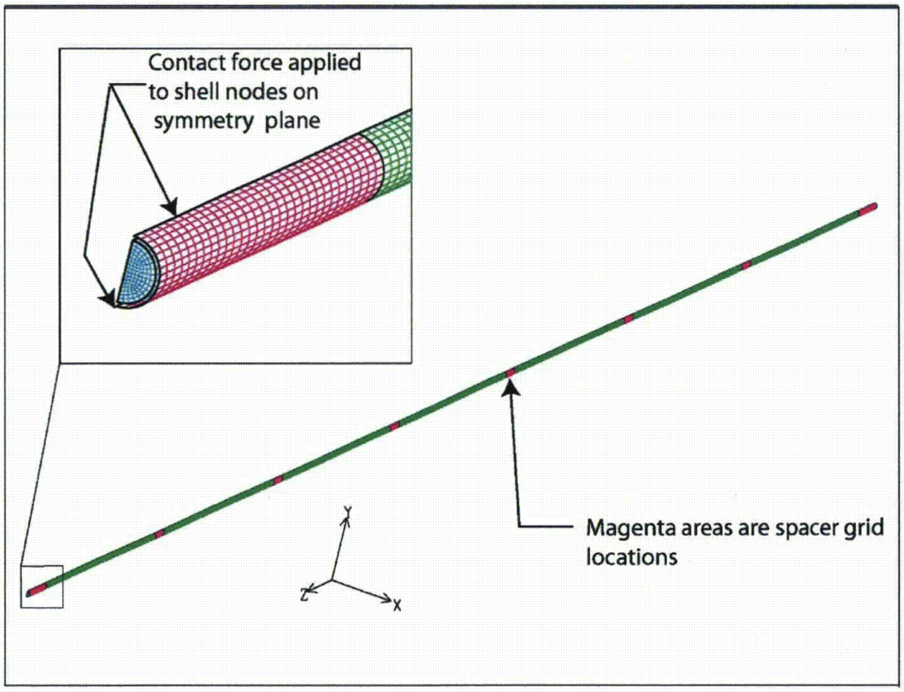


Figure 3-22. Detailed finite element model of a single fuel rod.

Figure 3-23 shows the maximum plastic strain at each location. The largest of these strains is slightly below two percent, which is half the plastic strain capacity of irradiated zircaloy at the maximum burn-up allowed in the Rail-Lead cask (45,000 MWD/MTU) (Sanders et al., 1992), so fuel rods will not crack. For cladding to fail, the peak acceleration of the cask would have to be above 200 G. The only impacts that are severe enough to crack the rods are those with impact speeds onto an essentially unyielding target of 145 kph or higher. A detailed description of the fuel assembly modeling is included in Appendix III.

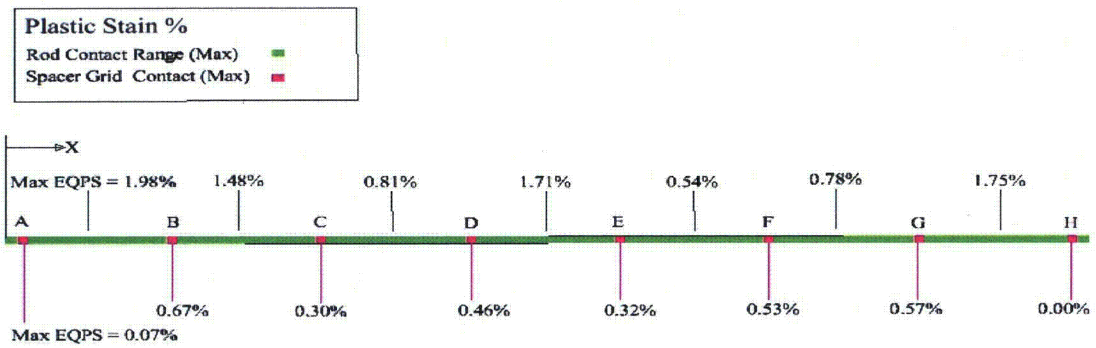


Figure 3-23. Maximum strains in the rod with the highest loads.

3.7 Conclusions

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

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

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

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

CHAPTER 4

CASK RESPONSE TO FIRE ACCIDENTS

4.1 Introduction

Certified Type B casks are designed to survive a fully-engulfing fire for thirty minutes. Certification analyses of the hypothetical accident condition (HAC) fire environment specified in 10 CFR 71.73 generally impose a thermal environment on the package that is similar to or more severe than a real fully-engulfing fire. This is more severe than the majority of the thermal environments a cask may be exposed to in an actual transportation accident that results in a fire (Fischer et al. 1988). Large open pool fires can burn at temperatures higher than the average temperature of 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 a certification analysis could be more severe for seal and fuel rod response than the exposure to an actual fire.

For this risk study, computer codes capable of modeling fires and the thermal response of casks exposed to fires in a realistic[15] fashion are used to analyze the response of the Rail-Steel and the Rail-Lead casks to three different fire configurations. These configurations are described in this chapter and the temperature responses of the casks are presented and discussed. An analysis of the thermal performance of the Truck-DU cask when exposed to a severe fire scenario is also presented.

The thermal response of each cask is compared to two characteristic temperature limits. These are the seal failure temperature (350°C for elastomeric seals used in the Rail-Lead 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 SNF transportation industry. Section 7.2.5.2 of 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 fire accident scenarios are analyzed for each rail cask and one for the truck cask. A hydrocarbon fuel pool that conforms to the HAC fire described in 10 CFR 71.73 is used as the basis for each scenario. This regulation specifies a hydrocarbon fuel pool that extends between

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

one and three meters horizontally beyond the external surface of a cask. To ensure the casks analyzed in this study are fully engulfed by the fire, all fuel pools were assumed to extend three meters from the sides of the cask (a pool fire that extends less than three meters can be sufficient to ensure full engulfment of smaller packages).

4.2.2 Fire Duration

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

In the case of the Truck-DU cask, the fire duration is based on the fuel capacity of a typical petroleum tank truck. About 9,000 gallons (34,070 liters) of gasoline can be transported on the road by one of these tank trucks. Provided that the overall dimensions of the Truck-DU cask are 2.3 m × 6 m (7.5 ft × 19.7 ft), a regulatory pool that extends horizontally 3 meters (10 feet) beyond the outer surface of the cask would be 8.3 m × 12 m (27.2 ft × 39.4 ft). To pool 9,000 gallons (34,070 liters) of gasoline in a pool of this area, the pool would need to be 0.3 m (1 ft) deep, a configuration that is difficult to obtain in an accident scenario and therefore unlikely to occur. Such a pool fire would burn for a little more than an hour. As discussed for the rail cask pool fire, the other possibility of maintaining a fire that can be engulfing and that can burn for that duration is if, for example, gasoline were to flow at the right rate to maintain the necessary fuel pool conditions. This scenario is also very unlikely. Nevertheless, one hour is used as the duration of a fire that is not moving over time for the conservative analysis of the Truck-DU cask.

4.2.3 Hypothetical Accident Configurations for the Rail Casks

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

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

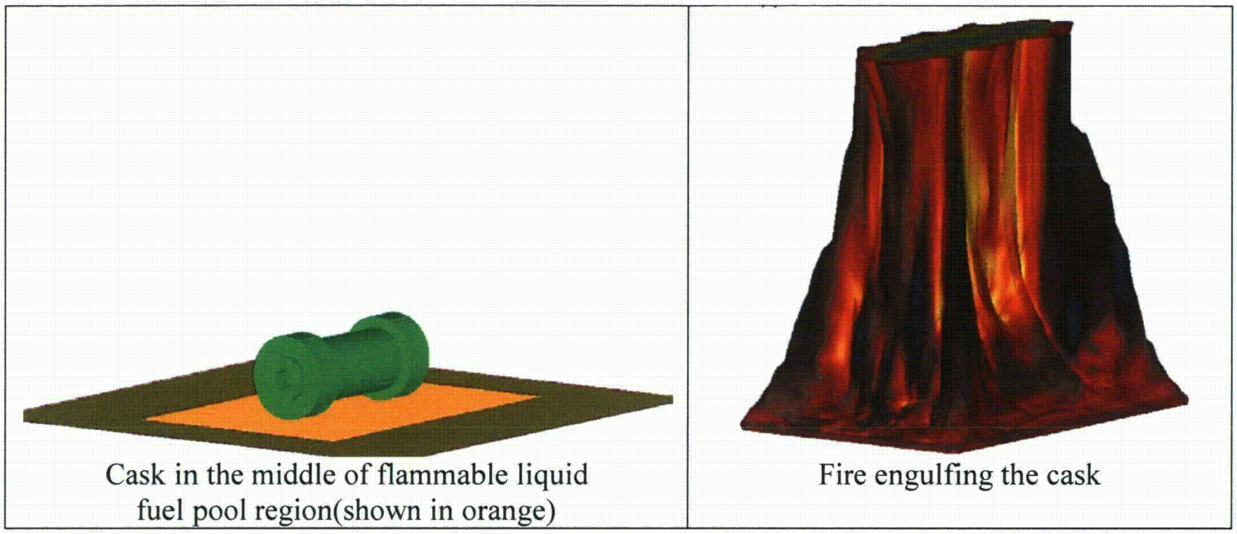


Figure 4-1. Cask lying on ground concentric with fuel pool.

2. Cask lying on the ground three meters (10 feet) away from the pool of flammable liquid (with the side of the cask aligned with the 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 the width of one rail car. This could be the case in an accident in which the rail cars derail in an "accordion" fashion.

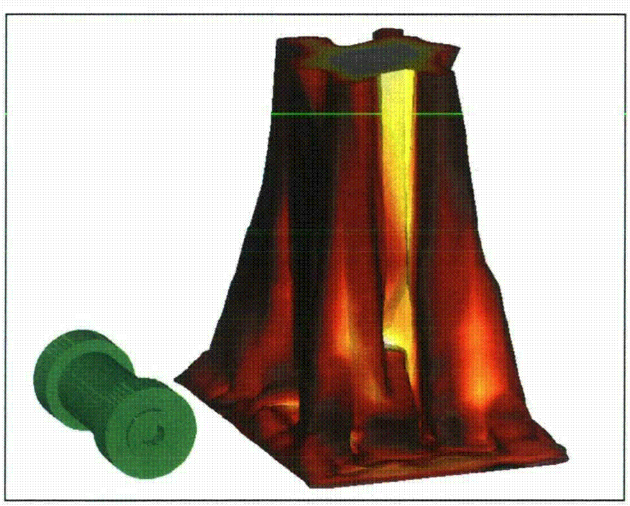


Figure 4-2. Cask lying on ground 3 meters from pool fire.

3. Cask lying on the ground 18 meters (60 feet) from the pool of flammable liquid (with the side of the package aligned with the side of fuel pool) as depicted in Figure 4-3. This scenario represents the hypothetical case in which the pool of flammable liquid and the cask are separated by the length of one rail car. This represents an accident in which the separation between a tank car carrying flammable liquid and the railcar carrying the SNF package is maintained (the distance of a buffer rail car) 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.

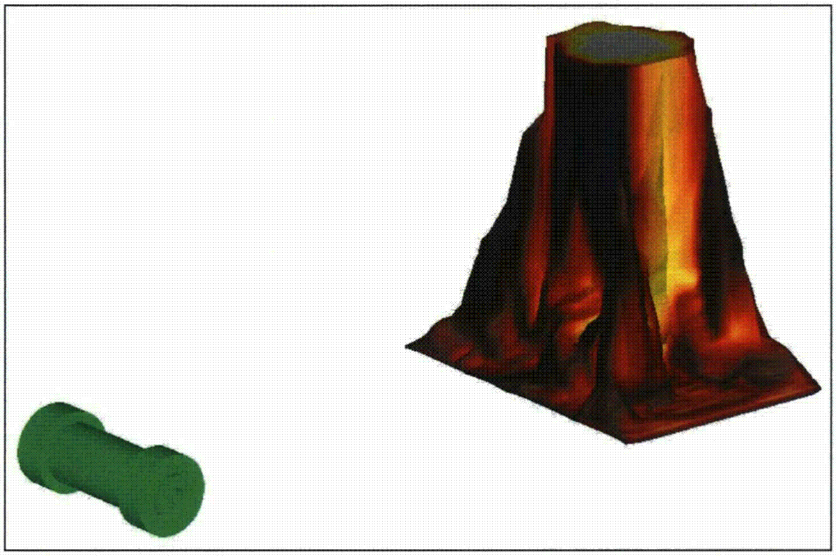
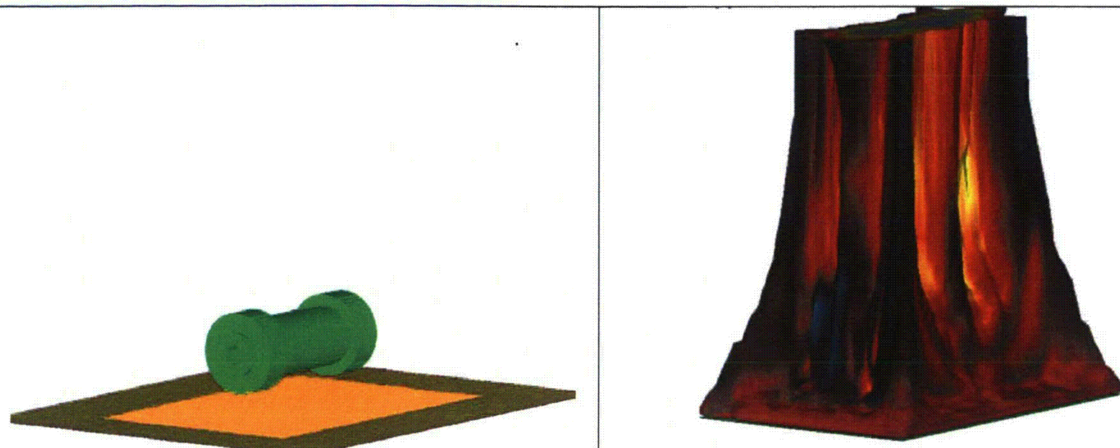


Figure 4-3. Cask lying on ground 18 meters from pool fire.

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

Before these accident scenarios are analyzed, two additional 30-minute regulatory HAC fire analyses are performed for each rail cask based on the conditions described in 10 CFR 71.73. In the first analysis, a commercially-available FE heat transfer code is used to apply an 800°C (1475°F) uniform-heating fire condition to the casks. In the second analysis, a benchmarked computational fluid dynamics (CFD) computer model with radiation heat transfer is used. In this model, each cask is positioned one meter above the fuel pool (as described in 10 CFR 71.73) and a realistic fire fully engulfs the cask as shown in Figure 4-4. The results from FE uniform heating analyses were compared to those in the safety analysis reports for the respective casks to ensure that the cask models used in these analyses are representative. The results from the CFD fire analyses are compared to the results obtained from the uniform-heating FE analyses to demonstrate that the realistic CFD fire does impose conditions that are similar to the uniform heating.



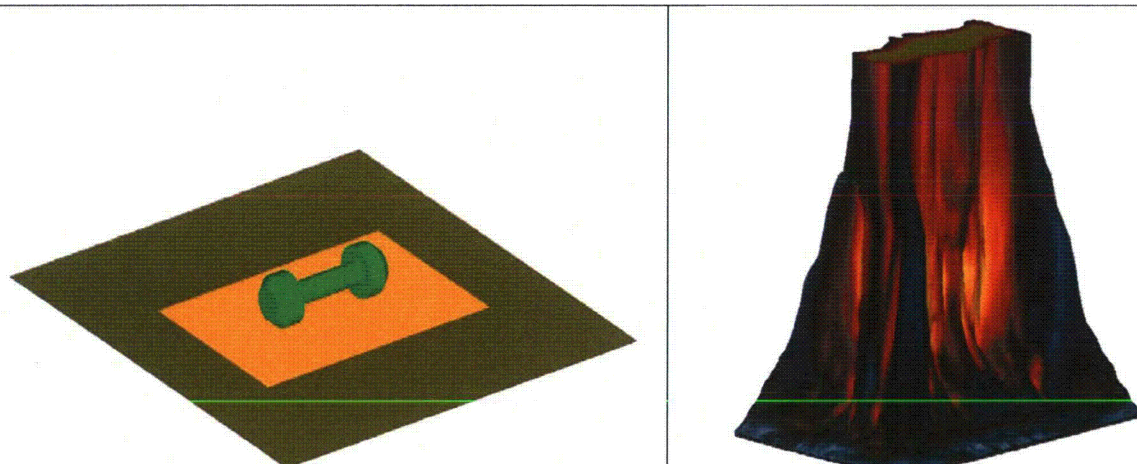
Cask elevated 1m above flammable liquid fuel pool region (shown in orange)

Regulatory fire engulfing the cask

Figure 4-4. Regulatory pool fire configuration.

4.2.4 Hypothetical Accident Configuration for the Truck Cask

In the case of the truck cask, solely the hypothetical accident configuration in which the cask is assumed to be concentric with a flammable fuel pool and is fully engulfed by a fire is analyzed. This hypothetical accident configuration is presented in Figure 4-5.



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

Fire engulfing the cask

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

4.3 Analysis of Fire Scenarios Involving Rail Casks

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

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

Results are presented in the following order:

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

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

4.3.2 Simulations of the Rail Casks

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

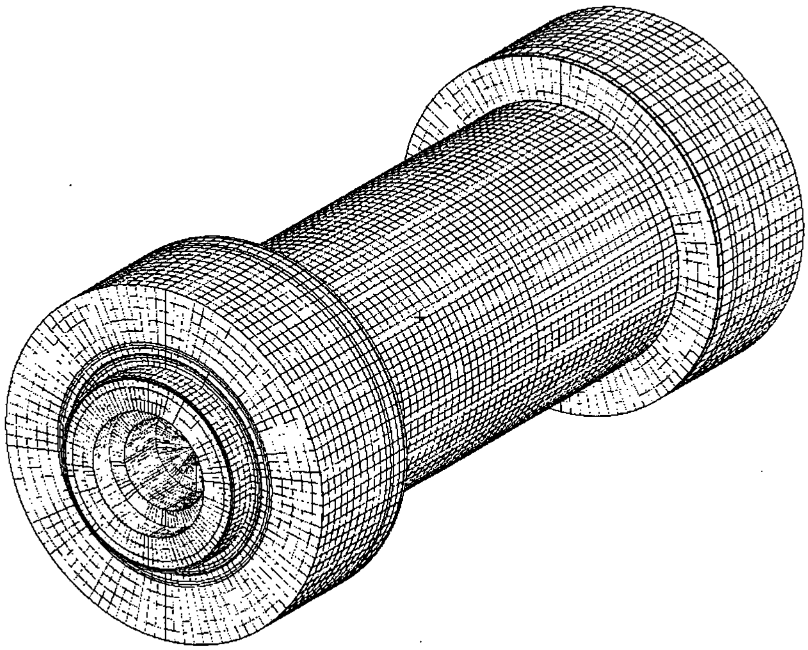
Both the Rail-Steel and the Rail-Lead casks have a polymeric neutron shield that is assumed to melt completely and be replaced by air at its operational temperature limit (see Appendix IV).

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

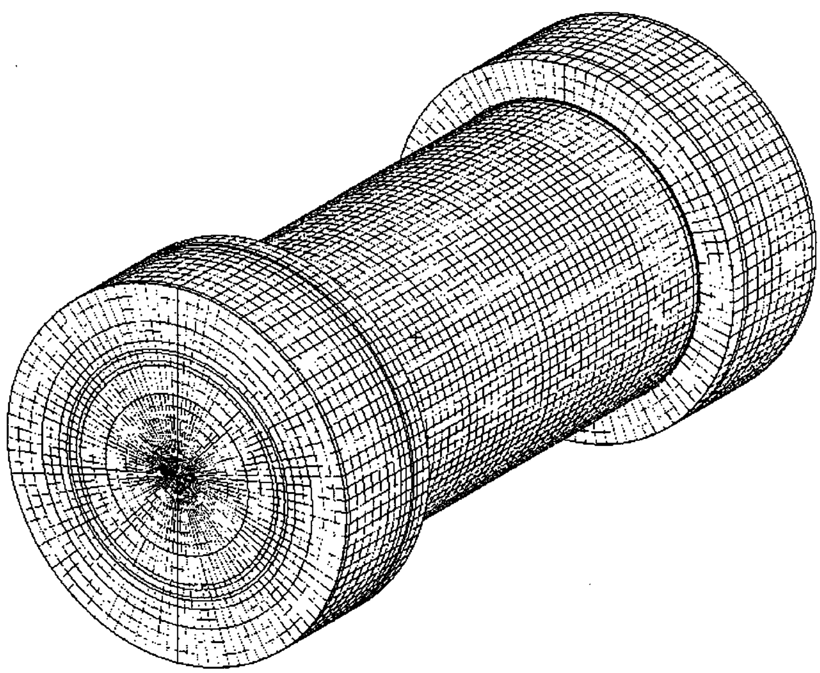
Impact limiters are modeled as undamaged (not deformed). The Rail-Steel cask has aluminum honeycomb impact limiters and the Rail-Lead cask has wood impact limiters. Spaces between components are explicitly modeled in both casks as these could have a significant effect on the thermal response of the cask.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

4.3.3 Simulation of the Spent Nuclear Fuel Region

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



Rail-Steel cask



Rail-Lead cask

Figure 4-6. Finite element models of the two rail casks analyzed.

4.3.4 Rail-Steel Cask Results

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

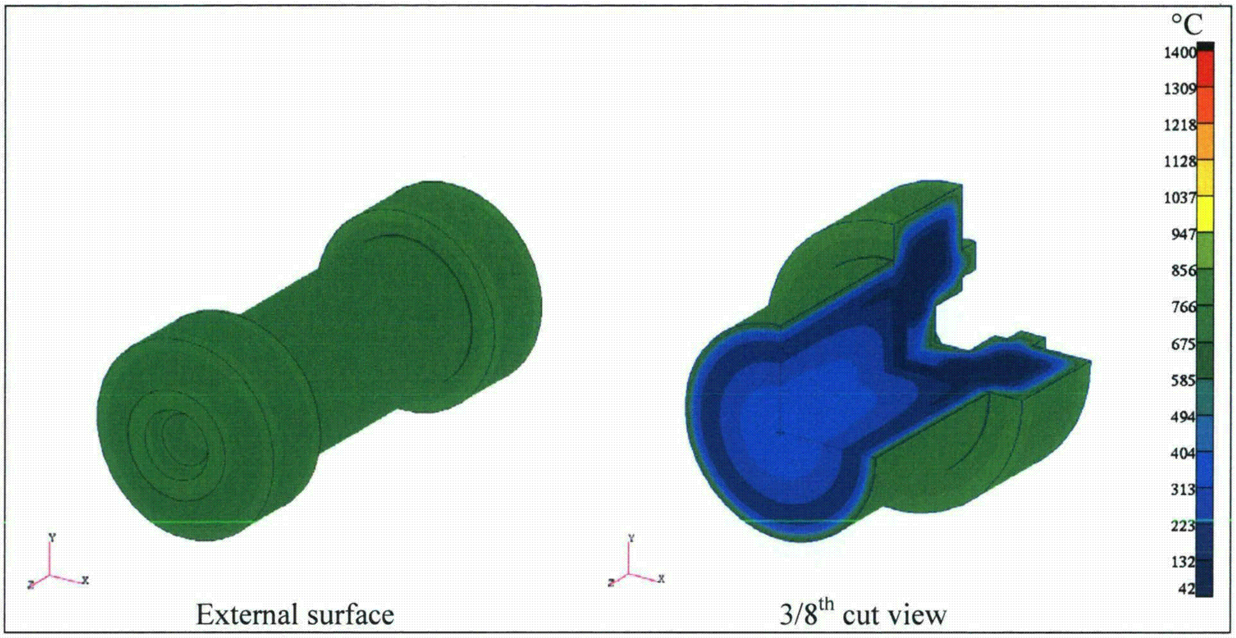


Figure 4-7. Temperature distribution of the Rail-Steel cask at the end of the 30-minute 800°C regulatory uniform heating.

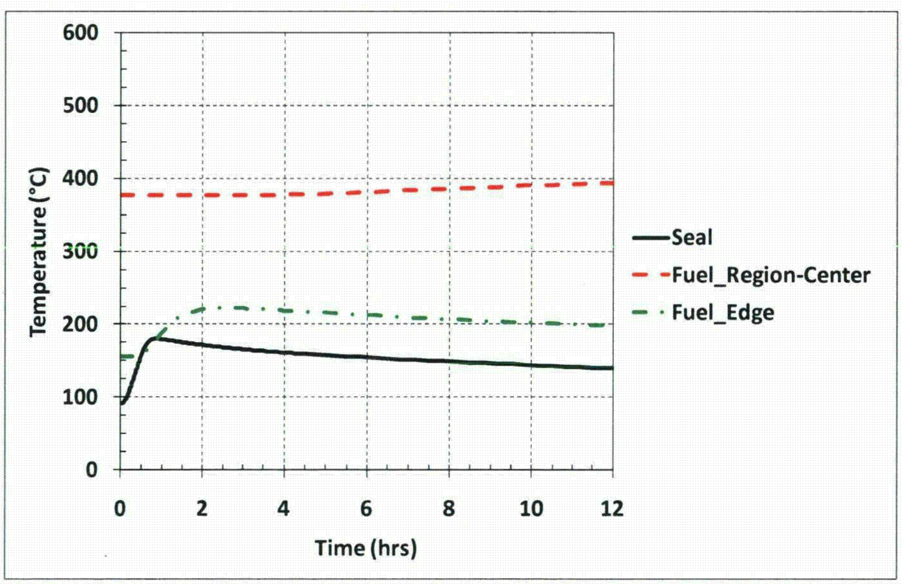


Figure 4-8. Temperature of key cask regions, Rail-Steel cask undergoing regulatory uniform heating.

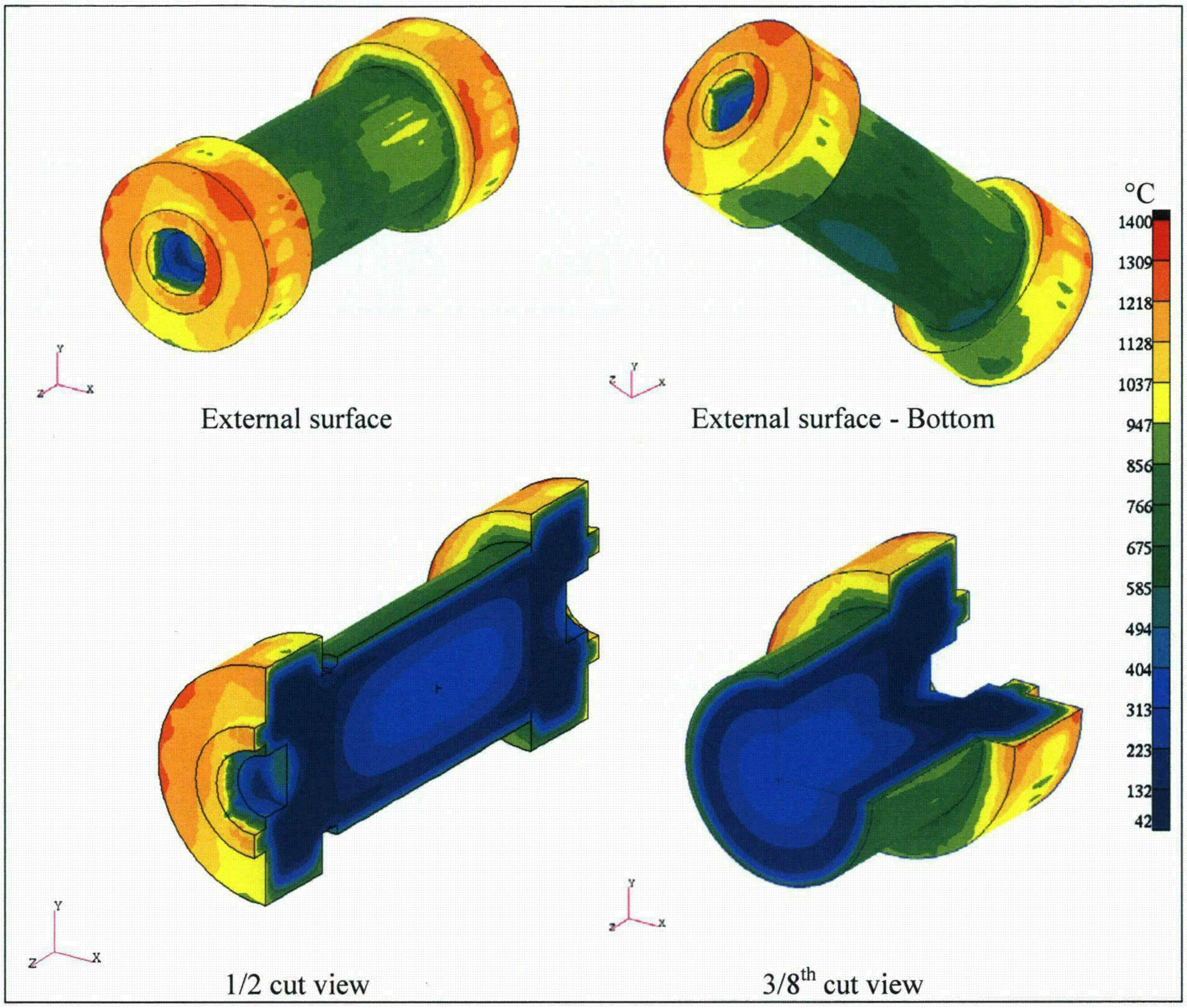


Figure 4-9. Temperature distribution of the Rail-Steel cask at the end of the 30-minute regulatory CAFE fire.

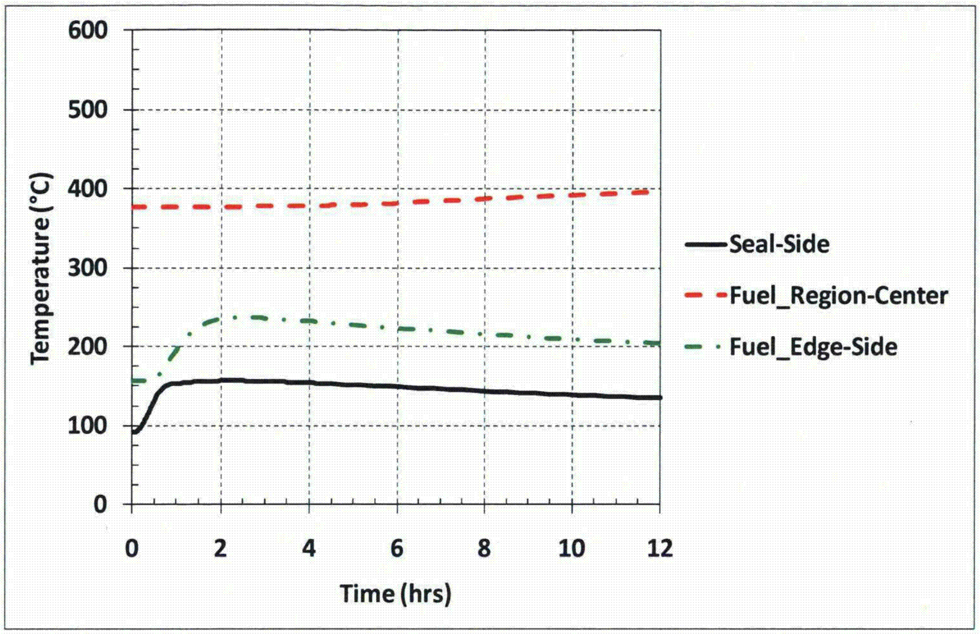


Figure 4-10. Temperature of key cask regions, Rail-Steel cask undergoing REGULATORY CAFE fire.

The uniform external heating produces an even temperature response around the circumference of the cask. However, the realistic uneven fire heating of the exterior produces temperatures 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 4-11.

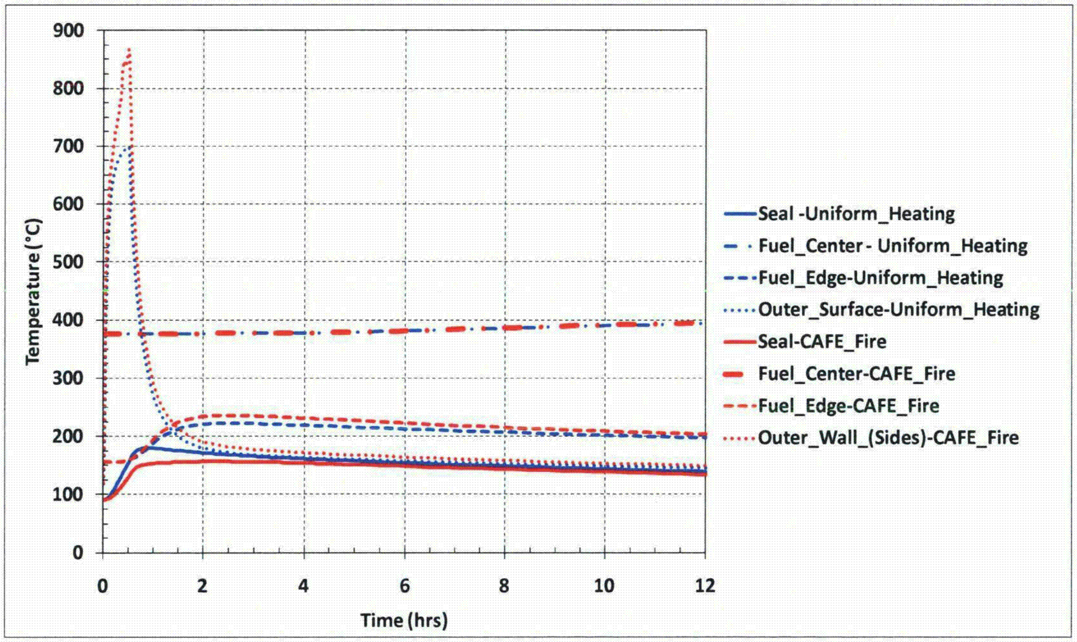
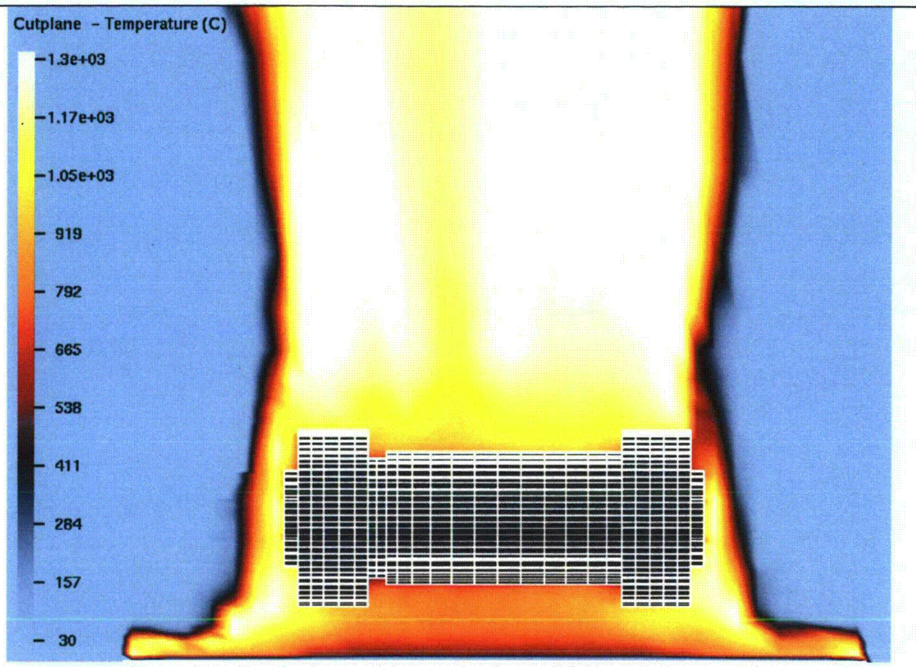


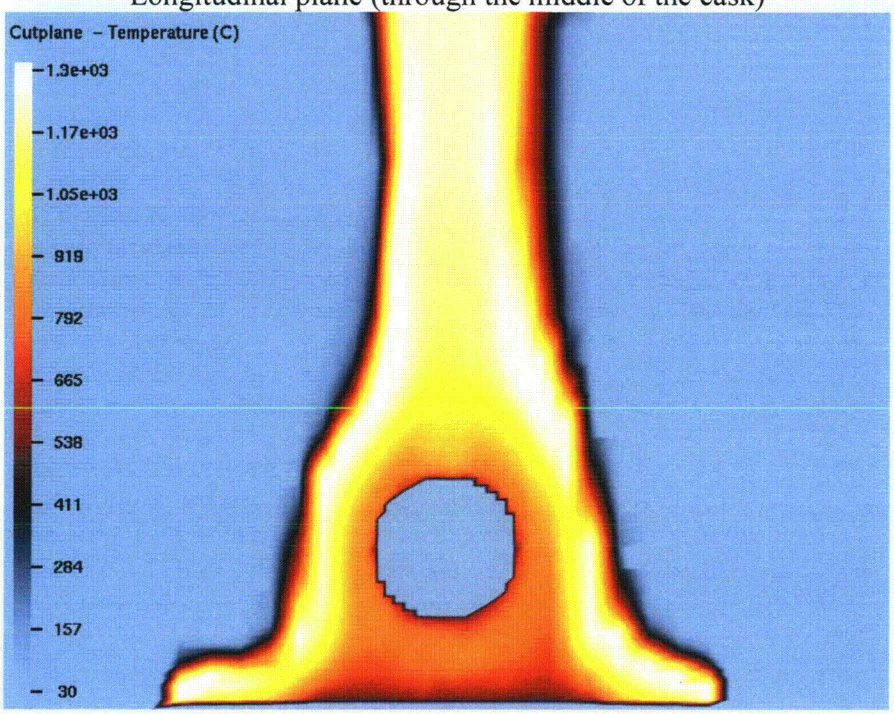
Figure 4-11. Comparison of regulatory fire analysis for Rail-Steel cask: Uniform heating vs. CAFE fire.

Figure 4-11 illustrates that the uniform heating thermal environment described in 10 CFR 71.73 heats up the seal region of the Rail-Steel cask more than a real fire may, even though a real fire can impart to the cask a temporary and localized thermal environment that is hotter than 800°C. A real fire applies a time- and space-varying thermal load to an object engulfed by it. In particular, large fires have an internal region where fuel 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 pool fires, where much of the oxygen is consumed in the perimeter of the plume region. Since combustion is inefficient inside the vapor dome, this region stays cooler than the rest of the fire 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 4-12 and Figure 4-13.



Longitudinal plane (through the middle of the cask)



Transverse plane (at mid-length of the cask)

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

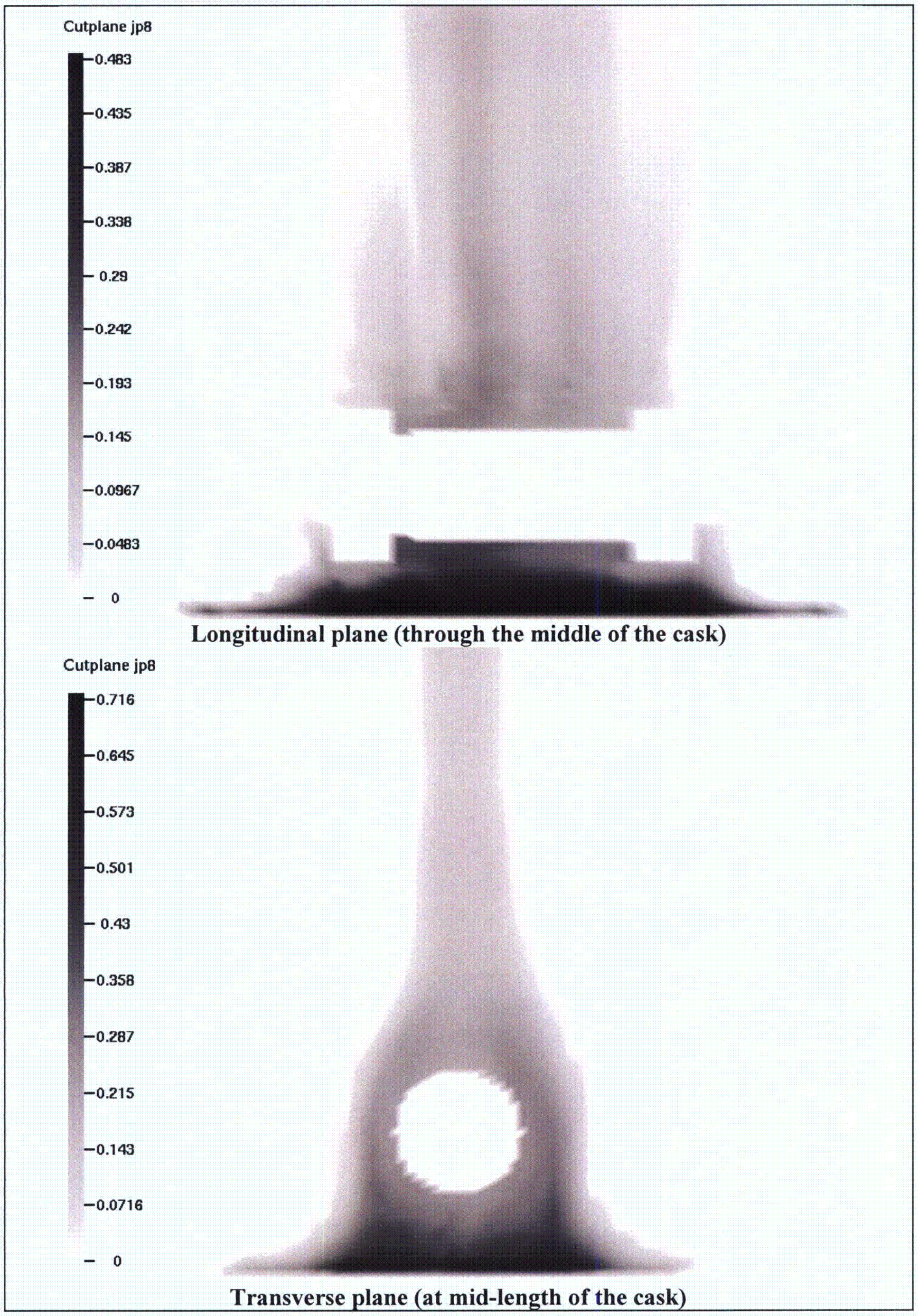


Figure 4-13. Fuel concentration plots from the regulatory CAFE fire analysis.

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

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

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

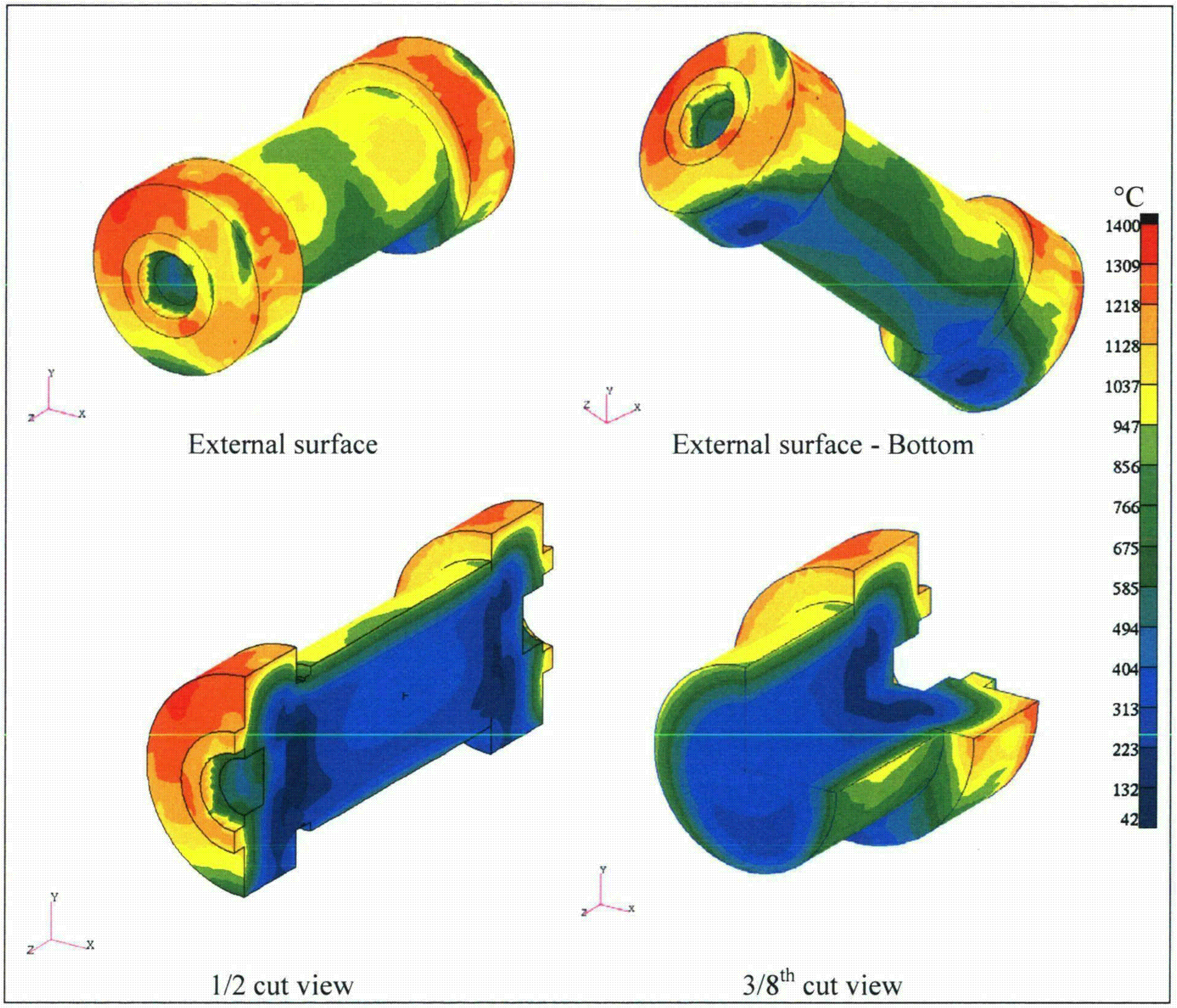


Figure 4-14. Temperature distribution of the Rail-Steel cask at the end of the 3-hour concentric CAFE fire with cask on ground.

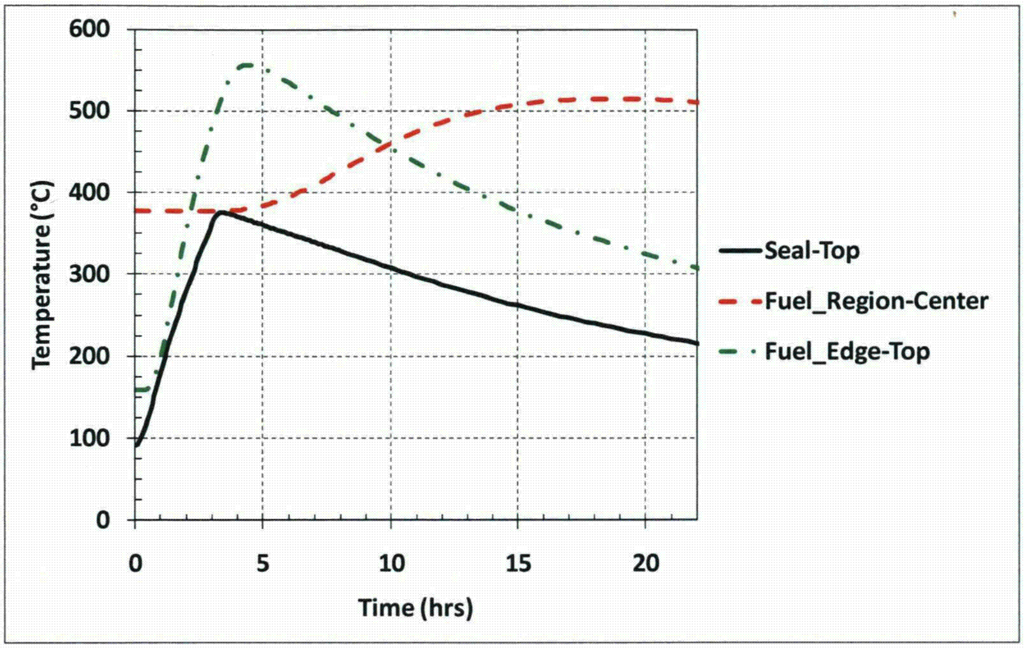
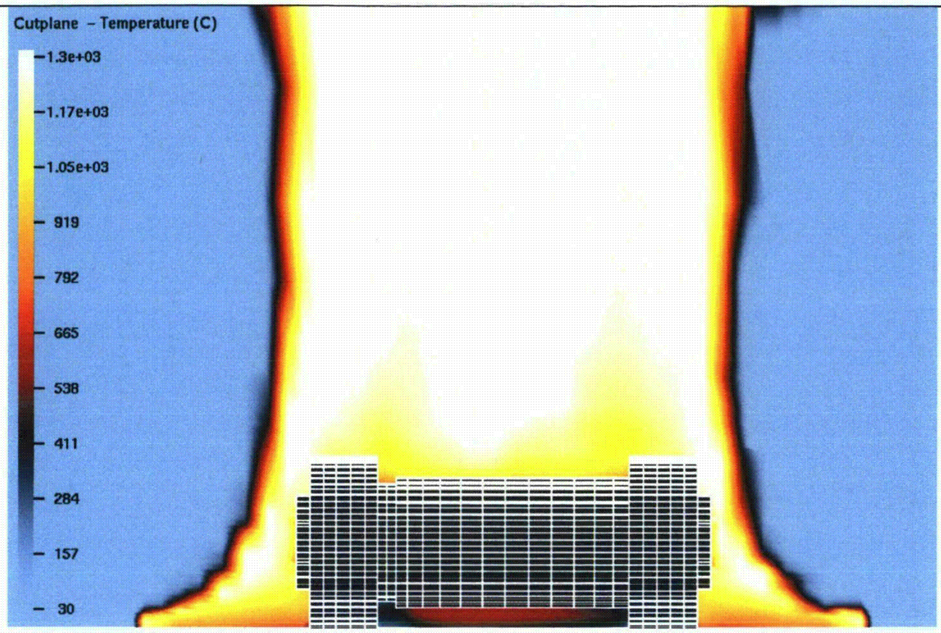
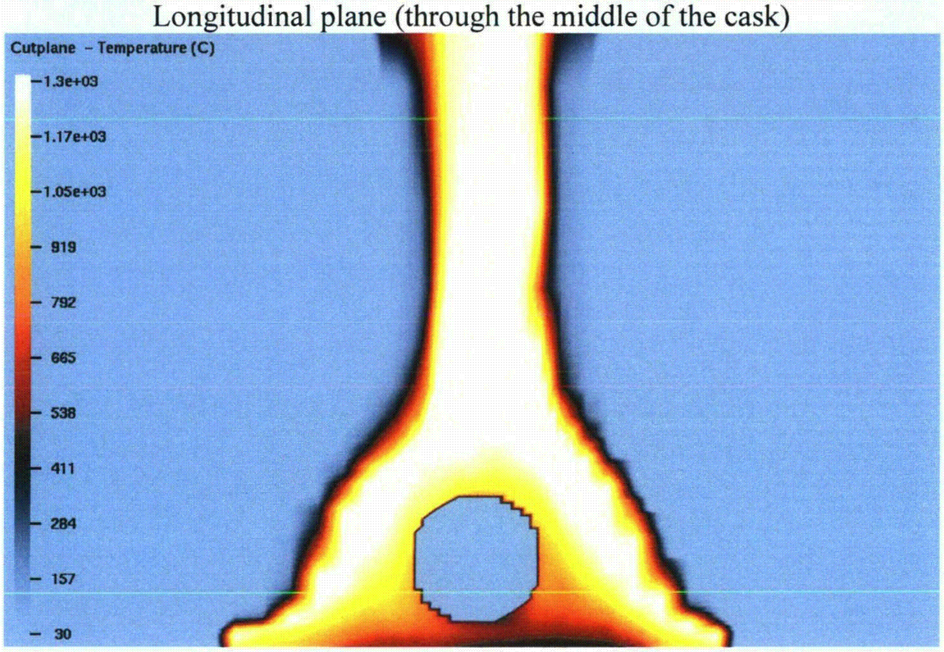


Figure 4-15. Temperature of key cask regions, Rail-Steel cask with cask on ground, concentric fire.

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



Longitudinal plane (through the middle of the cask)



Transverse plane (at mid-length of the cask)

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

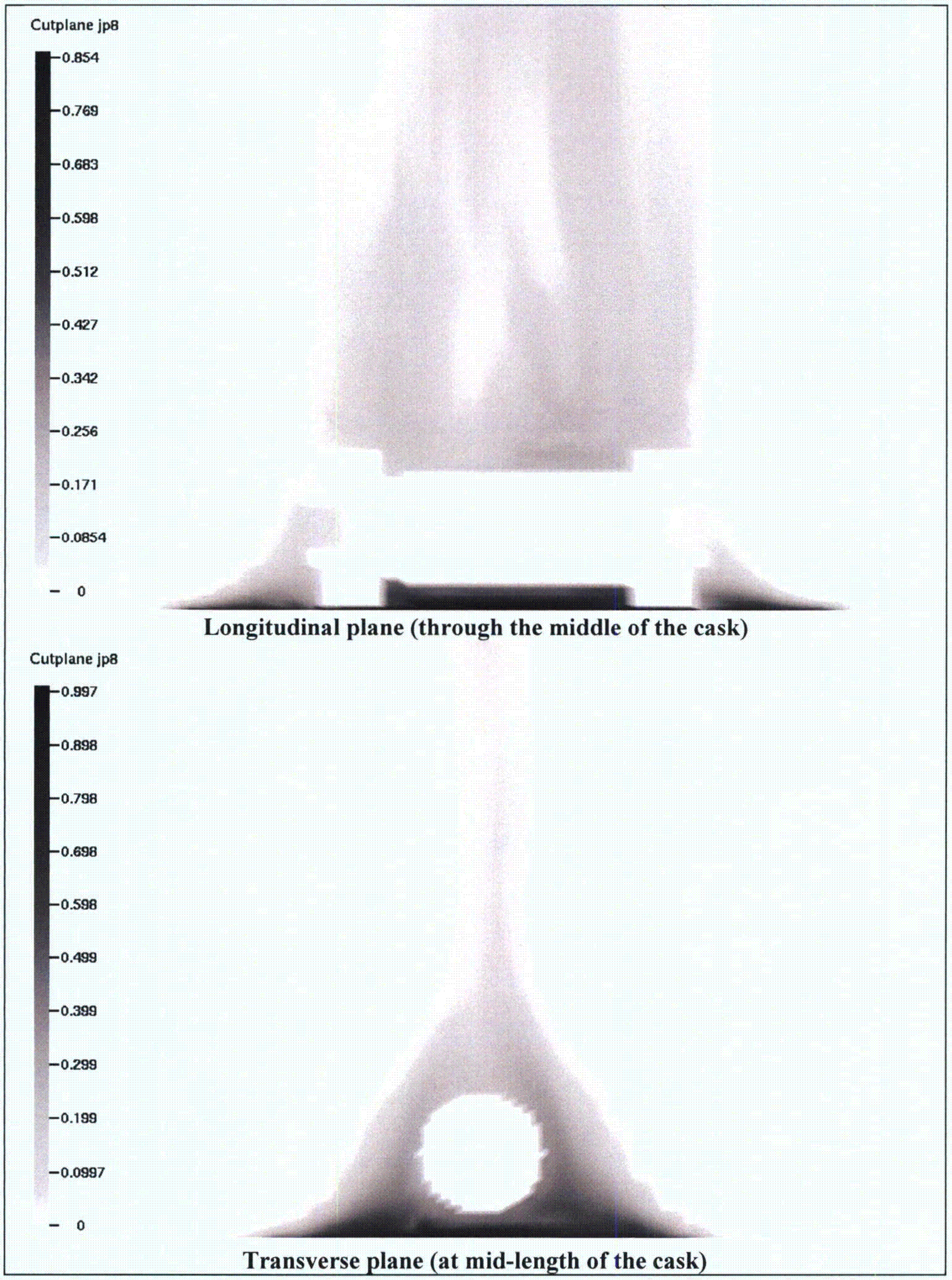


Figure 4-17. Fuel concentration plots from the CAFE fire analysis of the cask on ground.

The results of the offset fire analyses are summarized in Figure 4-18 through Figure 4-21. In the case of the 3-meter offset, the side of the cask facing the fire received heat by thermal radiation. The heat absorbed by the cask during the 3-hour exposure caused the temperature of the cask to rise as depicted in Figure 4-18 and Figure 4-19. Similarly, the 18-meter offset fire caused the cask temperature to rise as illustrated in Figure 4-20 and Figure 4-21. These results show that offset fires, even as close to the cask as three meters, do not represent a threat to this thermally massive SNF transportation cask. The maximum temperatures observed in the seal and fuel SNF region did not reach their temperature limits. Therefore, offset fire scenarios will not cause this package to release radioactive material.

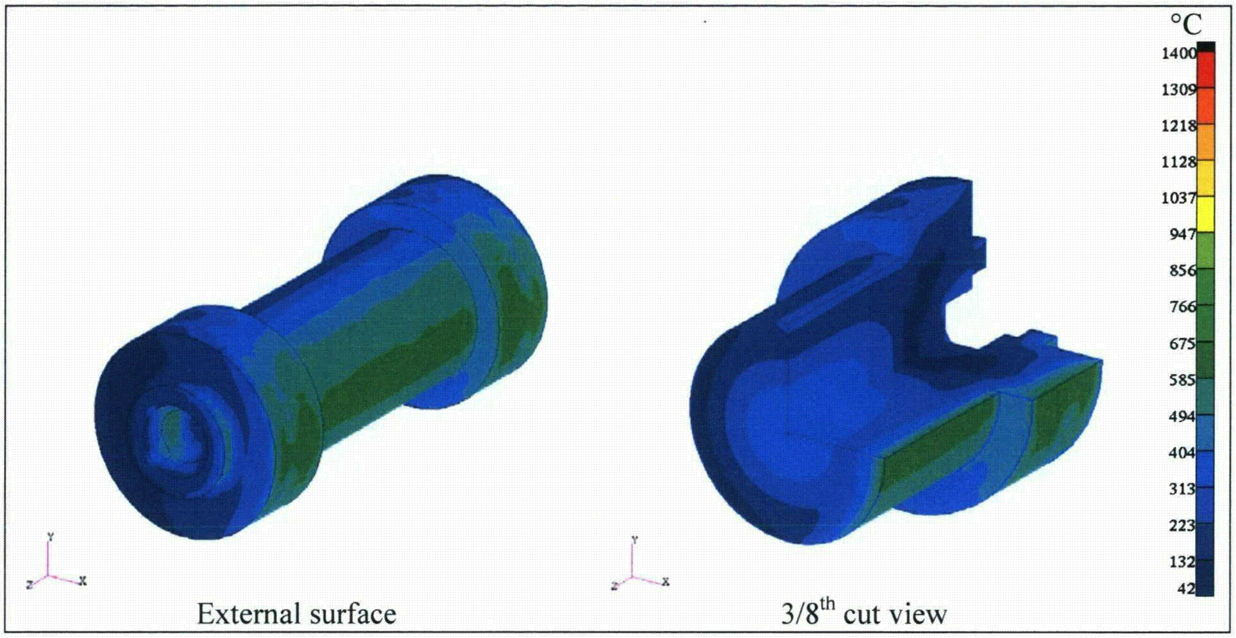


Figure 4-18. Temperature distribution of the Rail-Steel cask at the end of the 3-hour, 3m offset CAFE fire with cask on ground.

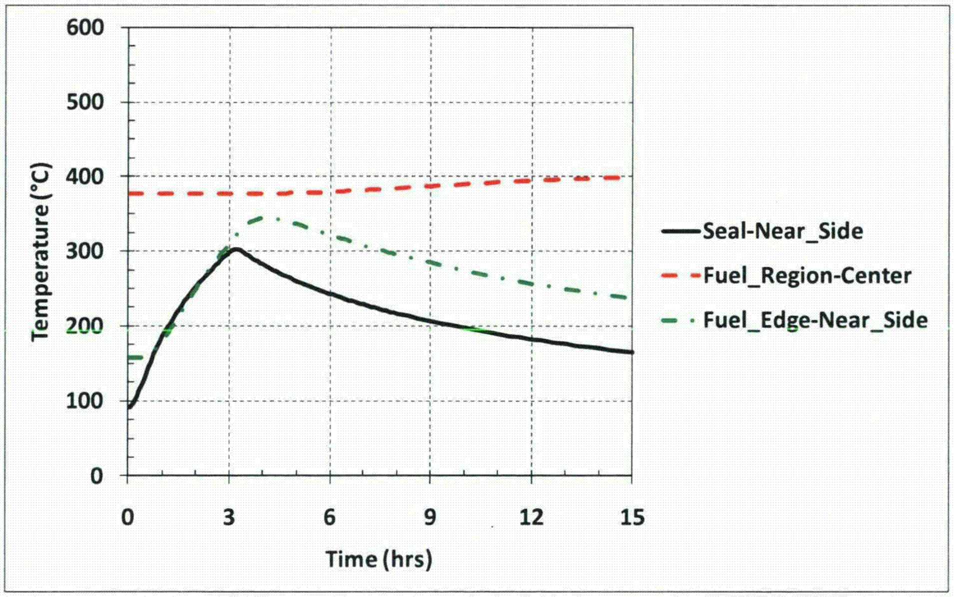


Figure 4-19. Temperature of key cask regions, Rail-Steel cask with Cask on ground, 3m offset fire.

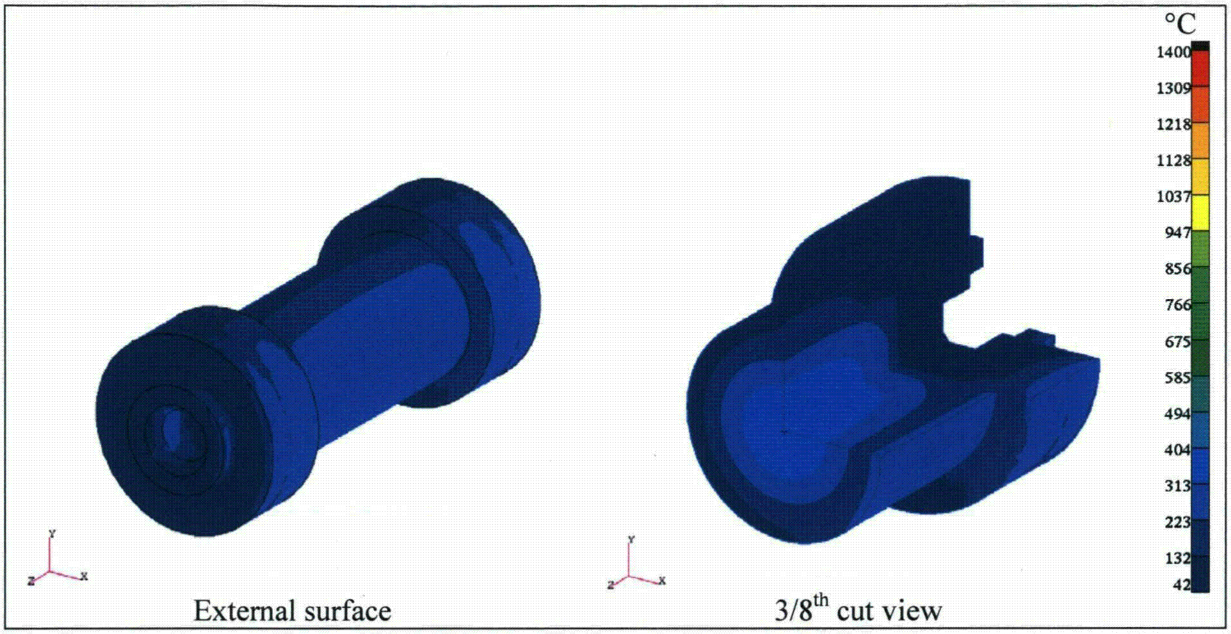


Figure 4-20. Temperature distribution of the Rail-Steel cask at the end of the 3-hour 18m offset CAFE fire with cask on ground.

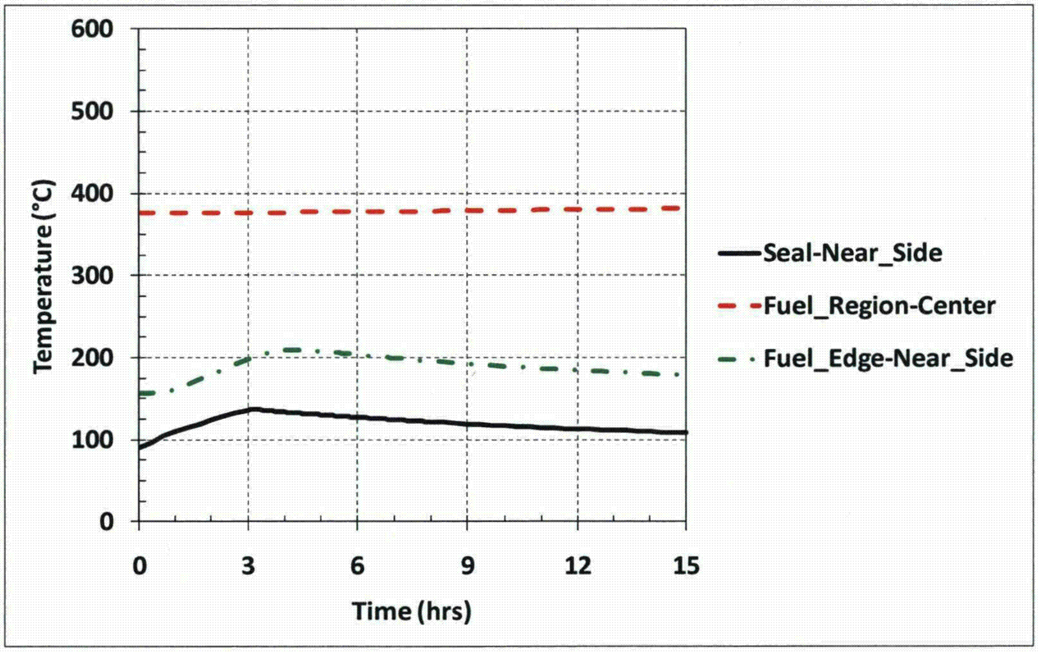


Figure 4-21. Temperature of key cask regions, Rail-Steel cask with cask on ground, 18m offset fire.

Summary of Rail-Steel Cask Analysis Results

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

4.3.5 Rail-Lead Cask Results

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

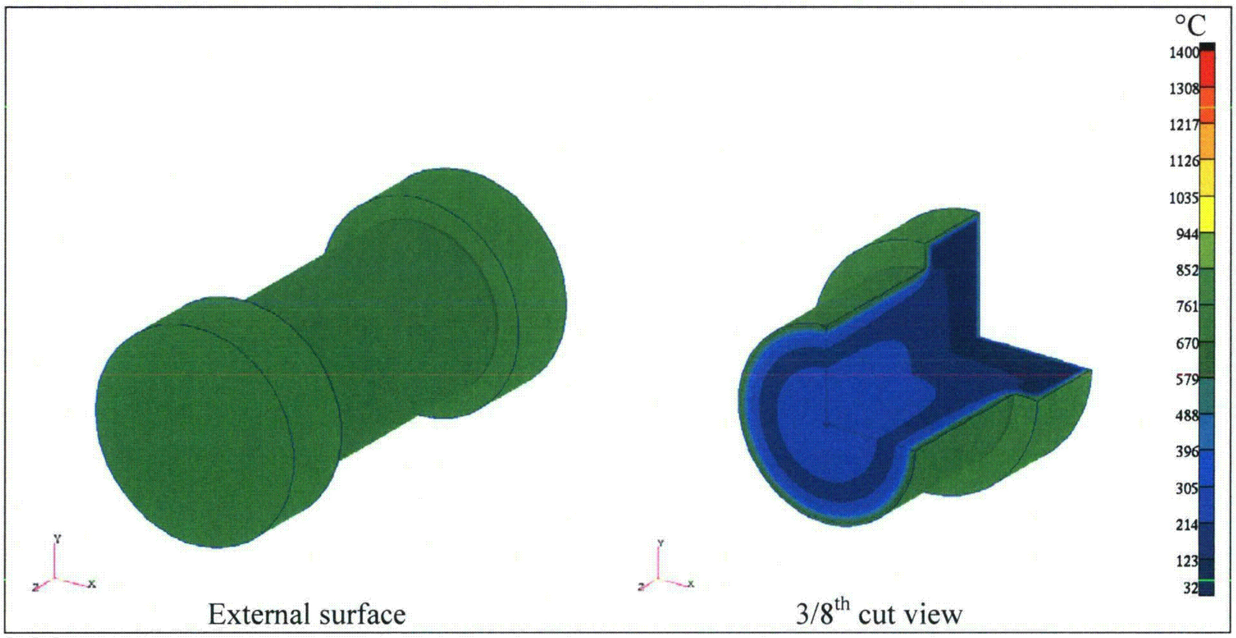


Figure 4-22. Temperature distribution of the Rail-Lead cask at the end of the 30-minute 800°C regulatory uniform heating.

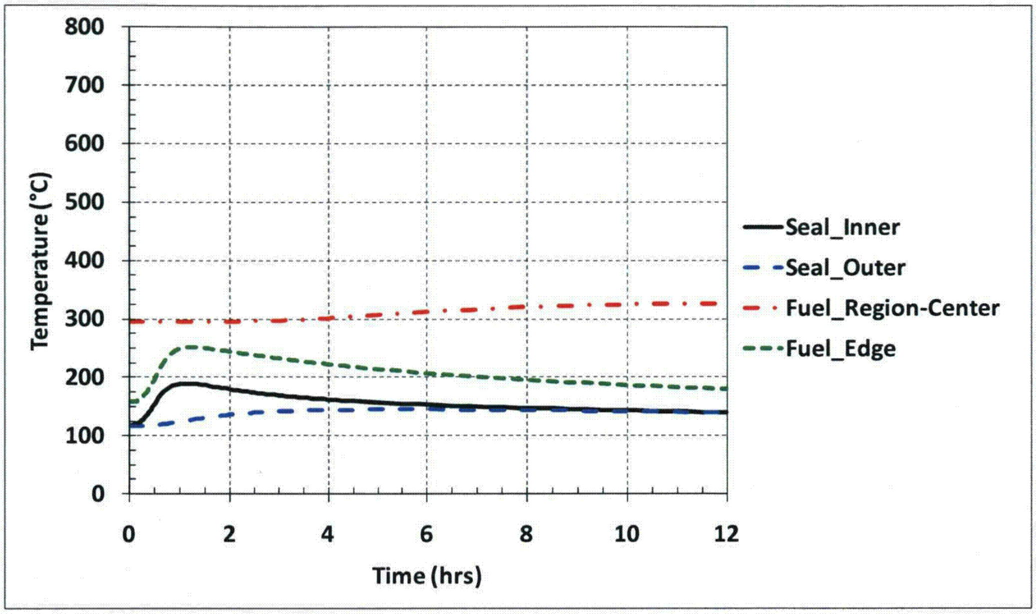


Figure 4-23. Temperature of key cask regions, Rail-Lead cask undergoing regulatory uniform heating.

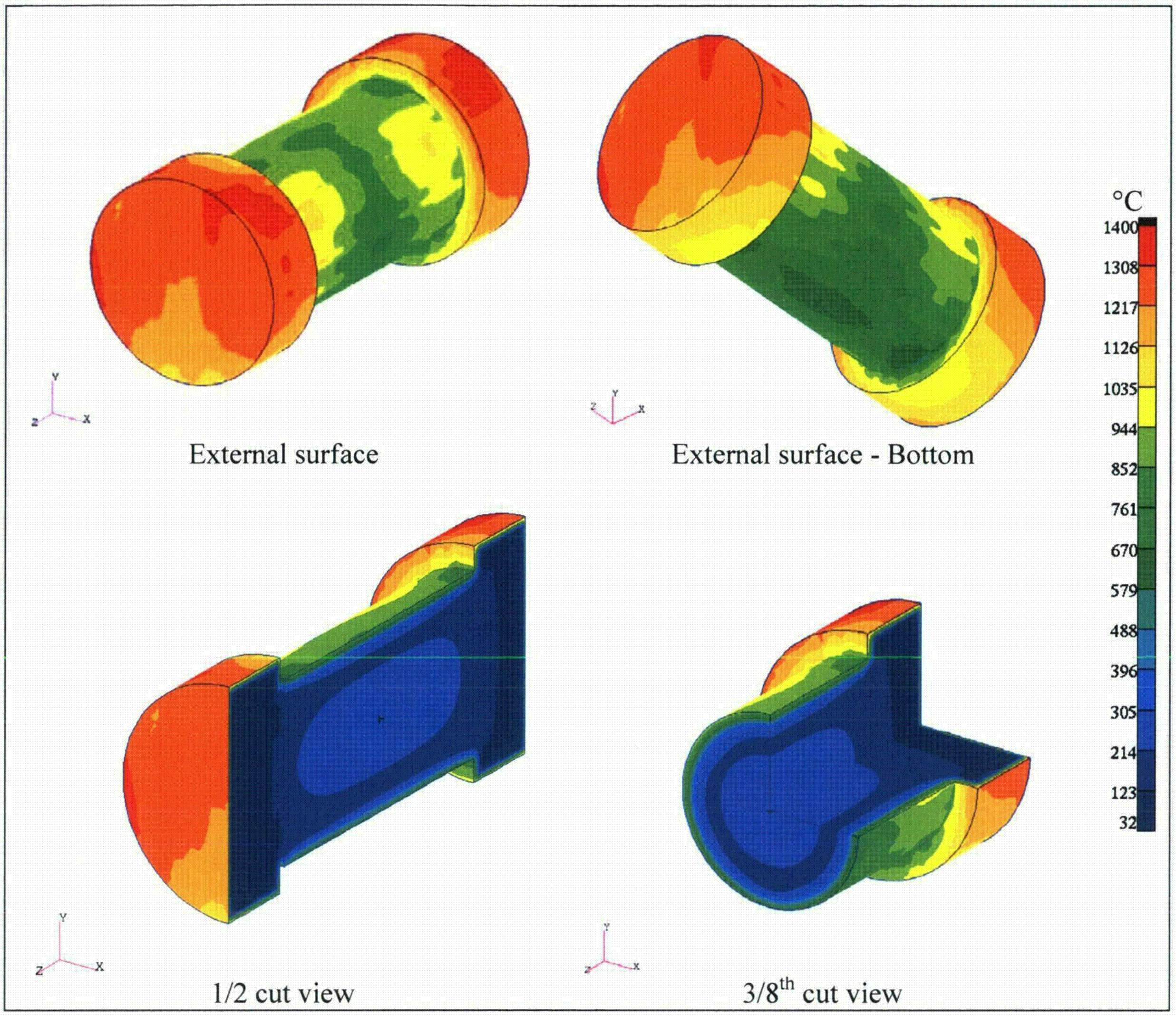


Figure 4-24. Temperature distribution of the Rail-Lead cask at the end of the 30-minute regulatory CAFE fire.

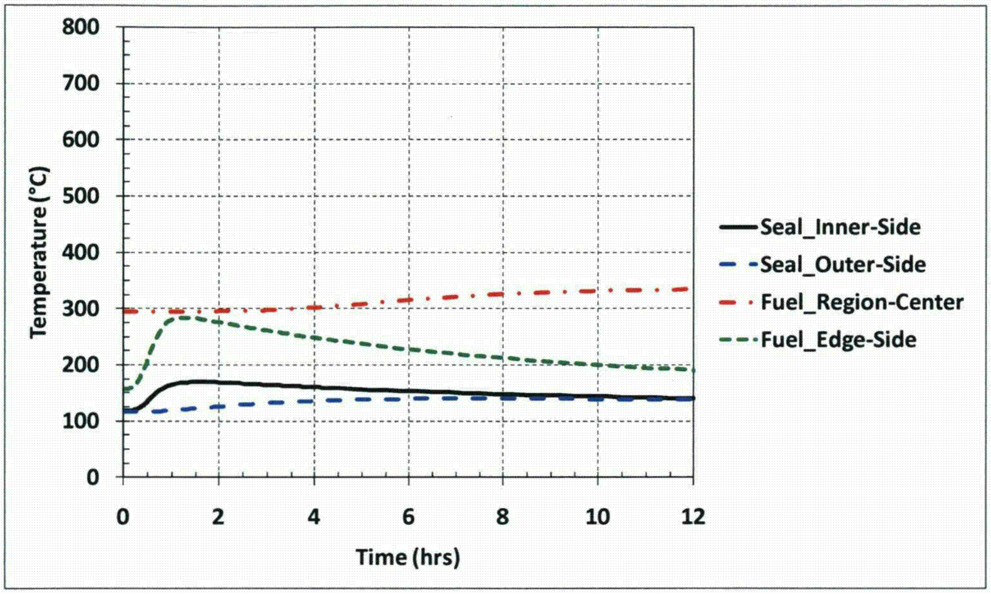


Figure 4-25. Temperature of key cask regions, Rail-Lead cask in regulatory CAFE fire.

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

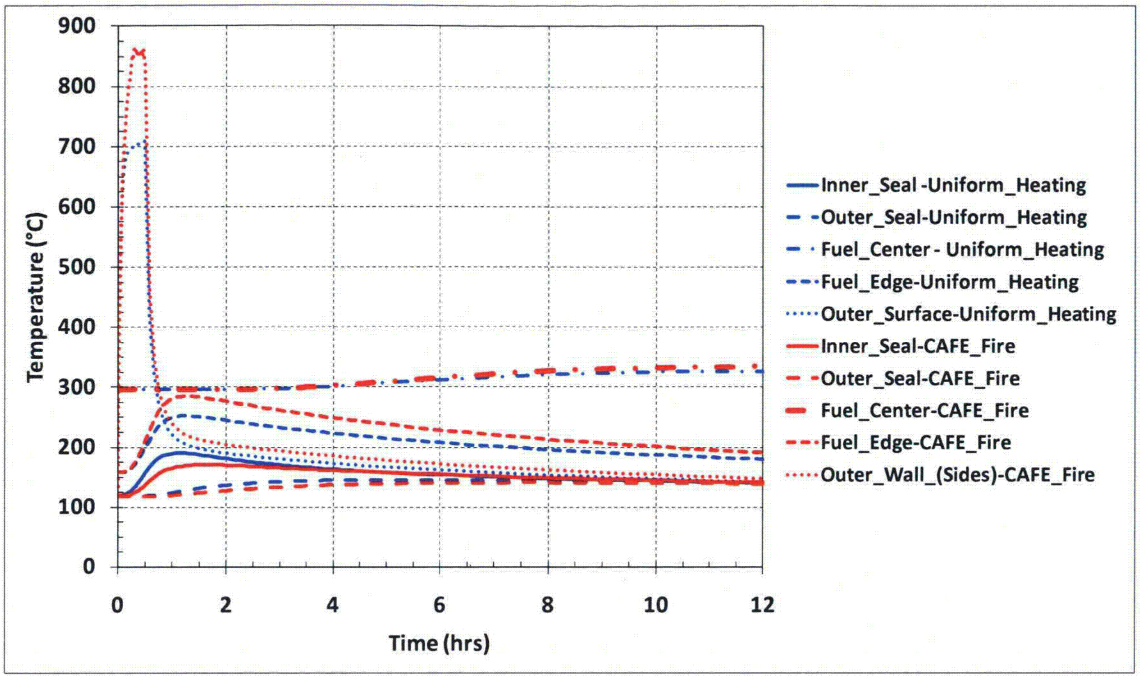


Figure 4-26. Comparison of regulatory fire analysis, Rail-Lead cask: Uniform heating vs. CAFE fire.

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

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

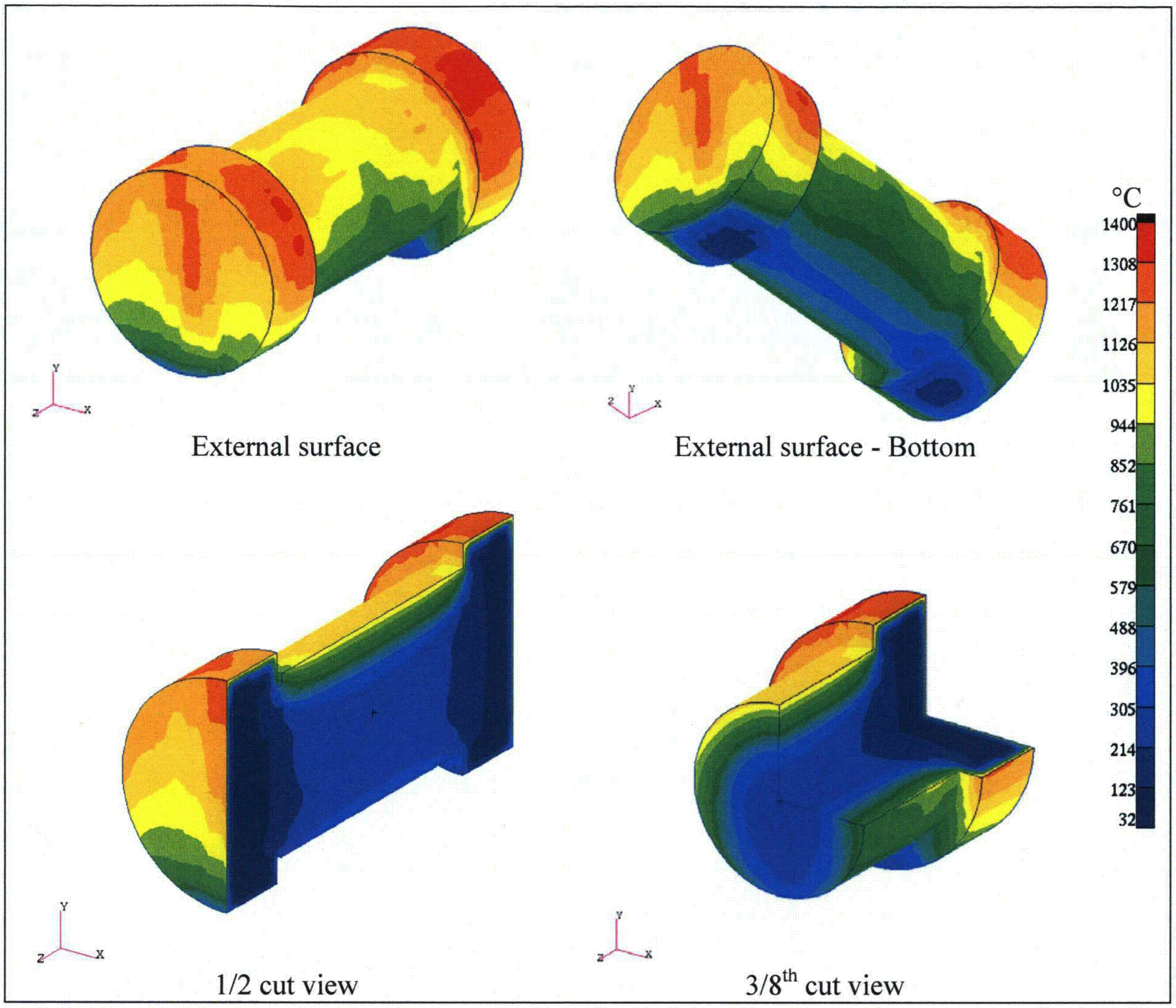


Figure 4-27. Temperature distribution of the Rail-Lead cask at the end of the 3-hour concentric CAFE fire with cask on ground.

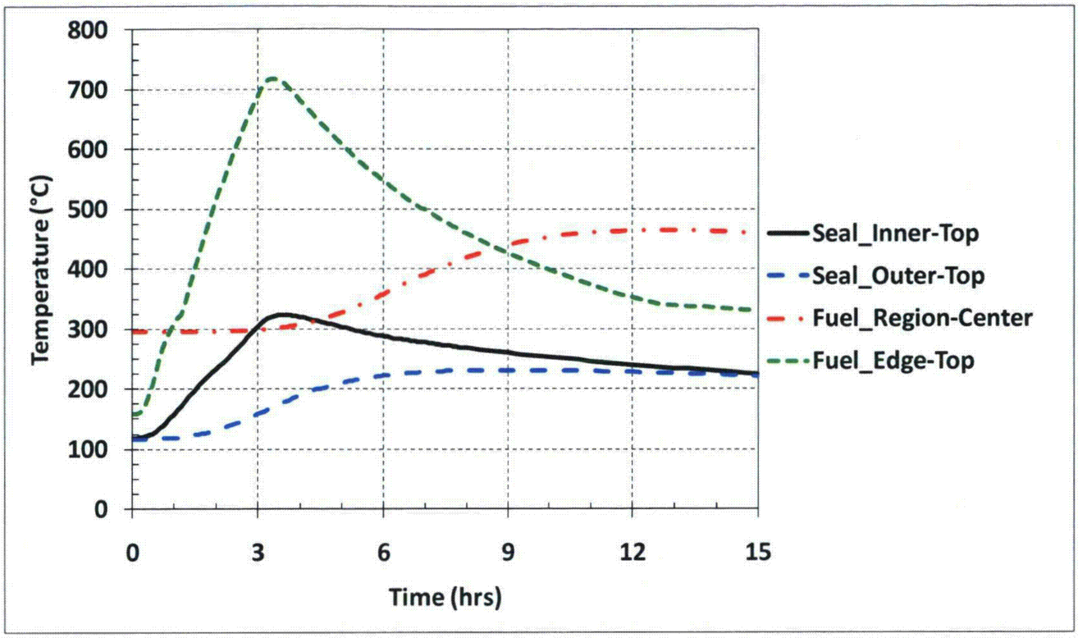


Figure 4-28. Temperature of key cask regions, Rail-Lead cask with cask on ground, concentric fire.

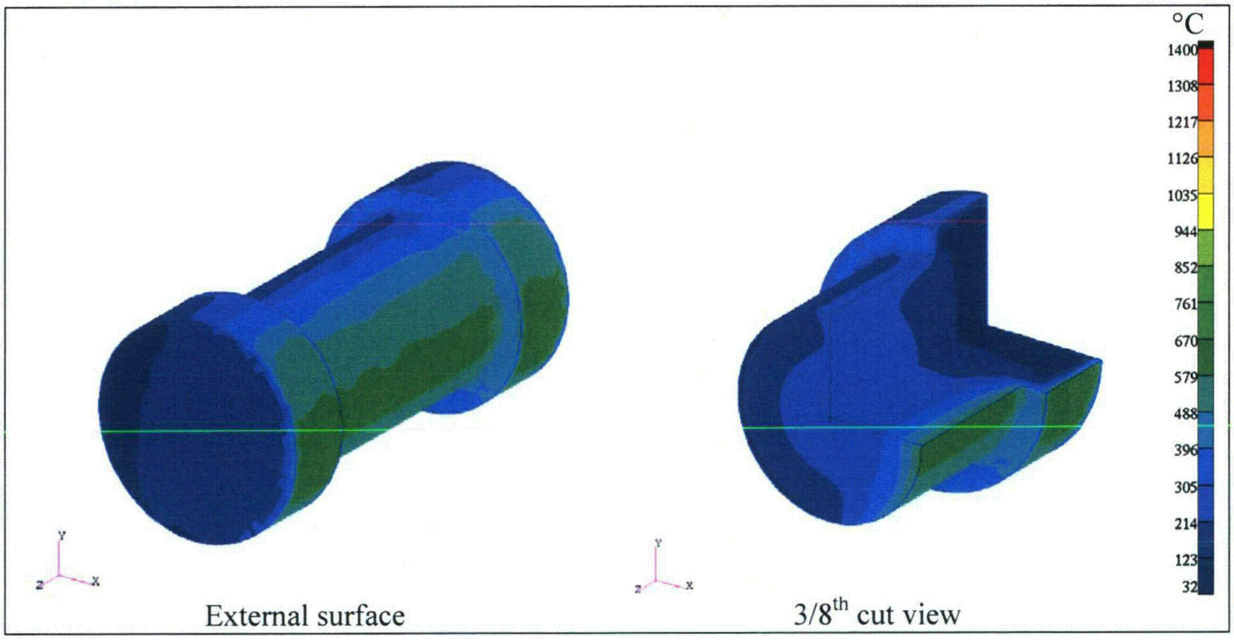


Figure 4-29. Temperature distribution of the Rail-Lead cask at the end of the 3-hour 3m offset CAFE fire with cask on ground.

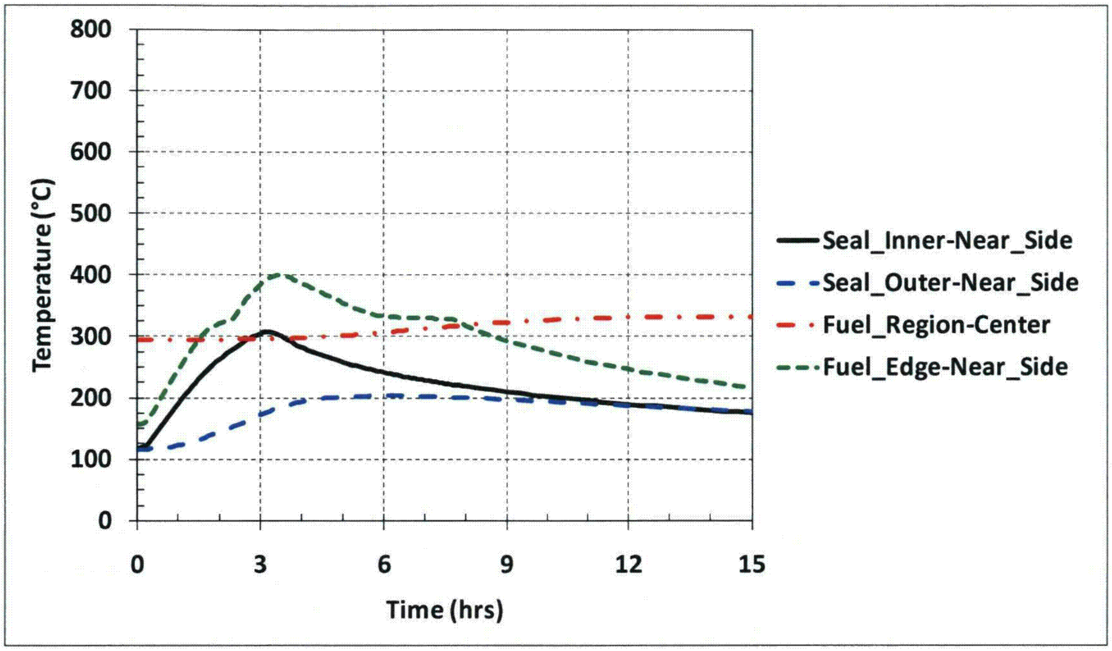


Figure 4-30. Temperature of key cask regions, Rail-Lead cask with Cask on ground, 3m offset fire.

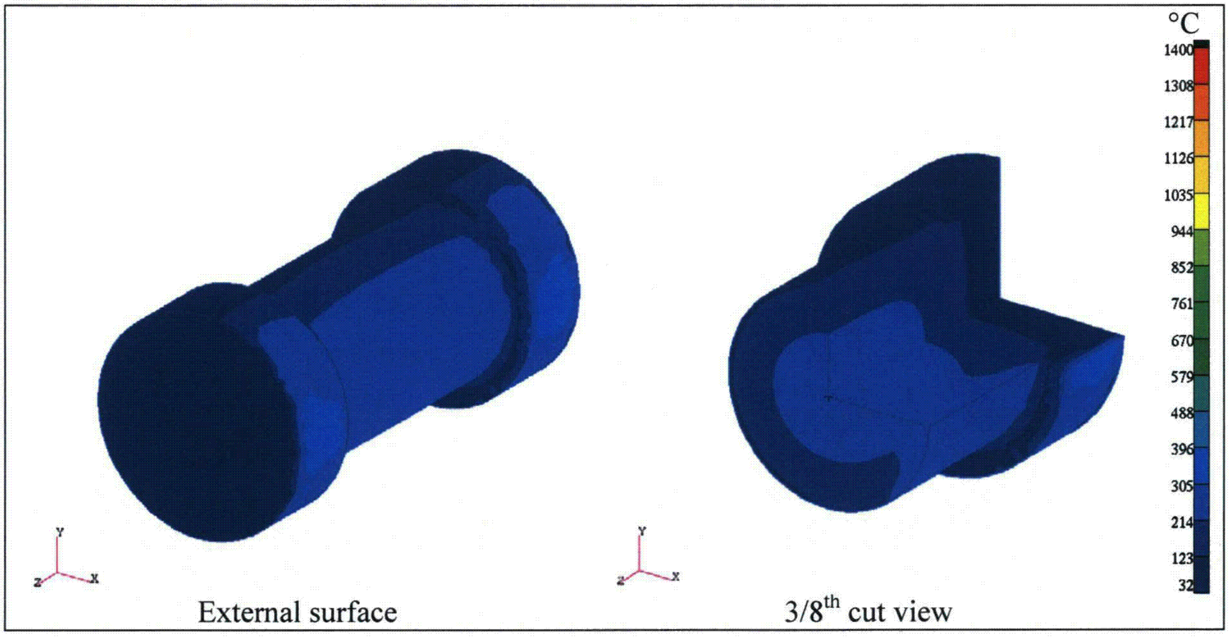


Figure 4-31. Temperature distribution of the Rail-Lead cask at the end of the 3-hour 18m offset CAFE fire with cask on ground.

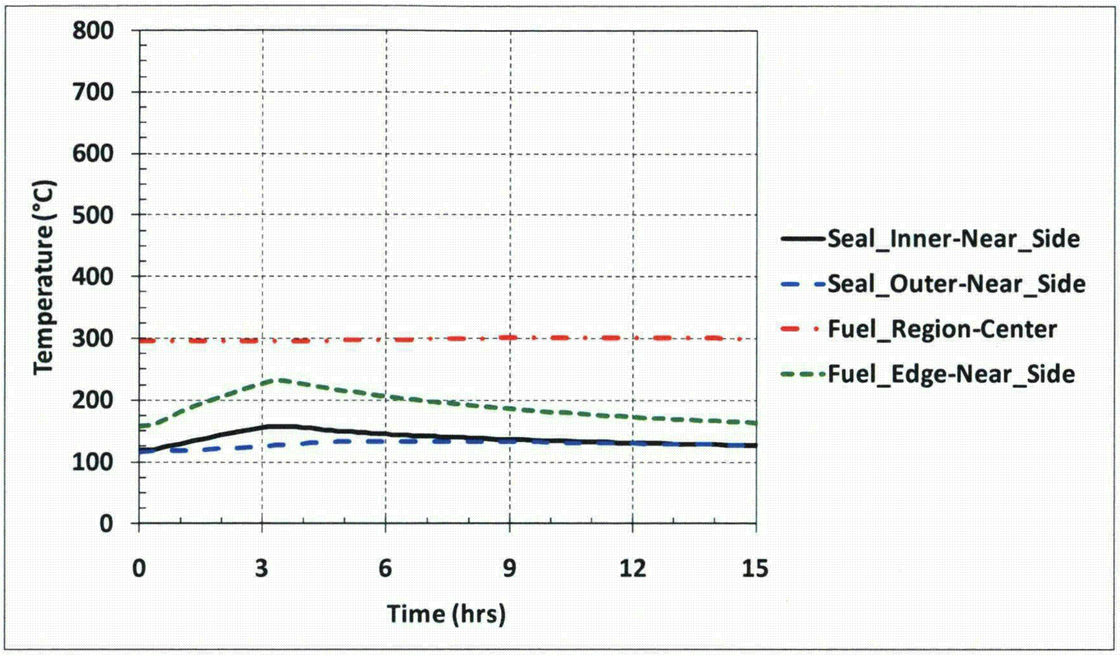


Figure 4-32. Temperature of key cask regions, Rail-Lead cask with Cask on ground, 18m offset fire.

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

Melting of the Lead Gamma Shield

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

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

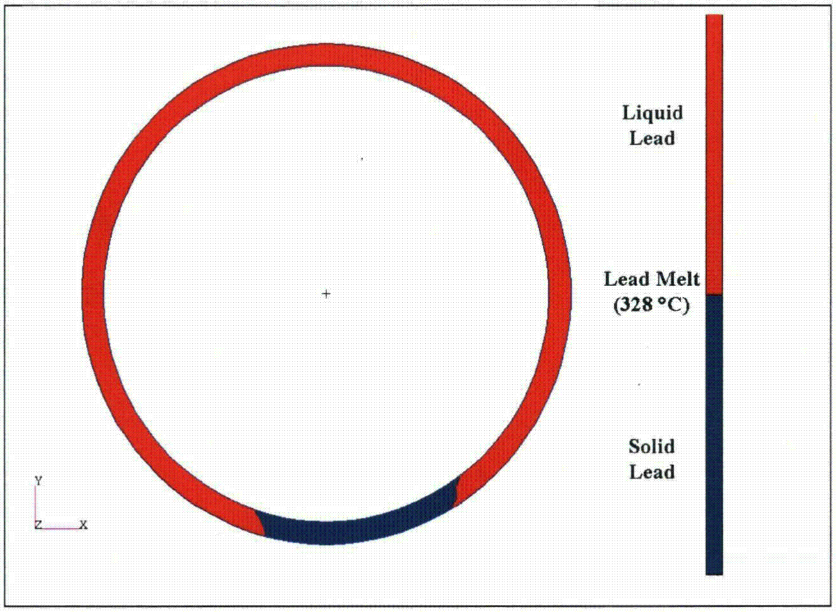


Figure 4-33. Rail-Lead cask lead gamma shield region – maximum lead melt at the middle of the cask. – Scenario: Cask on ground, 3-hour concentric pool fire.

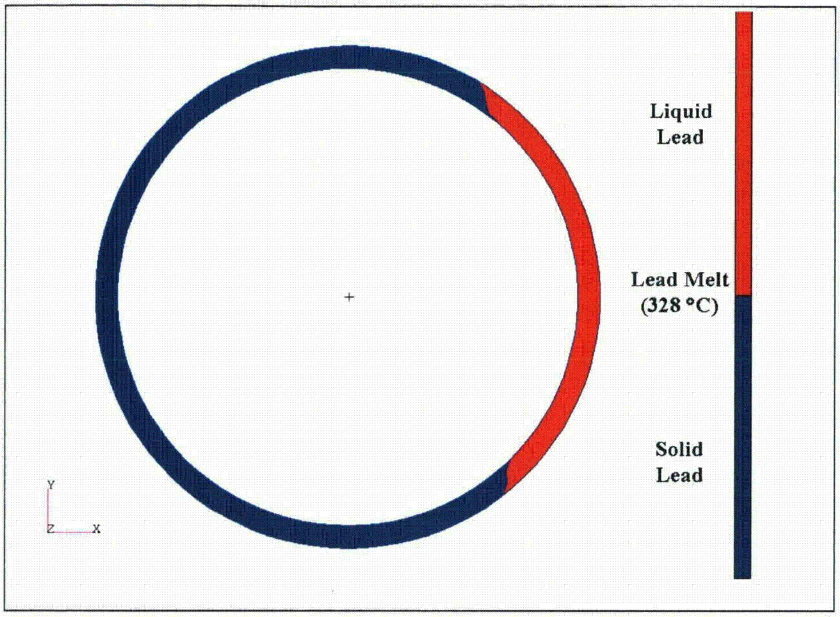


Figure 4-34. Rail-Lead cask lead gamma shield region – maximum lead melt at the middle of the cask. – Scenario: Cask lying on ground, 3-hour 3m offset pool fire.

Summary of Rail-Lead Cask Analysis Results

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

4.4 Truck Cask Analysis

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

4.4.1. Simulation of the Truck Cask

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

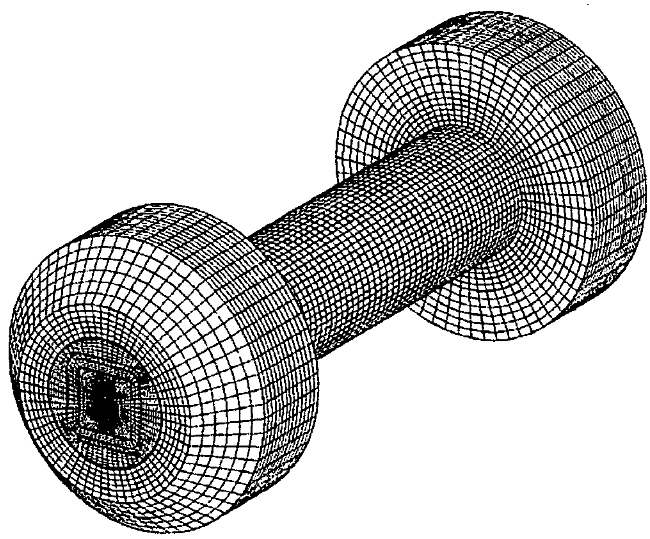


Figure 4-35. Finite element model of the Truck-DU cask.

4.4.2. Simulation of the Spent Nuclear Fuel Region

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

4.4.3. Truck-DU Cask Results

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

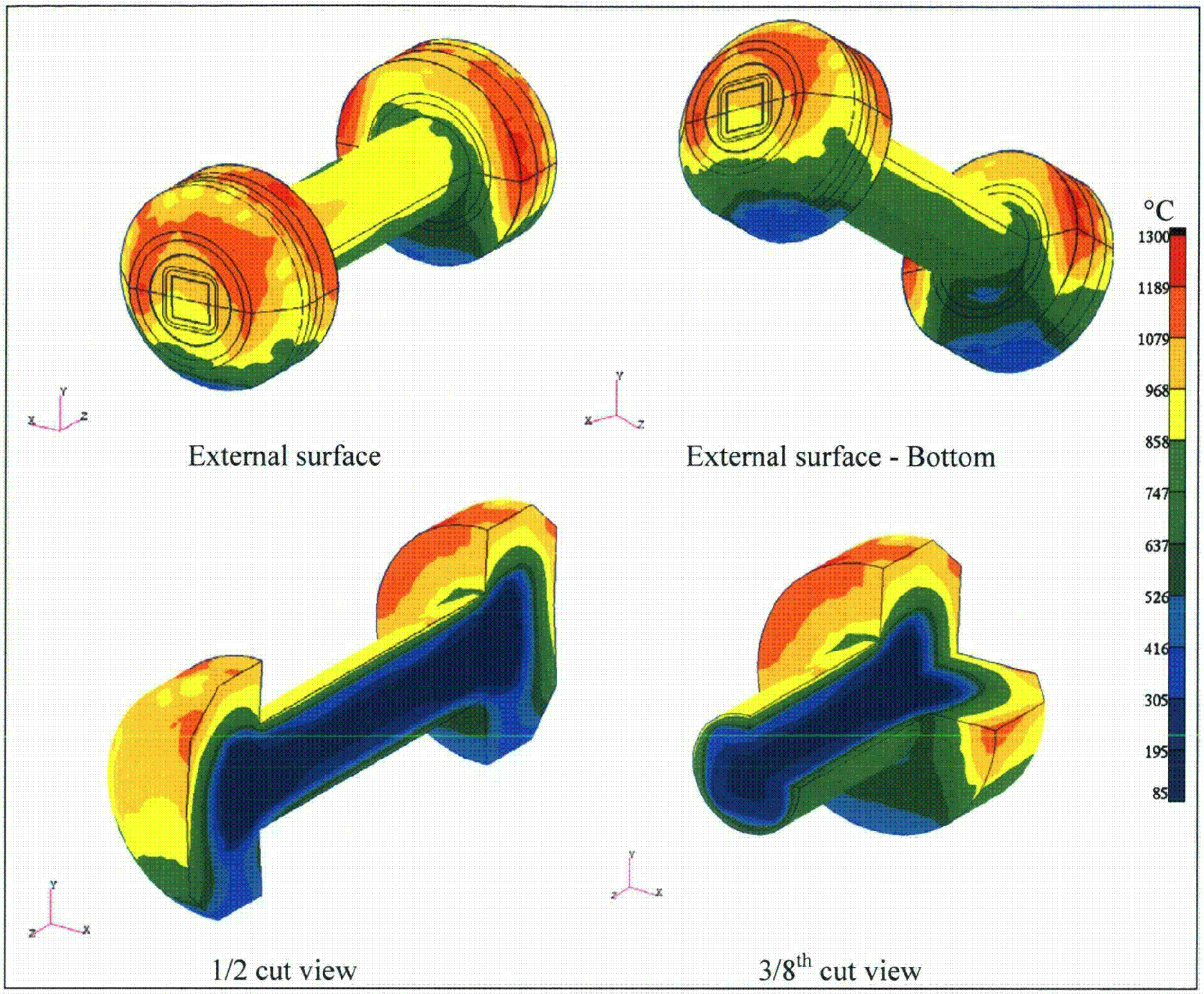


Figure 4-36. Temperature distribution of the Truck-DU cask at the end of the 1-hour concentric CAFE fire with cask on ground.

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

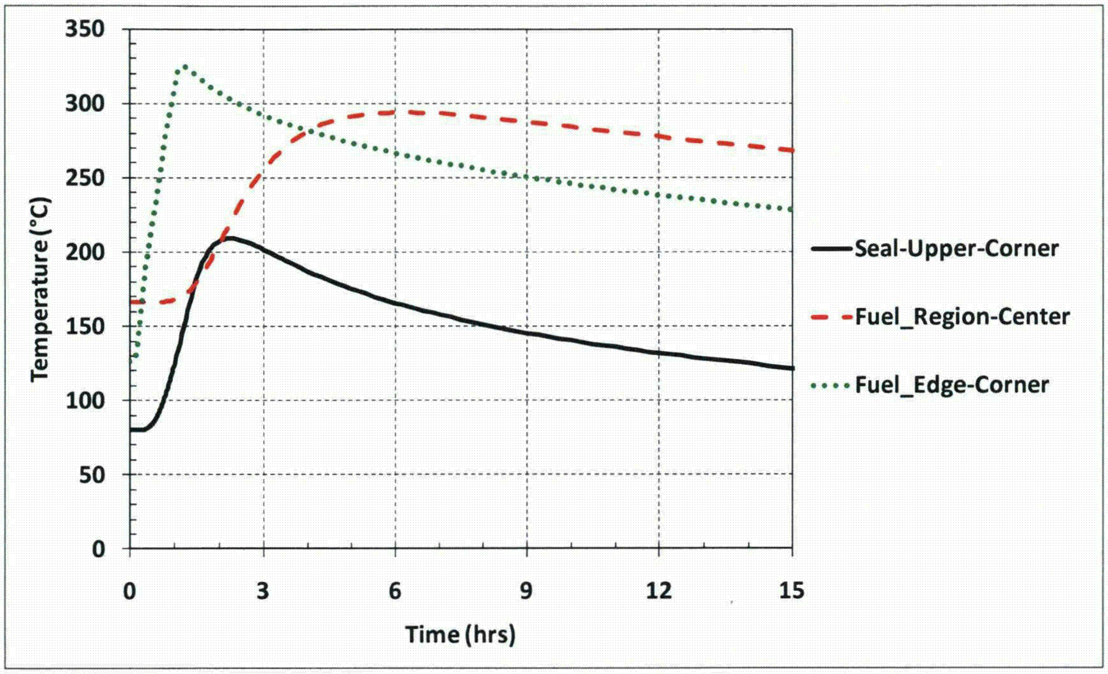
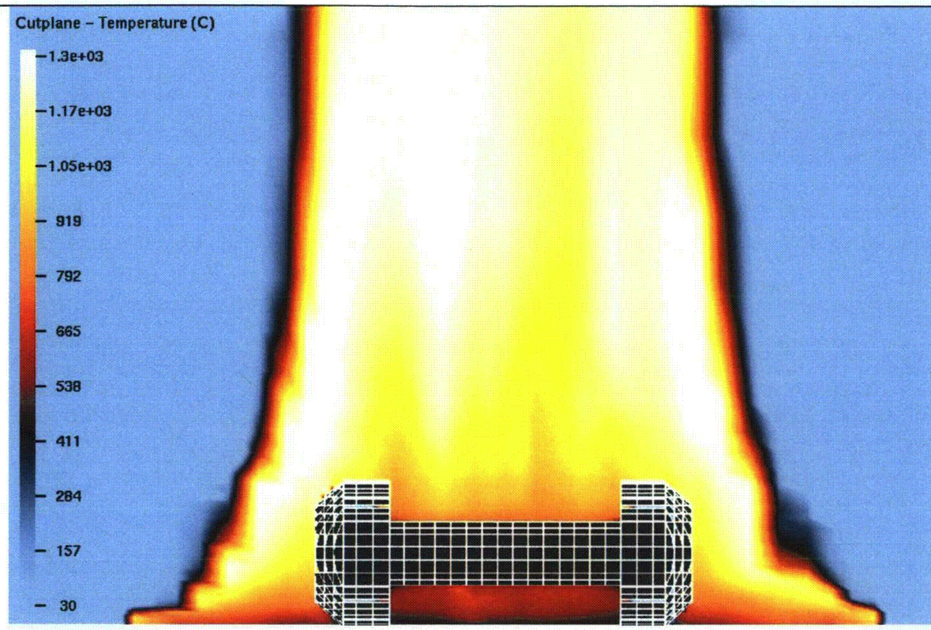
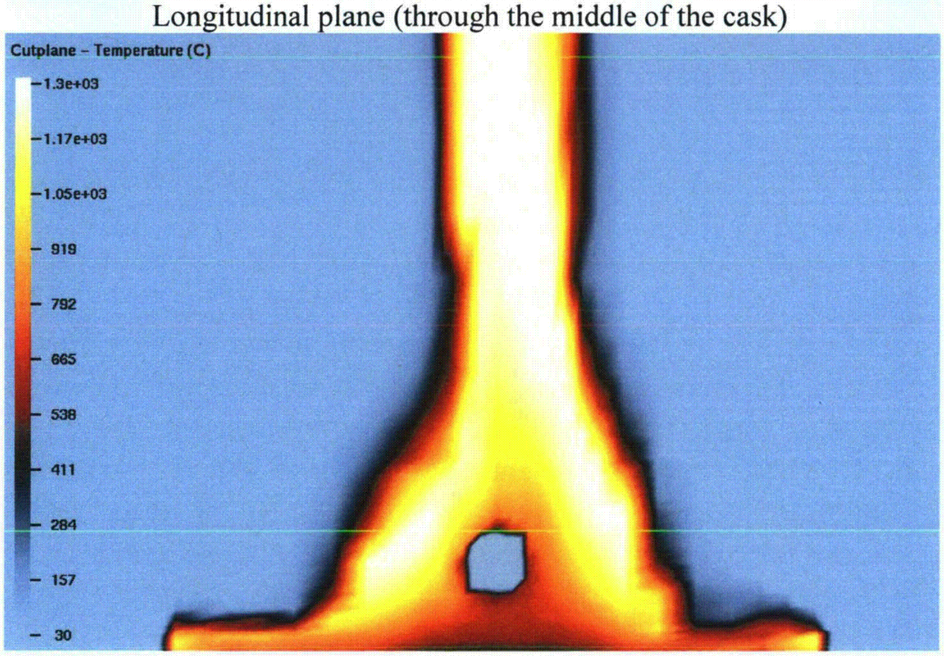


Figure 4-37. Temperature of key cask regions, Truck-DU cask with cask on ground, concentric fire.

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

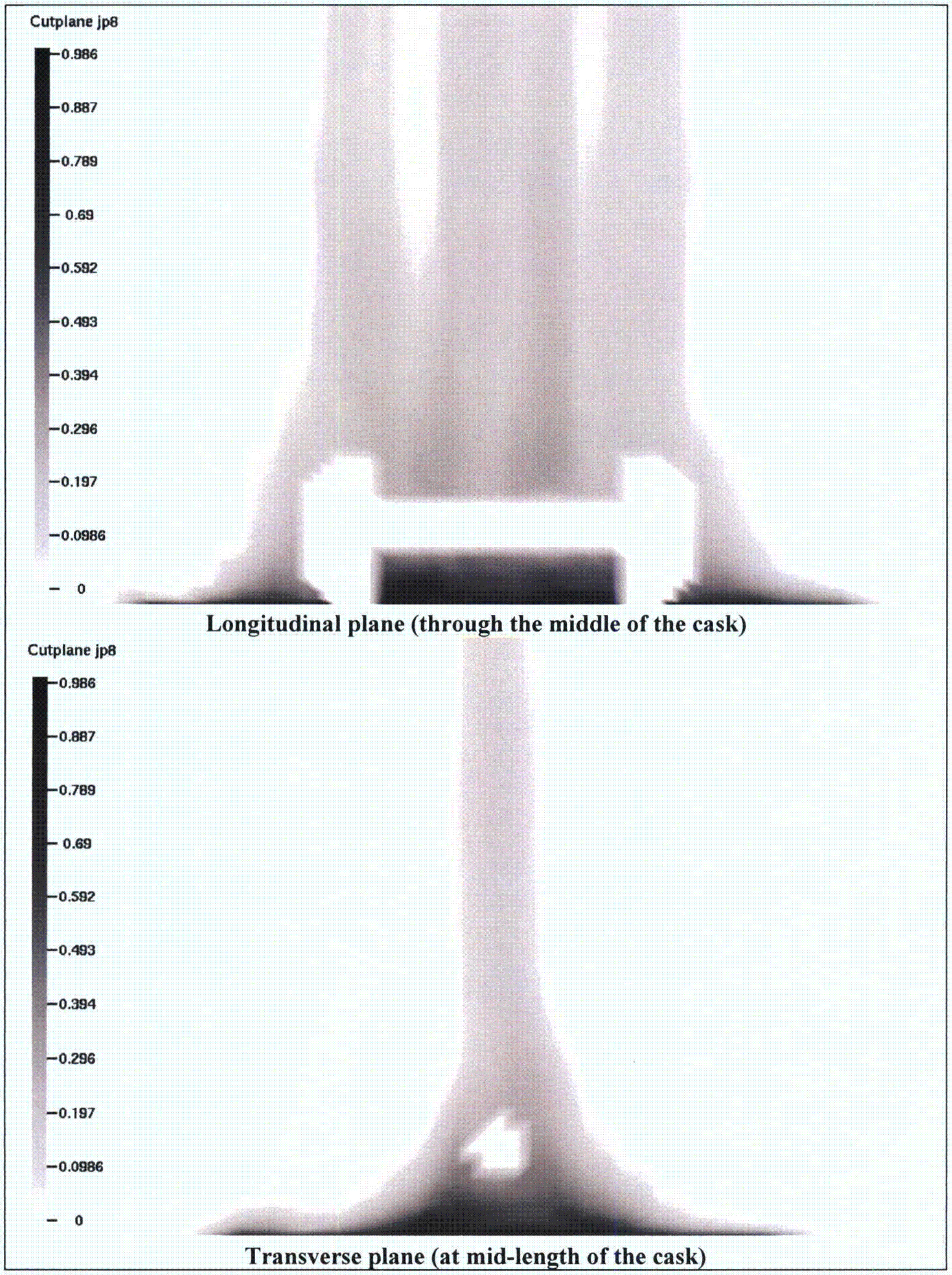


Longitudinal plane (through the middle of the cask)



Transverse plane (at mid-length of the cask)

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



Longitudinal plane (through the middle of the cask)

Transverse plane (at mid-length of the cask)

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

Summary of Truck-DU Cask Analysis Results

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

4.5 Conclusions

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

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

Analyses of these four fire accident scenarios are performed for the Rail-Steel and the Rail-Lead casks. An analysis of a Truck-DU cask on the ground concentric with a hydrocarbon fuel pool sufficiently large to engulf the cask is also performed. Probable worst-case fire accident scenarios for a rail cask transported by railway and for a truck cask transported by roadway were represented within the cases analyzed.

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

CHAPTER 5

TRANSPORTATION ACCIDENTS

5.1 Types of Accidents and Incidents

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

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

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

An accident happens at a particular spot on the route. When the accident happens, the vehicle carrying the spent fuel cask stops. Thus, there can be no more than one accident for a shipment. 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. Accidents damaging the fuel but not damaging the cask, and potential consequent risk to workers are not included in this study.

5.2 Accident Probabilities

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

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

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

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

Table 5-1. Illustrations of net probability

Accident Probability for a 3000-Mile Cross-Country Trip[a]	Accident	Conditional Probability[b]	Net Probability Of Accident
0.0165	Truck collision with a gasoline tank truck	$0.82 \times 0.003 = 0.0025$	$0.82 \times 0.003 \times 0.0165 = 0.000041$
0.00138	Rail/truck 50 mph collision at grade crossing	$0.7355 \times 0.985 \times 0.0604 \times 0.0113 = 0.00049$	$0.7355 \times 0.985 \times 0.0604 \times 0.0113 \times 0.00138 = 0.0000068$
0.00087	Railcar falling off bridge at 30 mph	$0.7355 \times 0.2665 \times 0.9887 = 0.194$	$0.7355 \times 0.2665 \times 0.9887 \times 0.00087 = 0.00017$

[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 kilometer (accident frequency) for a particular type of vehicle as recorded by the DOT and reported by the Bureau of Transportation Statistics. Large truck accidents and freight rail accidents are the two data sets used in this analysis. The accident frequency varies somewhat from state to state. The U.S. average for large trucks for the period 1991 to 2007 is 0.0035 accidents per thousand kilometers (km). For rail accidents, the average is 0.00024 per thousand railcar-km (DOT, 2008). The DOT has compiled and validated national accident data for truck and rail from 1971 through 2007, but the accident rates declined definitively between 1971 and the 1990s. For this analysis, rates from 1996 through 2007 are used: 0.0019 accidents per thousand large truck-km and 0.00011 accidents per thousand railcar-km.

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

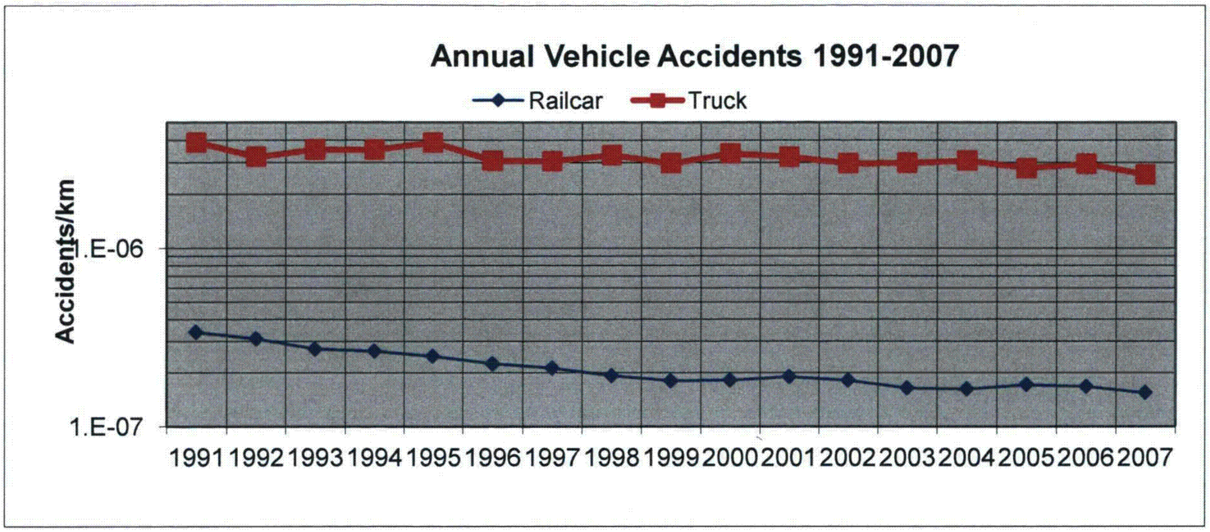


Figure 5-1. Accident frequencies in the U.S. from 1991 until 2007.

As Chapters 3 and 4 show, however, the only accidents that could result in either the loss of radiation shielding or release of radioactive material are rail accidents involving the Rail-Lead cask. These are

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

Whether or not these accidents happen depends on the likelihood (conditional probability) of the accident scenario as well as on the accident frequency. The event trees for truck and rail, Figures V-1 and V-2 of Appendix V, show some of the elements of accident scenarios in each branch of the respective event tree. The dependence on probability is illustrated by Figure V-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.

Table 5-2 shows the conditional probabilities of accidents that could result in a radiation dose to a member of the public and of accidents in which there is neither loss of lead shielding nor a release of radioactive material. 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-Lead cask

Accident Scenario for the Rail-Lead Cask	Conditional probability of gamma shield loss or radioactive material content release exceeding 10 CFR 71.51 quantities
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 loss of lead shielding and no release of radioactive material: Truck-DU and Rail-Steel accidents	0.999991

Loss of lead shielding and radioactive material release from a fire both depend on the same sequence of events that would result in a hot enough fire close enough to the cask to cause the damage. Therefore the conditional probabilities are the same. A more detailed discussion is in Appendix V.

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

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-Lead 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-Steel cask carries canistered fuel and would not release any radioactive material. Neither Truck-DU casks nor Rail Steel casks are lead-shielded, so that shielding loss would not occur.

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

- The external dose rate from the cask (Table 2-1).
- A ten-hour stop (DOE, 2002) at the scene of the accident, until the vehicle and/or cask can be moved safely. Ten hours is a conservative estimate.
- An average distance of five meters between the cask and the first responders and others who remain with the cask.

- For collective doses, the average rural, urban, and suburban population densities for each route.

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

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

Cask	Dose in Sv	Ten-hour allowed dose in Sv derived from the one-hour dose in 10 CFR 71.51
Truck-DU	1.0 E-03	0.10
Rail-Lead	9.2E-04	0.10
Rail-Steel	6.9E-04	0.10

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

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

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

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

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	Truck-DU Suburban	Truck-DU 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. Average background doses for the 16 routes analyzed are

- Rural: 6.9×10^{-4} person-Sv
- Suburban: 0.019 person-Sv
- Urban: 0.11 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: 6.93×10^{-4} person-Sv
- Suburban: 0.0191 person-Sv
- Urban: 0. 110008 person-Sv

The background and accident 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.

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 adequate to meet the external dose regulation of 10 CFR Part 71. Spent nuclear fuel is extremely radioactive and requires shielding that absorbs both gamma radiation and neutrons. The sum of the external radiation doses from gamma radiation and neutrons should not exceed 0.0001 Sv per hour at two meters from the cask, by regulation.

Each spent-fuel transportation cask analyzed uses a different gamma shield. Each may use different neutron shielding as well, but since no credit is taken for the neutron shield, it is not usually part of the accident analysis. The Rail-Steel cask has a stainless steel wall thick enough to attenuate gamma radiation to acceptable levels. The Truck-DU cask uses metallic DU. Neither of these shields would be damaged, or even affected by, an accident. The Rail-Lead cask has a lead gamma shield which could 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 in the lead shield. In the figures, the dose estimates for the large, gaps are depicted on the left end of the graph, and the fraction of lead shield lost (gap size) decreases from left to right. Figure 5-2 and Figure 5-3 show that doses larger than the external dose that would be allowed by the regulation of 10 CFR 71.51 occur when the lead shielding gap is more than two percent of the shield.

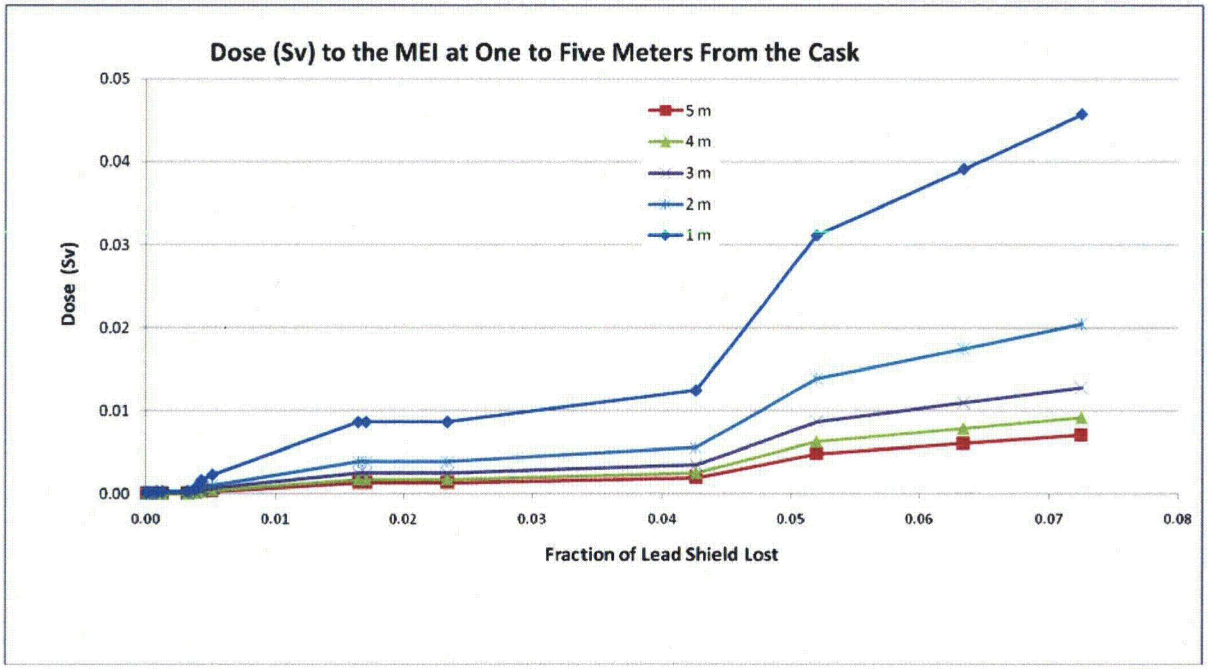


Figure 5-2. Radiation dose to the maximally exposed individual (MEI) from loss of lead gamma shielding at distances from one to five meters from the cask carrying spent fuel. The horizontal axis represents the fraction of shielding lost—the shielding gap—and is not to scale.

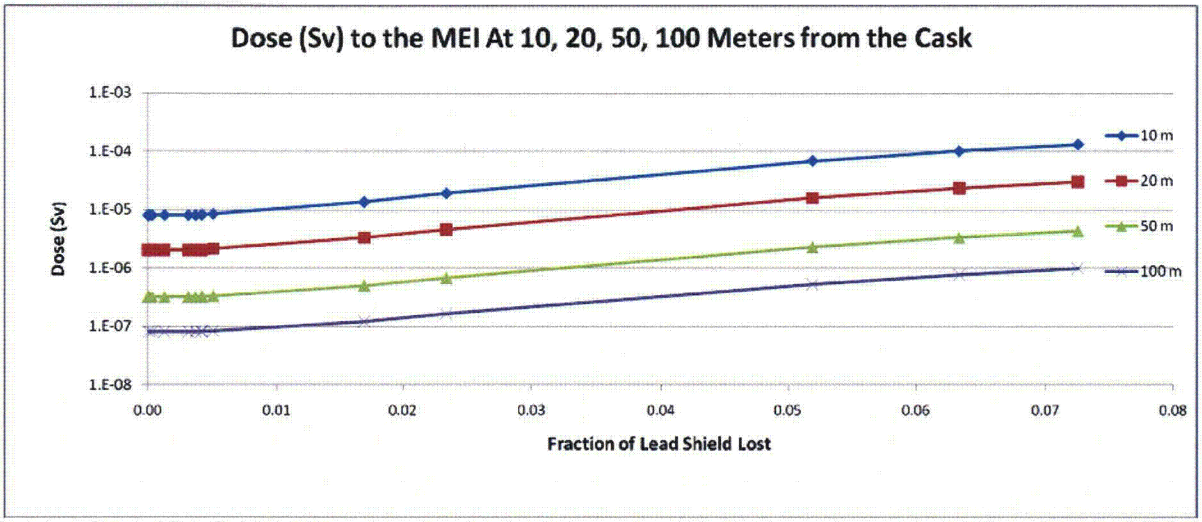


Figure 5-3. Radiation dose to the maximally exposed individual from loss of lead gamma shielding at distances from 20 to 100 meters from the cask carrying spent fuel. The vertical axis is logarithmic so that all of the doses can be shown on the same graph. The horizontal axis represents the fraction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

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 Maine Yankee Nuclear Plant site to Hanford is:

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

or about one in 100 million per Main Yankee to Hanford shipment.

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

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

ORIGIN	DESTINATION	AVERAGE ACCIDENTS PER KM	AVERAGE ACCIDENTS FOR THE TOTAL ROUTE
MAINE YANKEE	ORNL	6.5×10^{-7}	0.00139
	DEAF SMITH	5.8×10^{-7}	0.00194
	HANFORD	4.2×10^{-7}	0.00214
	SKULL VALLEY	5.1×10^{-7}	0.00218
KEWAUNEE	ORNL	4.3×10^{-7}	0.00594
	DEAF SMITH	3.3×10^{-7}	0.00487
	HANFORD	2.4×10^{-7}	0.00468
	SKULL VALLEY	3.7×10^{-7}	0.00103
INDIAN POINT	ORNL	8.8×10^{-6}	0.0112
	DEAF SMITH	6.2×10^{-7}	0.00192
	HANFORD	5.1×10^{-7}	0.00212
	SKULL VALLEY	5.5×10^{-7}	0.00217
INL	ORNL	3.6×10^{-7}	0.0012
	DEAF SMITH	3.5×10^{-7}	0.00067
	HANFORD	3.2×10^{-7}	0.00034
	SKULL VALLEY	2.8×10^{-7}	0.00013

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 accidents whose resultant dose rates would be 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 a gap in lead shielding will occur after a fire involving the cask is about 10^{-19}. The conditional probability is so small because the following has to happen before a fire is close enough to the cask, and 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.

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

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

5.4.2 Neutron Shielding

The type of fuel that can be transported in the three casks considered has relatively low neutron emission but does require neutron shielding. This is usually a hydrocarbon or carbohydrate polymer of some type that often contains a boron compound. All three of the casks studied have polymer neutron shields. Table 5-8 shows the neutron doses to individuals who are about five meters from a fire-damaged cask for ten hours. Neutrons are absorbed by air much better than is gamma radiation, so that external neutron radiation would impact receptors close to the cask but not members of the general public. 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	0.1
Rail-Lead	0.0076	0.1
Rail-Steel	0.0076	0.1

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 fire 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 PWR fuel that has "burned" 45,000 MWD/MTU and has been cooled for nine years. The Rail-Lead 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-Lead 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-Lead cask (TBq)

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

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

5.5.2 Conditional Probabilities and Release Fractions

Seven accident scenarios involving the Rail-Lead cask, described in Chapter 3, could result in releases of material to the environment. The details of these scenarios that are important to calculating the resulting doses are shown in Table 5-10. A detailed description of the movement of radionuclide particles from fuel rods to the cask interior and from the cask interior to the environment is found in Appendix V, Sections V.5.4.1 and V.5.4.2.

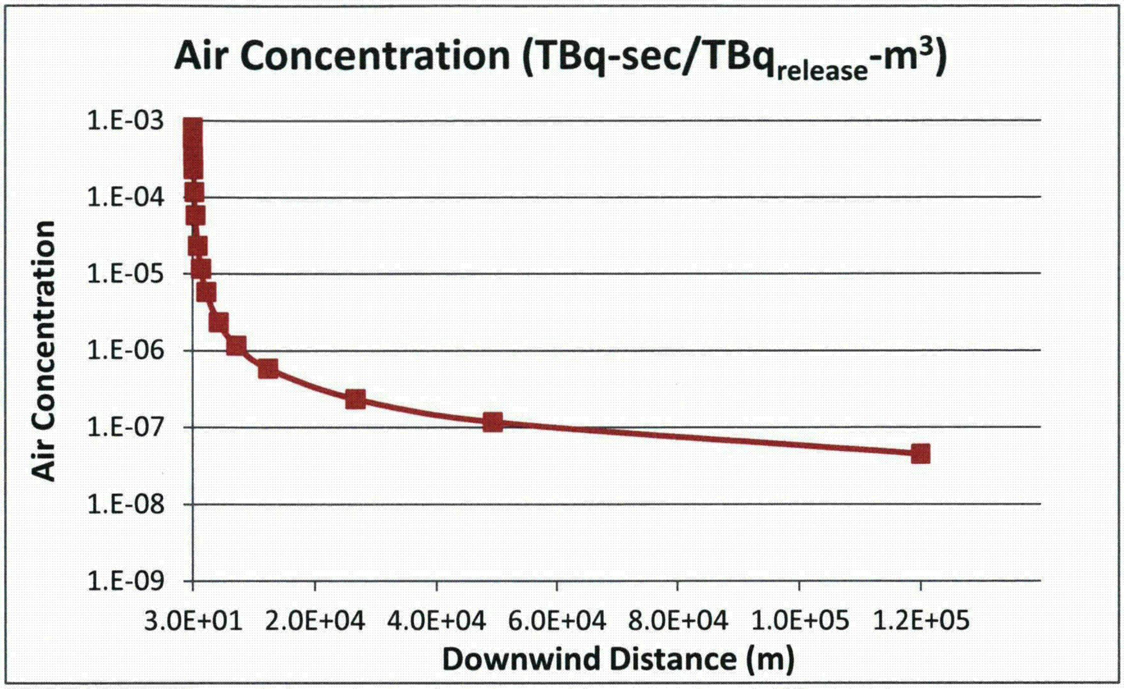
Table 5-10. Parameters for determining release functions for the accidents that would result in release of radioactive material

	Cask Orientation	End	Corner	Side	Side	Side	Side	Corner
	Rigid Target Impact Speed (kph)	**193**	**193**	**193**	**193**	**145**	**145**	**145**
	Seal	metal	metal	elastomer	metal	elastomer	metal	metal
Cask to Environment Release Fraction	**Gas**	0.800	0.800	0.800	0.800	0.800	0.800	0.800
	Particles	0.70	0.70	0.70	0.70	0.70	0.70	0.64
	Volatiles	0.50	0.50	0.50	0.50	0.50	0.50	0.45
	Crud	0.001	0.001	0.001	0.001	0.001	0.001	0.001
Rod to Cask Release Fraction	**Gas**	0.005	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	1.52E-06	1.52E-06	5.81E-05

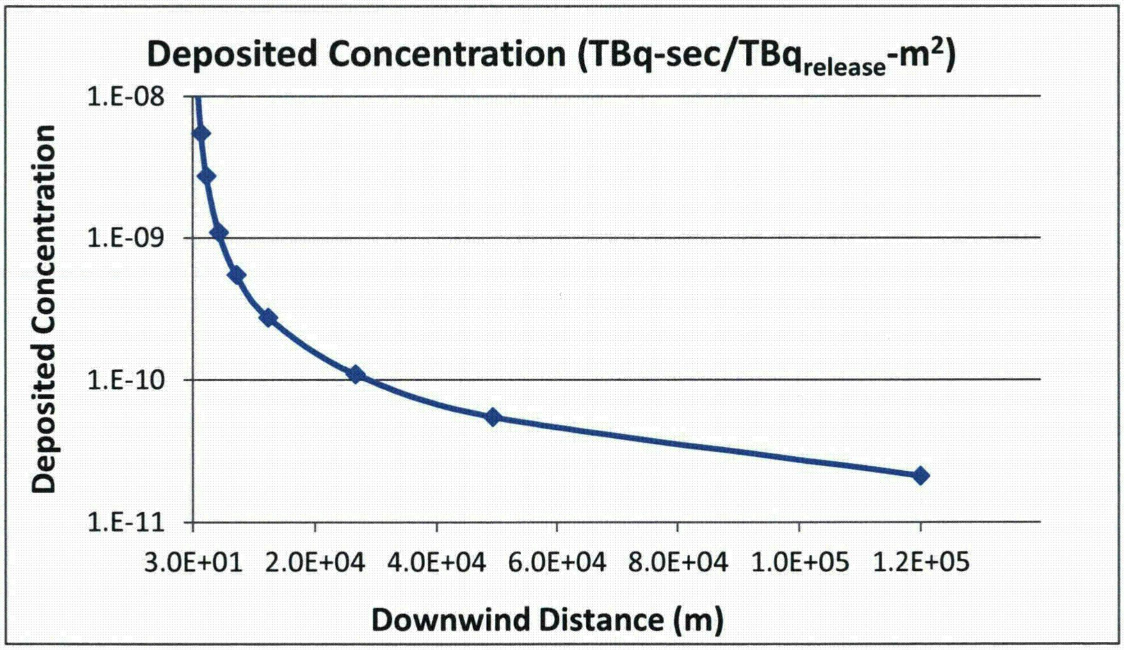
5.5.3 Dispersion

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



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



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

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

5.5.4 Consequences and Risks from Accidents Involving Release of Radioactive Material

The dose from each of the accidents that would involve a release is shown in Table 5-11.

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

Cask Orientation	Impact Speed (kph)	Seal	Inhalation	Re-suspension	Cloud-shine	Ground-shine	Total
End	193	metal	1.59	0.0137	0.0001	0.0009	1.60
Corner	193	metal	1.59	0.0137	0.0001	0.0009	1.60
Side	193	elastomer	1.59	0.0137	0.0001	0.0009	1.60
Side	193	metal	1.59	0.0137	0.0001	0.0009	1.60
Side	145	elastomer	1.58	0.0137	4.53E-06	3.61E-05	1.59
Side	145	metal	1.59	0.0137	8.78E-05	9.42E-04	1.60
Corner	145	metal	0.7270	0.0063	0.0001	0.0009	0.73

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 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 receive a dose only as long as he or she is exposed to the deposited or airborne material. If people near the accident are evacuated, and evacuation can take as much as a day, then they only receive an external dose for a day.

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 considers the total effective dose equivalent: the sum of the inhalation and external doses.

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 conditional 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. Of the three casks in this study, only the Rail-Lead cask could result in a release in each kind of accident considered.

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

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	3.6E-09	2.2E-09	1.9E-09	9.6E-10
KEWAUNEE	1.5E-09	7.4E-10	7.2E-10	5.1E-10
INDIAN POINT	6.1E-08	2.3E-09	2.4E-09	7.7E-10
IDAHO NATIONAL LAB	3.7E-10	6.0E-10	1.6E-10	1.1E-09

These dose risks are negligible by any standard.

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

Table 5-13. Total collective dose risk (person-Sv) for each route from a loss of 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	4.0E-09	2.5E-09	2.1E-09	1.1E-09
KEWAUNEE	1.7E-09	8.3E-10	8.1E-10	5.9E-10
INDIAN POINT	6.8E-08	2.6E-09	2.7E-09	8.7E-10
IDAHO NATIONAL LAB	4.3E-10	7.0E-10	1.8E-10	1.2E-9

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

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

	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

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. There is no expectation of any release from spent fuel shipped in inner welded canisters from any impact or fire accident analyzed.
- The collective dose risk from loss of lead shielding is comparable to the collective dose risk from a release, though both are very small. The doses and collective dose risks from loss of lead shielding are larger than were calculated in NUREG/CR-6672 as a result of better precision in the finite element modeling and a more accurate model of the dose from a gap in the lead shield.
- The conditional risk of either a release or loss of shielding from a fire is negligible.
- The consequences (doses) of some releases and some loss of shielding scenarios 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 and are in agreement with previous studies.

CHAPTER 6

OBSERVATIONS AND CONCLUSIONS

The present document is an assessment of the risks of transporting spent nuclear fuel, updating the assessment performed for NUREG-0170, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes*, published in 1977. Both NUREG-0170 and this document provide a technical basis for the regulations of 10 CFR Part 71. Other studies, like the Modal Study (Fischer, et al., 1987) and NUREG/CR-6672 (Sprung, et al., 2000), support the conclusions of NUREG-0170.

Regulations and regulatory compliance analyses are different from risk assessments. A regulation must be conservative because its purpose is to ensure safety, and 10 CFR Part 71, which regulates transportation, requires a conservative estimate (i.e., overestimate) of the damage to a cask in an accident and the radiation emitted from the cask during routine transportation. The original 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.

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

The present study is a more accurate 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 spent nuclear fuel transportation radiological safety, and is an addition to the technical basis for 10 CFR Part 71. Also, it represents the current state of the art for such analyses. The results of this study are compared with preceding risk assessments in the figures that follow.

6.1 Routine Transportation

Figure 6-1 and Figure 6-2 show results of routine truck and rail transportation of a single shipment of spent nuclear fuel. Figure 6-1 plots average collective radiation dose (person-Sv) from truck transportation, and Figure 6-2 plots average collective radiation dose from rail transportation. These average doses include the doses to the population along the route, doses to occupants of vehicles sharing the route, doses at stops, and doses to vehicle crew.

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

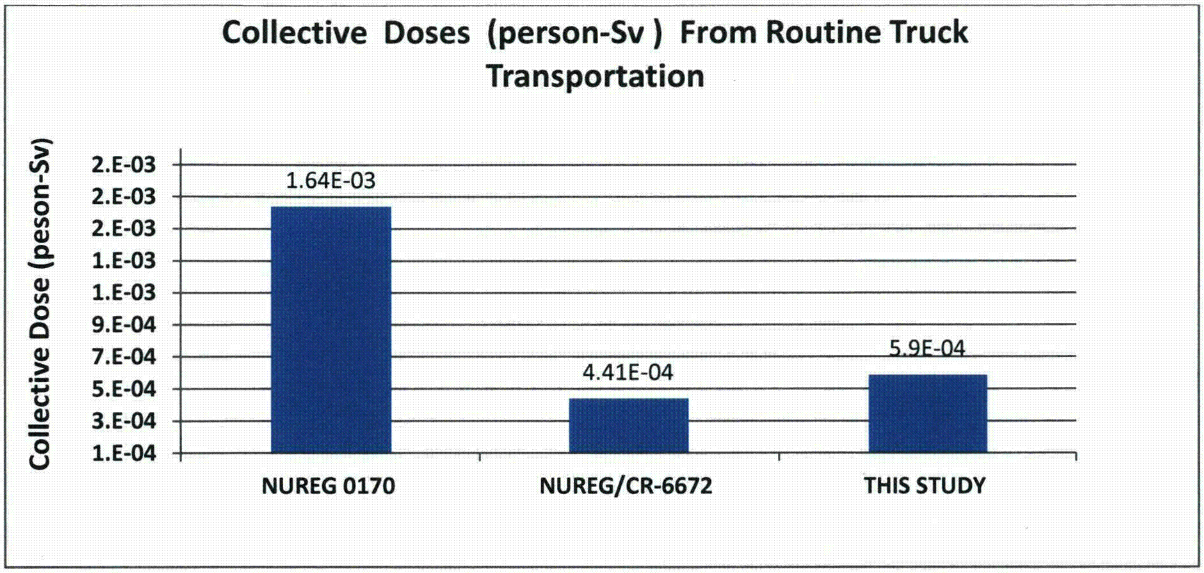


Figure 6-1. Collective doses (person-Sv) from routine truck transportation.

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

The collective average dose in the present study is larger than the NUREG/CR-6672 result because present populations are generally larger, particularly along rural routes, and the vehicle densities are much larger (see Chapter 2). These increases were offset by the greater vehicle speeds used in the present study.

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

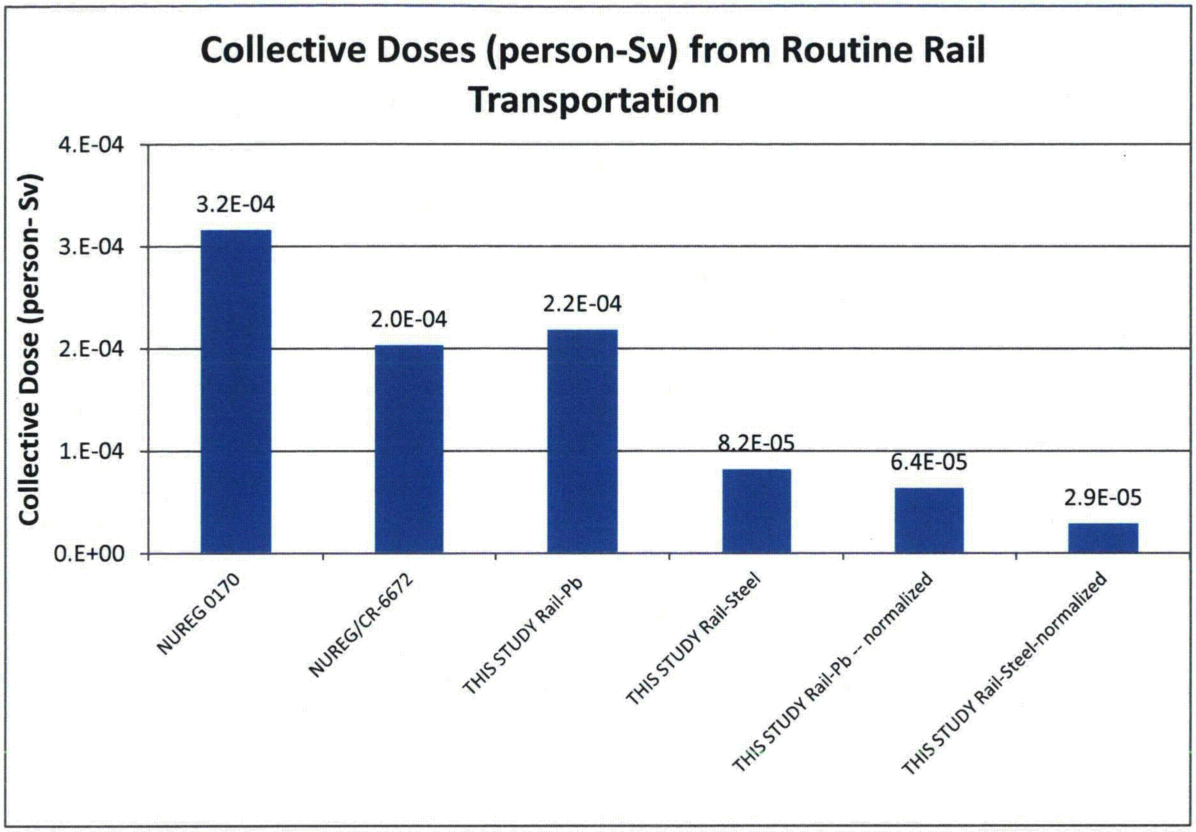


Figure 6-2. Collective doses (person-Sv) from routine rail transportation.

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

The differences in the collective doses from routine transportation between the cited studies are not the result of differences in external radiation from the spent fuel casks. The 1975 version of 10 CFR Part 71[18] specified the same limit on external radiation (the transport index) as Part 71 specifies today.

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

- 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 rural and suburban routes and 24 kph on

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

urban routes. The present speeds are based on data instead of the estimated values used in the previous studies.

- Differences in populations along the routes. NUREG-0170 used six persons per km^2 for rural populations, 719 per km^2 for suburban routes, and 3861 per km^2 for urban routes. NUREG/CR-6672 used 1990 census data provided by the codes HIGHWAY and INTERLINE and used the mean values of Gaussian distributions of population densities on 200 routes in the United States. This study uses 2000 census data provided by TRAGIS (Johnson and Michelhaugh, 2002), with some updates based on 2008 census data, for the rural, suburban, and urban truck and rail route segments in each state traversed in each of the sixteen routes studied. The variation from the NUREG-0170 values is considerable.
- Differences in vehicles per hour on highways. NUREG-0170 and NUREG/CR-6672 both used the 1975 values of 470 vehicles per hour on rural routes, 780 on suburban routes, and 2800 on urban routes. This study used 2002 state vehicle density data for each state traversed. The national average vehicle density is 1119 vehicles per hour on rural routes, 2464 on suburban routes, and 5384 on urban routes. This large difference in vehicle density probably explains the difference in collective doses for routine truck transportation between NUREG/CR-6672 and this study.
- Differences in calculating doses to rail crew. NUREG-0170 calculated doses to rail and railyard crew by estimating the distance between the container carrying radioactive material and the crew member. NUREG/CR-6672 used the Wooden (1980) calculation of doses to railyard workers, and did not calculate a dose to the crew on the train. This study calculated all doses using the formulations in RADTRAN 6, calculated an in-transit crew dose, used an updated value for the time of a classification stop (27 hours instead of 30 hours), and used in-transit stop times from TRAGIS rather than the stop dose formula, pegged to total trip length, used in NUREG/CR-6672. The in-transit crew dose calculated in this study was small enough that it contributed a negligible amount to these doses.

Dose to the maximally exposed individual is a better indication of the radiological effect of routine transportation than collective dose. The same event results in different collective doses depending on the population affected, which varies both spatially and temporally. 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.

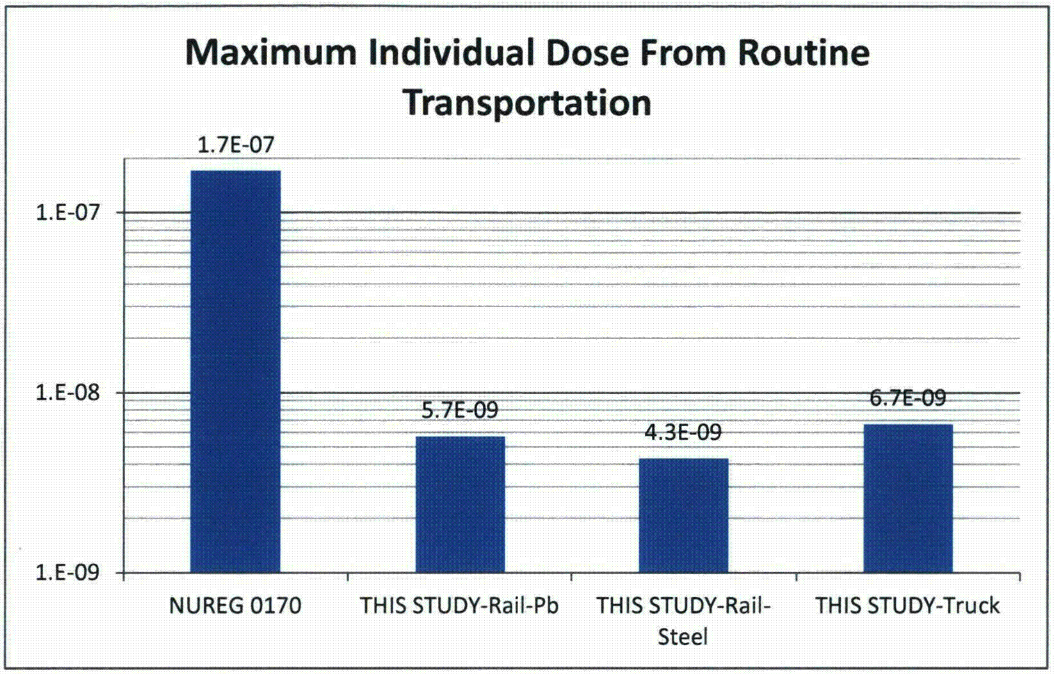


Figure 6-3. Maximum individual dose (Sv) from routine transportation.

6.2 Transportation Accidents

Radiological accident risk is expressed in units of "dose risk" that include the probability of an accident and the conditional probability of certain types of accidents. The units used are dose units (Sv) because probability is a unitless number. NUREG-0170, NUREG/CR-6672, and this study all used the version of RADTRAN available at the time of the study to calculate dose risk, but the input parameters differed 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 and Figure 6-6 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-Lead cask in this study, is the only cask type of the three studied that can either release radioactive material or can lose lead gamma shielding in a rail or highway traffic accident.

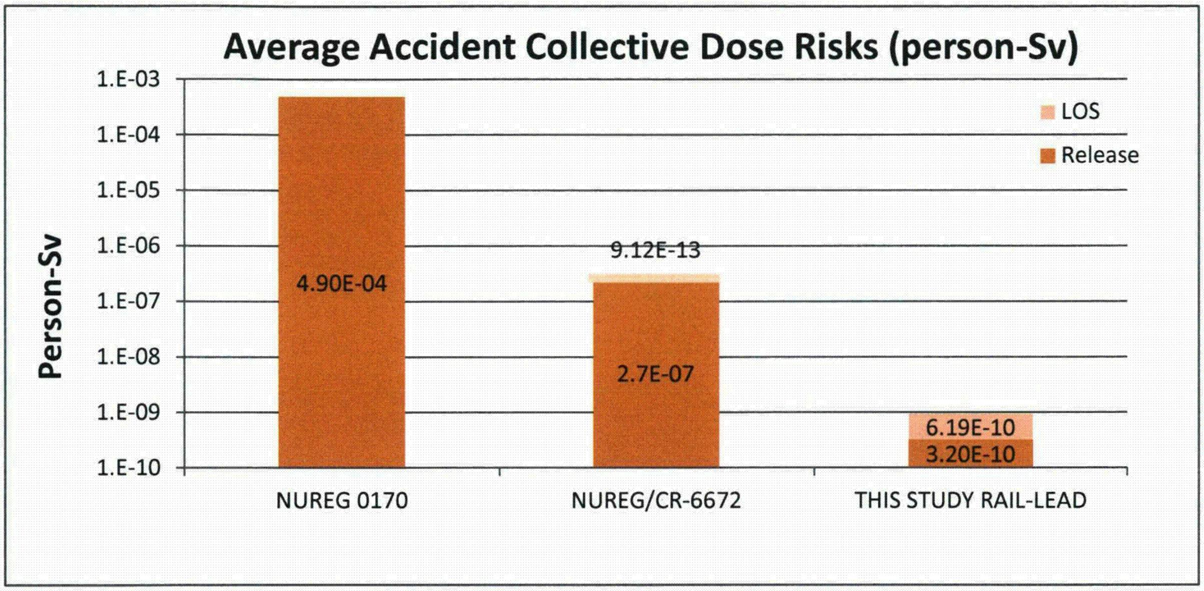


Figure 6-4. Accident collective dose risks from release and LOS accidents. The LOS bar representing the NUREG/CR-6672 collective dose is not to scale.

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 postulated a one percent release. The range of conditional probabilities was from 1×10^{-5} for the most severe (100 percent release) accident to 80 percent for the two no-release accident scenarios. The NUREG-0170 "universe" of accidents and their consequences was based primarily on engineering judgment and was clearly conservative.

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

The present study followed the analytical outline of the NUREG/CR 6672 analysis, but analyzed the structural and thermal behavior of a certified lead-shielded cask design loaded with fuel that the cask is certified to transport. Instead of the 19 truck scenarios and 21 rail scenarios that included potential releases of radioactive material, the current study resulted in only seven rail scenarios that included releases, as described in Chapters 3 and 5. The only parts of the cask structure that could be damaged enough to allow a release are the seals. Release could take place through the seals only if the seals fail and if the cask is carrying uncanistered fuel. No potential truck accident scenario resulted in seal failure, nor did any fire scenario. In the present study, only the Rail-Lead cask response to accident conditions resulted in a release. A comparison of the collective dose risks from potential releases in this study to both NUREG-0170 and NUREG/CR-6672 is appropriate, since the latter two studies considered only potential releases. The collective dose risks decrease with each succeeding study as expected, since the overall conditional probability of release and the quantity of material potentially released decreases with each successive study.

The 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, public interest in the transportation of spent fuel focuses on the consequences of possible accidents (without regard to their likelihood). The average estimated consequences (collective doses) from potential accidents involving release for the present study is 2 person-Sv. This consequence is orders of magnitude less than the 110 person-Sv in NUREG-0170 and the 9000 person-Sv estimated from Figure 8.27 in NUREG/CR-6672.

NUREG-0170 did not consider loss of spent fuel cask lead shielding, which can result in a significant increase the dose from gamma radiation being emitted by the cask contents. NUREG/CR-6672 analyzed 10 accident scenarios in which the lead gamma shield could be compromised and calculated a fractional shield loss for each. An accident dose risk was calculated for each potential fractional shield loss. The present study followed the same general calculation scheme, but with a more sophisticated model of gamma radiation from the 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 delay in this study. The resulting collective dose risk from this accident is shown in Figure 6-6 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.

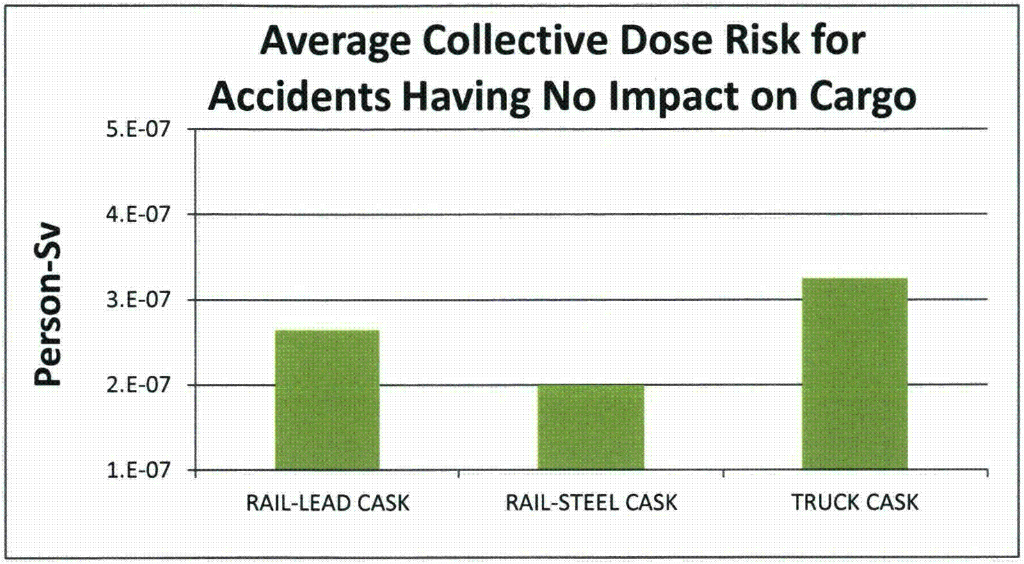


Figure 6-5. Average collective dose from accidents that have no impact on the cargo.

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 serious but not dire. Moreover, these consequences depend on the size of the population exposed rather than on the radiation or radioactive material released. The consequences (doses) to the maximally exposed individual, 1.6 Sv to a member of the public from a release and 1.1 Sv from loss of lead shielding to a possible first responder, could result in latent health consequences rather than immediate health effects.

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. Average collective doses from this type of accident for the 16 routes studies are shown in Figure 6-6. 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.

APPENDIX I

CASK DETAILS AND CERTIFICATES OF COMPLIANCE

Table of Contents

I.1 Cask Descriptions

This appendix provides a listing and brief description of the spent fuel transport casks that were considered for evaluation in this risk analysis. Also provided are the certificates of compliance for those casks selected for evaluation.

I.1.1 Truck Casks

GA-4	The Steel-DU-Steel design is stiffer than lead casks and has 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) allow larger closure deformations before leakage.
	Truck casks have polymer neutron shielding.
	Larger capacity allows for larger radioactive material inventory and possible larger consequences from an accident.
	The design is from the late 80s; General Atomics used finite element analyses in certification.
	The DU shielding is made from 5 segments, which could possibly result in segment-to-segment problems.
	The cask body has a square cross-section, which provides more possible orientations.
	The cask has an Aluminum honeycomb impact limiter.
NAC-LWT	The steel-lead-steel design is relatively flexible, which should result in plastic deformation of the body before seal failure.
	The NAC-LWT contains either a single PWR assembly or two BWR assemblies.
	The cask has both elastomeric and metallic seals. The low compression of the elastomeric seal (metallic is primary) allows little closure movement before leakage but may have better performance in a fire.
	The lead shielding could melt during severe fires, leading to loss of shielding.
	With liquid neutron shielding, the tank is likely to fail in extra-regulatory impacts.
	The bottom end impact limiter is attached to the neutron shielding tank, making side drop analysis more difficult.
	The NAC-LWT has an aluminum honeycomb impact limiter.
	The cask is very similar to the generic steel-lead-steel cask from NUREG 6672.
	The cask is being used for Foreign Research Reactor shipments.

I.1.2 Rail Casks

NAC-STC	The cask has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure
	The NAC-STC is certified for both direct loaded fuel and for fuel in a welded canister.
	The cask can contain either 26 directly loaded PWR assemblies or 1 Transportable Storage Container (3 configurations, all for PWR fuel).
	The cask can have either elastomeric or metallic seals. A configuration must be chosen for analysis.
	The lead shielding used could melt during severe fires, leading to loss of shielding.
	The NAC-STC has polymer neutron shielding.
	The cask has a wood impact limiter (redwood and balsa).
	This cask is similar to the steel-lead-steel rail cask from NUREG 6672.
	Two casks have been built and are being used outside of the U.S.
NAC-UMS	The NAC –UMS has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure.
	The fuel is in a welded canister.
	Baskets for 24 PWR assemblies or 56 BWR assemblies are available.
	Elastomeric seals allow larger closure deformations before leakage.
	The lead shielding could melt during severe fires, leading to loss of shielding.
	The cask has polymer neutron shielding.
	The cask has a wood impact limiter (redwood and balsa).
	The cask is similar to the steel-lead-steel rail cask from NUREG 6672.
	The NAC-UMS cask has never been built.
HI-STAR 100	The HI-STAR 100 cask has a layered all-steel design.
	The fuel is in a welded canister.
	Baskets for 24 PWR assemblies or 68 BWR assemblies are available.
	The cask has metallic seals, resulting in smaller closure deformations before leakage.
	The cask has polymer neutron shielding.
	The cask has aluminum honeycomb impact limiters.
	At least 7 of these casks have been built and are being used for dry storage; no impact limiters have been built.
	The HI-STAR 100 is proposed as the transportation cask for the Private Fuel Storage facility (PFS).

TN-68	The TN-68 has a layered all-steel design.
	Directly loaded fuel is used in the cask.
	The TN-68 has 68 BWR assemblies.
	Metallic seals result in smaller closure deformations before leakage.
	The cask has polymer neutron shielding.
	The cask h as a wood impact limiter (redwood and balsa).
	At least 24 TN-68 casks have been built and are being used for dry storage; no impact limiters have been built.
MP-187	The MP-187 has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure.
	The fuel in a welded canister.
	There are 24 PWR assemblies.
	Metallic seals result in smaller closure deformations before leakage.
	The MP-187 has hydrogenous neutron shielding.
	The cask has aluminum honeycomb/polyurethane foam impact limiters (chamfered rectangular parallelepiped).
	This cask has never been built
MP-197	The MP-197 has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure.
	The fuel in a welded canister.
	There are 61 BWR assemblies.
	Elastomeric seals allow larger closure deformations before leakage.
	The MP-197 has hydrogenous neutron shielding.
	The cask has a wood impact limiter (redwood and balsa).
	This cask has never been built.
TS125	The TS125 has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure.
	The fuel in a welded canister.
	There are basket designs for 21 PWR assemblies or 64 BWR assemblies.
	Metallic seals result in smaller closure deformations before leakage.
	The TS125 has polymer neutron shielding.
	The cask has aluminum honeycomb impact limiters.
	This 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

1. 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.

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)*

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.

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.

INDEX TO APPENDIX A

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 $\geq$ 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

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 ≥ 18 years, an average burnup ≤ 30,000 MWD/MTIHM, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 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:	$\leq$ 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

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 $\geq$ 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:	$\leq$ 176.2 inches (nominal design)
f. Fuel assembly width:	$\leq$ 5.85 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 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 $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U for 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

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 [illegible])	≤ 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