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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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 assuring that radioactive material is transported safely. To achieve this, they have enacted the rules of Title 10 of the Code of Federal Regulations, Part 71. In 1977 the NRC published NUREG-0170, an assessment of the adequacy of those regulations to provide the assurance of safety. 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 from spent nuclear fuel transportation come from both the radiation given off from the fuel, which is greatly attenuated but not eliminated, by the transportation casks, and the possibility of release of some radioactive material during a severe accident. This investigation shows the risk from the external radiation 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 the original risk assessment and this one, the calculated risk due to the external radiation of the cask is very similar to that calculated in NUREG-0170. 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 five orders of magnitude less than was estimated in NUREG-0170. Significant conclusions are that no realistic accident will lead to release of radioactive material for transportation in a rail cask with an inner welded canister or transportation in a truck cask, that no realistic fire will lead to a release for any of the casks studied, that for a rail cask without an inner canister an accident that leads to any release of radioactive material will occur less than once in 25 million shipments, and that in the worst-case accident maximum individual dose is less than two sieverts. These results strongly demonstrate that the regulations of the NRC definitely assure the safety of the transportation of spent nuclear fuel.

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ACRONYMS

ALARA	As Low As Reasonably Achievable [worker dose]
AMAD	activity median aerodynamic diameter
Bq	becquerel
BWR	boiling water [nuclear] reactor
CAFE	Container Analysis Fire Environment
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
Ci	curie
CoC	Certificate of Compliance
DOE	[U.S.] Department of Energy
DOT	[U.S.] Department of Transportation
DU	depleted uranium
EPA	[U.S.] Environmental Protection Agency
EQPS	equivalent plastic strain
FE	finite element
HAC	hypothetical accident conditions
HLW	high-level [radioactive] waste
IAEA	International Atomic Energy Agency
ISCORS	[U.S.] Interagency Steering Committee on Radiation Standards
kph	kilometers per hour
LCF	latent cancer fatalities
LOS	loss of [lead] shielding
m	meter
MEI	maximally exposed individual
MN	mega newton
MPC	multi-purpose canister
mph	miles per hour
mrem	millirem
MTU	metric tons of uranium
MWD	megawatt-days
NRC	[U.S.] Nuclear Regulatory Commission
PWR	pressurized water [nuclear] reactor
rem	Roentgen equivalent man
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
Sv	sievert
TEDE	total effective dose equivalent
TI	transport index

CHEMICAL SYMBOLS

Cd	cadmium
Cm	curium
Co	cobalt
Cs	cesium
Eu	europium
I	iodine
Kr	krypton
Pa	protactinium
Pb	lead
Pu	plutonium
Ru	ruthenium
U	uranium
UO_2	uranium dioxide

PUBLIC SUMMARY

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

- The average radiation dose to members of the public from routine transportation of radioactive materials is a fraction of the background dose.
- The radiological risk from accidents in transporting radioactive materials is very small compared to the non-radiological risk from accidents involving large trucks or freight trains.

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

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

A 1987 study applied actual accident statistics to projected spent fuel transportation (Fisher, 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 spent fuel shipments had not had enough accidents to support projections or predictions. However, 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 a computer modeling technique (Sprung, et al., 2000). Semi-detached trailer truck and rail accident statistics for general freight shipments were used because even by 2000 there had been too few accidents involving fuel cycle shipments to provide statistically valid data.

The dispersion of material released from the cask in an accident was also modeled with increasing refinement. NUREG-0170 assumed that most very severe accidents would result in release of all of the releasable cask contents to the environment; this engineering judgment overstated the release but was nevertheless used because analytical capabilities at the time did not permit a more accurate assessment of the release. The 2000 study analyzed the physical properties of spent fuel rods in a severe accident, and revised estimates of material released to one percent or less of the NUREG-0170 estimates. Accordingly, risk estimates were revised downward. The 2000 study also verified that an accidental release of radioactive material could only be through the seals at the end of the cask where the lid is attached: an accident could cause seal failure, but no breach in the cask body.

The present study models real casks and the commercial spent nuclear fuel that these casks are certified to transport. Two rail casks and a truck cask are evaluated. Rail casks are believed likely to be used for most future shipments.

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 kept 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.

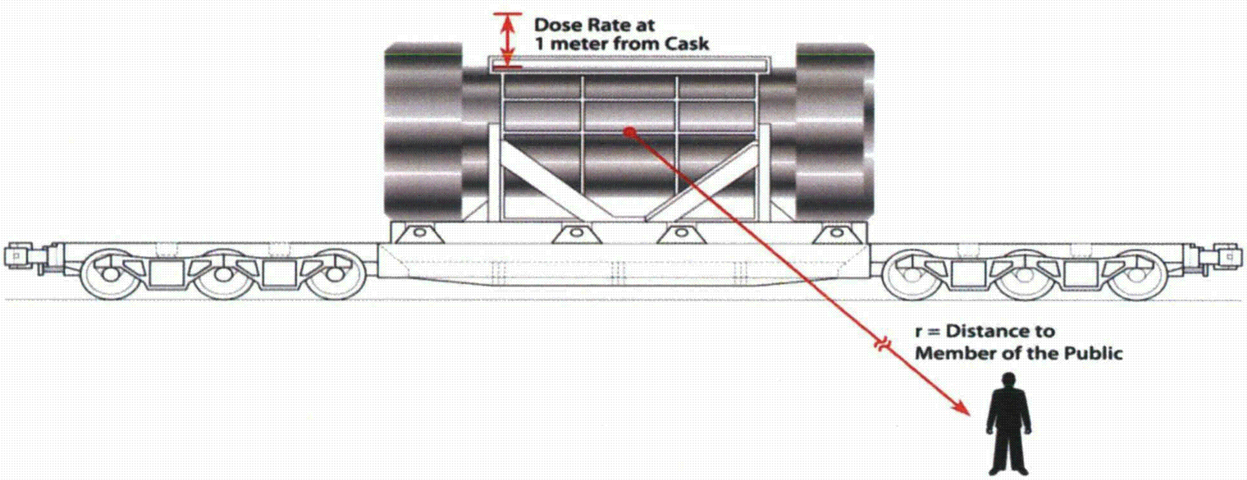


Figure PS-1. Model of a spent fuel cask in routine, incident-free transportation and radiation dose to a member of the public. Relative sizes of the cask and person are approximately to scale.

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. Figure PS-2 shows the total dose 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. The background radiation dose to exposed workers and members of the public during the time of the shipment is included in Figure PS-2.

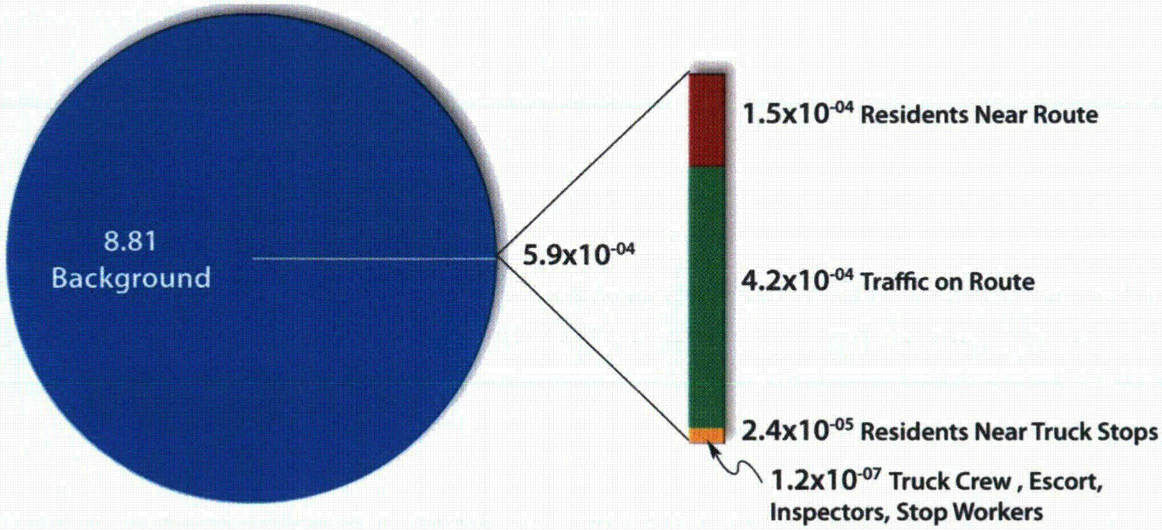


Figure PS-2. Collective Doses from Background and from a Truck Shipment of Spent Nuclear Fuel (Person-Sv)

The collective doses calculated for routine transportation are approximately the same for the 2000 study and for this study of the risks from spent fuel transportation, and are about 40 percent of the doses reported in NUREG 0170. Figure PS-3 shows a comparison of the collective doses from truck transportation, averaged over the sixteen routes studied.

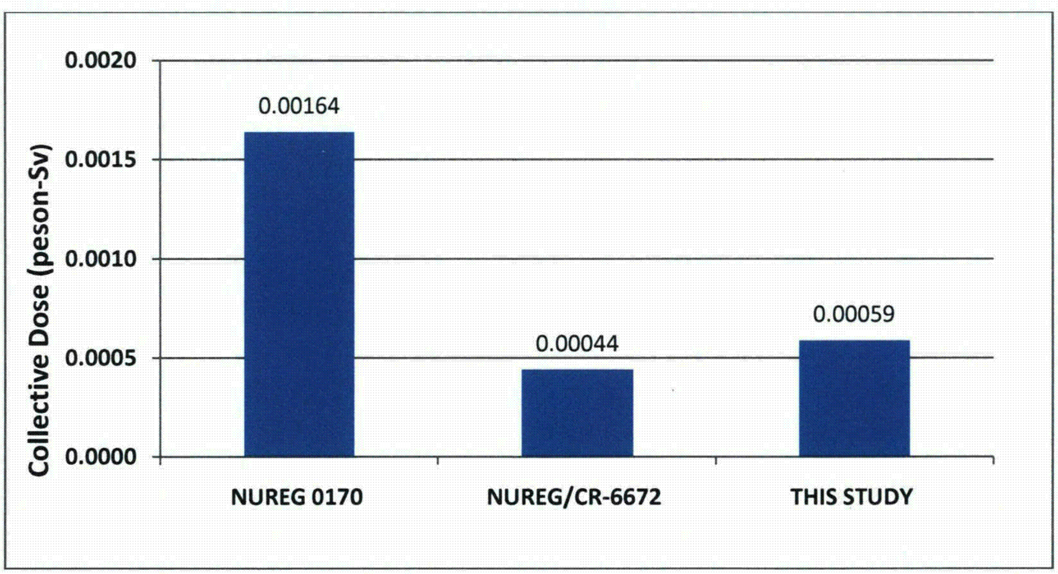


Figure PS-3. Collective doses (person-Sv) from routine truck transportation.

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 cask responds to the accident. Figure PS-4 shows one impact scenario, a 97 kph (60 mph) corner impact onto a rigid target, and the resulting deformations. Almost all of the deformation is in the impact limiter, a device that is added to the cask to absorb energy, much like the bumper of a car. Similar analyses were performed for impacts at 48, 97,145, and 193 kilometers per hour (kph)—equal to 30, 60, 90, and 120 mph—in end-on, corner, and side-on orientations for two cask designs. These impact speeds encompass all accidents for truck and rail transportation. Figure PS-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 encompass all fire-related accidents in rail transportation. The longest duration for an engulfing fire during truck transportation is one hour, due to the smaller amount of fuel that is carried on board a tanker truck.

The detailed simulations were performed for two spent fuel casks that are intended for transportation by railroad, the NAC-STC and the HI-STAR 100, and one truck transportation cask, the GA-4.

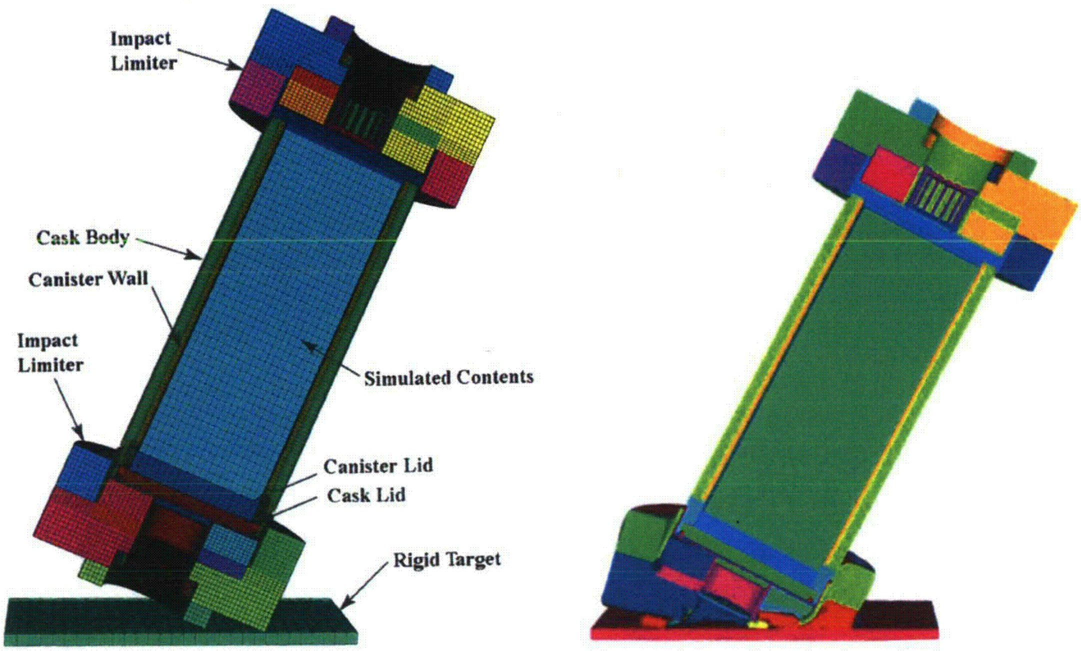
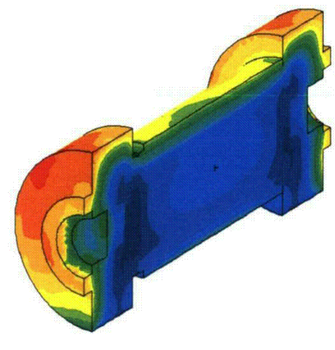


Figure PS-4. Corner impact onto a rigid target at 97 kph (60 mph) accident scenario for a spent fuel cask and the deformations produced by the impact.

a) During fire



b) After fire

Figure PS-5. Engulfing fire scenario and the temperature contours in the rail cask model following a 3-hour fire duration. The transparency of the flames has been increased so the cask can be seen. In the actual fire simulation, and in a real fire, the flames are opaque.

The impact and thermal analysis results indicate that 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 a longer duration stop associated with the accident. The stop would require the emergency responders to clear the accident scene and to arrange for shipment resumption. In this situation the emergency responders will receive a radiation dose due to the radiation emitted from the cask. However, because there is no loss in effectiveness of the gamma shielding, the radiation dose to these responders is quite small.

For rail transport of spent fuel that is in an inner welded canister, the detailed impact and thermal analyses show there would be no release of radioactive material in any accident scenario evaluated. Only for very improbable impacts and long duration fires could there be a small reduction in the effectiveness of the lead gamma shielding, leading to an elevated external radiation level. This loss of lead shielding (LOS) would result in a maximum dose to a person 20 meters from the cask of $3x10^{-5}$ Sv and a collective population dose risk of less than $6x10^{-9}$ person-Sv per shipment.

For rail transport of spent fuel that is not enclosed within an inner welded canister, there is the possibility for a small release of radioactive material following exceptionally severe (and improbable) impacts. The maximum dose risk to an individual from this release would be about 4.5×10^{-13} Sv. The maximum dose to an individual would be 1.6 Sv if the release actually happened. The collective population dose risk is $5x10^{-7}$ person-Sv per shipment along the route with the largest population.

Similar to the routine transportation collective doses, 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). 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 studied are shown in Figure PS-7.This scenario is important because over 99.999 percent of all accident scenarios do not lead to either release of radioactive material or loss of shielding.

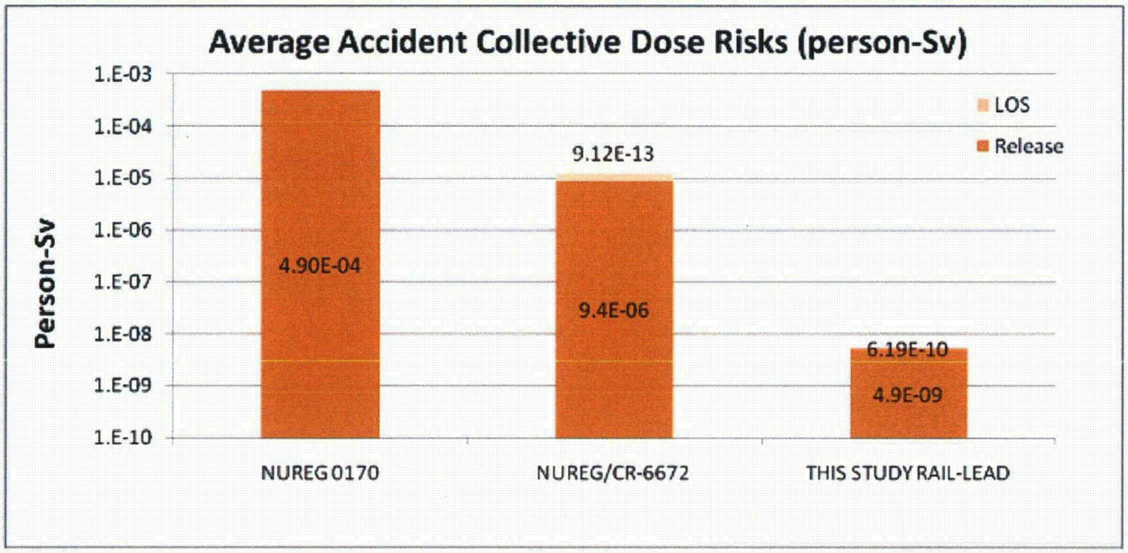


Figure PS-6. Accident collective dose risks from release and LOS accidents. The LOS bar representing the NUREG/CR-6672 collective dose is not to scale.

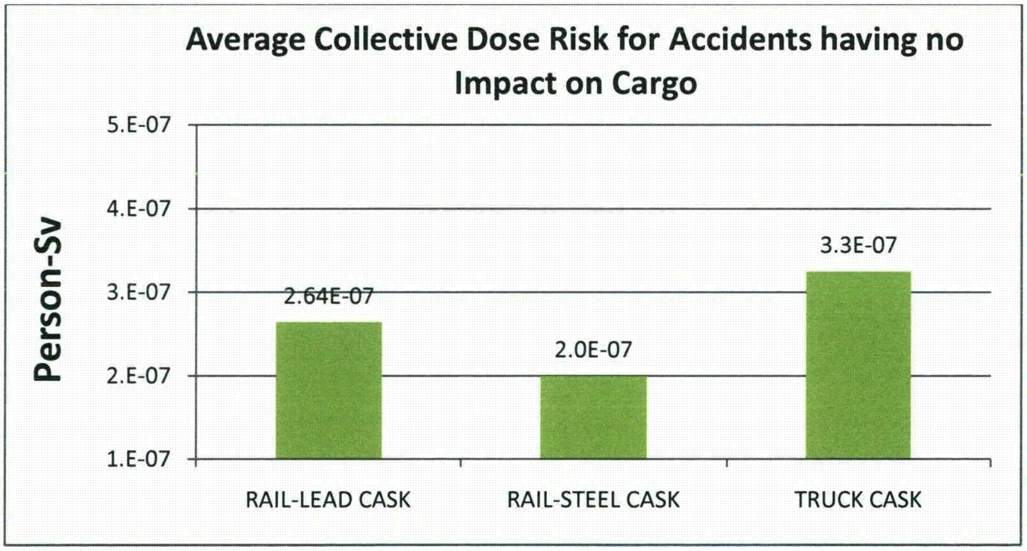


Figure PS-7. Average collective dose risk from accidents that have no impact on the cargo.

In summary:

- The sixteen routes selected for this study are an adequate representation of U.S. routes for spent nuclear fuel. There were relatively little variations 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 will not be released in an accident if the fuel is in a canister in the cask, eliminating an already small transportation risk.
- Radioactive material will not be released for any accident involving a truck cask.
- Only rail casks without inner welded canisters can have any release in extremely severe accidents—99.999 percent of potential accidents will not result in a release of radioactive material from any cask.
- 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.
- These results are not unexpected and are in agreement with previous studies.

The analyses and results described in this report provide assurance that spent fuel shipments can be completed safely, that the NRC regulations governing transportation of spent nuclear fuel are effective, and it is expected that the excellent safety record for spent fuel shipments will be maintained.

CHAPTER 1

INTRODUCTION

1.1 History and Purpose of this Analysis

The purpose of this study is analysis of the radiological risks of transporting spent nuclear fuel, in both routine transportation and transportation accidents. This study will use the latest available data and modeling techniques. It is primarily a study of 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 rounds out this series by analyzing the behavior of certified casks carrying fuel of known isotopic composition and burnup. The studies that preceded this one were based on conservative and generic assumptions.

This study is not intended to be a risk assessment for any particular transportation campaign, like transportation from reactors to a permanent repository. It also does not include the probabilities or consequences of malevolent acts.

The Nuclear Regulatory Commission (NRC) certifies casks used to transport spent nuclear fuel under Title 10 of the Code of Federal Regulations Part 71 (10 CFR Part 71). Part of the technical basis for this regulation was NUREG-0170, the Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes (NRC, 1977), an environmental impact statement for transportation of all types of radioactive material by road, rail, air, and water. The conclusions drawn 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 (1981) were "adequate to protect the public against unreasonable risk from the transport of radioactive materials." (46 FR 21629, April 13, 1981)

The risk assessment of NUREG-0170 was based on very conservative estimates of risk parameters, and on the imprecise models available at the time. The NRC concluded that the regulations were adequate because even very conservative estimates of risk parameters did not result in unacceptable risk. NRC also recognized that the agency's policies on radioactive materials transportation should be "subject to close and continuing review." In the spirit of continuing review, two comprehensive contractor reports dealing with spent fuel transportation have been issued since 1977: the Modal Study (Fischer et al., 1987) and NUREG/CR-6672 (Sprung et al., 2000)[1]. Both studies were advances in transportation risk assessment. The Modal Study was the first intensive examination of vehicle accident statistics and was the first to organize the frequency of severe accidents by structural and thermal response of the cask. Using documented accident frequencies of large trucks and railcars, the Modal Study organized the

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

probability of accidents by the structural and thermal response of the casks in the accident. The Modal Study concluded that the frequency of accidents severe enough to produce significant 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 impact speed and temperature. In addition, NUREG/CR-6672 developed expressions for the behavior of spent fuel in accidents and potential release of this material and analyzed the potential releases. The enhanced modeling capability available for NUREG/CR-6672 allowed analysis of the detailed structural and thermal damage to transportation casks. NUREG/CR-6672 also used results of experiments by Lorenz et al. (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 of initial uranium [GWD/MTU]) and relatively high burnup (60 GWD/MTU) pressurized water reactor (PWR) and boiling water reactor (BWR) fuel, although the transportability of the high burnup fuel was not considered. NUREG/CR-6672 studied the behavior of two generic truck casks and two generic rail casks which were each composites of several certified casks.

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

The present study analyzes the behavior of three currently certified casks carrying Westinghouse 17x17 (PWR) fuel assemblies with 45GWD/MTU burnup, the highest burnup that any of the three casks are certified to carry. The resulting radiological risks are less than those reported in NUREG/CR-6672. For routine transportation, the risks are slightly higher than those estimated in NUREG/CR-6672, because population densities along transportation routes have increased and traffic densities on the highways used for truck transportation have increased. 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 NURGE-0170 and NUREG/CR-6672 is the result of new data and observations and improved modeling techniques.

1.2 Risk

Risk provides an understanding of events that might happen in the future. Because risks are projections of potential future events, calculations of risk are estimates and based on approximations taken from historical or analytical data.

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. However, 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 are projections of potential accidents and events, when and where they will happen, and how severe they will be. Risk estimates include estimating the likelihood and the severity of transportation accidents and 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. An event with 100% probability of occurrence is an event that is certain to happen. In reality, very few events are certain to happen or certain not to happen (zero probability). The probability of most events is between these two extremes. For example, transportation accidents involving large trucks have a very low probability (or we would hesitate to drive on the same freeway as a large truck). The probability of a traffic accident is about 1/100,000 per mile according to the Department of Transportation Bureau of Transportation Statistics (DOT, 2007), and the probability of a particular traffic accident scenario that includes vehicles carrying casks of radioactive material is much smaller still, as shown in the event trees in Appendix II of this document.

1.2.1 Accident Data

The only data available to estimate the future probability of a scenario 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.[2] Even accidents involving all hazardous materials transportation do not provide a large enough data base for statistical validity. The database used in this study is the frequency of highway accidents involving large semi-detached trailer trucks and the frequency of freight rail accidents (DOT, 2007). Freight rail accident frequency is based on accidents per railcar-mile.

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

1.2.2 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. The risk associated with 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:

- 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 referred to as "incidents."[3]
  - Accidents where damage to the vehicle and/or trailer is enough that the vehicle cannot move from the scene of the accident under its own power, but does not result in damage to the spent fuel cask.
  - Accidents involving a death or injury, but there is 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 no release of radioactive material.
  - Accidents in which there is a release of radioactive material.

Traffic accident statistics (accident frequencies) are used in the accident analysis to calculate risks from accidents. Average traffic accident frequencies since 1996 for large semi-detached trailer trucks are about two accidents per million highway kilometers (0.0019 per thousand km) and for freight rail, about one accident per ten million railcar kilometers (0.00011 per thousand railcar-km). The overall accident scenario probability is the product of the probability that an accident will happen (accident rate times distance) 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."

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

1.3 Regulation of Radioactive Materials Transportation

Transportation of radioactive materials on public rights of way is regulated by the NRC under 10 CFR Part 71 and by the Department of Transportation (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. DOT regulations include labeling, occupational and vehicle standards, registration requirements, reporting requirements, and packaging regulations. In general, only the DOT packaging regulations apply to industrial and Type A packages, while both the DOT and the NRC regulations apply to Type A(F) fissile materials packages and Type B packages. Industrial and Type A non-fissile packages are designed for routine transportation and are not certified to maintain their integrity in accidents, though many do. Type B packages are used to transport very radioactive materials. They are designed to maintain their integrity in severe accidents because the NRC recognizes that any transport package and vehicle may be in traffic accidents. This study addresses the transportation of spent nuclear fuel, and thus concerns itself only with Type B packages.

Nuclear fuel that has undergone fission ("burned") in a reactor is both extremely hot and extremely radioactive when it is removed from the reactor. In order to cool thermally and to allow the very radioactive and short-lived fission products in the fuel to decay, the fuel is moved from the reactor into a large pool of water. The fuel remains in this pool for at least three to five years, until it can be remotely handled safely. The fuel usually remains in the pool as long as there is space for it. After the fuel has cooled sufficiently it can be removed from the pool to dry surface storage, either at the nuclear power plant or other location. Fuel is almost never transported before it has cooled for five years. The transportation casks used are rated for heat load. This rating 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. Therefore ransportation containers for very radioactive materials like spent nuclear fuel are 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 cask.

Type B containers are designed to pass the series of tests described in 10 CFR 71.73, summarized below:

- A 30-foot drop onto an unyielding target. "Unyielding" in the test context means that the target is hard enough and heavy enough that the cask absorbs all the impact energy when it drops and the target does not absorb any impact energy. This is a test condition, meant to be more severe than an actual transportation accident. This drop is followed by

[4] Although release of a specific quantity of each radionuclide is allowed by regulation, Type B casks are typically designed to not release any material.

- A 40-inch drop onto a 6-inch-diameter steel cylinder to test resistance to punctures. This test is followed by
- A 1475 °F fire that fully engulfs the cask for 30 minutes.
- Immersion for eight hours under three feet of water.

Figure 1-1 illustrates the four types of tests the cask must be demonstrated to resist prior to being certified.

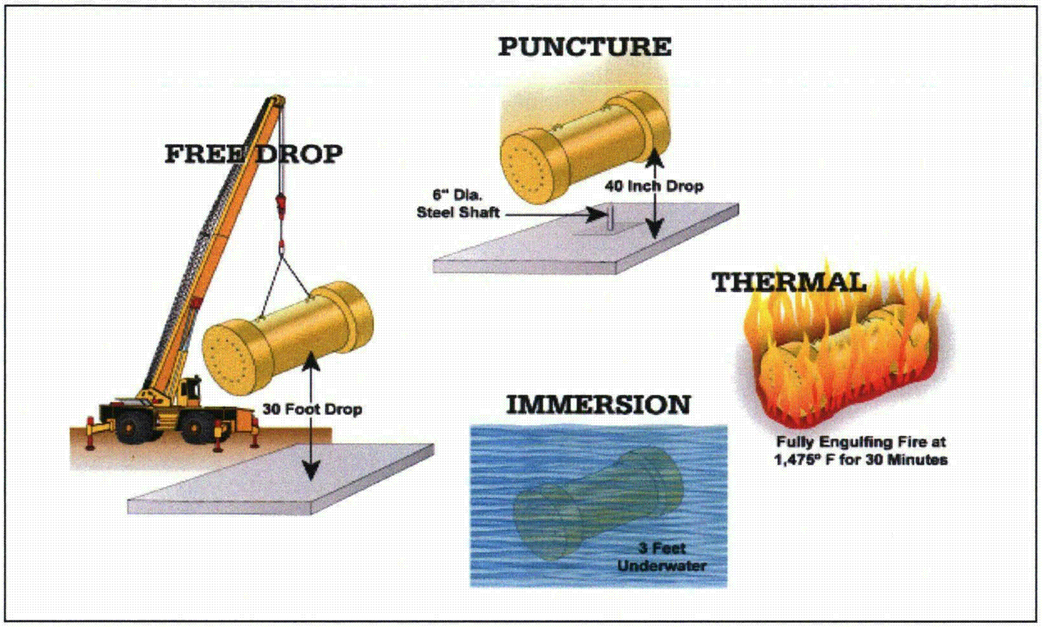


Figure 1-1. The four tests for Type B casks

The package tests in 10 CFR 71.73 are generally representative of hypothetical accident conditions. These tests are not intended to represent any specific transportation route, any specific historical transportation accident, or "testing to failure" for a specific package. These tests are intended to simulate the damaging effects of a severe transportation accident (but not all conceivable accidents) in a manner that provides international acceptability, uniformity, and repeatability.

The tests are performed on a package design. 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 reference to previous satisfactory demonstrations of a similar nature. In practice, the safety analysis performed for Type B casks often incorporates a combination of physical testing, computer modeling, and engineering evaluation. The packaging Safety Analysis Report contains the information about the package design's performance in the tests and an evaluation against the acceptance criteria in 10 CFR 71.

In addition to confirming that the tests were performed and that the package performance in the tests met the acceptance criteria, NRC staff reviews drawings to confirm that the safety-related aspects of the package are accurately described and conform to the design that was tested. Staff also reviews the acceptance criteria for packaging fabrication, to confirm that appropriate tests and criteria are specified for materials and components that affect package performance. Staff also reviews the operating instructions and maintenance program, which is a part of the application. The operating instructions typically identify tests or packaging components which must be inspected prior to making a shipment in the package. For example, visual inspection of welds, examination for dents or damage, and leak testing of the package are specified. The maintenance program typically addresses those items which must be periodically inspected, repaired or replaced.

NRC relies on all of these types of activities to ensure the safety of transportation. "Production" model packages – those intended to be used for transportation in commerce – would not each go through the Part 71 design tests. After the package had been through the tests, it is no longer in the pre-test condition. It is the pre-test condition package that is authorized for transport (IAEA, 2008).[5]

NRC regulations specify that total release of material from the cask over a period of one week following the tests can be no more than that allowed to be shipped in a non-accident resistant Type A package. The regulation also permits a maximum post-test external radiation dose rate of 0.01 Sv per hour.

10 CFR Part 20

This section of the Code of Federal Regulations prescribes the largest radiation dose that a member of the public should receive from NRC-licensed facilities, exclusive of background radiation, diagnostic or therapeutic radiation, or material that has been discharged to the environment in accordance with NRC regulation. These doses are:

- 1 mSv per year total effective dose equivalent (TEDE), including both external and committed internal dose.
- 0.02 mSv per hour in any unrestricted area from external sources. As Table 2-12 shows, the doses from routine, incident-free transportation are considerably below these limits.
- 5 mSv 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 provides 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 TEDE
- 0.15 Sv/year to the lens of the eye
- 0.5 Sv/year to the skin.

[5] The authors are indebted to Michele Sampson of the NRC staff for the information on cask testing.

1.4 Selection of Casks

Past generic risk assessments for the transportation of spent fuel have used generic casks with features similar to real casks, but generally without all of the conservatisms that are part of real cask designs. In this effort, the generic risk assessment is 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 sub-set of casks was selected. Appendix I lists the various spent fuel casks that were certified by the NRC at the time the study began, provides the method of selecting the casks to be used, identifies the important features of the various cask designs, and concludes with the chosen casks.

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

Table 1-1. NRC Certified Commercial Light Water Power Reactor Spent Fuel Casks

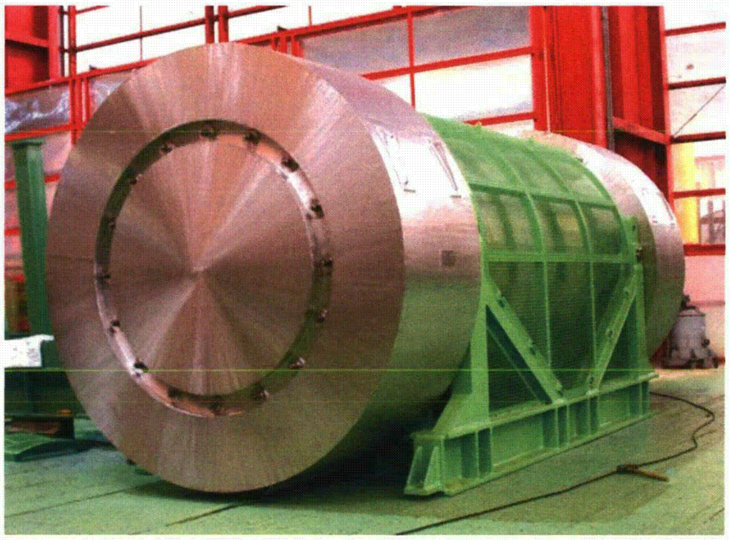
Cask	Package ID	Canister	Contents (Number of assemblies)	Type
IF-300	USA/9001/B( )F	No	7 PWR, 17 BWR	Rail
NLI-1/2	USA/9010/B( )F	No	1 PWR, 2 BWR	Truck
TN-8	USA/9015/B( )F	No	3 PWR	Overweight[a]
TN-9	USA/9016/B( )F	No	7 BWR	Overweight[a]
NLI-10/24	USA/9023/B( )F	No	10 PWR, 24 BWR	Rail
NAC-LWT	USA/9225/B(U)F-96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

[a]Overweight truck

The casks chosen for detailed analysis are the NAC-STC (Figure 1-2) and the Holtec HI-STAR 100 (Figure 1-3) rail casks. The GA-4 cask (Figure 1-4) will be used to evaluate truck shipments, but detailed impact analyses of this cask will not be 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. are used to assess the response of the GA-4 cask. The complete Certificates of Compliance (as of April 12, 2010) for each of these casks is included in Appendix I. The NAC-STC cask was chosen because it is certified for transport of spent fuel either with or without an internal welded canister. Its certificate of compliance allows use of either elastomeric o-rings or metallic o-rings.

Even though there are five casks in the group that use lead as their gamma shielding, of this group only the NAC-STC can transport fuel that is not contained within an inner welded canister. The HI-STAR 100 rail cask is chosen because it is the only all-steel cask in the group that is certified for transport of fuel in an inner welded canister. The GA-4 truck cask is chosen because it has a larger capacity than the NAC-LWT truck cask, and therefore is more likely to be used in any large transportation campaign.

The choice of rail casks allows comparison between directly loaded and canistered fuel, comparison between a steel-lead-steel cask and an all-steel cask, and comparison between elastomeric o-ring seals and metallic o-ring seals.



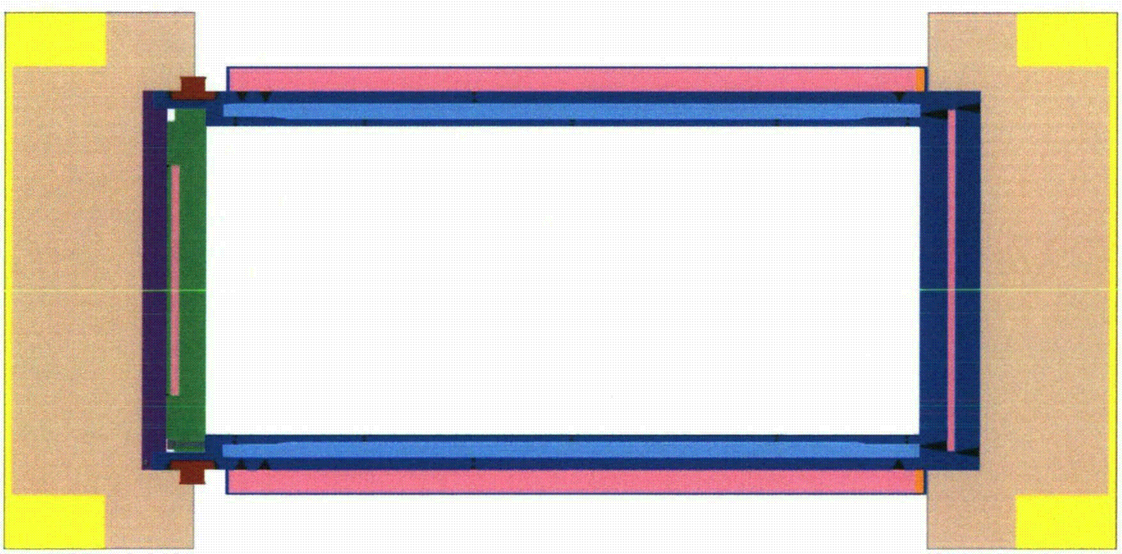


Figure 1-2. NAC-STC cask (courtesy of NAC International)

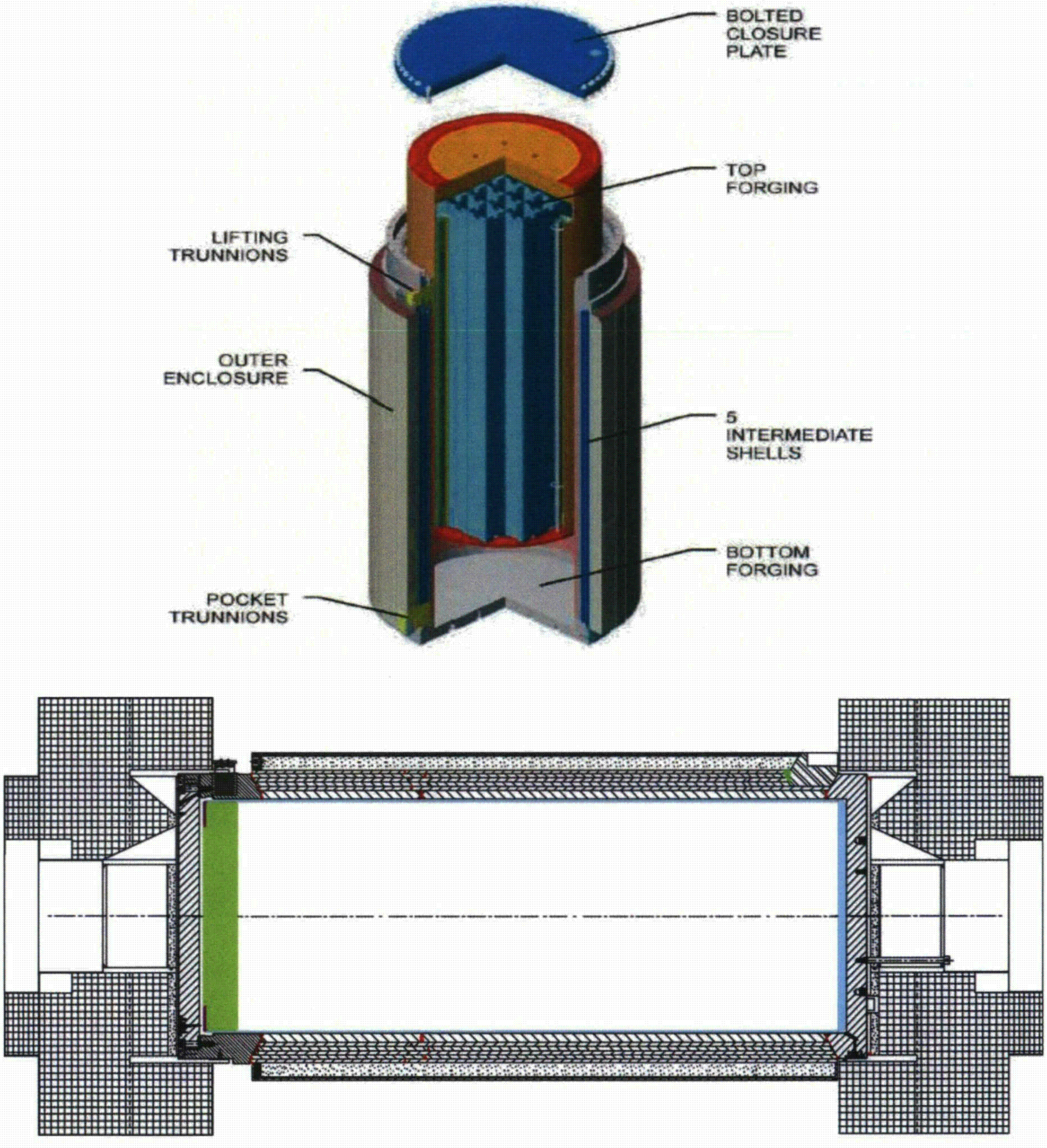


Figure 1-3. Basic layout of the HI-STAR 100 rail transport cask (from Haire and Swaney, 2005, and Holtec International, 2004)

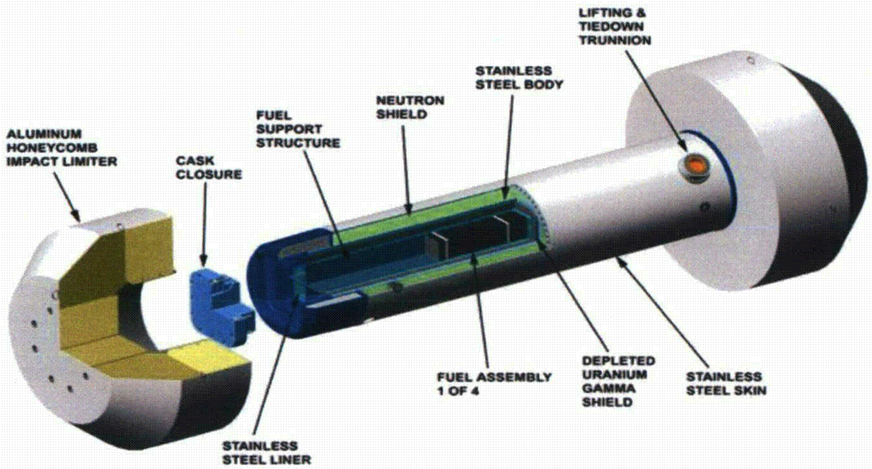


Figure 1-4. GA-4 cask (courtesy of General Atomics)

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

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.

1.5 Organization of this Report

Each chapter in this study, except the 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.

1.5.1 Chapter 1 and Appendix I

This chapter provides an introduction to the study, a brief background discussion, a discussion of risk as applied to transportation of radioactive materials, a discussion of cask selection, and a review of the organization of the report. Appendix I contains details about the cask selections and the Certificates of Compliance of the selected casks.

1.5.2 Chapter 2 and Appendix II

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

1.5.3 Chapter 3 and Appendix III

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

1.5.4 Chapter 4 and Appendix IV

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

1.5.5 Chapter 5 and Appendix V

Chapter 5 and Appendix V address the RADTRAN analysis of transportation accidents, development of accident event trees and conditional probabilities, development of the radionuclide inventory and radioactive materials releases and dispersion of released material in the environment. The chapter also discusses accidents in which there are no releases – the most likely accidents – in which the radioactive cargo is not affected at all, and an essentially undamaged cask and its conveyance (which may be damaged) sit for many hours at the accident location.

1.5.6 Chapter 6

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

1.5.7 Bibliography

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

CHAPTER 2

RISK ANALYSIS OF ROUTINE TRANSPORTATION

2.1 Introduction

NUREG-0170 (NRC, 1977) was the first comprehensive assessment of the environmental and health impact of transporting radioactive materials, and documented estimates of the radiological consequences and risks associated with the shipment by truck, train, plane, or barge of about 25 different radioactive materials, including power reactor spent fuel. However, 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 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[9]. The risks and consequences of accidents and incidents interfering with routine transportation are discussed in Chapter 5.

This chapter includes the following:

- A brief discussion of ionizing radiation emitted during transportation.
- A description of the RADTRAN model of routine transportation.

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[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.
[9] Non-radiological vehicular risks are not compared to radiological risks because the radiological risks are expressed as doses rather than health effects.

- Radiation doses from a single routine shipment to:
  - Members of the public who live along the transportation route and near stops
  - Occupants of vehicles that share the route with the radioactive shipment
  - Various groups of people at stops
  - Workers

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

2.2 Radiation Emitted during Routine Transportation

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

Spent nuclear fuel, the subject of this analysis, is extremely radioactive, emitting ionizing radiation in the form of alpha, beta, gamma, and neutron radiation. The casks that are used to transport spent nuclear fuel have exceedingly thick walls that absorb most of the emitted ionizing radiation and thereby shield the public and the workers. Figure 2-1 shows two generic cask diagrams on which the shielding is identified.

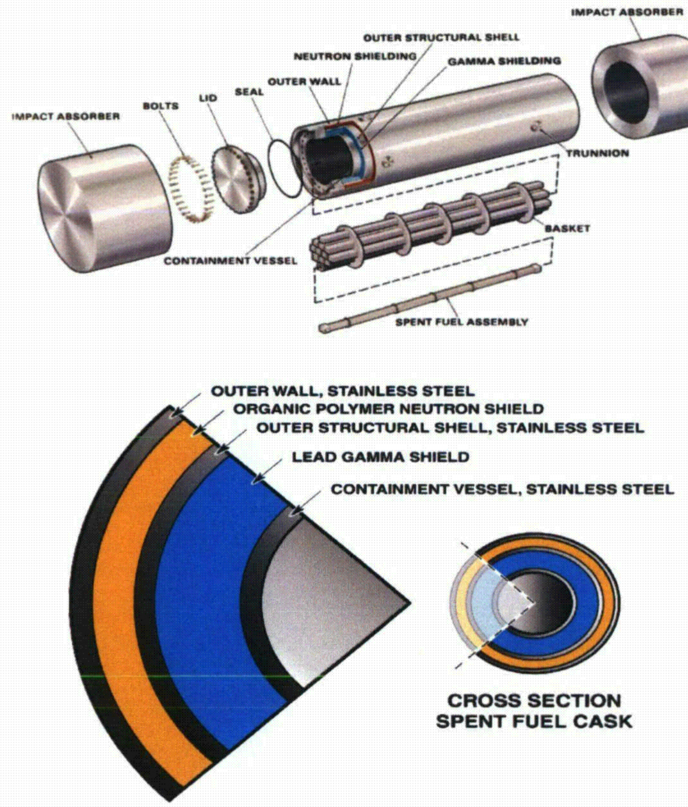

Figure 2-1. The upper figure is an exploded view of a generic spent fuel cask. The lower figure is a cross-section of the layers of the cask wall. (Sandia National Laboratories archive)

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

Absorbed radiation dose is measured in sieverts (Sv) in the Standard International system. Average U. S. background radiation from naturally occurring and some medical sources is 0.0036 Sv per year (Shleien et al., 1998, Figure 1.1),[10] a single dental x-ray delivers a dose of 4×10^{-5} Sv, and a single mammogram delivers 1.3×10^{-4} Sv (Stabin, 2009). The average radiation dose rate from a spent fuel cask allowed by regulation is 0.1 mSv per hour, measured at two meters from the outside of the cask (10 CFR71.47(b)(3)), or about 0.14 mSv/hour at one meter from a cask four to five meters long.

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

The external radiation doses from the casks in this study (Figures 1-3 to 1-5) at two meters from the cask, as reported in the cask Safety Analysis Reports, are shown in Table 2-1. Measured values for the Rail-Steel and 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 at 1 m	0.00014	0.00014	0.000103
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 divided by ten, 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 enter into the calculation of doses from routine transportation. Doses from the external radiation from the cask depend on the external dose rate (TI), 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. The diameter of this spherical model, called the "critical dimension," is the longest dimension of the actual spent fuel cask.

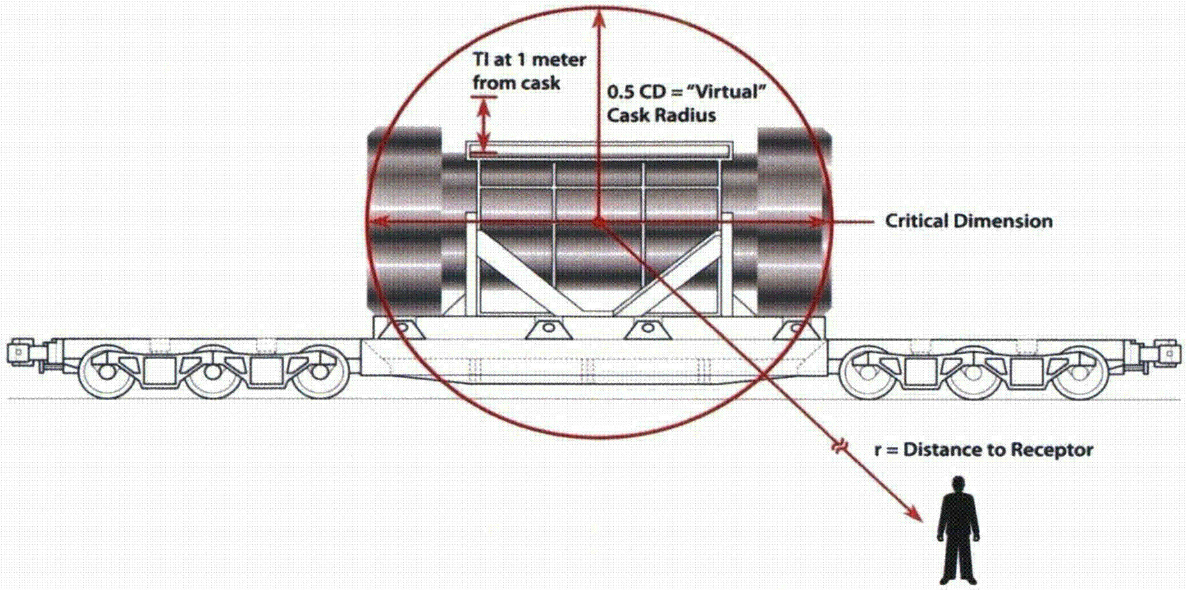

Figure 2-2. RADTRAN model of the vehicle in routine, incident-free transportation. The cask in this diagram is positioned horizontally, and the critical dimension is the cask length.

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, their 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 anyone along the vehicle's route is exposed. Therefore, an individual member of the public gets the largest dose from a moving vehicle when he or she is as close as possible to the vehicle, and the vehicle is traveling as slowly as possible. For trucks and trains carrying spent fuel, a speed of 24 km per hour (kph) and distance of 30 meters (about 100 feet) are assumed for maximum exposure.[11] Table 2-1 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.

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

Table 2-1. Maximum individual in-transit doses

Package	Dose
Rail-Lead	5.7E-09 Sv
Rail-Steel	4.3E-09 Sv
Truck-DU	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. A collective dose, sometimes called a population dose, is essentially an average individual dose multiplied by the number of people exposed.[12] As shown in Figure 2-3, RADTRAN calculates collective doses along transportation routes by integrating over the width of a band along the route where the population resides (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 (about a half mile) on either side of the route: from 30 meters (10 feet) from the center of the route to 800 meters.

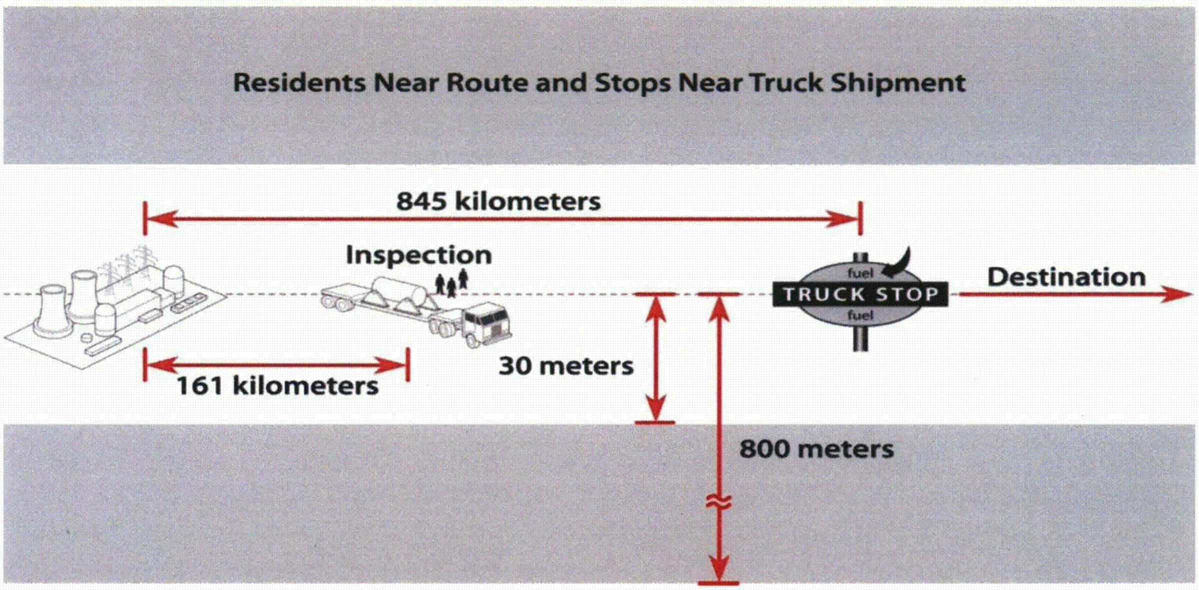

Figure 2-3. Diagram of a truck route as modeled in RADTRAN (not to scale)

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

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

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

rural, suburban, and urban segments according to the population per square mile (population density). Table 2-2 summarizes the characteristics of each population type that are part of the dose calculation by RADTRAN. References for these parameter values are in the Table 2-3 footnotes.

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

	Highway			Rail		
	Rural	Suburban	Urban	Rural	Suburban	Urban
Population density per km^2 (per mi^2)[a]	0 to 54 (0 to 139)	54 to 1286 (139 to 3326)	>1286 (>3326)	0 to 54 (0 to 139)	54 to 1286 (139 to 3326)	>1286 (>3326)
Nonresident/ resident ratio[b]	NA	NA	6	NA	NA	6
Shielding by buildings[b]	0	13%	98.2%	0	13%	98.2%
U.S. average vehicle speed[c] kph (mph)[c,d]	108 (67)	108 (67)	101(63)	40 (27)	40 (27)	24 (15)
U.S . average vehicles per hour[e,f]	1119	2464	5384	17	17	17
Occupants of other vehicles[e,g]	1.5	1.5	1.5	1	1	5

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

Each route clearly has a distribution of rural, urban and suburban areas, as shown by the example of the truck route in Figure 2-4.

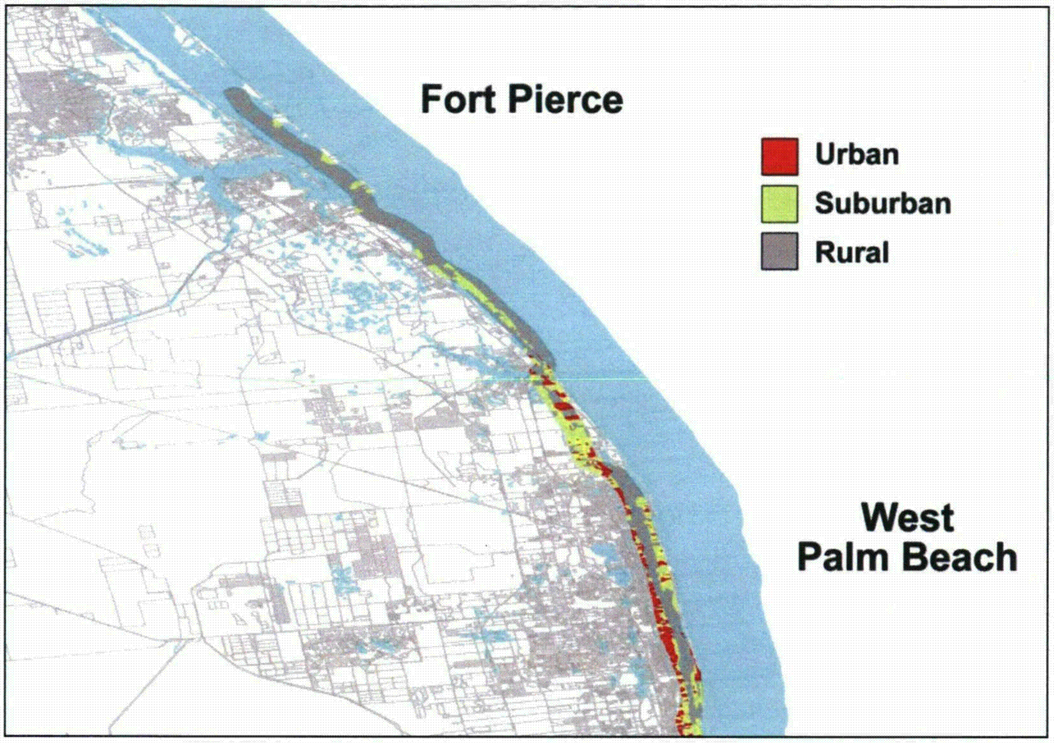

Figure 2-4. A segment of U.S.1/1A along the Florida coast. The gray band indicating a rural route is actually located on the western (land) side of the Intracoastal Waterway. (courtesy of G. Scott Mills)

Figure 2-4 shows a segment of U.S.1along the Florida coast and Intracoastal Waterway from West Palm Beach to Fort Pierce. The broad stripe along the coastline of the Intracoastal Waterway is the half-mile band on either side of the highway. The red areas represent urban populations, the yellow areas, suburban, and the gray areas, rural. Instead of analyzing each separate rural, urban, and suburban segment of this stretch of highway, each type of each is combined for RADTRAN dose calculations. The routing code WebTRAGIS (Johnson and Michelhaugh, 2003) provides these combinations for each state traversed by a particular route. Table 2-3 shows this WebTRAGIS output for a sample rail route from Kewaunee Nuclear Plant, WI to Oak Ridge National Laboratory.

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).

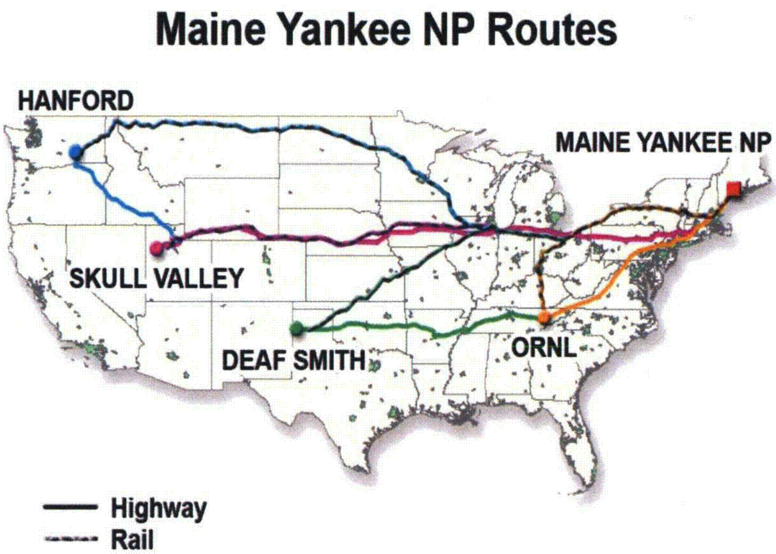

Figure 2-5. Highway and rail routes from Maine Yankee Nuclear Plant site.

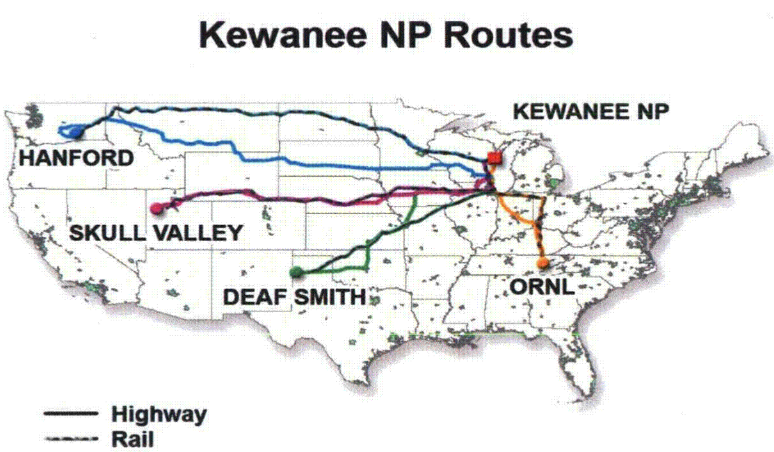

Figure 2-6. Highway and rail routes from Kewaunee Nuclear Plant.

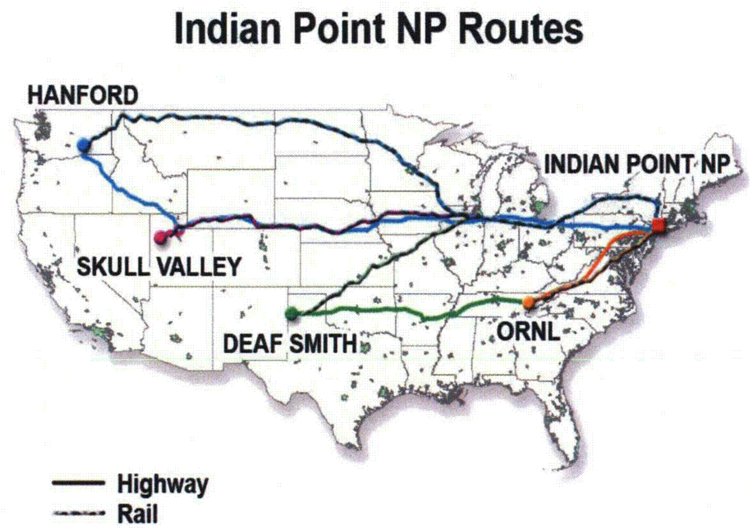

Figure 2-7. Highway and rail routes from Indian Point Nuclear Plant.

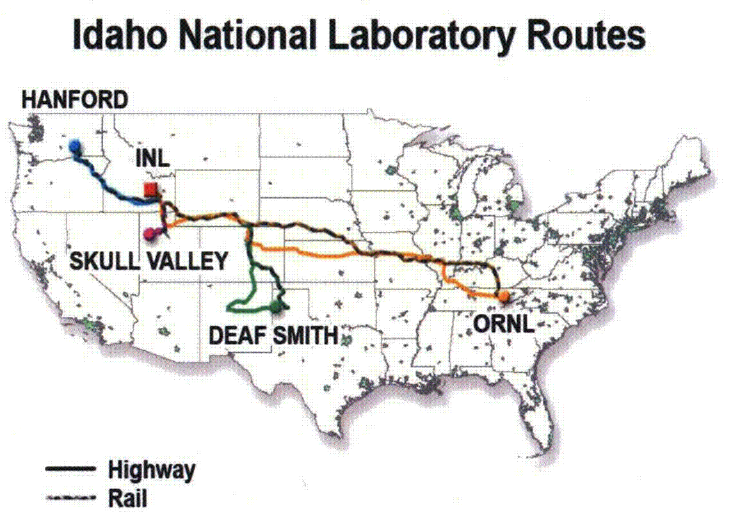

Figure 2-8. Highway and rail routes from Idaho National Laboratory.

The route segment lengths and population densities are entered into RADTRAN, which 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 32 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 a National Laboratory (Idaho National Laboratory). The routes modeled are shown in Table 2-4. Both truck and rail versions of each route were analyzed.

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. 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. The destination sites include two proposed repository sites (Deaf Smith County, TX and Hanford, WA) (DOE, 1986), the site of the proposed Private Fuel Storage facility (Skull Valley, UT), and a National Laboratory site (Oak Ridge, TN; ORNL). The chosen origin and destination sites do not represent any planned spent fuel transportation plans, but provide a good representation of the lengths and population densities that could be encountered during any shipment of spent fuel.

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 2-5 and Table 2-6 are in units of person-Sv.

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

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

Table 2-5 and Table 2-6 present collective doses for rail and truck, respectively, for each of the sixteen origin/destination pairs. 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	Total	Rural	Suburban	Urban	Total
MAINE YANKEE	ORNL	2.5E-05	2.9E-04	1.4E-05	3.3E-04	1.9E-05	2.2E-04	1.1E-05	2.5E-04
	DEAF SMITH	3.0E-05	3.5E-04	1.8E-05	4.0E-04	2.3E-05	2.7E-04	1.4E-05	3.0E-04
	HANFORD	3.8E-05	4.1E-04	2.1E-05	4.7E-04	2.9E-05	3.1E-04	1.6E-05	3.6E-04
	SKULL VALLEY	4.2E-05	4.1E-04	1.4E-05	4.7E-04	3.2E-05	3.2E-04	1.1E-05	3.6E-04
KEWAUNEE	ORNL	1.7E-05	1.7E-04	1.1E-05	2.0E-04	1.3E-05	1.3E-04	8.1E-06	1.5E-04
	DEAF SMITH	1.3E-05	1.5E-04	9.2E-06	1.7E-04	1.0E-05	1.2E-04	7.1E-06	1.3E-04
	HANFORD	1.6E-05	1.5E-04	4.7E-06	1.7E-04	1.2E-05	1.1E-04	3.6E-06	1.2E-04
	SKULL VALLEY	2.3E-05	1.9E-04	1.1E-05	2.2E-04	1.7E-05	1.4E-04	8.0E-06	1.7E-04
INDIAN POINT	ORNL	1.2E-05	2.3E-04	5.8E-05	3.0E-04	9.1E-06	1.7E-04	1.7E-05	2.0E-04
	DEAF SMITH	2.7E-05	2.3E-04	1.9E-05	2.8E-04	2.0E-05	2.1E-04	1.4E-05	2.5E-04
	HANFORD	3.5E-05	3.4E-04	2.1E-05	4.0E-04	2.5E-05	2.3E-05	9.0E-06	5.8E-05
	SKULL VALLEY	3.6E-05	3.2E-04	2.1E-05	3.8E-04	2.8E-05	2.4E-04	1.6E-05	2.9E-04
IDAHO NATIONAL LAB	ORNL	2.8E-05	1.8E-04	6.0E-06	2.1E-04	2.2E-05	1.4E-04	4.5E-06	1.7E-04
	DEAF SMITH	1.1E-05	8.9E-05	8.9E-06	1.1E-04	7.7E-06	6.8E-05	6.8E-06	8.2E-05
	HANFORD	8.5E-06	4.7E-05	1.7E-06	5.8E-05	6.5E-06	3.6E-05	1.3E-06	4.4E-05
	SKULL VALLEY	4.9E-06	4.1E-05	2.3E-06	4.8E-05	3.7E-06	3.1E-05	1.8E-06	3.6E-05

Table 2-6. Collective doses (person-Sv) for truck transportation

FROM	TO	Truck-DU				
		Rural	Suburban	Urban	Urban Rush Hour[a]	Total
MAINE YANKEE	ORNL	7.9E-06	1.4E-04	2.9E-06	2.6E-07	1.51E-04
	DEAF SMITH	1.4E-05	1.9E-04	3.3E-06	3.4E-07	2.08E-04
	HANFORD	2.2E-05	1.7E-04	2.3E-06	4.2E-07	1.95E-04
	SKULL VALLEY	1.8E-05	1.5E-04	2.3E-06	4.2E-07	1.71E-04
KEWAUNEE	ORNL	6.5E-06	7.4E-05	1.8E-06	2.7E-07	8.26E-05
	DEAF SMITH	1.1E-05	6.3E-05	1.2E-06	2.5E-07	7.55E-05
	HANFORD	1.4E-05	6.6E-05	1.1E-06	6.6E-08	8.12E-05
	SKULL VALLEY	1.2E-05	5.0E-05	1.1E-06	9.5E-08	6.32E-05
INDIAN POINT	ORNL	6.1E-06	8.9E-05	1.2E-06	2.6E-07	9.66E-05
	DEAF SMITH	1.1E-05	1.1E-04	1.6E-06	3.4E-07	1.23E-04
	HANFORD	2.1E-05	1.2E-04	1.9E-06	4.2E-07	1.43E-04
	SKULL VALLEY	1.7E-05	1.1E-04	1.9E-06	4.2E-07	1.29E-04
IDAHO NATIONAL LAB	ORNL	1.4E-05	8.4E-05	1.2E-06	2.7E-07	9.95E-05
	DEAF SMITH	7.3E-06	4.9E-05	1.1E-06	2.5E-07	5.77E-05
	HANFORD	4.2E-06	2.0E-05	3.0E-07	6.6E-08	2.46E-05
	SKULL VALLEY	2.0E-06	1.6E-05	4.3E-07	9.5E-08	1.85E-05

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

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

- Urban residents sustain a slightly larger dose from a single rail shipment than from a truck shipment on the same state route, even though urban population densities are similar and the external dose rates from the cask are nearly the same. As shown in Table 2-4, 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 2-45, 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 2-5 and Table 2-6 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 x 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 1.5×10^{-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. Everyone is exposed to this background all the time, whether a shipment occurs or not .
- A truck traveling at an average of 108 kph travels the 1747 km in 16 hours.
- During those 16 hours, the 1.34 million people will have received a collective background dose of 8.81 person-Sv, about 600,000 times the collective dose from the shipment.
- The total collective dose during a shipment to this 1.34 million people is not 1.5×10^{-5} person-Sv), but 8.810015 person-Sv.
- The NRC recommends that collective dose be used only in comparisons (NRC, 2008).
- The appropriate comparison between the collective dose from this shipment of spent fuel is thus not a comparison between 1.5×10^{-5} person-Sv from the shipment and zero dose if there is no shipment, but between 8.810015 person-Sv if there is a shipment and 8.810000 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 2-7 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.[13] 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.

[13] 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	2.6E-05	2.1E-05	2.1E-05	2.0E-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	8.0E-05	2.3E-05	2.9E-05	6.1E-05	1.8E-05	2.4E-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	6.1E-05	2.0E-05	2.9E-05	4.6E-05	1.5E-05	2.4E-05
	SKULL VALLEY	6.2E-05	1.9E-05	5.3E-06	4.7E-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 2-9, a model of the RADTRAN calculation. The occupants of any car or truck that passes the spent fuel cask in either direction will sustain a small radiation dose.

The radiation dose to occupants of other vehicles depends on the exposure distance and time, the number of other vehicles on the road, and the number of people in the other vehicles. 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. A more complete discussion of the calculation method is in Appendix II and 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[14] (Weiner, et al., 2009, Appendix D) were:

- 1119 on rural segments, about 2 ½ times the 1977 vehicle density.

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[14] 2004 is the most recent year for which data have been validated.

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

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

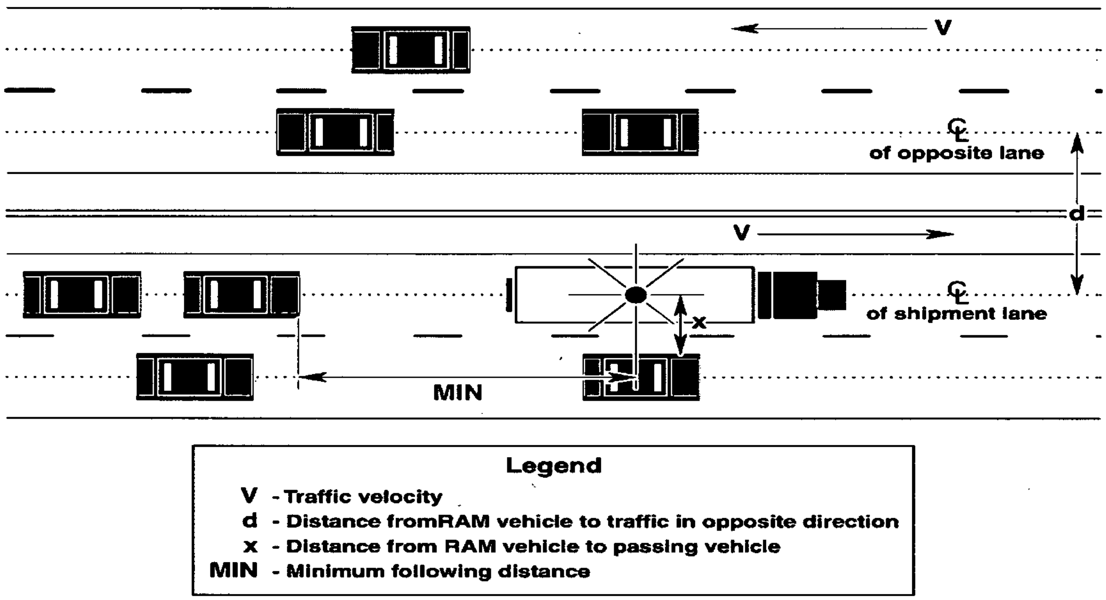


Figure 2-9. Diagram for calculating radiation doses to occupants of other vehicles (from Neuhauser et al., 2000)

Detailed discussion and state-by-state results are presented in Appendix II. The collective doses for truck traffic are shown in Table 2-8.

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

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

2.3.4 Doses at Stops

Both trucks and trains stop occasionally on long trips. Common carrier freight trains stop to exchange freight cars, to change crews, and, when necessary, to change railroads. The rail stops at the origin and destination of a trip are called "classification stops" and are 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. The shipments in this analysis are assumed to use dedicated rail. A dedicated train is a train that carries a single cargo from origin to destination; coal unit trains are a good example of dedicated trains.

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

- Railyard workers (including inspectors)
- Train crew
- Residents who live near the rail yard.

The 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 may be found in Appendix II.

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

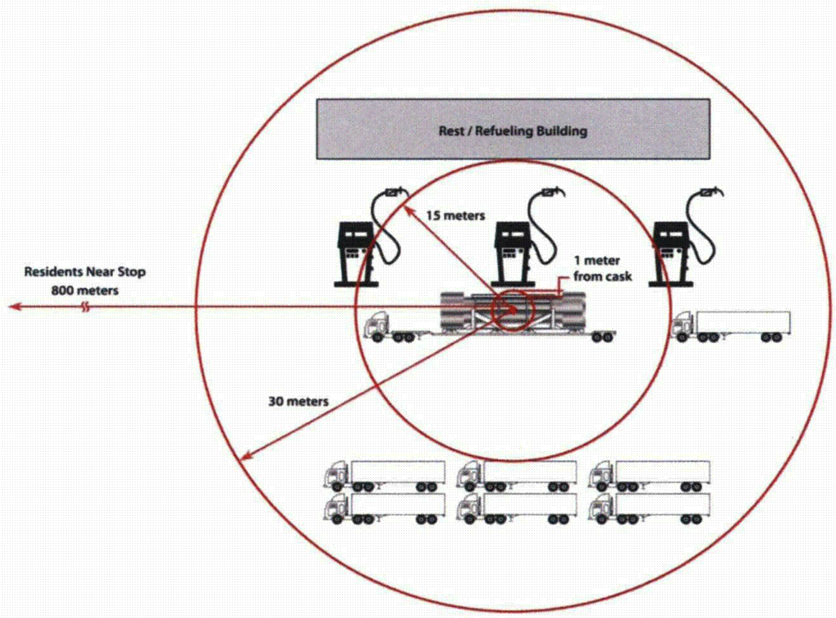


Figure 2-10. Diagram of truck stop model (not to scale)

Table 2-9 lists some sample input data used to calculate doses at stops.

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

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

[a]From Griego et al., 1996

Rail

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

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

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

- 0.35×10^{-5} Sv from the Rail-Lead
- 0.27×10^{-5} Sv from the Hi-STAR 100

Table 2-10 shows the doses at stops to yard workers and residents near the stop for the Maine Yankee-to Hanford rail route. Because different routes have different in-transit stops and stop times for crew changes and inspections, a representative result is given here instead of presenting results for an entire route or for all sixteen routes.

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

Stop	Route type and State	Time (hours)	Railyard worker		Residents near stop	
			Rail-Lead	Rail-All Steel	Rail-Lead	Rail-All Steel
1	Suburban, ME	4.0	2.2 E-05	1.6 E-05	3.4 E-05	2.6 E-05
2	Rural, NY	4.0	2.2 E-05	1.6 E-05	9.2 E-06	6.9 E-06
3	Suburban, IL	2.0	1.1 E-05	8.1 E-06	1.2 E-04	9.4 E-05

Truck

Table 2-11 shows the collective doses to residents near stops for the rural and suburban segments of the 16 routes studied. Urban stops were not modeled because trucks carrying Truck - DU casks of spent fuel are unlikely to stop in urban areas. A more detailed discussion of these calculations is in Appendix II.

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

Origin	Route	Type	Persons/km^2	Number of stops	Dose
MAINE YANKEE	ORNL	Rural	19.9	1.73	1.1E-06
		Suburban	395	2.09	2.3 E-05
	Deaf Smith	Rural	18.6	2.47	1.5 E-06
		Suburban	371	1.6	1.7 E-05
	Hanford	Rural	15.4	4.33	2.2 E-06
		Suburban	325	1.5	1.4 E-05
	Skull Valley	Rural	16.9	3.5	1.9 E-06
		Suburban	332.5	1.3	1.2 E-05
KEWAUNEE	ORNL	Rural	19.8	0.81	5.2 E-07
		Suburban	361	0.59	6.0 E-06
	Deaf Smith	Rural	361305	2.0	8.6 E-07
		Suburban	339	0.52	5.0 E-06
	Hanford	Rural	10.5	3.4	1.2 E-06
		Suburban	316	0.60	5.4 E-06
	Skull Valley	Rural	12.5	2.6	1.1 E-06
		Suburban	324.5	0.44	4.1 E-06
INDIAN POINT	ORNL	Rural	20.5	0.71	4.7 E-07
		Suburban	388	0.71	7.8 E-06
	Deaf Smith	Rural	17.1	2.3	1.3 E-06
		Suburban	370	1.2	1.3 E-05
	Hanford	Rural	13.0	4.1	1.8 E-06
		Suburban	338	1.1	1.1 E-05
	Skull Valley	Rural	14.2	3.3	1.5 E-06
		Suburban	351	0.93	9.3 E-06
IDAHO NATIONAL LAB	ORNL	Rural	12.4	3.1	1.3 E-06
		Suburban	304	0.72	6.3 E-06
	Deaf Smith	Rural	7.8	2.3	5.8 E-07
		Suburban	339	0.35	3.4 E-06
	Hanford	Rural	6.5	0.43	9.0E-08
		Suburban	200	0.57	3.2 E-06
	Skull Valley	Rural	10.1	0.42	1.4 E-07
		Suburban	343	0.11	1.1 E-06

The rural and suburban population densities in Table 2-12 are the averages for the entire route. An analogous calculation can be made for each state traversed. However, in neither case can one determine beforehand exactly where the truck will stop to refuel. In some cases (e.g., INL to Skull Valley) the truck may not stop at all; the total distance from INL to the Skull Valley site is only 466.2 km (290 miles). The route from Indian Point to ORNL illustrates another situation. This route is 1028 km (639 miles) long, and would thus include one truck stop, which could be in either a rural or a suburban area. The results shown in Table 2-11 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.

Table 2-12 summarizes the occupational doses. Workers who handle spent fuel containers in storage, loading and unloading casks from vehicles or during intermodal transfer are not addressed in this analysis. Truck refueling stops in the U.S. no longer have attendants who refuel trucks.[15] Gas station and truck stop workers are in concrete or brick buildings and would be shielded from the radiation with the same shielding as in urban housing (98% shielded).

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

* Not applicable for this combination of cask type and dose type.

2.5 Unit Risk

RADTRAN, the model used for the calculation of transportation risk, multiplies numbers. The only calculation that RADTRAN makes which is not a simple multiplication is calculating emissions from the spherical model shown in Figure 2-2. 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 are calculated for a single shipment and, even though overestimated, they are uniformly very small. Maximally exposed individual doses are comparable to background and less than doses from many medical diagnostic procedures. Collective doses are orders of magnitude less than collective background dose, as shown in Figure 2-11. The NRC recommends that collective doses (average doses integrated over a population) only be used only for comparisons (NRC, 2008). The proper comparison for collective doses is between the background collective dose plus the shipment dose and the background dose if there is no shipment. The collective dose is not zero in the absence of a shipment.

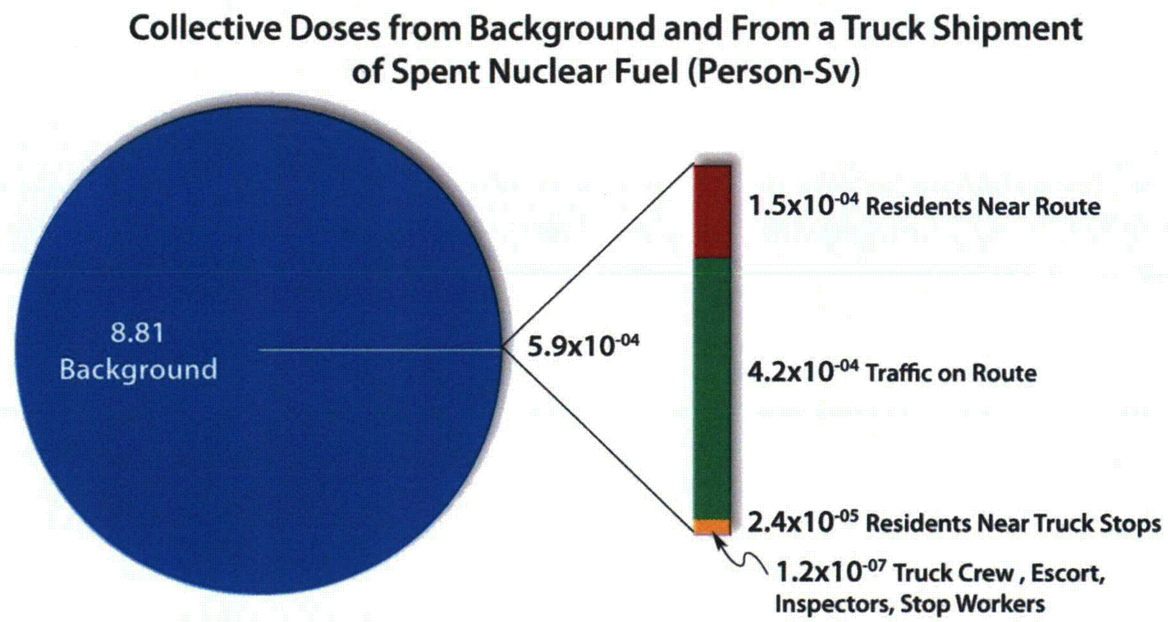


Figure 2-11. Collective doses from background and from one of the truck shipments of spent nuclear fuel (person-Sv)

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 minimum material properties, allowing only small amounts of yielding, and requiring materials with high ductility. These approaches ensure that the casks will not only survive a nine-meter drop, but will also survive much higher speed impacts.

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

3.2 Finite Element Analyses of Casks

Previous risk studies have been carried out using generic casks. In the case of the Modal Study (Fischer et al, 1987) it was assumed any accident that was more severe than the regulatory hypothetical impact accident would lead to a release from the cask. In NUREG/CR-6672 (Sprung et al., 2000) the impact limiters of the generic casks were assumed to be unable to absorb more energy than the amount from the regulatory hypothetical impact accident (a nine-meter free fall onto an essentially rigid target). Modeling limitations at the time of the studies required both of these assumptions. In reality, casks and impact limiters each have excess capacity to resist impacts. In this study, three NRC certified casks were used instead of generic casks, and the actual excess capacity of those cask designs was included in the analyses.

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

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

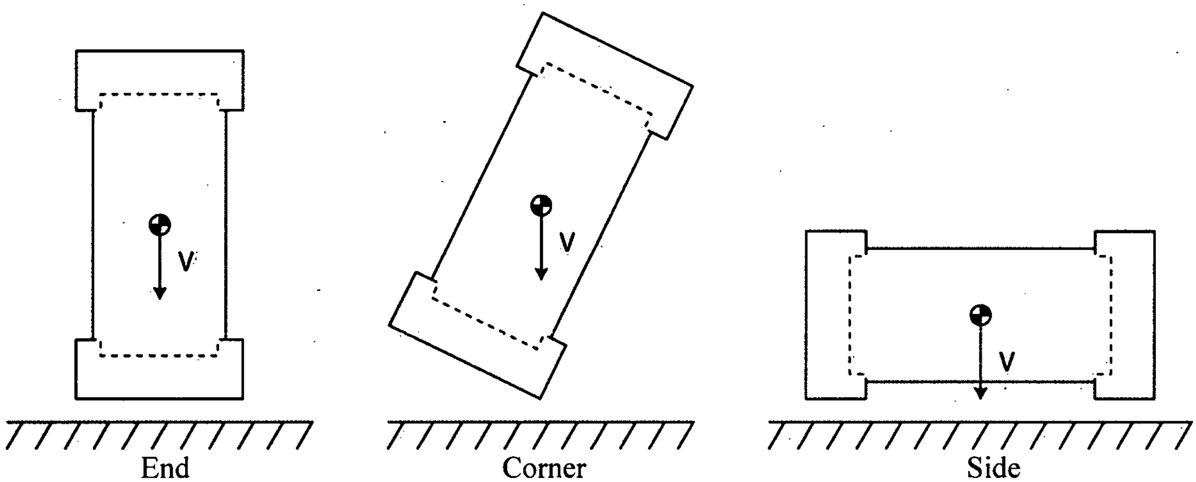


Figure 3-1. Impact orientations analyzed

3.2.1 Rail-Steel Cask

Finite element model

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

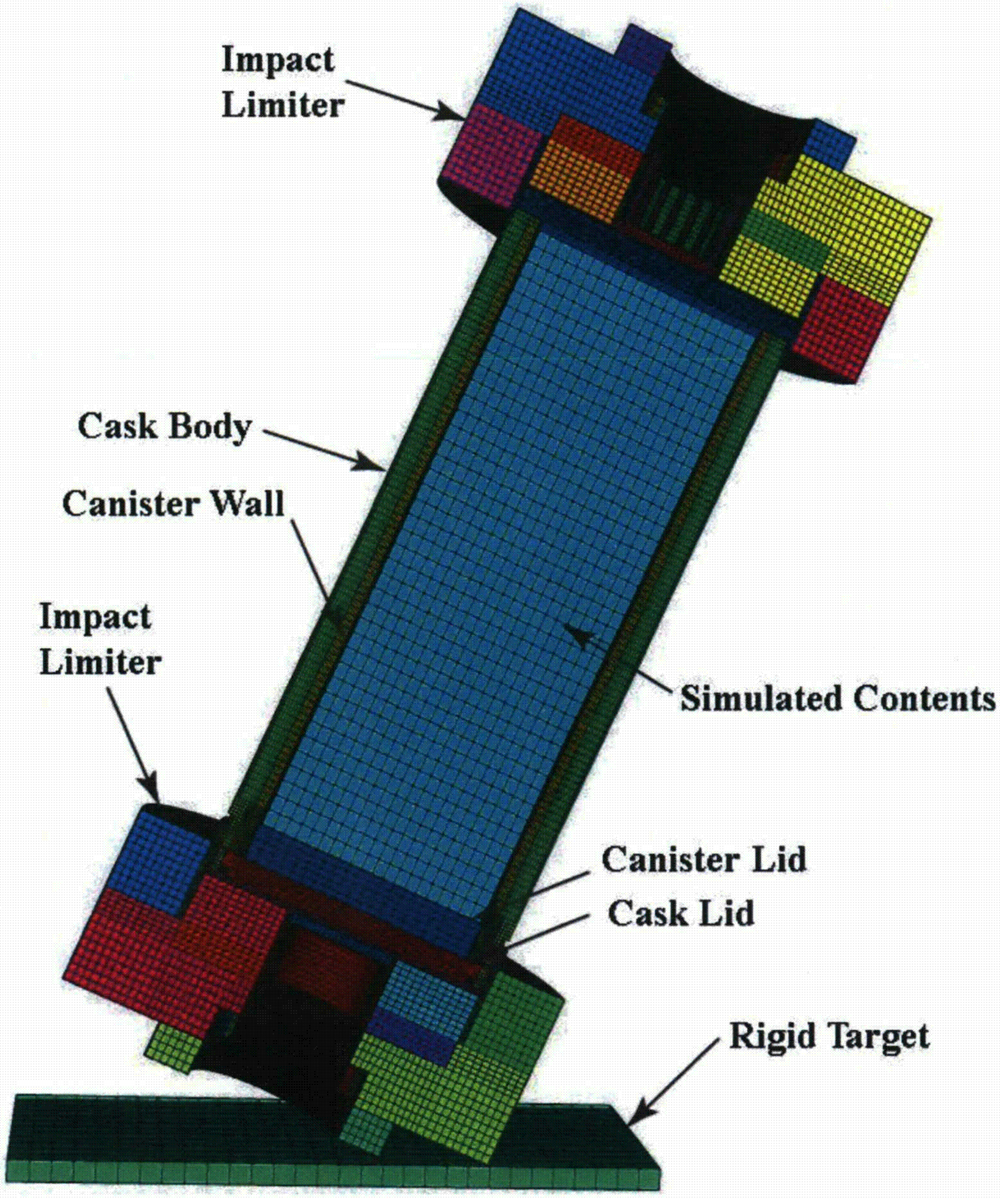
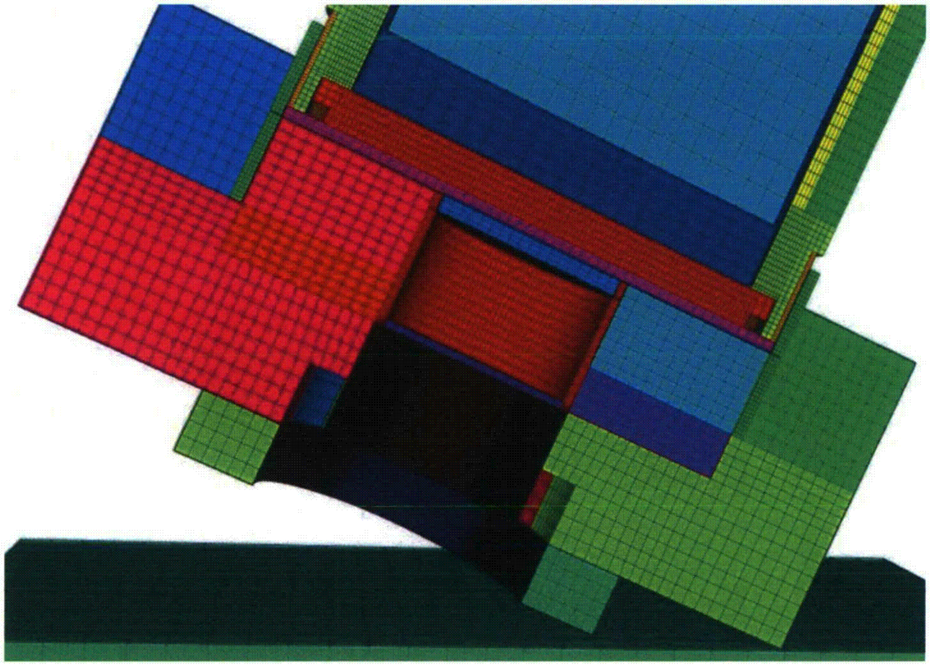
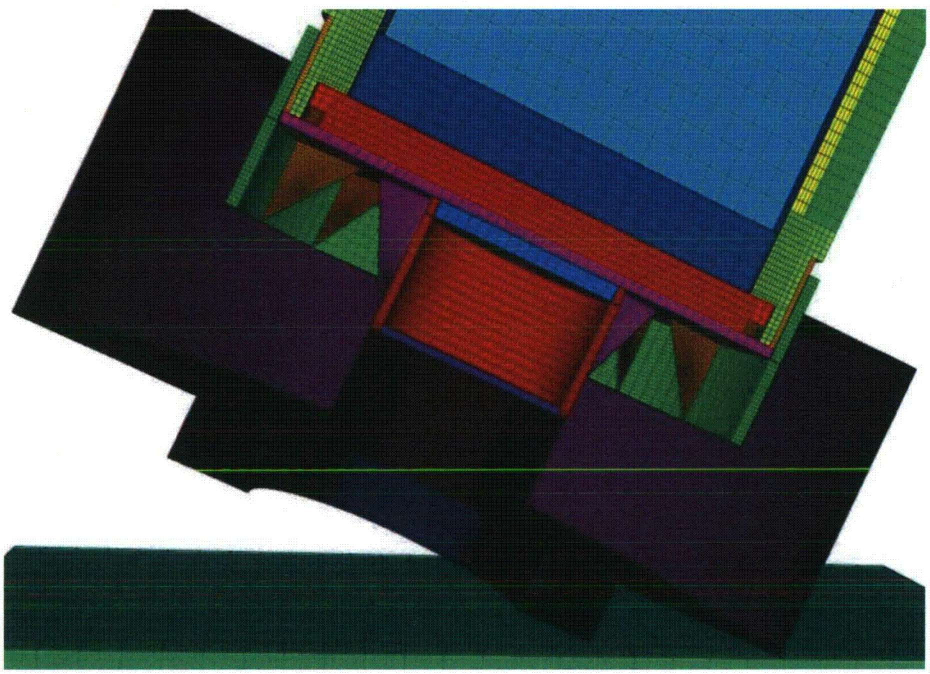


Figure 3-2. Finite element mesh of the Rail-Steel cask



Impact limiter showing the various blocks of honeycomb



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

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

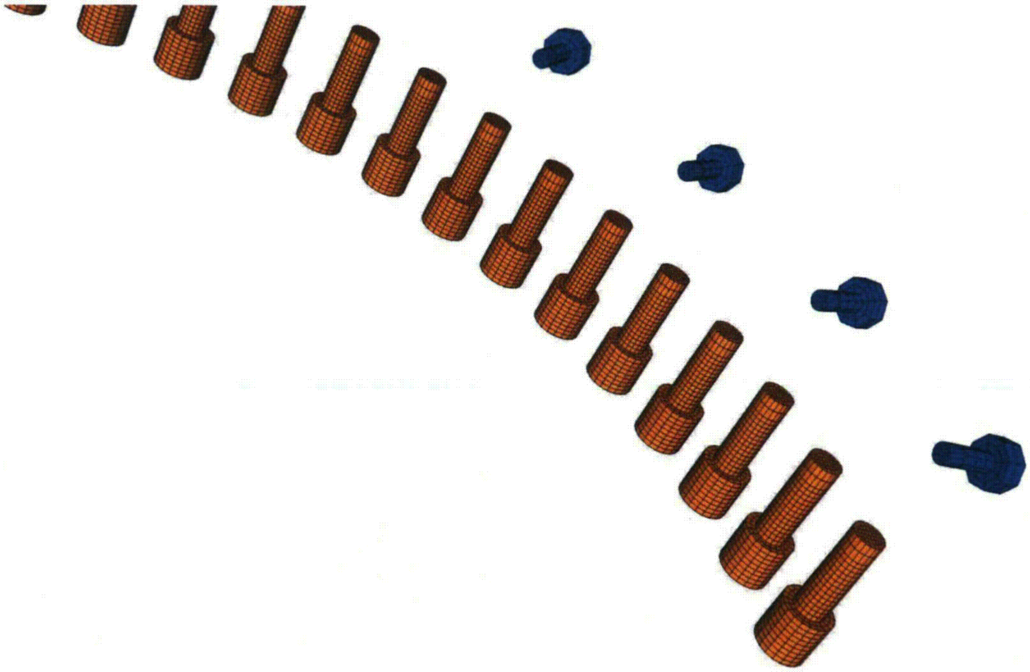


Figure 3-4. Finite element mesh of the Rail-Steel closure bolts and the closure end impact limiter attachment bolts. The highly refined mesh in these critical parts assures an accurate assessment of the closure response.

Analysis results

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

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

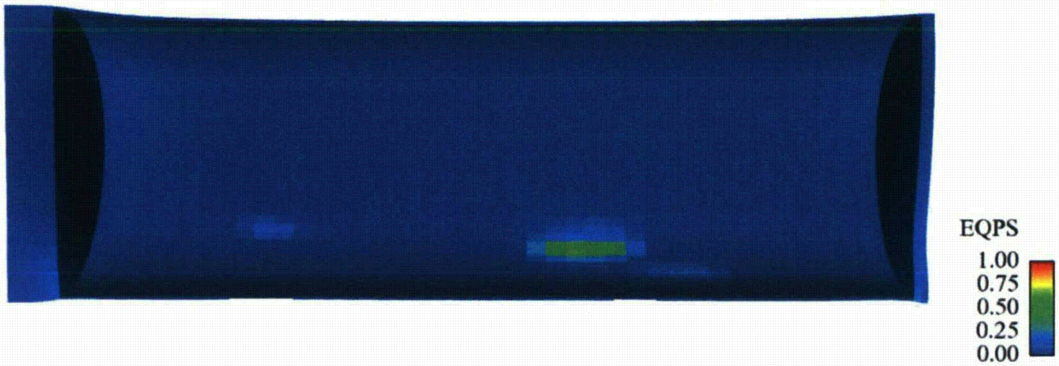


Figure 3-5. Plastic strain in the welded canister of the Rail-Steel for the 193 kph side impact case

3.2.2 Rail-Lead Cask

Finite Element Model

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

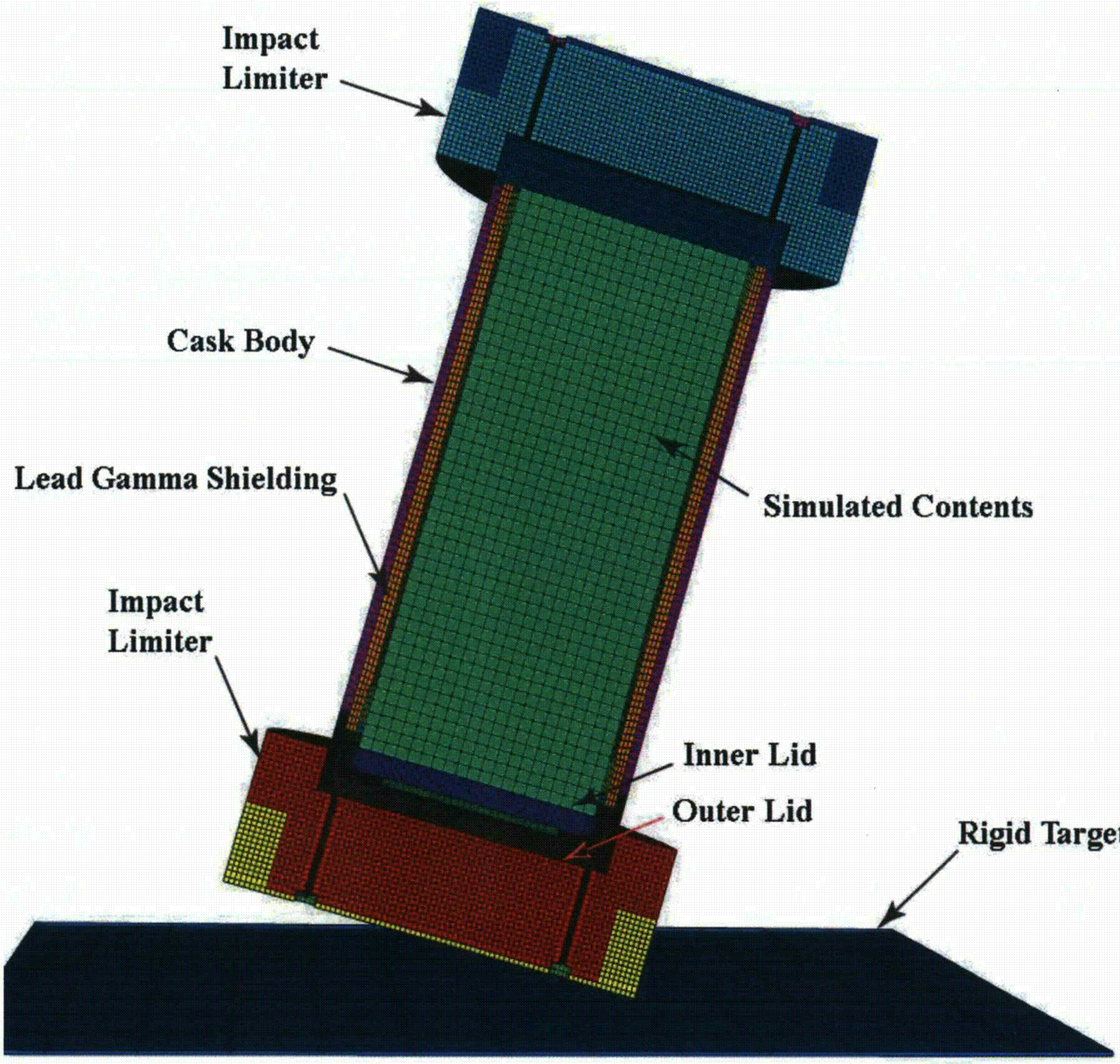
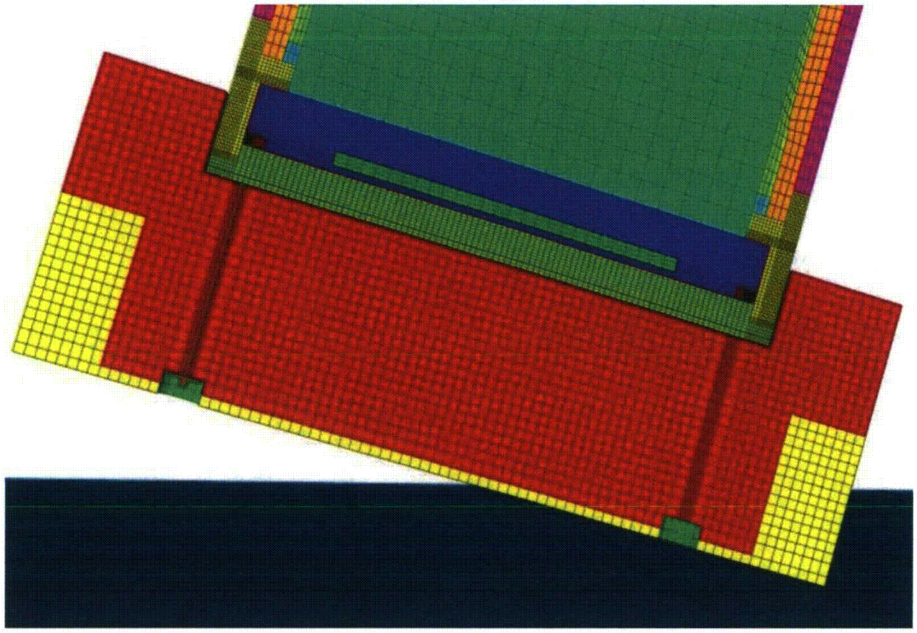
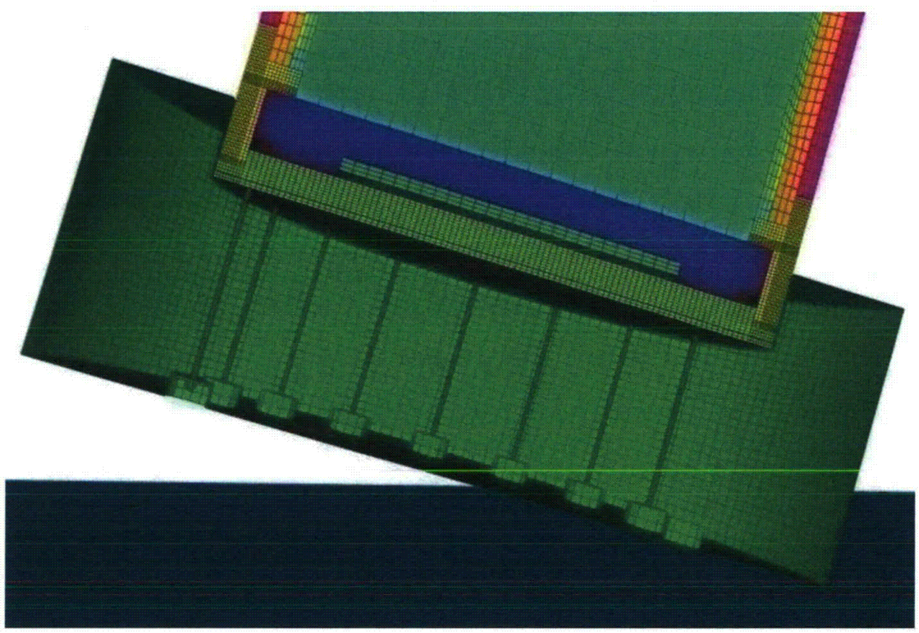


Figure 3-6. Finite element mesh of the Rail-Lead cask



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

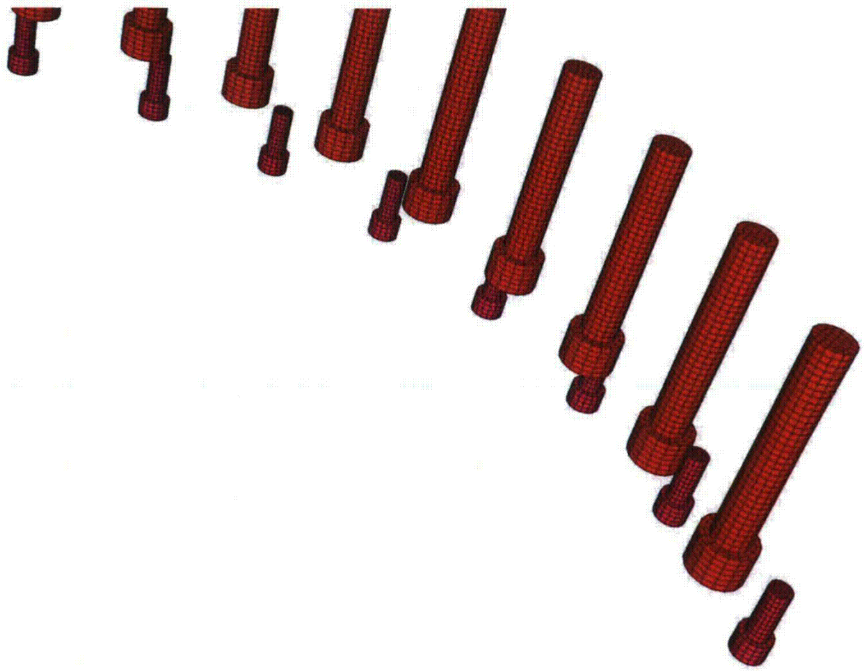


Figure 3-8. Finite element mesh of the Rail-Lead closure bolts for both the inner and outer lids. The longer bolts are for the inner lid and the shorter ones for the outer lid.

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

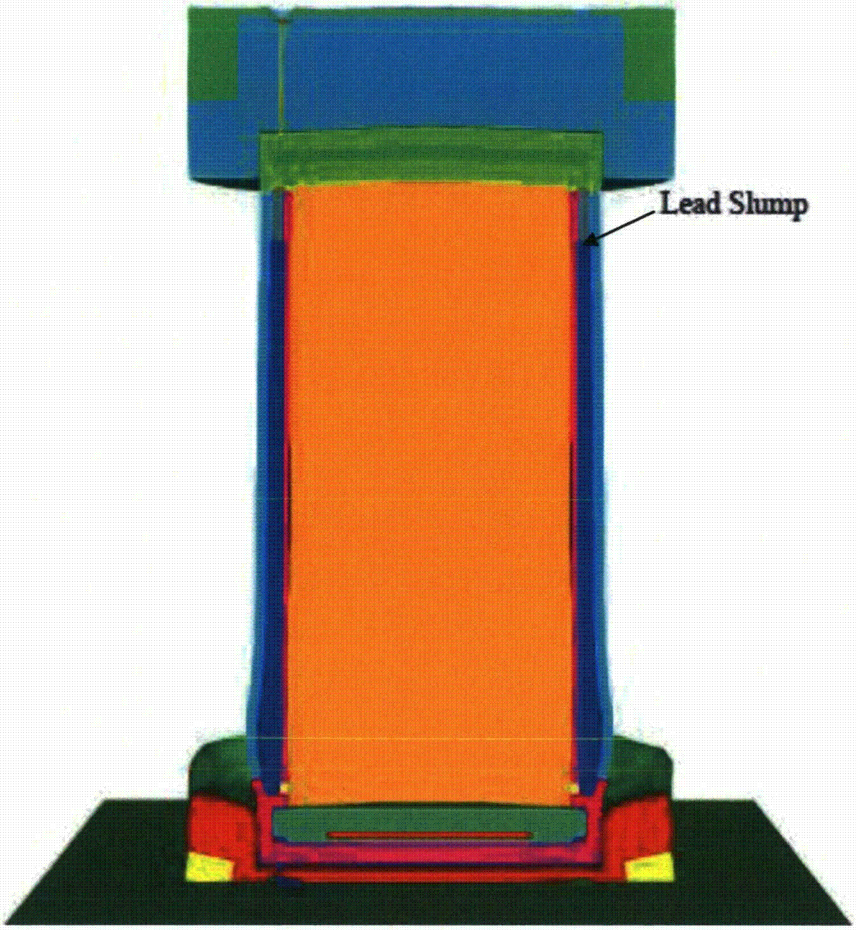


Figure 3-9. Deformed shape of the Rail-Lead cask following the 193 kph impact onto an unyielding target in the end-on orientation

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

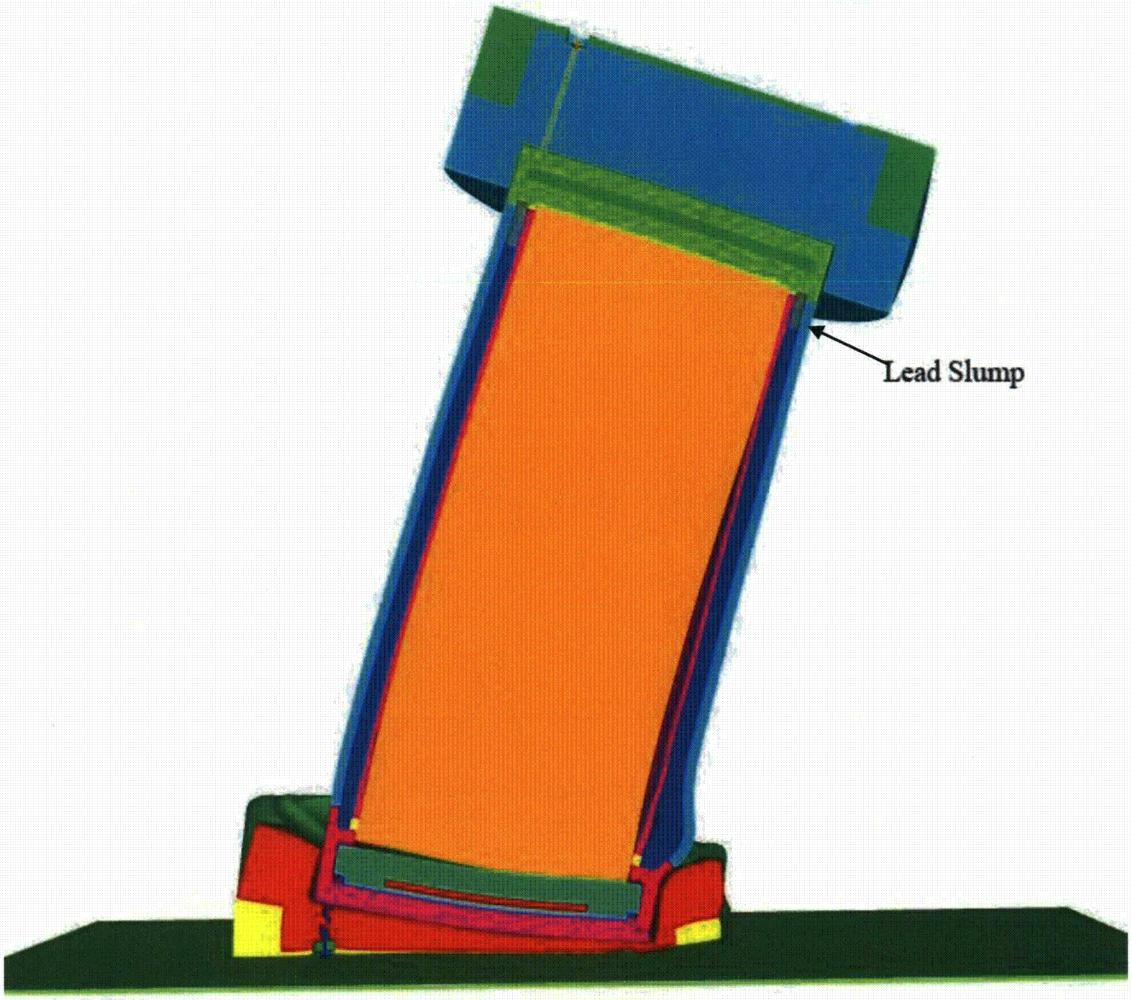


Figure 3-10. Deformed shape of the Rail-Lead following the 193 kph impact onto an unyielding target in the corner orientation

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

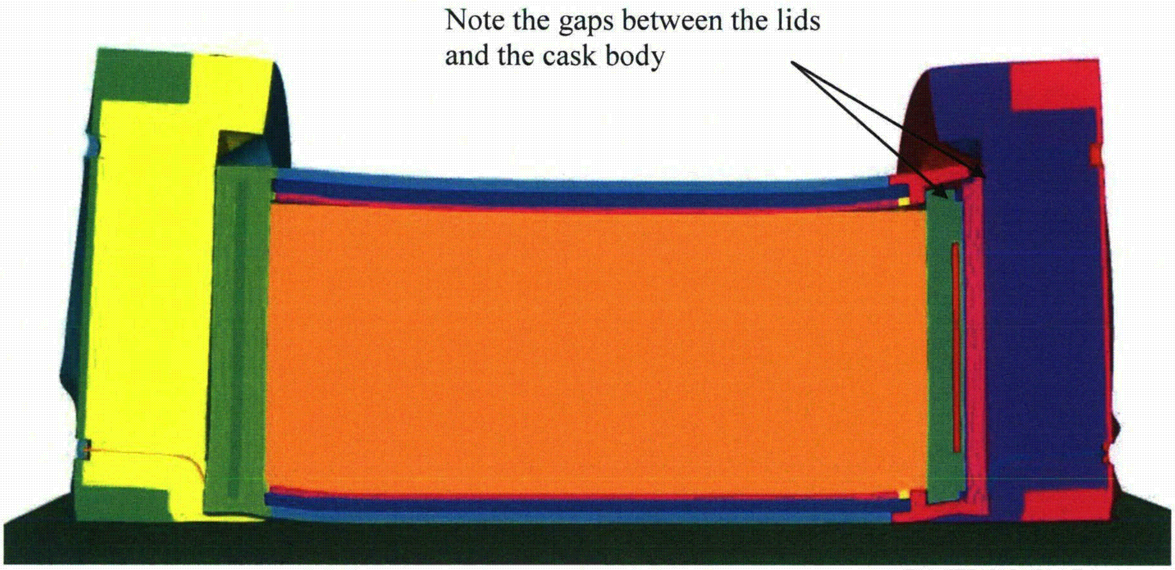


Figure 3-11. Deformed shape of the Rail-Lead following the 145 kph impact onto an unyielding target in the side orientation

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. None of the analyses show sufficient deformation into the interior volume of the cask to cause a failure of the internal welded canister. So for this cask, like the Rail-Steel cask, if the spent fuel is transported in an inner welded canister there would be no release from any of the impacts.

In the cases without an inner canister the cask can be used for dry spent fuel storage before shipment or to transport fuel that is removed from pool storage and immediately shipped. In the first of these two cases metallic o-rings provide the seal between each of the lids and the cask body. This type of seal is less tolerant to movement between the lids and the cask, and closure opening greater than 0.25 mm will cause a leak. If the cask is used for direct shipment of spent fuel, elastomeric o-rings provide the seal between each of the lids and the cask body. This type of seal can withstand closure openings of 2.5 mm without leaking (Sprung et al., 2000). Table 3-2 gives the calculated axial gap in each analysis and the corresponding leak area for both metallic and elastomeric seals.

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

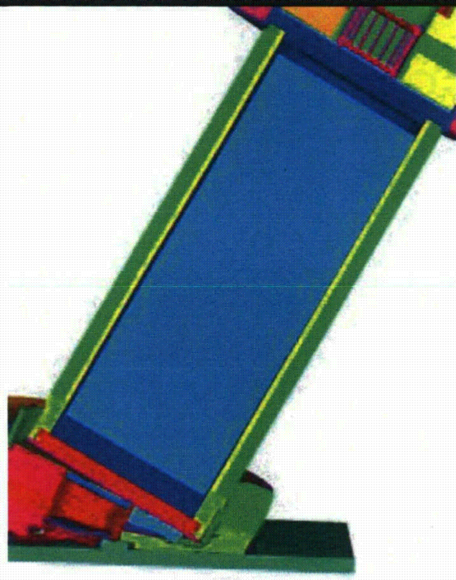
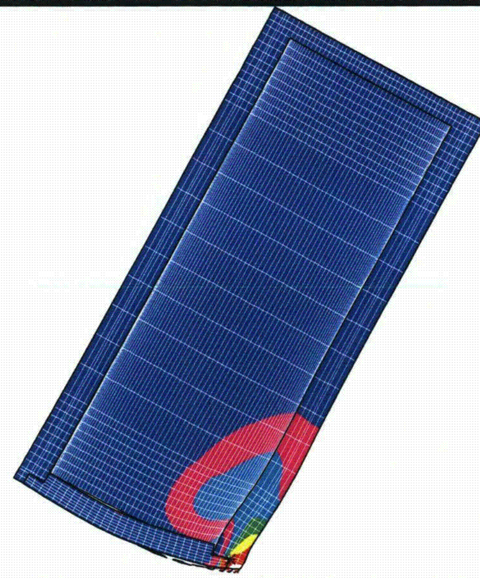
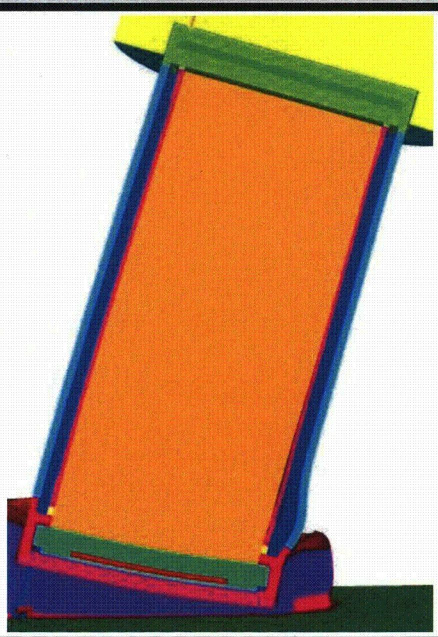
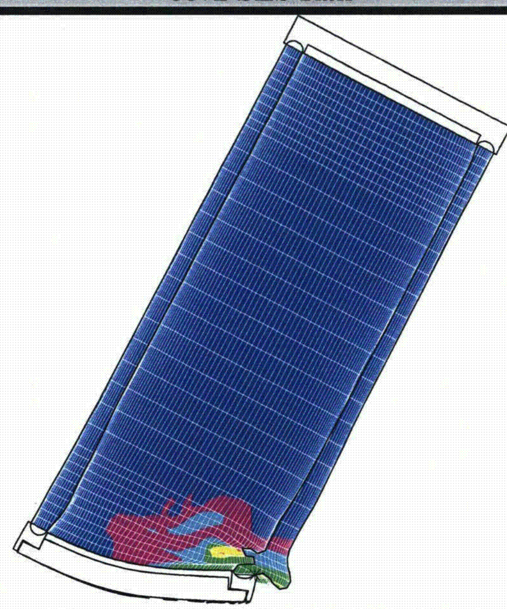
*Estimated; the method used to calculate the gaps for the other cases is explained in Appendix III.

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

3.2.3 Truck-DU Cask

Detailed finite element analyses of the Truck-DU cask were not performed for this study, because the response of the truck casks in NUREG/CR-6672 indicated no gaps between the lid and the cask body at any impact speed. Therefore, the results discussed here are based upon the finite element analysis of the generic steel-DU-steel truck cask performed for NUREG/CR-6672. In general, the results from the analyses performed for this study have shown that the analyses performed for NUREG/CR-6672 were conservative (see Table 3-3), so the results discussed below are likely to be an overestimate of the damage to the Truck-DU cask from severe impacts. Figure 3-12 shows the deformed shape and plastic strain contours for the generic steel-DU-truck cask from Appendix A of NUREG/CR-6672 (Figures A-15, A-19, and A-22). None of the impacts caused strains that are great enough to fail the cask wall, and in all cases the deformation in the closure region was insufficient to cause seal failure. Table 3-4 (extracted from Table 5.6 of NUREG/CR-6672) provides the deformation in the seal region for each case. For all of these cases there would be no release of radioactive contents.

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

Item/Cask	Rail-Steel	6672 Monolithic Steel
Deformed Shape 145 kph		 (Figure A-35 of NUREG/CR-6672)
Failed Bolts	No	Yes
Item/Cask	**Rail-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

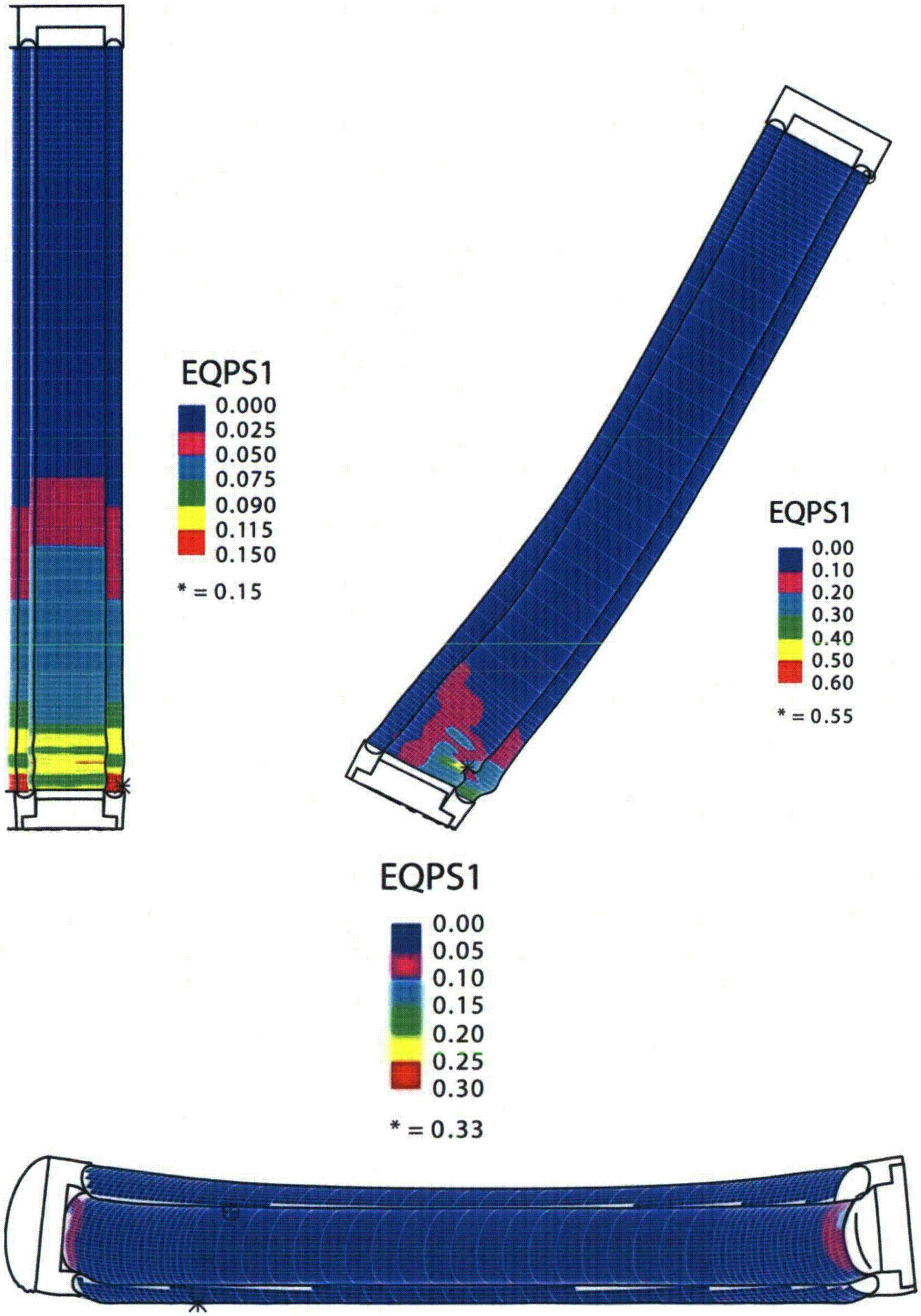


Figure 3-12. Deformed shapes and plastic strains in the generic steel-DU-steel truck cask from NUREG/CR-6672 (impact limiter removed)

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

Cask	Analysis Velocity	Corner Impact		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 MN (33 million pounds). Very few real targets are capable of applying this amount of force.

If the cask hits a flat target, such as the ground, roadway, or railway, it will penetrate into the surface. The greater the contact force between the cask and the ground, the greater the penetration depth. Figure 3-13 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 3-13. As an example, the end impact at 97 kph onto an unyielding target requires a contact force of 123.9 MN. A penetration depth of approximately 2.2 meters will cause the soil to exert this amount of force. The soil absorbs 142 MJ of energy in being penetrated this distance. Adding the energy absorbed by the soil to the 41 MJ of energy absorbed by the cask gives a total absorbed energy of 183 MJ. For the cask to have this amount of kinetic energy it would have to be traveling at 205 kph. Therefore, a 205 kph impact onto hard desert soil causes the same amount of damage as a 97 kph impact onto an unyielding target. A similar calculation can be performed for other impact speeds, orientations and target types. Table 3-6 provides the resulting equivalent velocities. Where the calculated velocity is more than 250 kph the value in the table is listed as greater than 250. No accident velocities are more than this. The concrete target used is a 23 cm thick slab on engineered fill. This is typical of many concrete roadways and concrete retaining walls adjacent to highways. Details on the calculation of equivalent velocities are included in Appendix III.

Table 3-5. Peak contact force for the Rail-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**

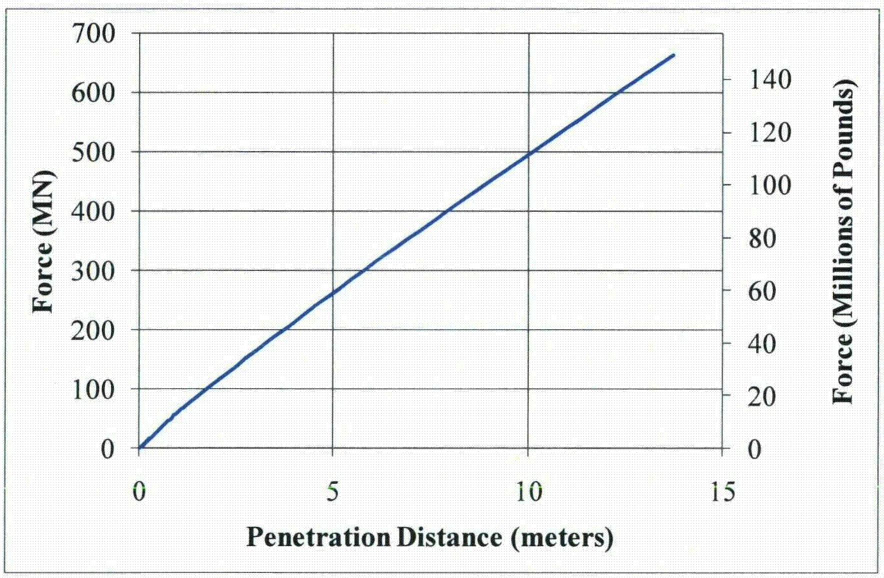


Figure 3-13. Force generated by the Rail-Lead cask penetrating hard desert soil

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

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

3.4 Effect of Impact Angle

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

Accident databases do not include impact angle as one of their parameters, so there is no information on the relative frequency of impacts at various angles. Given that vehicles usually travel parallel to the nearby surfaces, for this study a triangular distribution of impact angles was used. Figure 3-14 shows the assumed step-wise distribution of impact angle probabilities. For impacts onto hard targets, which are necessary to damage the cask, the component of the velocity that is parallel to the impact surface has very little effect on the amount of damage to the cask. This requires the accident speed to be higher for a shallow angle impact then a perpendicular one in order to achieve the same amount of damage. Figure 3-15 depicts an example of an impact at a shallow angle and the components of the velocity parallel and perpendicular to the surface. Table 3-7 provides the cumulative probability of exceeding an impact angle range and the accident speeds that are required to have the velocity component in the direction perpendicular to the target.

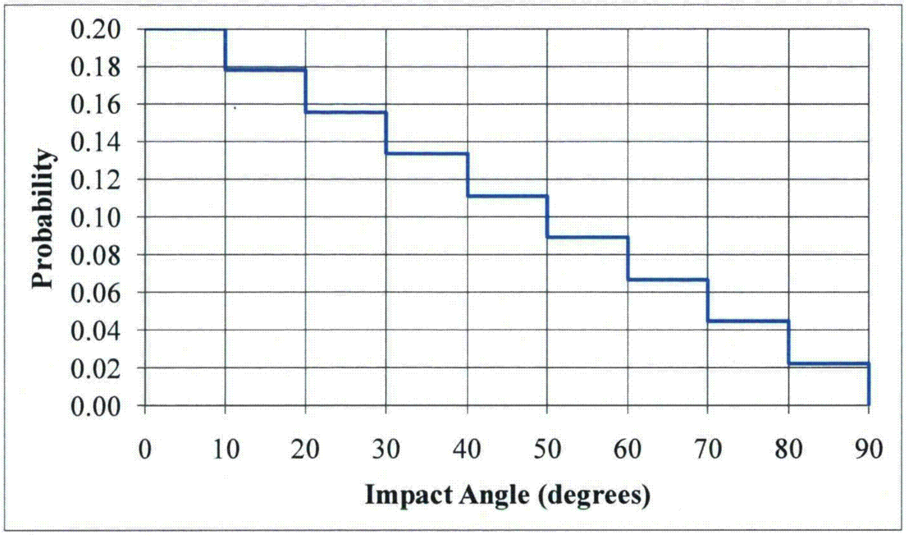


Figure 3-14. Probability distribution for impact angles

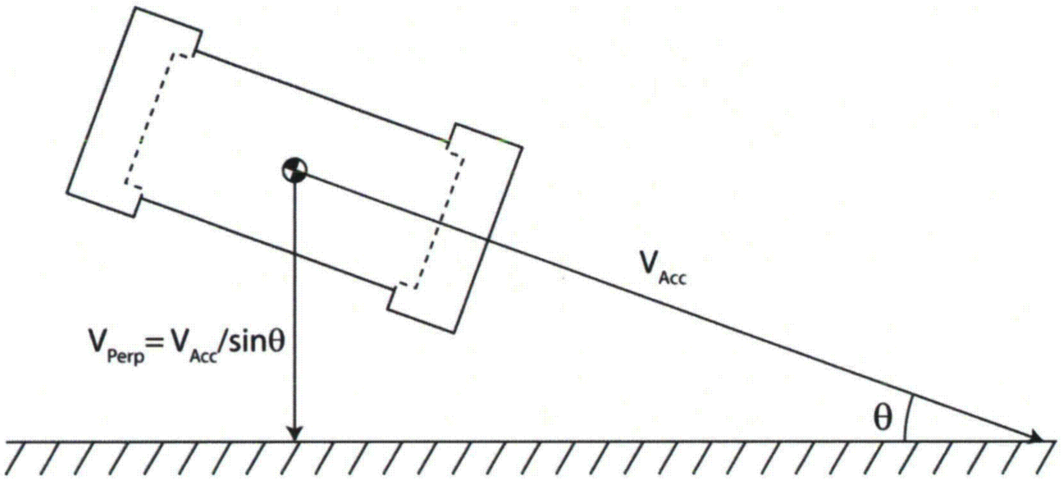


Figure 3-15. Influence of impact angle on effective velocity

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

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

3.5 Impacts with Objects

The discussions in the preceding sections all dealt with impacts onto flat surfaces. A large number of impacts deal with surfaces that are not flat. These include impacts into columns and other structures, impacts by other vehicles, and, more rarely, impacts by collapsing structures. These types of impacts were not explicitly included in this study, but recent work by Sandia National Laboratories (NRC, 2003b, Ammerman and Gwinn, 2004, Ammerman et al., 2005) has shown the response of the GA-4 cask to some of these impacts. The result of an impact into a large, semi-circular, rigid column is shown in Figure 3-16 (NRC, 2003b). While this impact led to significant permanent deformation of the cask, the level of strain was not high enough to cause tearing of the containment boundary and there was no permanent deformation in the closure region and no loss of containment.

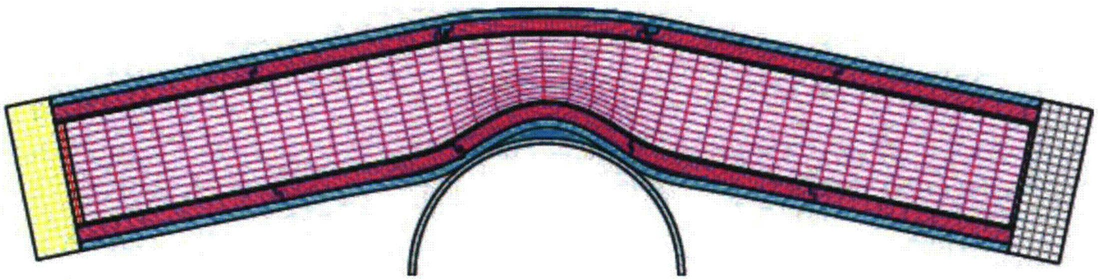


Figure 3-16. Deformations to the GA-4 truck cask after a 96 kph side impact onto a rigid semi-circular column, from (NRC, 2003b).

Another type of accident that could potentially damage a cask is the collision by a railroad locomotive. This is probably the most severe type of collision with another vehicle that is possible. Several different scenarios of this type of collision were investigated by Ammerman et al. (2005). The overall configuration of the general analysis case is shown in Figure 3-17. Variations on the general configuration included using the two most common types of locomotives, having a level crossing (such that the tires of the truck and the wheels of the locomotive are at the same elevation), having a raised crossing where the bottom of the main beams of the trailer at the same elevation as the top of the tracks, and having a skewed crossing so the impact is at 67° instead of at 90°. For all analyses the truck was assumed to be stopped. Train velocities of 113 kph and 129 kph were considered.

None of the analyses led to deformations that would cause a release of radioactive material from the cask and none of them resulted in cask accelerations that were high enough to fail the fuel rod cladding. Figure 3-18 shows a sequence of the impact. The front of the locomotive is severely damaged and the trailer is totally destroyed, but there is very little deformation of the cask—only minor denting where the collision posts of the locomotive hit.

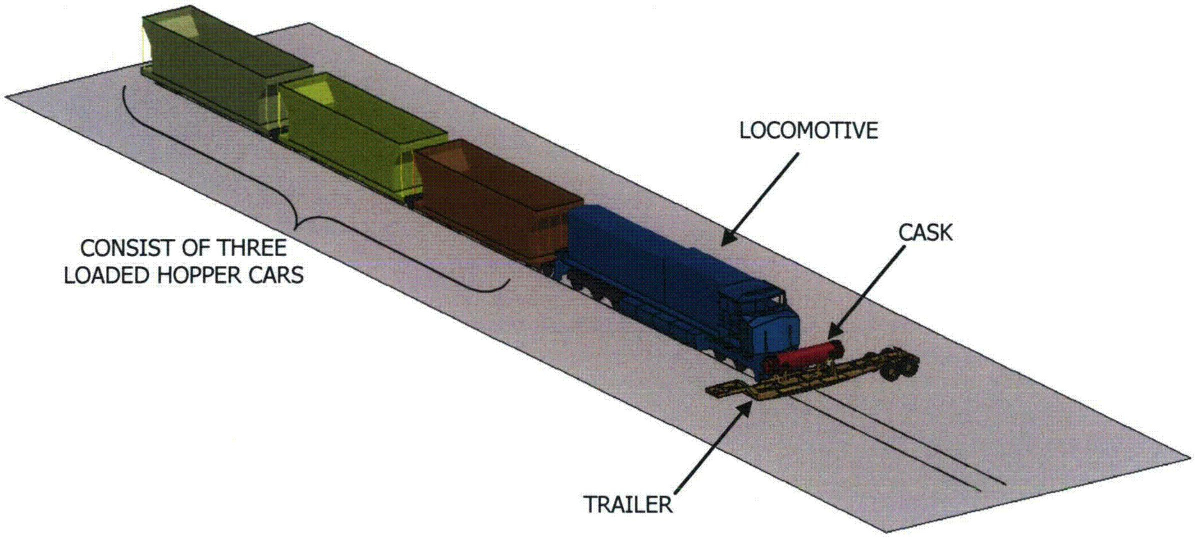


Figure 3-17. Configuration of locomotive impact analysis (from Ammerman et al., 2005)

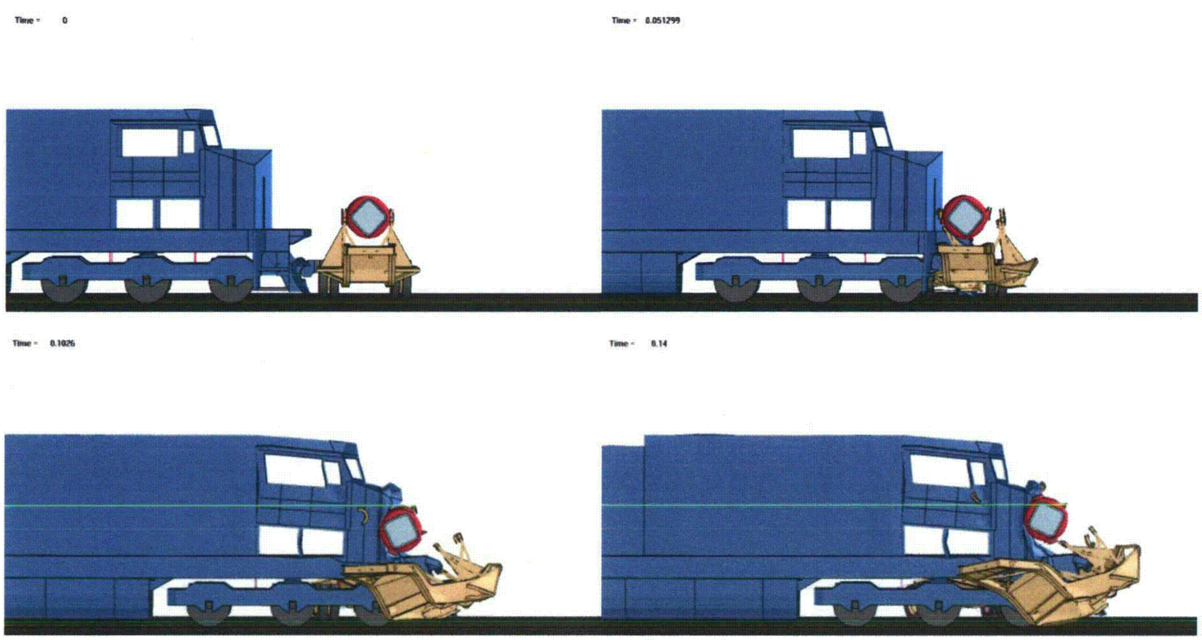


Figure 3-18. Sequential views of a 129 kph impact of a locomotive into a GA-4 truck cask (from Ammerman et al., 2005)

A type of accident that occurs less frequently, but also has the potential to damage a cask is the collapse of a bridge onto the cask. This type of accident occurred when an elevated portion of the Nimitz Freeway collapsed during the Loma Prieta earthquake. This accident scenario was analyzed to determine if it would cause a release of spent fuel from the GA-4 truck cask

(Ammerman and Gwinn, 2004). The analysis assumed the cask was lying directly on the roadway (neglecting the cushioning effect of the trailer and impact limiters) and one of the main beams of the elevated freeway fell and impacted the middle of the cask. The stresses in the cask and damage to the beam are shown in Figure 3-19. As in the other analyses for impacts with objects, there would be no loss of containment from this accident.

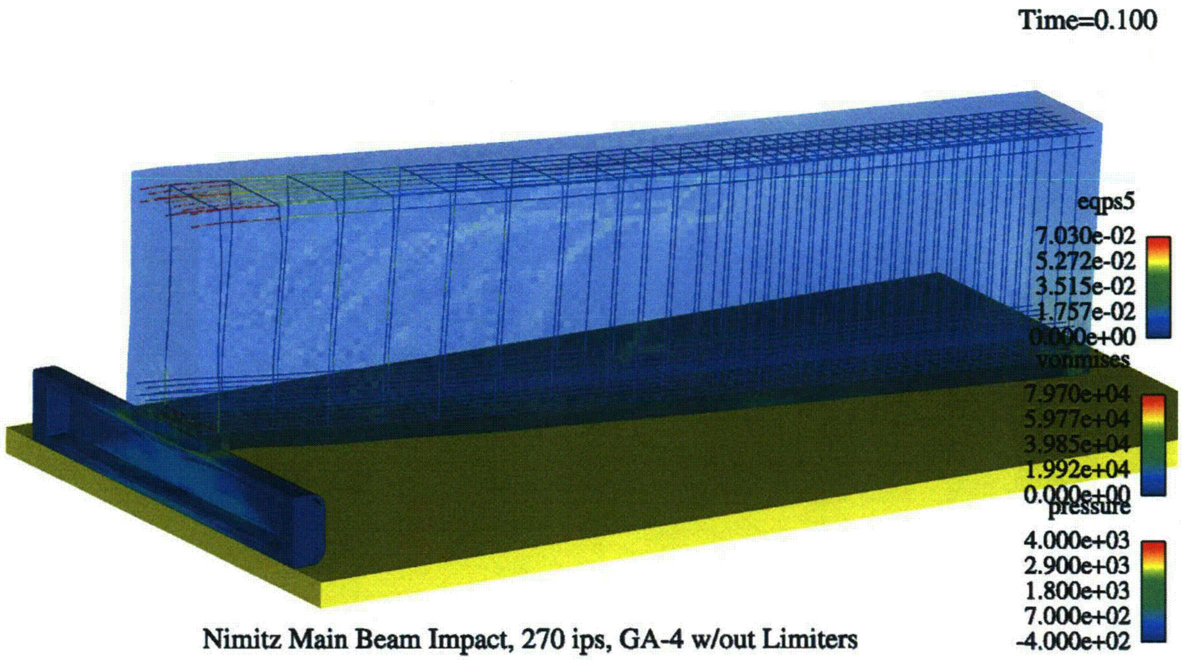


Figure 3-19. Results of a finite element simulation of an elevated freeway collapse onto a GA-4 spent fuel cask (from Ammerman and Gwinn 2004)

3.6 Response of Spent Fuel Assemblies

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 3-20 shows this model. The loads associated with a 100 G cask impact in a side orientation were then applied to this detailed model. Kalan et al. only analyzed side impacts 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 3-21. There is slight yielding of the rod at each support location and slightly more yielding where the rods impact each other.

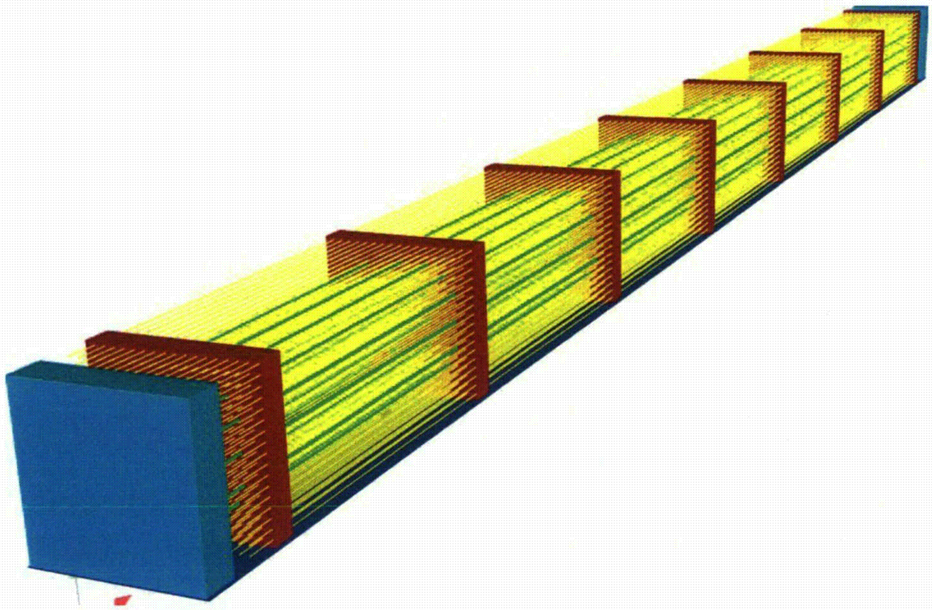


Figure 3-20. Finite element model of a PWR fuel assembly.

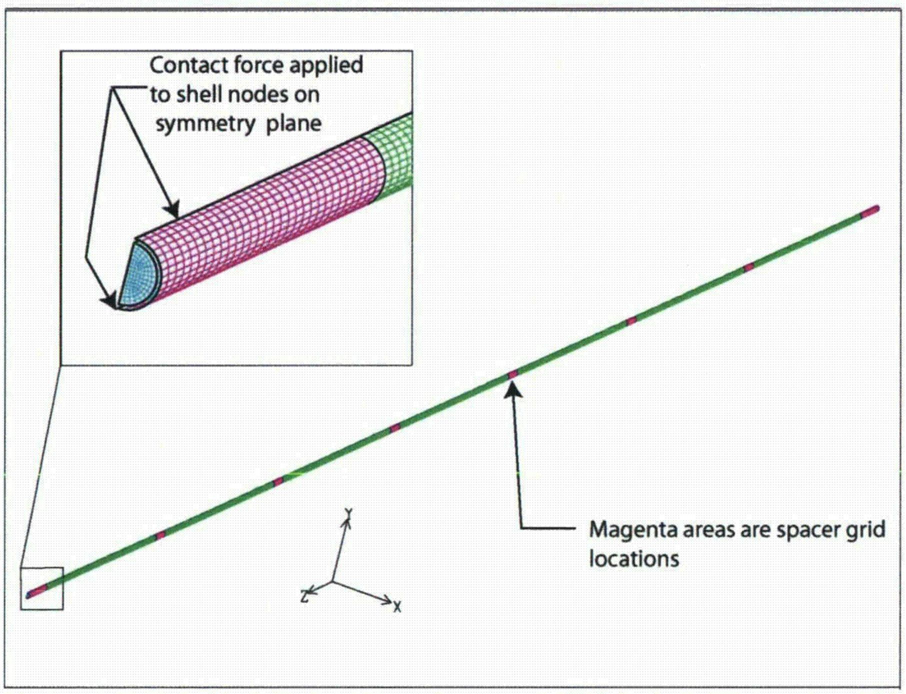


Figure 3-21. Detailed finite element model of a single fuel rod.

Figure 3-22 shows the maximum plastic strain at each location. The largest of these strains is slightly below 2%, which is half the plastic strain capacity of irradiated zircaloy at the maximum burn-up allowed in the Rail-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 200G. The only impacts that are severe enough to crack the rods are those with impact speeds onto an essentially unyielding target of 145 kph or higher. A detailed description of the fuel assembly modeling is included in Appendix III.

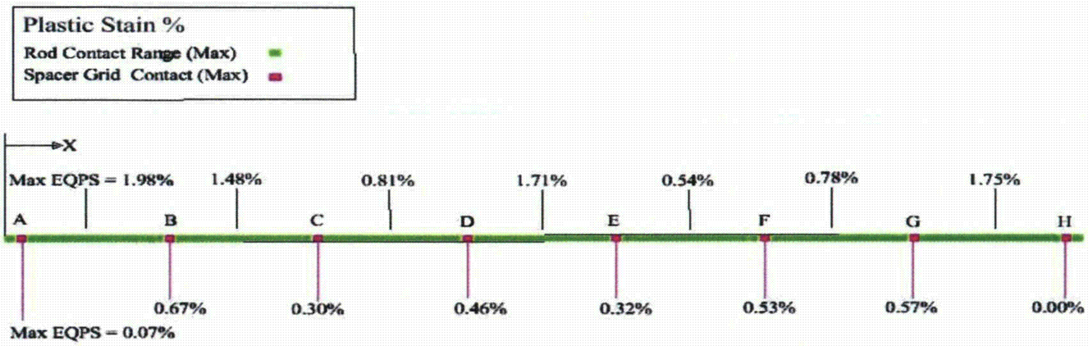


Figure 3-22. Maximum strains in the rod with the highest loads.

3.7 Conclusions

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

In the worst orientation (side impact) an impact speed onto a rigid target more than 97 kph is required to cause seal failure 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.

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. This is more severe than the majority of the thermal environments a cask may be exposed to in an accident that results in a fire. Certification analyses of the hypothetical 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. Large open pool fires can burn at temperatures higher than the 800°C specified in the regulations. Real fire plumes have location- and time-varying temperature distributions that vary from about 600°C to more than 1200°C (Koski, 2000; Lopez et al., 1998). Therefore, the evenly-applied 800°C fire environment used in 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 fashion are used to analyze the response of the Rail-Steel and the Rail-Lead casks to three different hypothetical fire configurations. These configurations are described in this chapter and the temperature responses of the casks are presented and discussed. A discussion of the thermal performance of the Truck-DU cask exposed to a severe hypothetical fire 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 spent nuclear fuel transportation industry. NUREG/CR-6672 also provides the rationale for the use of 750°C as the fuel rod burst rupture temperature. These temperature limits are used in this study to determine if the cask seals or fuel rods would be compromised, allowing release of radioactive material under any of the accident scenarios analyzed.

4.2 Description of Accident Scenarios

4.2.1 Pool size

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

4.2.2 Fire duration

The duration of the hypothetical fires for the rail cask analysis 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. To estimate the duration of the fires, this amount of fuel is assumed to form a pool with the dimensions of a regulatory pool fire for the rail casks that were analyzed. That is, fuel pools that extend horizontally three meters beyond the surfaces of the casks are used in the computer models. Provided that there are relatively small differences between the overall dimensions of the Rail-Steel cask and the Rail-Lead cask, these fuel pools are similar in size and are nominally 14m x 9m. A pool of this size would need to be 0.9m deep to pool 30,000 gallons of liquid fuel, a condition that is extremely unlikely to be met in any accident scenario. If the fuel in such pool were to ignite, this pool fire would burn for about 3 hours. This fire duration is estimated using a nominal hydrocarbon fuel recession (evaporation) rate of 5mm per minute, typical of large pool fires (SFPE, 2002; Lopez et al., 1998; Quintiere, 1998). Another way this large pool area could burn for up to three hours would be the even more unlikely case in which fuel flows at exactly the right rate to feed and maintain the pool area for the duration of the fire. Provided that both of these pooling conditions are very difficult to obtain, the fire duration presented here is considered to be conservative. Nevertheless, a 3-hour fire that is not moving over time and is capable of engulfing a rail cask over the duration of the fire is conservatively used for the analysis of the two rail casks considered in this study.

In the case of the Truck-DU cask, the fire duration is based on the fuel capacity of a typical petroleum tank truck. About 9,000 gallons (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 2.3m x 6m, a regulatory pool that extends horizontally 3 meters beyond the outer surface of the cask would be 8.3m x 12m. To pool 9,000 gallons of gasoline in a pool of this area, the pool would need to be 0.3m deep, a configuration that is difficult to obtain in an accident scenario and therefore unlikely to occur. Such a pool fire would burn for a little more than an hour. As discussed for the rail cask pool fire, the other possibility of maintaining a fire that can be engulfing and that can burn for that duration is if gasoline were to flow at the right rate to maintain the necessary fuel pool conditions. 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 hypothetical fire accident scenarios different 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.

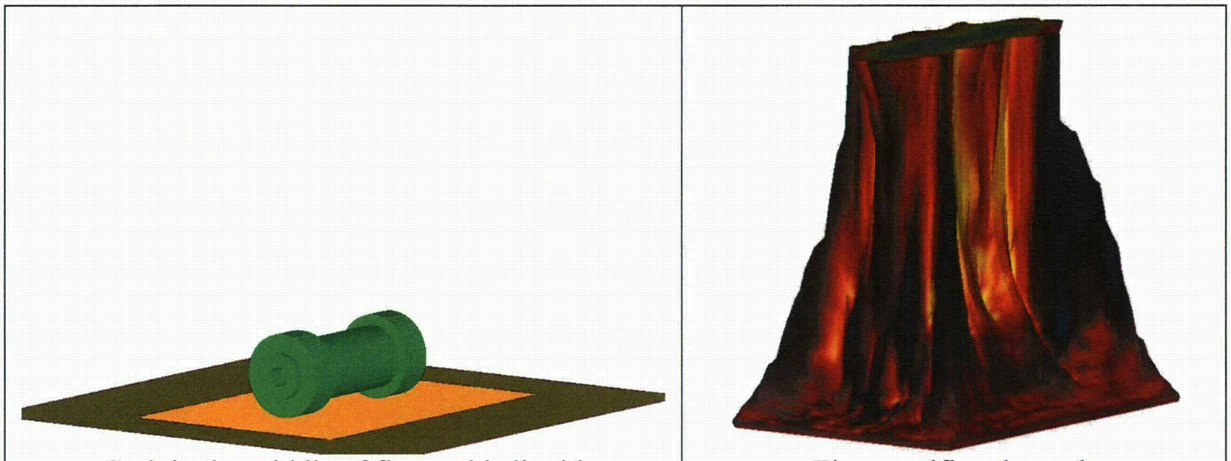
   This scenario represents the hypothetical case in which the liquid fuel spilled as a consequence of the accident flows to the location where the cask comes to rest following the accident and forms a large pool under (and concentric with) the cask.

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.

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.



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

Fire engulfing the cask

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

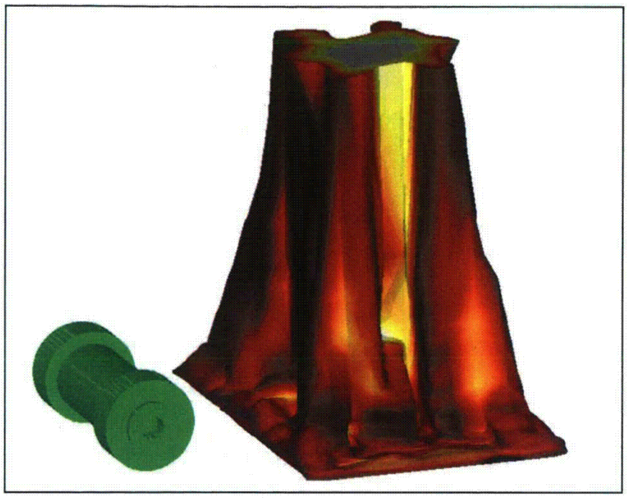


Figure 4-2. Cask lying on ground 3 meters from pool fire

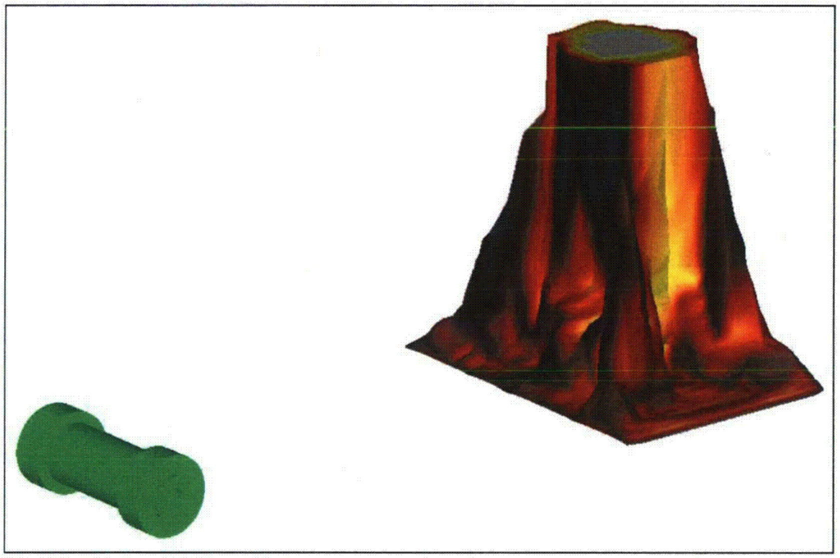
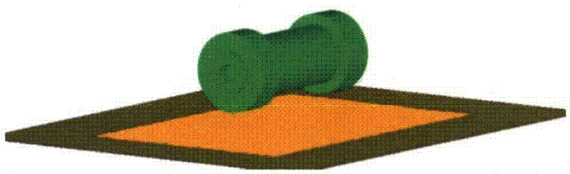


Figure 4-3. Cask lying on ground 18 meters from pool fire

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.

In addition to these hypothetical accident scenarios, two 30-minute regulatory HAC fire analyses are performed based on the conditions described in 10 CFR 71.73. In the first analysis a commercially-available finite element (FE) heat transfer code is used to apply an 800°C (1475°F) uniform-heating fire condition to the casks. In the second analysis, a benchmarked computational fluid dynamics (CFD) and radiation heat transfer computer model is used. In this model, the cask is positioned one meter above the fuel pool and the fire is realistically modeled as shown in Figure 4-4.



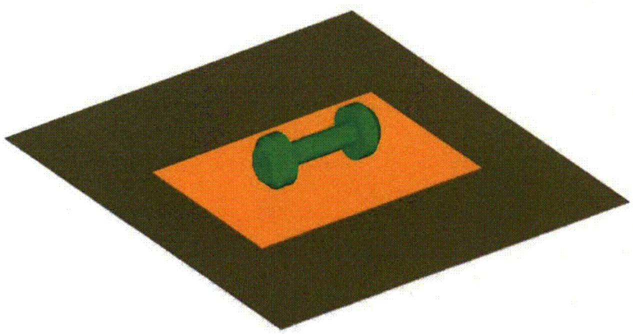
Cask elevated 1m above flammable liquid fuel pool region (shown in orange)

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 body. 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. However, the temperature scale for the Rail-Steel cask differs slightly from the scale for the Rail-Lead cask.

Results of the analyses are presented in the following order:

1. Regulatory 800°C (1475°F) uniform heating (30 minutes)
2. Regulatory CFD fire (30-minute fire)
3. Cask lying on the ground in the middle of a 3-hour pool fire
4. Cask lying on the ground 3 meters from a 3-hour pool fire
5. Cask lying on the ground 18 meters from a 3-hour pool fire

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.

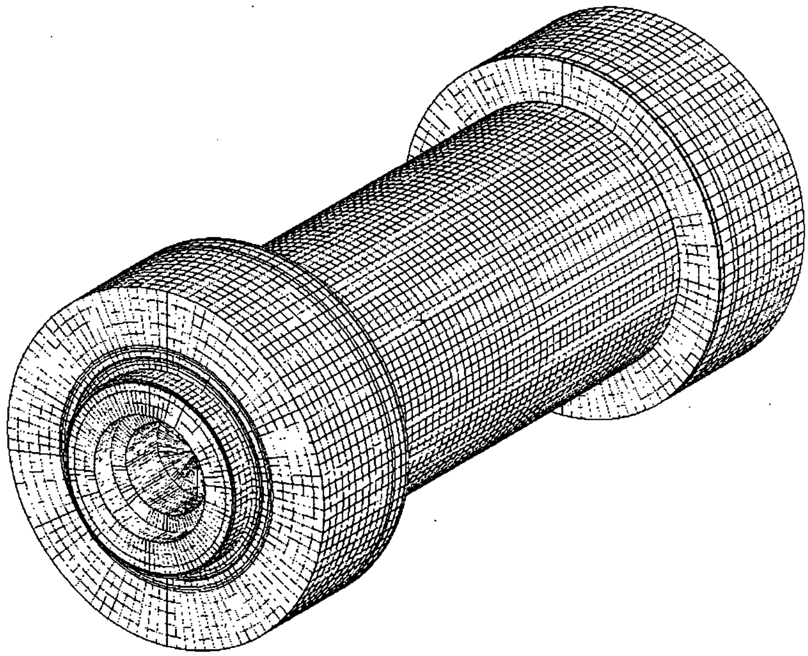
4.3.2 Simulations of the rail casks

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 is used to solve a wide variety of heat transfer problems. P-Thermal has been coupled with CAFE, allowing for a refined heat transfer calculation within complex objects, such as spent fuel casks, with realistic external fire boundary conditions.

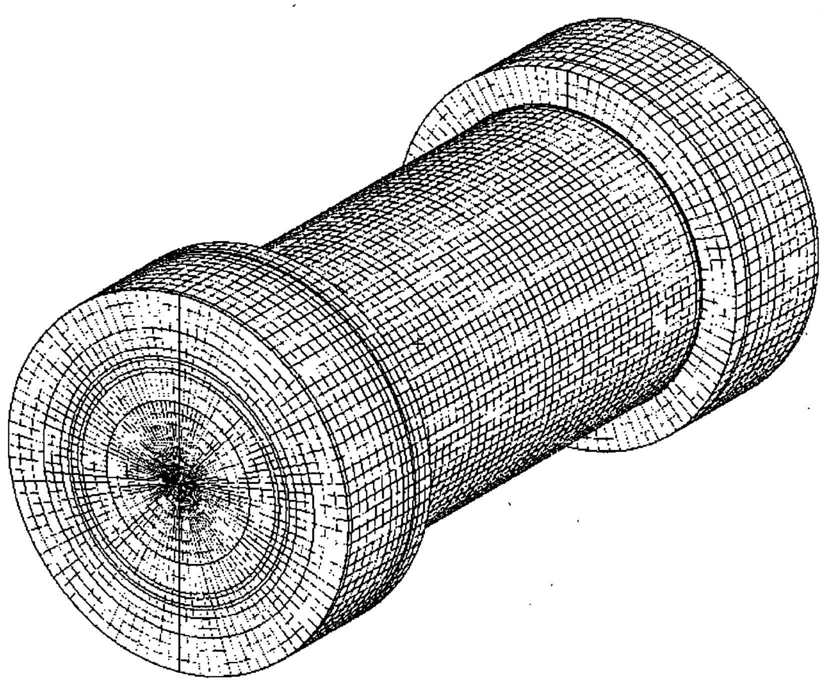
Both the Rail-Steel and the Rail-Lead casks have a polymeric neutron shield that is assumed to melt completely and be replaced by air at its operational temperature limit (see Appendix IV).

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.



Rail-Steel cask



Rail-Lead cask

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

4.3.3 Simulation of the spent fuel region

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

4.3.4 Rail-Steel cask results

The results for the Rail-Steel cask are presented in the order specified at the beginning of Section 4.3 in Figure 4-7 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. 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. This figure illustrates that the uniform heating thermal environment described in 10 CFR 71.73 heats up the seal region of the Rail-Steel cask more than a real fire may, even though a real fire can impart to the cask a localized thermal environment that is hotter than 800°C. A real fire applies a time- and space-varying thermal load to an object engulfed by it. In particular, large fires have an internal region where fuel in the form of gas exists but sufficient oxygen for that fuel to burn is not available. This region is typically called the "vapor dome." The lack of oxygen in the vapor dome is attributed to poor air entrainment in larger diameter 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. Note that these plots are snapshots of the distributions at an arbitrary time during the fire simulation. In reality, the fire moves slightly throughout the simulation causing these distributions to vary over time. Nevertheless, these plots show representative distributions for the cask and fire configuration shown.

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

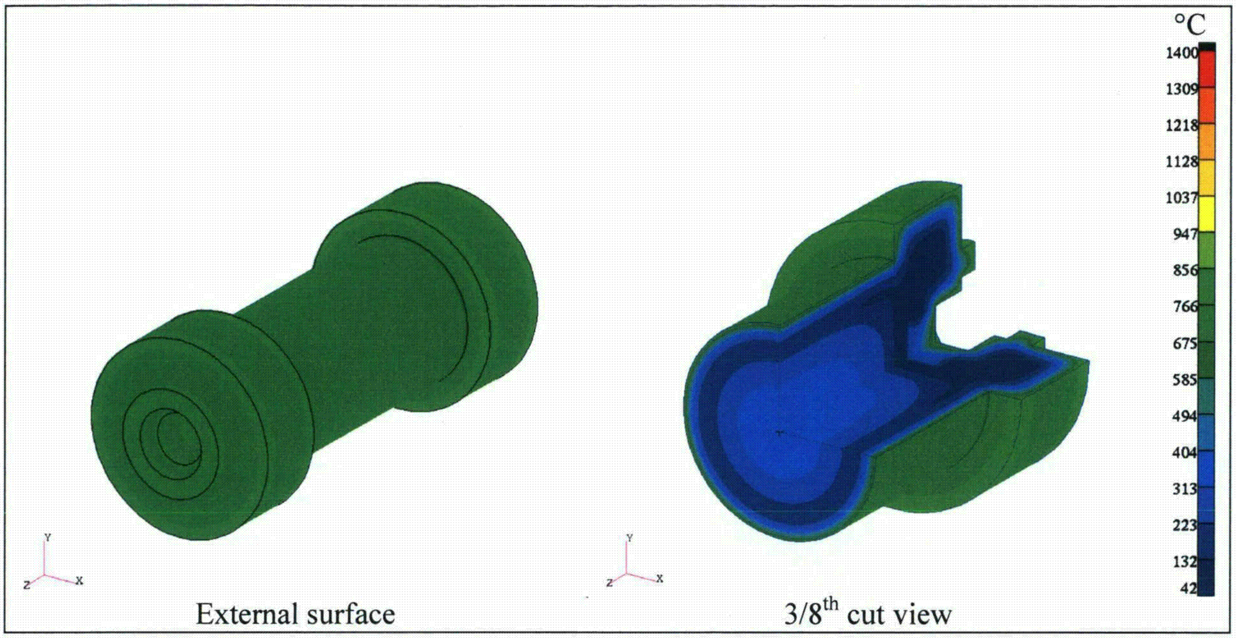


Figure 4-7. Temperature distribution of the Rail-Steel cask at the end of the 30-minute 800°C regulatory uniform heating

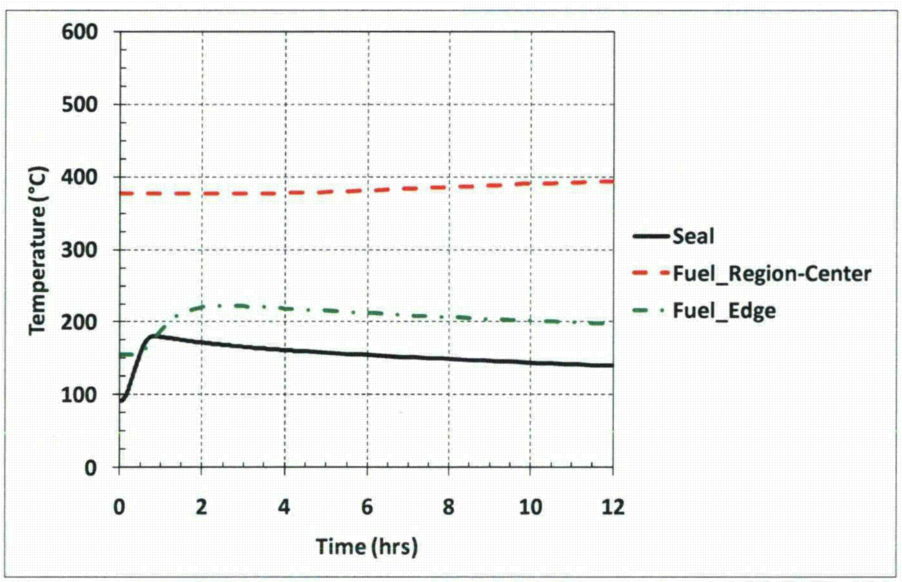


Figure 4-8. Temperature of key cask regions, Rail-Steel cask – Regulatory uniform heating

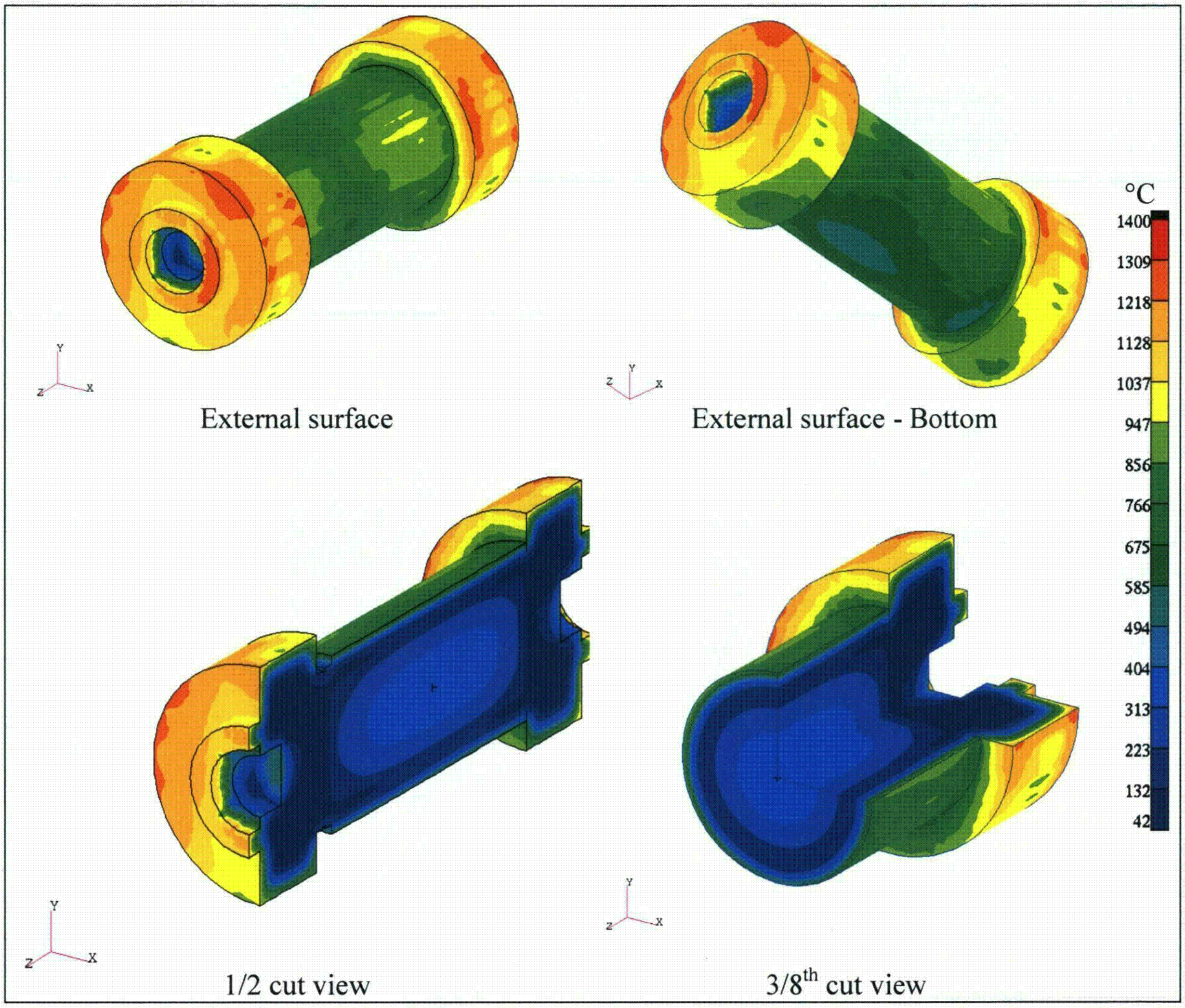


Figure 4-9. Temperature distribution of the Rail-Steel cask at the end of the 30-minute regulatory CAFE fire

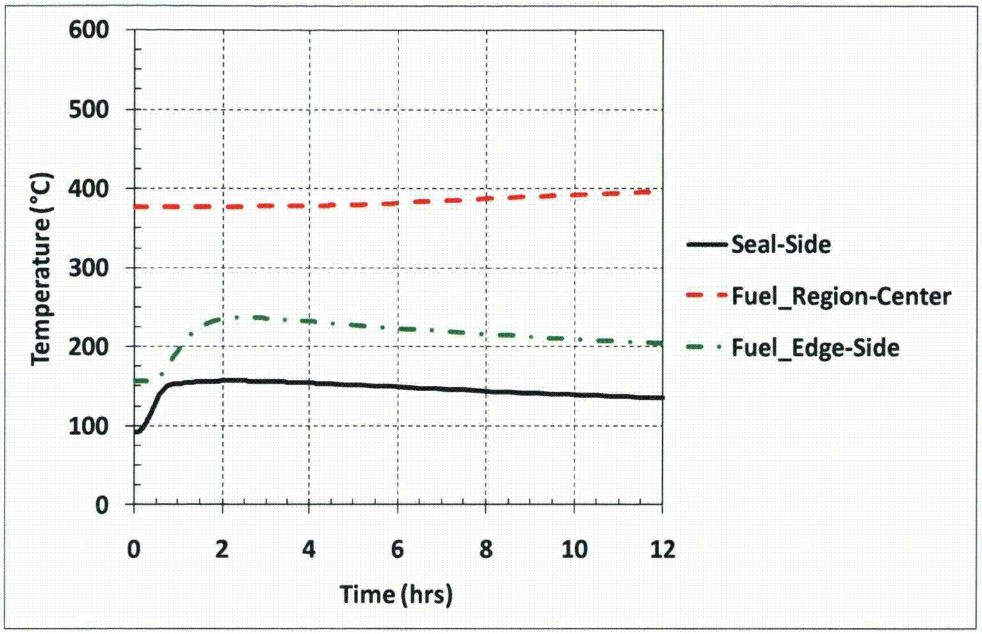


Figure 4-10. Temperature of key cask regions, Rail-Steel cask – Regulatory CAFE fire

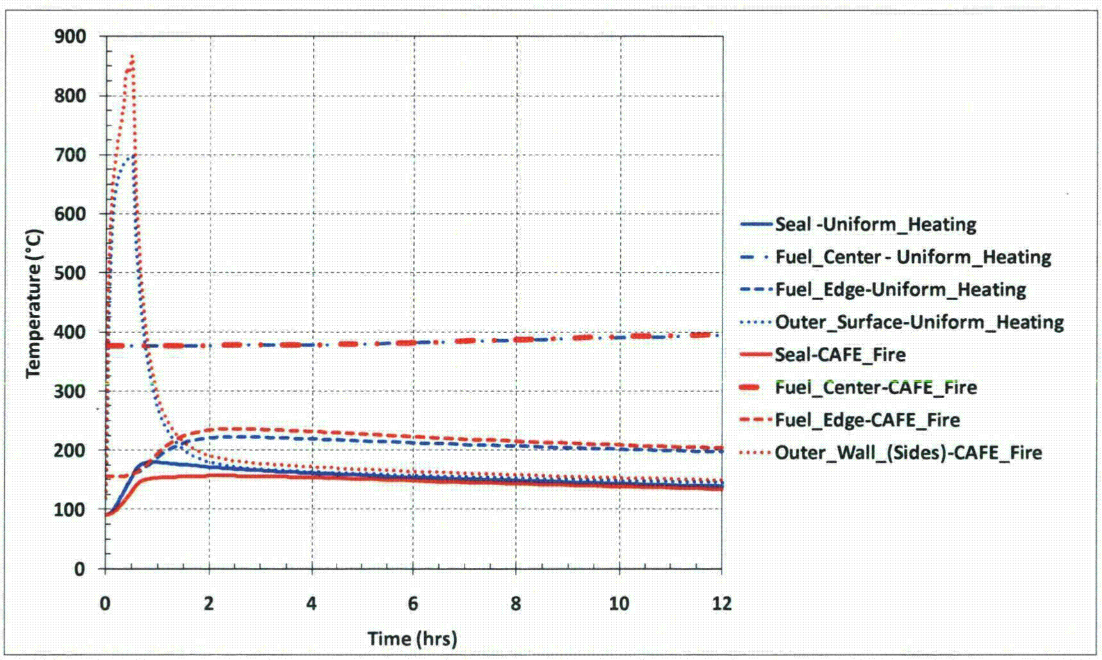
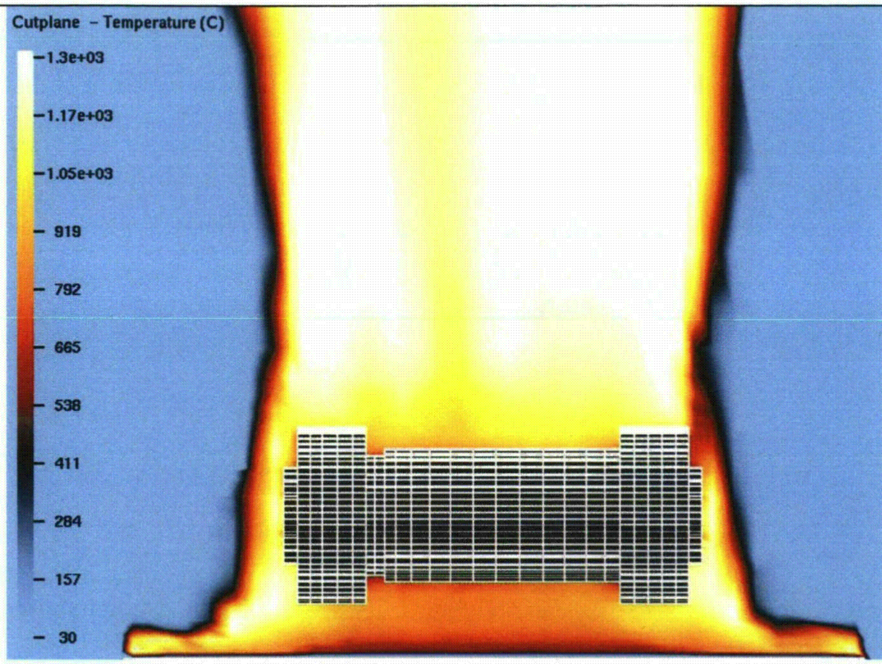
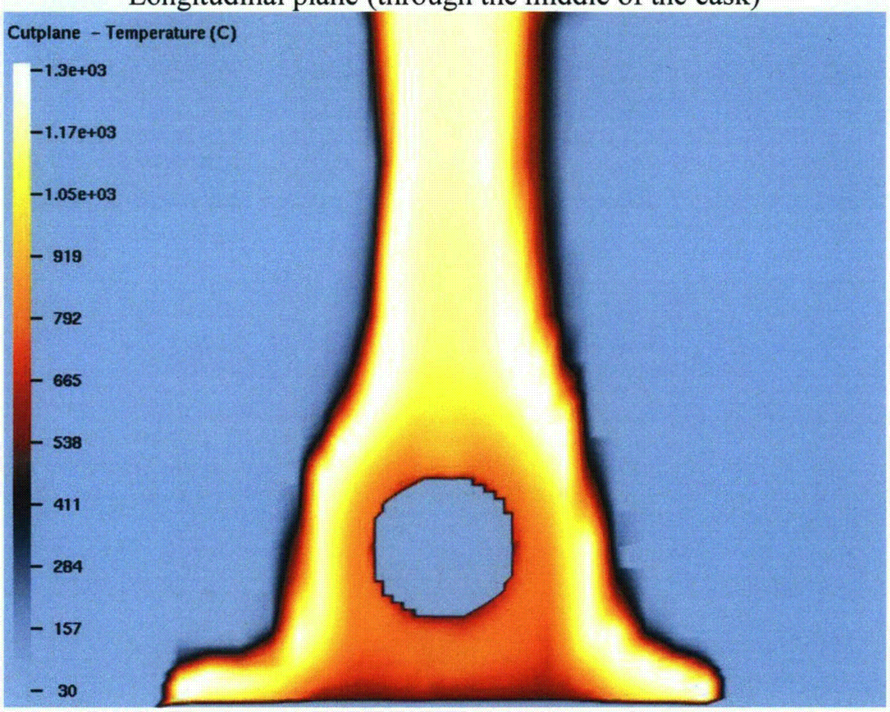


Figure 4-11. Comparison of regulatory fire analysis - Rail-Steel cask: Uniform heating vs. CAFE fire

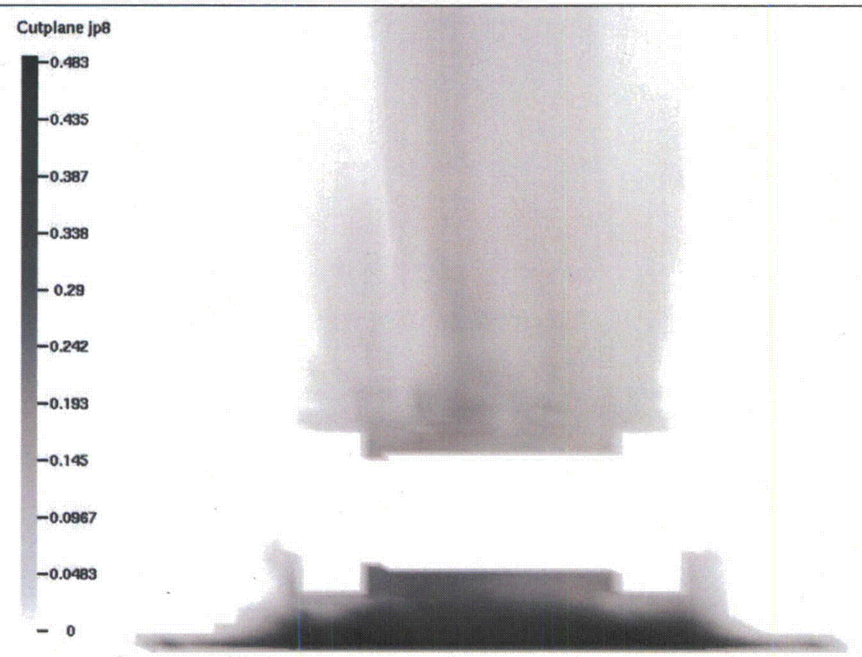


Longitudinal plane (through the middle of the cask)

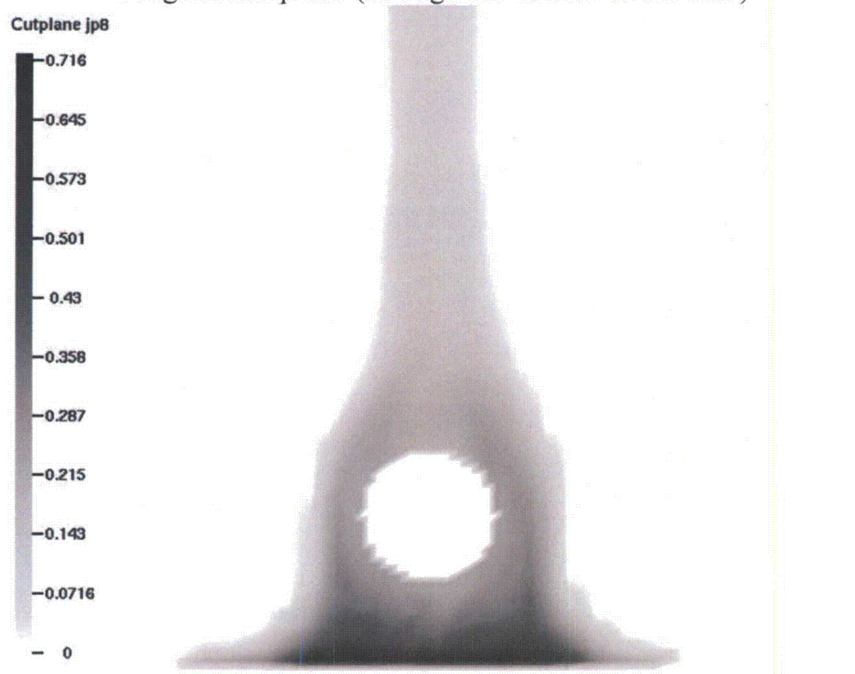


Transverse plane (at mid-length of the cask)

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



Longitudinal plane (through the middle of the cask)



Transverse plane (at mid-length of the cask)

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

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 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.

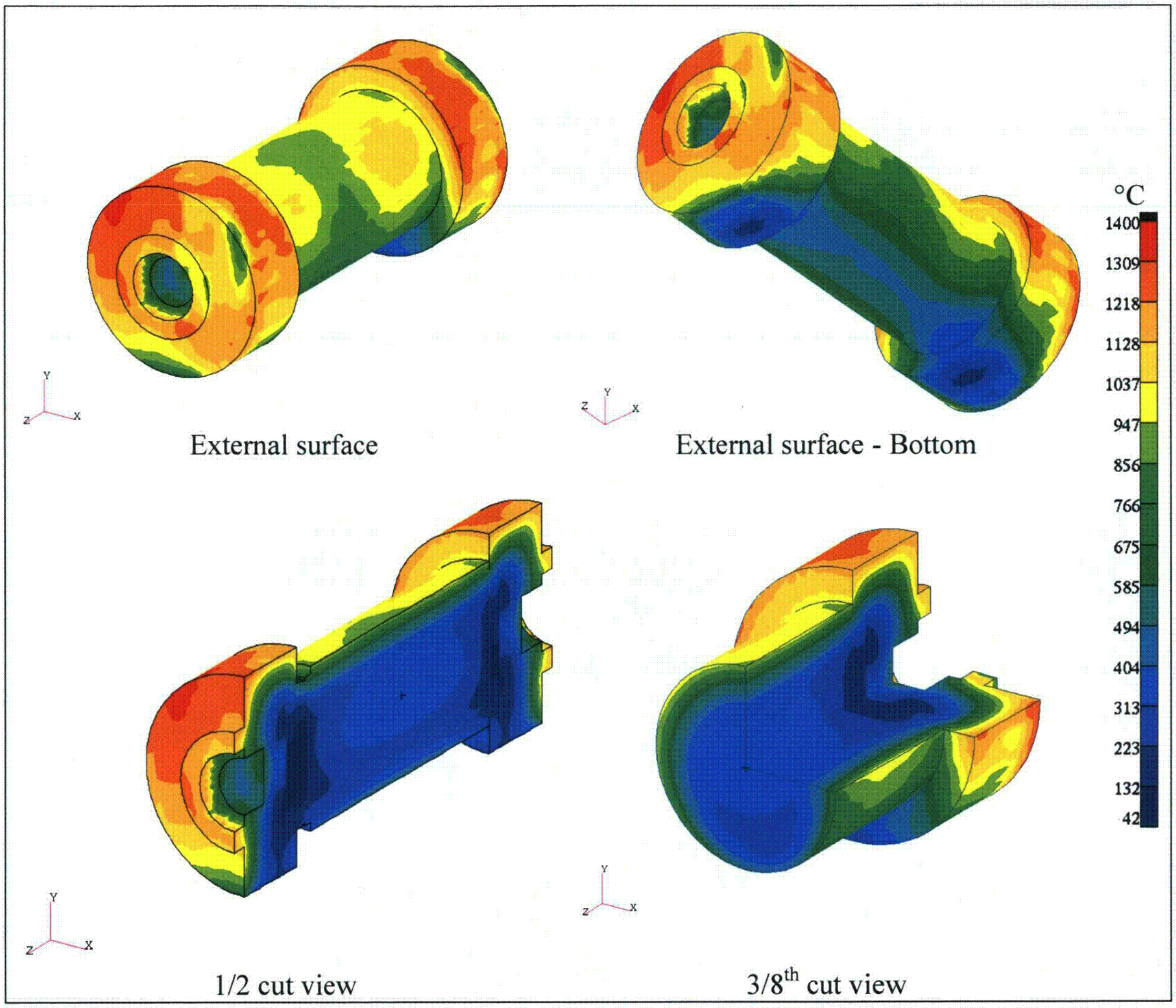


Figure 4-14. Temperature distribution of the Rail-Steel cask at the end of the 3-hour concentric CAFE fire - cask on ground

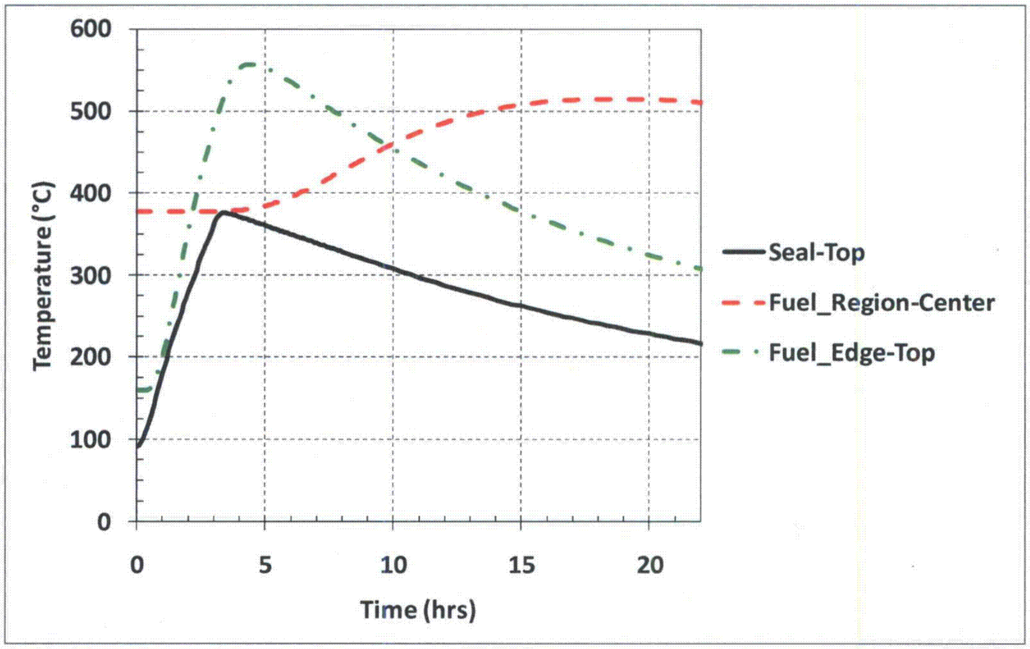
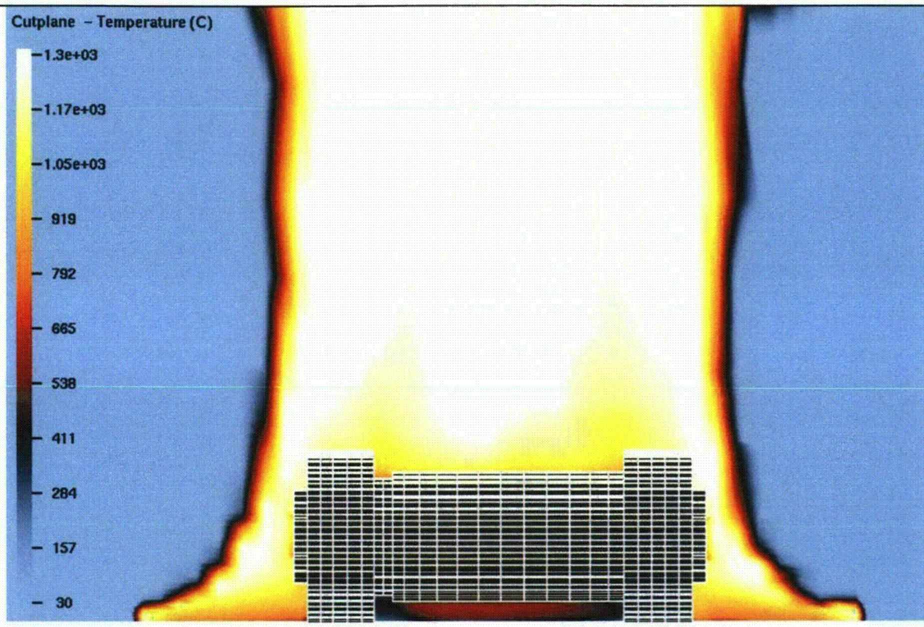
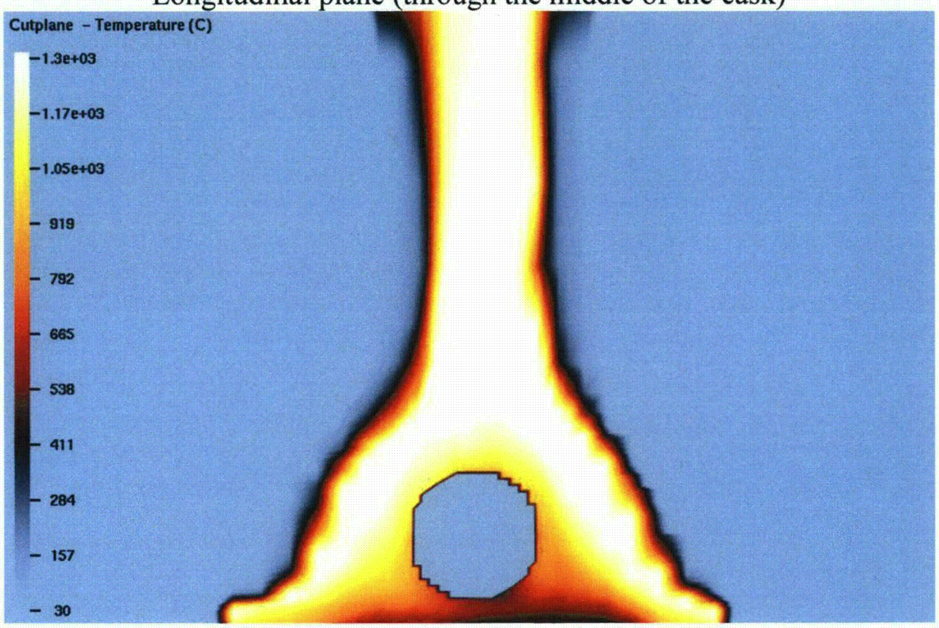


Figure 4-15. Temperature of key cask regions, Rail-Steel cask – Cask on ground, concentric fire

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.

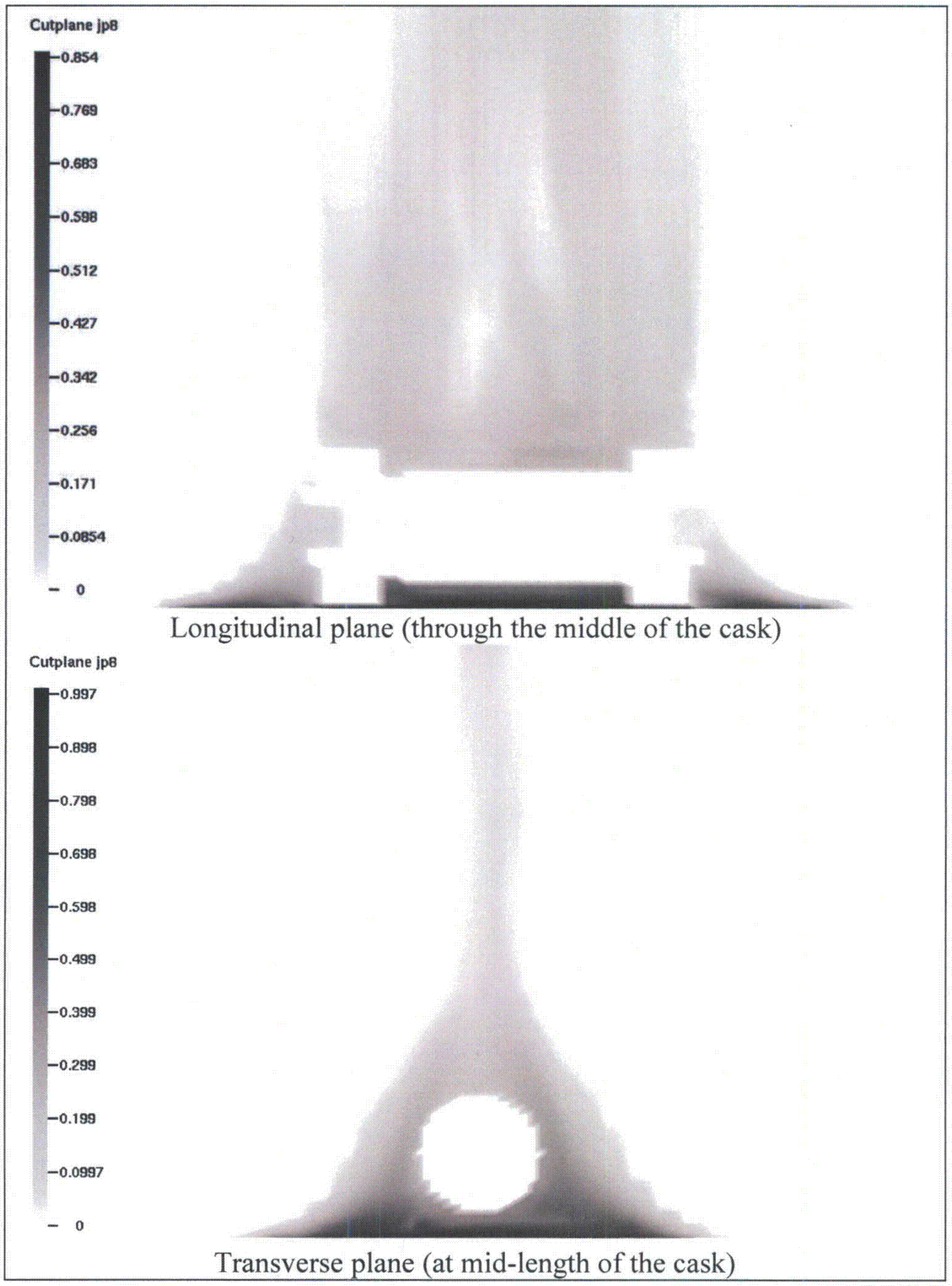


Longitudinal plane (through the middle of the cask)



Transverse plane (at mid-length of the cask)

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



Longitudinal plane (through the middle of the cask)

Transverse plane (at mid-length of the cask)

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 three-meter offset, the side of the cask facing the fire received heat by thermal radiation. The heat absorbed by the cask during the 3-hour exposure caused the temperature of the cask to rise as depicted in Figure 4-18 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 region did not reach their temperature limits. Therefore, offset fire scenarios will not cause this package to release radioactive material.

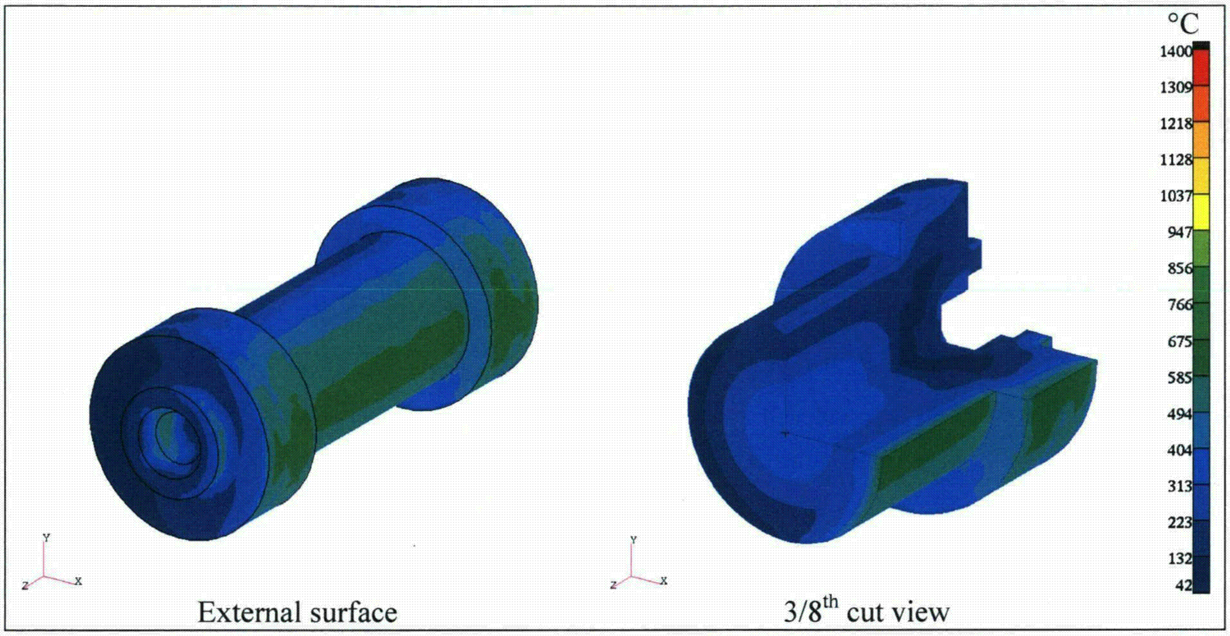


Figure 4-18. Temperature distribution of the Rail-Steel cask at the end of the 3-hour 3m offset CAFE fire - cask on ground

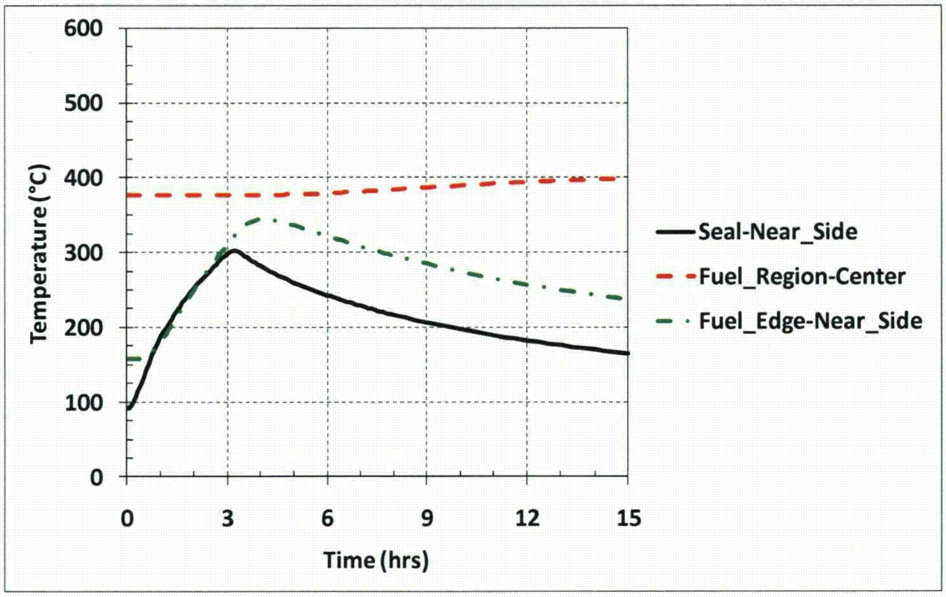


Figure 4-19. Temperature of key cask regions, Rail-Steel cask – Cask on ground, 3m offset fire

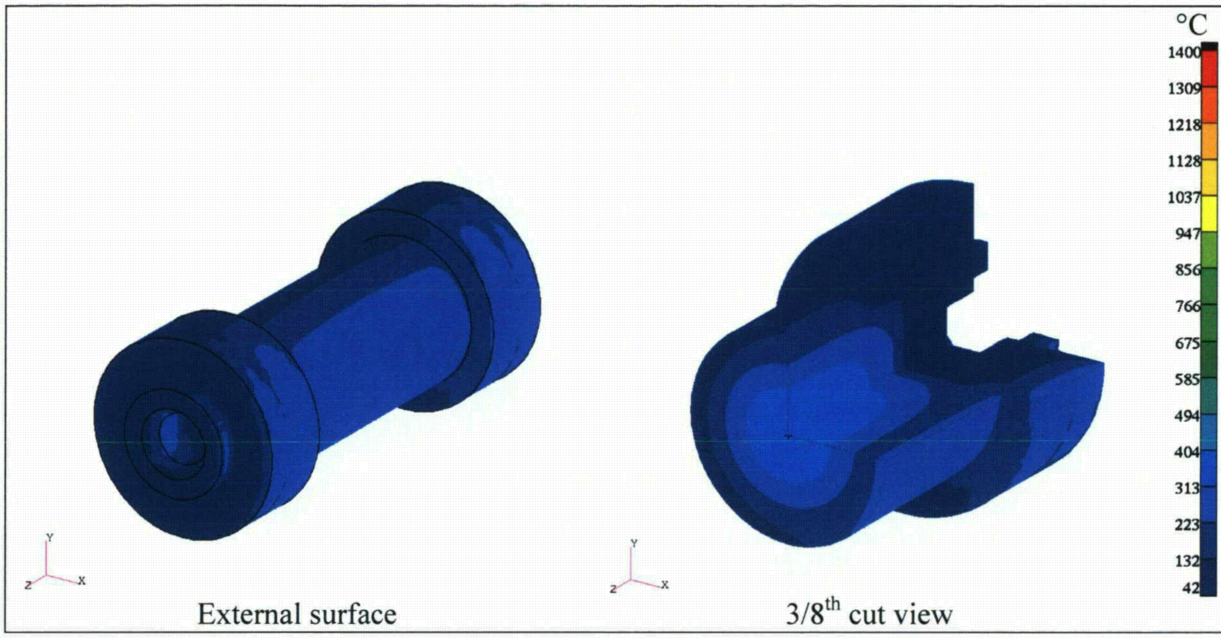


Figure 4-20. Temperature distribution of the Rail-Steel cask at the end of the 3-hour 18m offset CAFE fire - cask on ground

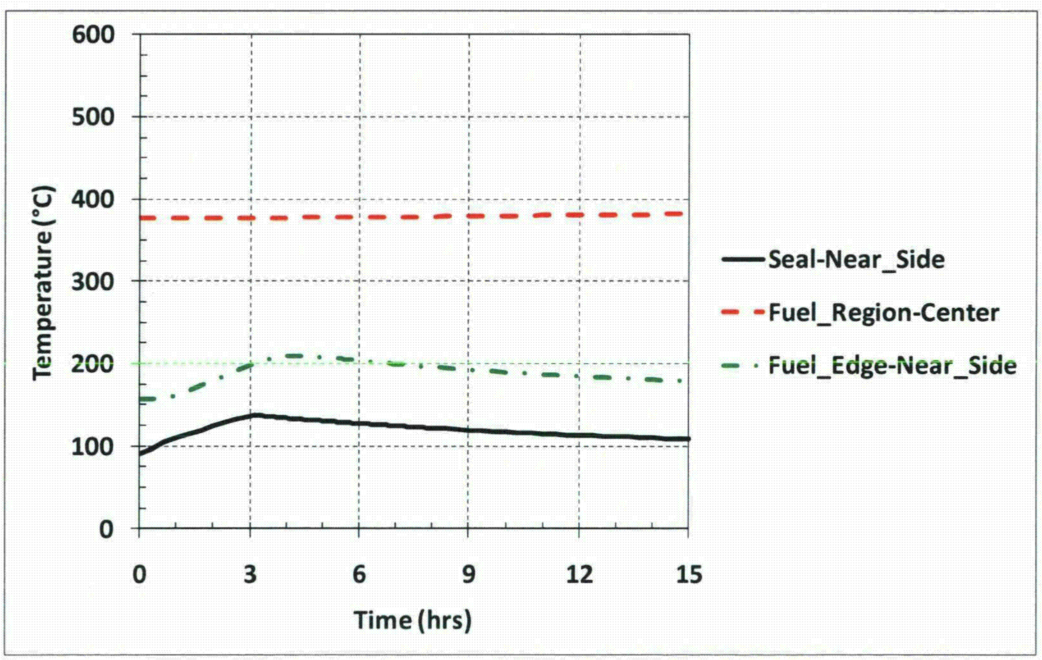


Figure 4-21. Temperature of key cask regions, Rail-Steel cask – Cask on ground, 18m offset fire

Summary of Rail-Steel cask analysis results

The results presented here show that the Rail-Steel cask is capable of protecting the fuel rods from burst rupture and is also capable of maintaining containment when exposed to the severe fire environments that are analyzed as part of this study. That is, the fuel region stayed below 750°C (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.

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.

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 and during that process, it absorbs (stores) heat while maintaining its temperature relatively constant at 328°C. As a result, the heat-up rate of portions of the cask slows down while the lead melts. That is why the curve of the region inward from the gamma shield region (i.e., the edge of the fuel region) in Figure 4-28 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.

Another effect considered in the cases where lead melted is the gradual thermal expansion and contraction of the gamma shield region during the heating and cooling of the cask. This effect is discussed in the next subsection.

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

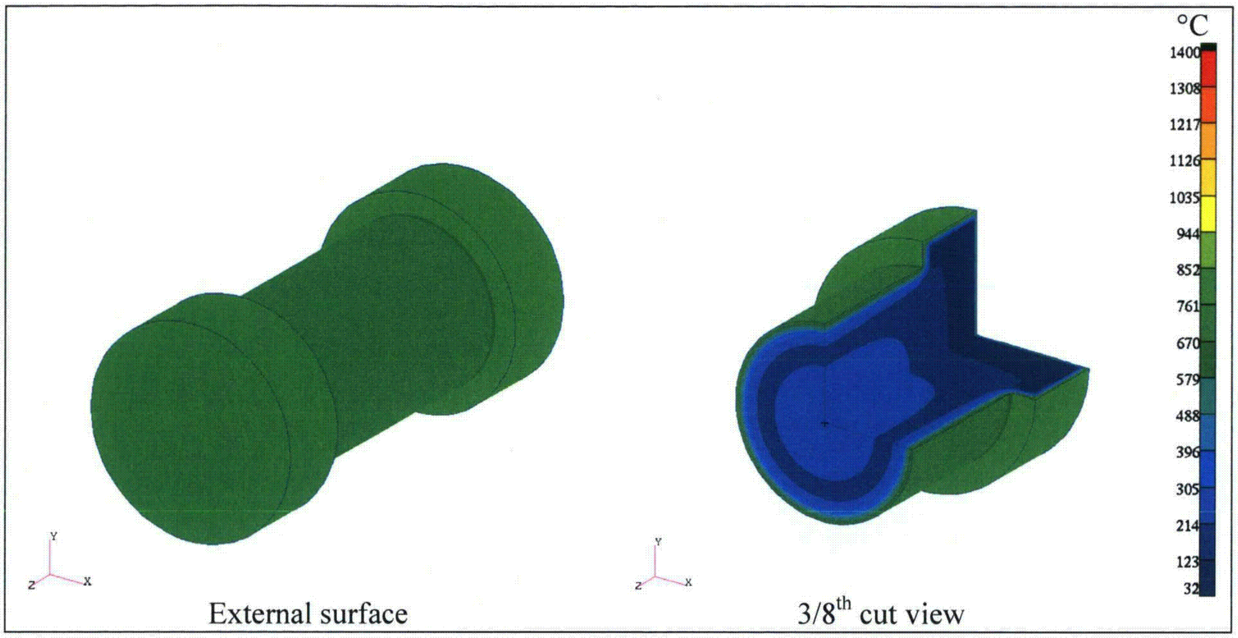


Figure 4-22. Temperature distribution of the Rail-Lead cask at the end of the 30-minute 800°C regulatory uniform heating

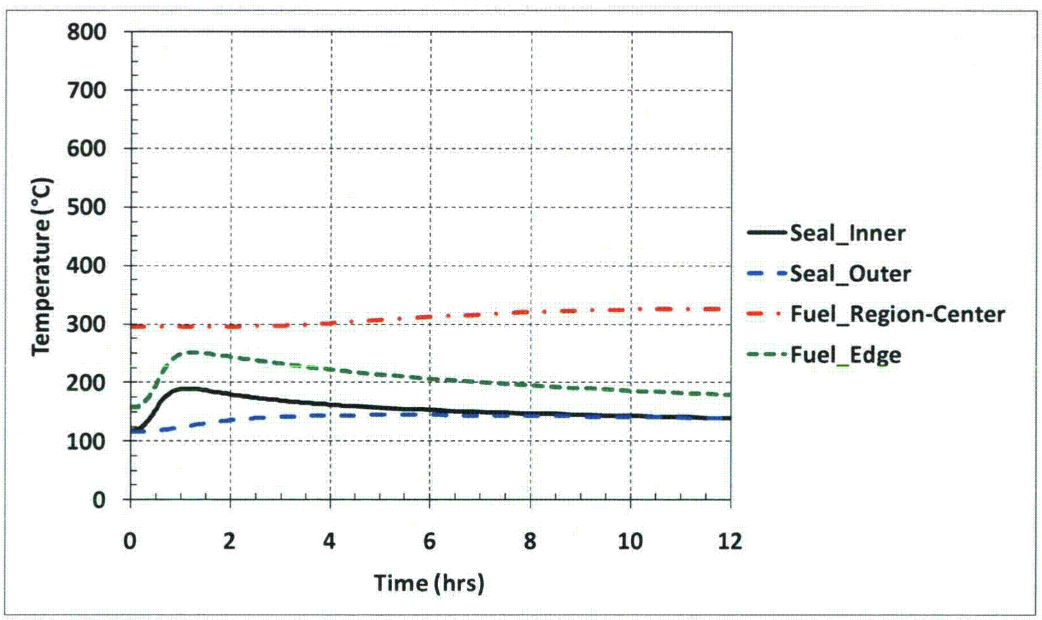


Figure 4-23. Temperature of key cask regions, Rail-Lead cask – Regulatory uniform heating

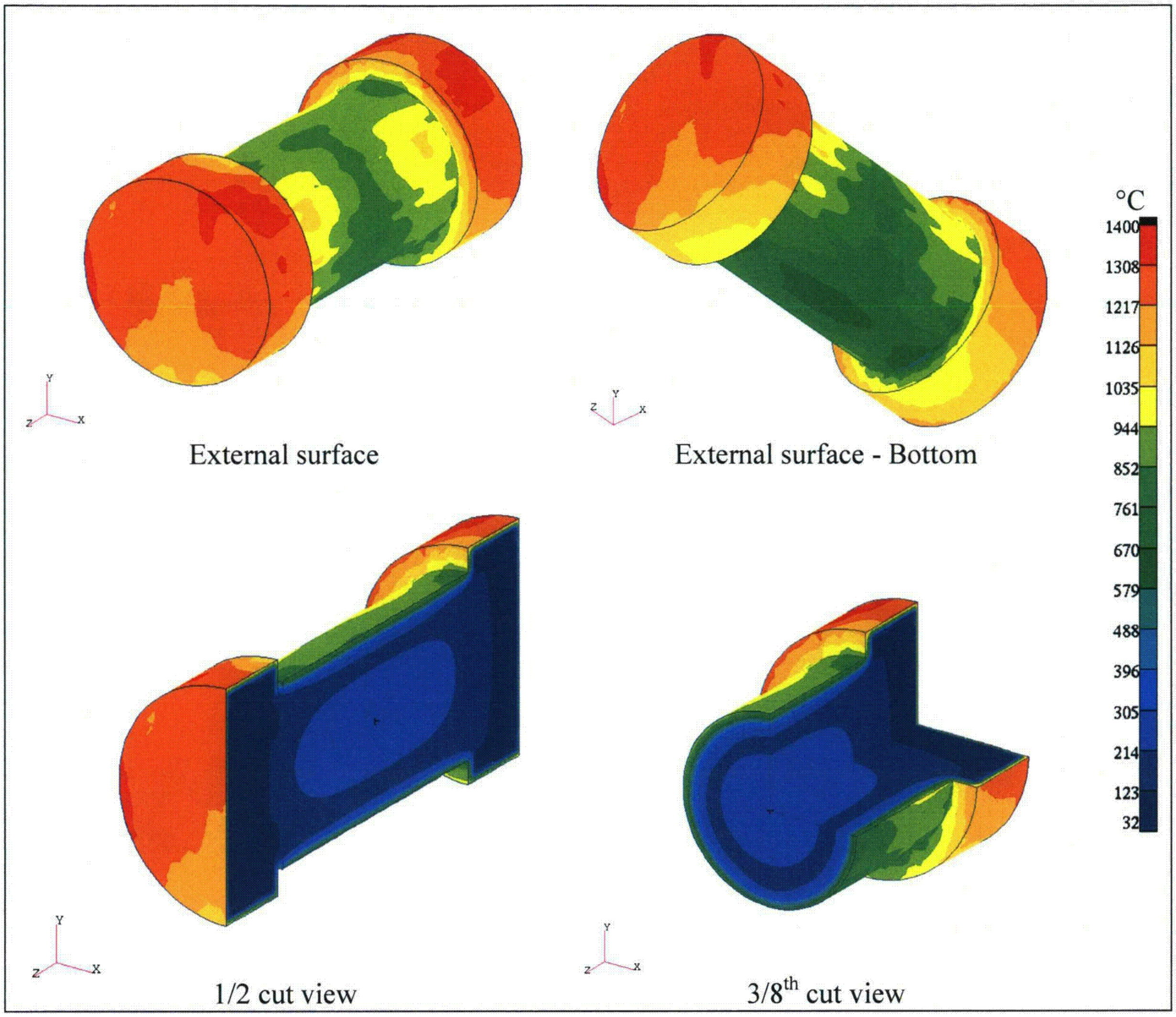


Figure 4-24. Temperature distribution of the Rail-Lead cask at the end of the 30-minute regulatory CAFE fire

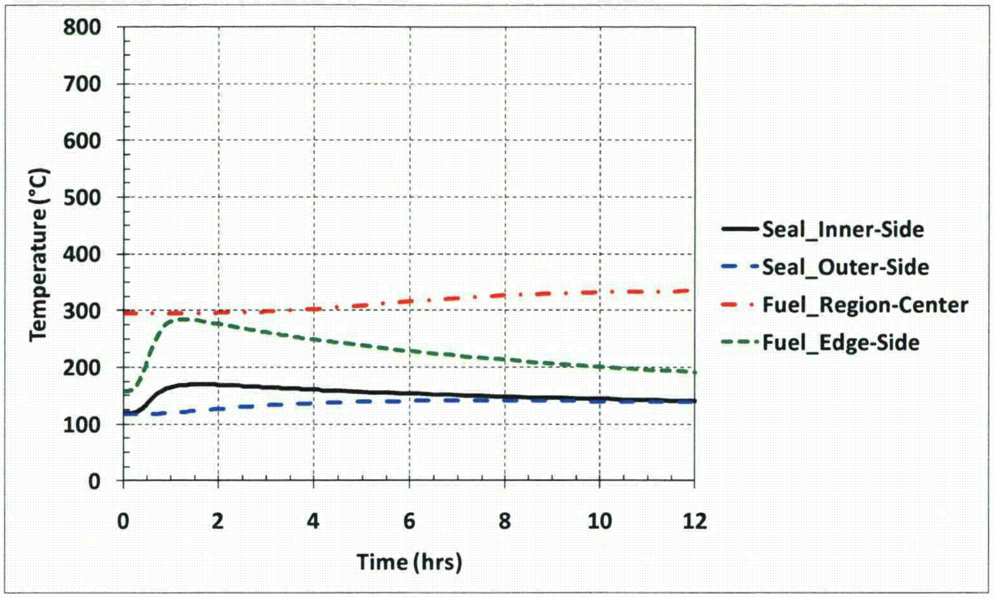


Figure 4-25. Temperature of key cask regions, Rail-Lead cask – Regulatory CAFE fire

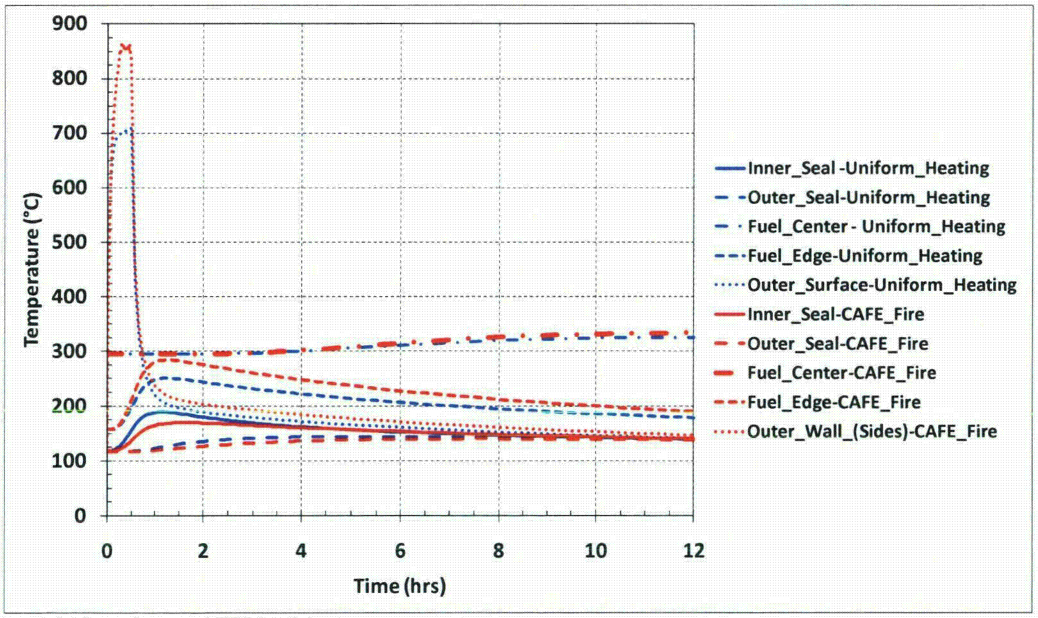


Figure 4-26. Comparison of regulatory fire analysis – Rail-Lead cask: Uniform heating vs. CAFE fire

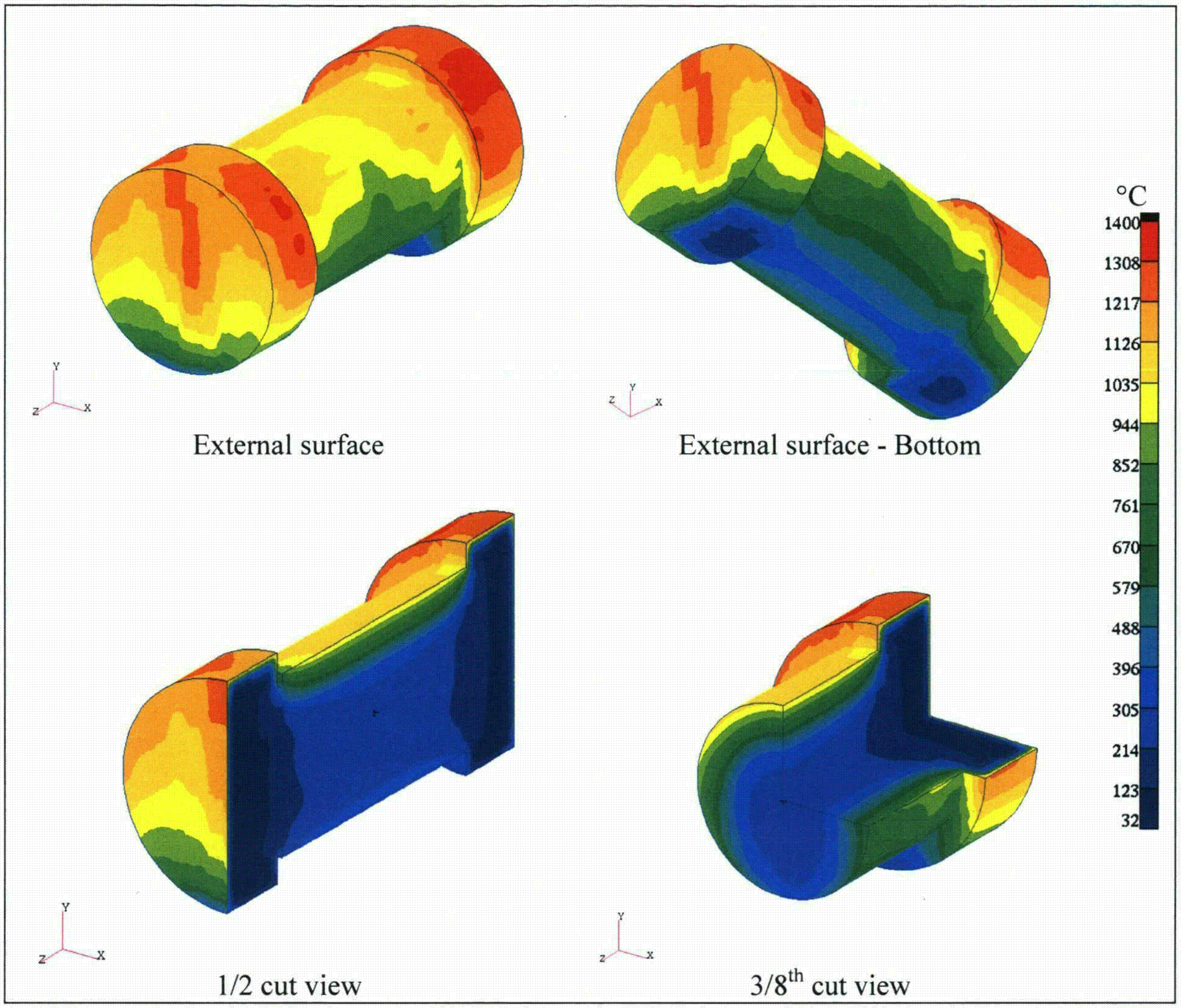


Figure 4-27. Temperature distribution of the Rail-Lead cask at the end of the 3-hour concentric CAFE fire - cask on ground

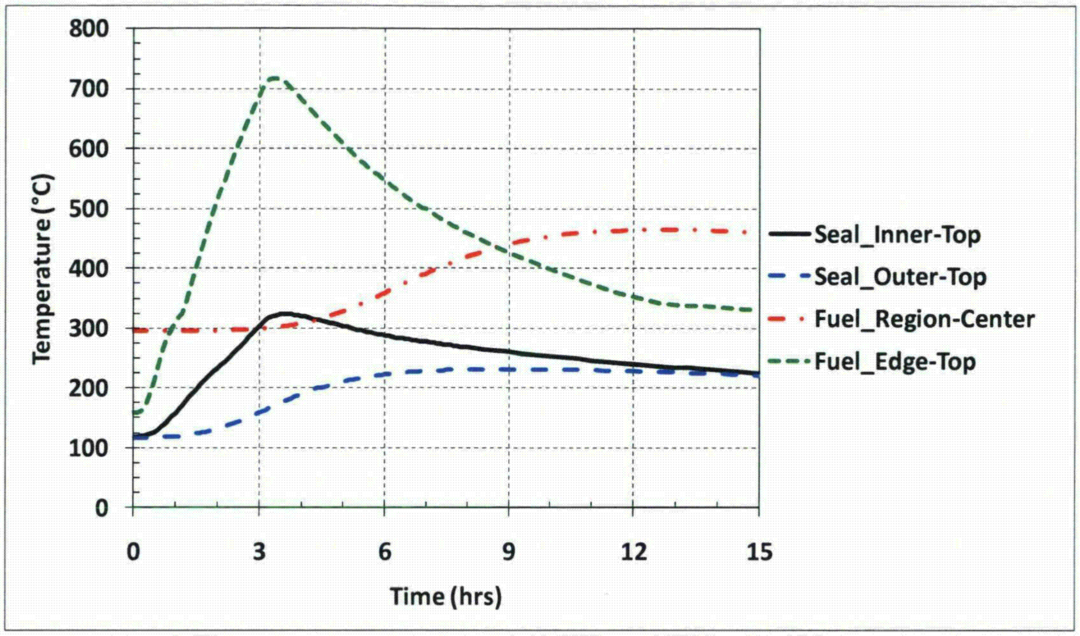


Figure 4-28. Temperature of key cask regions, Rail-Lead cask – Cask on ground, concentric fire

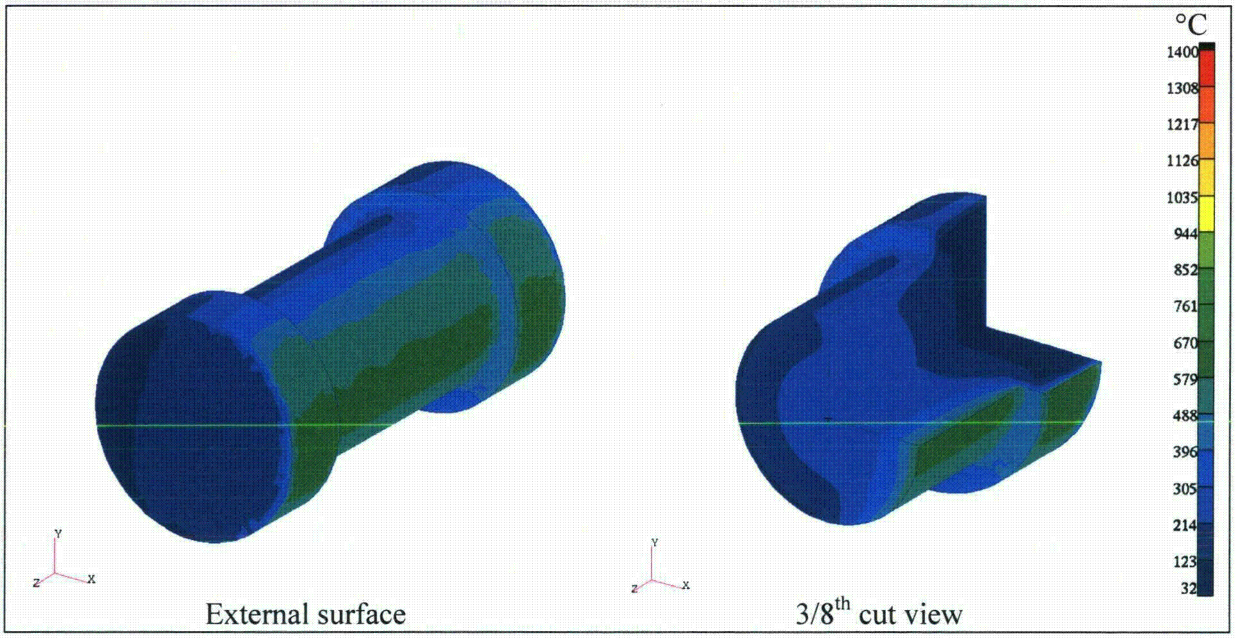


Figure 4-29. Temperature distribution of the Rail-Lead cask at the end of the 3-hour 3m offset CAFE fire - cask on ground

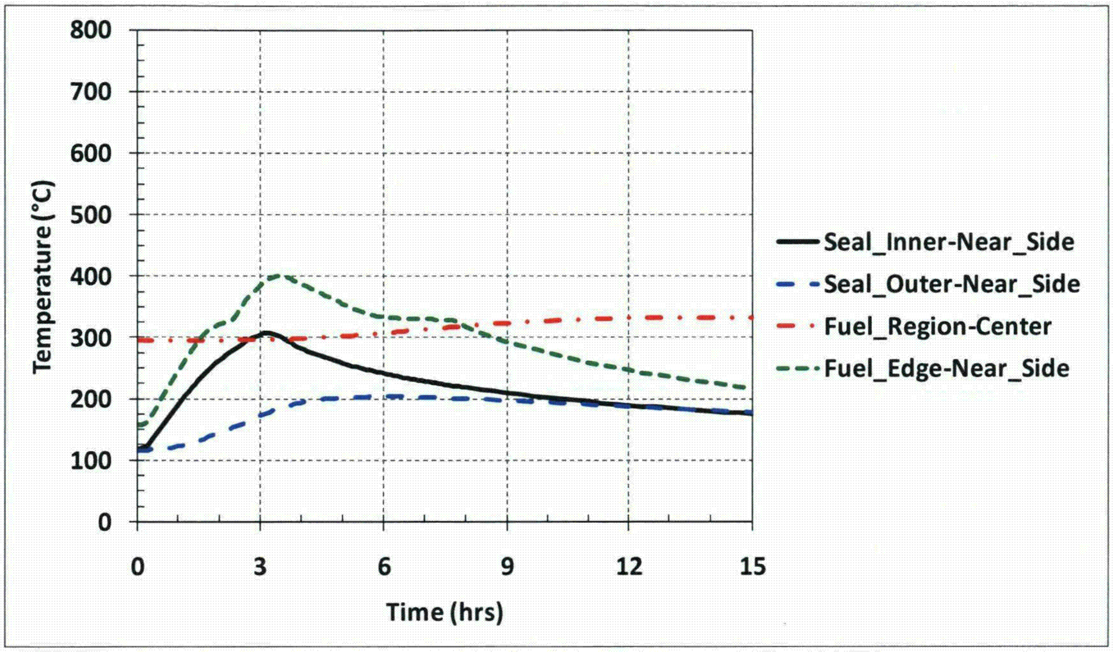


Figure 4-30. Temperature of key cask regions, Rail-Lead cask – Cask on ground, 3m offset fire

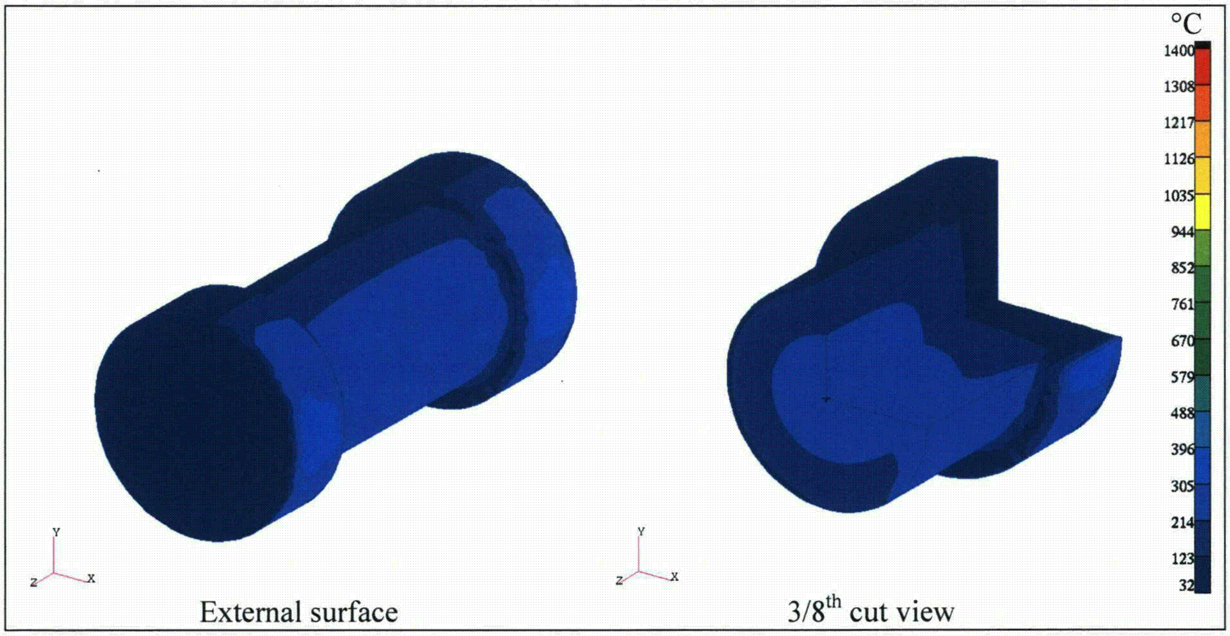


Figure 4-31. Temperature distribution of the Rail-Lead cask at the end of the 3-hour 18m offset CAFE fire - cask on ground

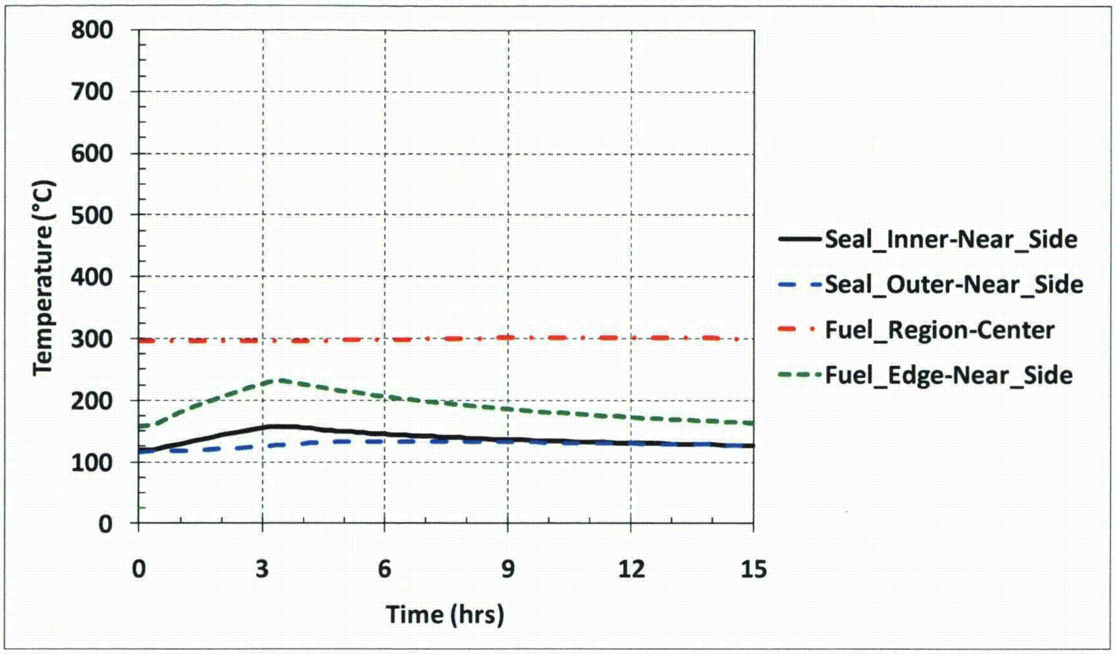


Figure 4-32. Temperature of key cask regions, Rail-Lead cask – Cask on ground, 18m offset fire

Melting of the lead gamma shield

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

The gap in the lead region caused by the concentric fire case is assumed to appear on the top portion of the cask. That is, after the lead melts and buckles the interior wall due to its thermal expansion, molten lead is assumed to flow to the lower portions of the gamma shield region of the cask, which allows a gap to be formed on the top portion of the cask. 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 0.5m (20 inches), which translates to an 8.1% loss of shielding. In the case of the three-meter offset fire, the gap is assumed to form on the top portion of the molten lead region shown in Figure 4-34. For this case, the gap is estimated to be about 0.127m (5 inches), which translates to a 2% loss of shielding. These loss-of-shielding fractions are used as part of the work presented in Chapter 5 to estimate the consequences.

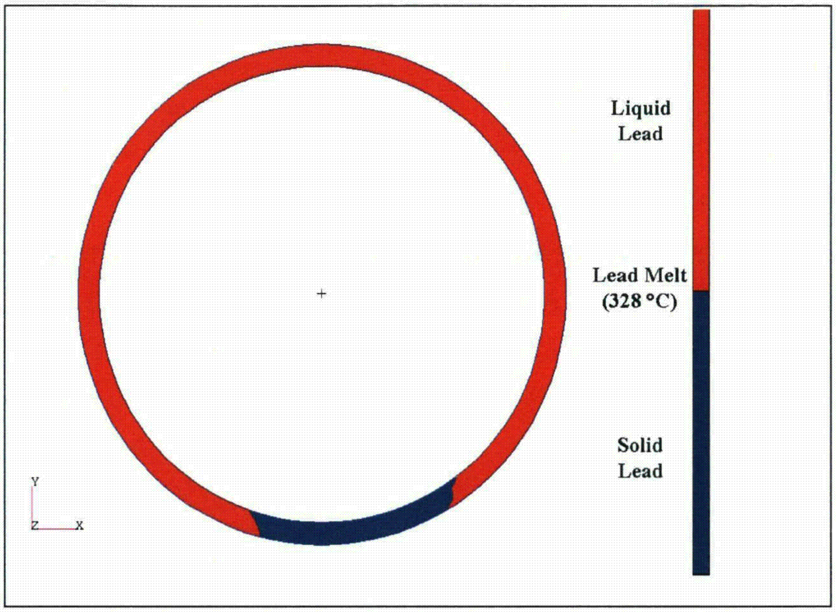


Figure 4-33. Rail-Lead cask lead gamma shield region - maximum lead melt at the middle of the cask – Scenario: Cask on ground, 3-hour concentric pool fire

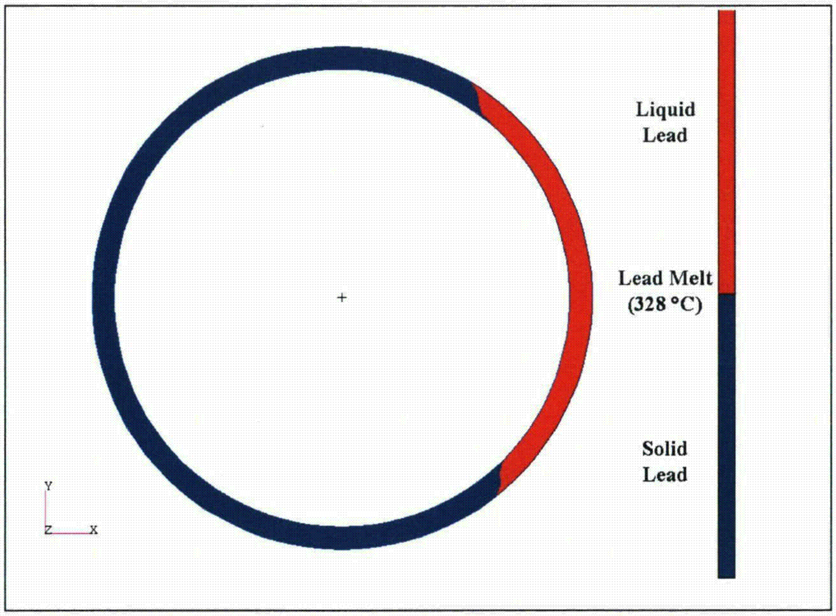


Figure 4-34. Rail-Lead cask lead gamma shield region - maximum lead melt at the middle of the cask – Scenario: Cask lying on ground, 3-hour 3-meter offset pool fire

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. However, some reduction of gamma shielding is estimated to occur in two cases. Partial loss of shielding is expected for the case in which the cask is exposed to an engulfing fire that burns for longer than 65 minutes and for the case in which the cask receives heat from a fire that is offset by three meters and burns for longer than two hours and 15 minutes. Nevertheless, no release of radioactive material is expected if this cask were to be exposed to any of these severe thermal environments, as the elastomeric seals did not reach their temperature limit. This ensures that the cask is capable of maintaining containment under any of the fire environments that are analyzed.

4.4 Truck Cask Analysis

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 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 consideration 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.

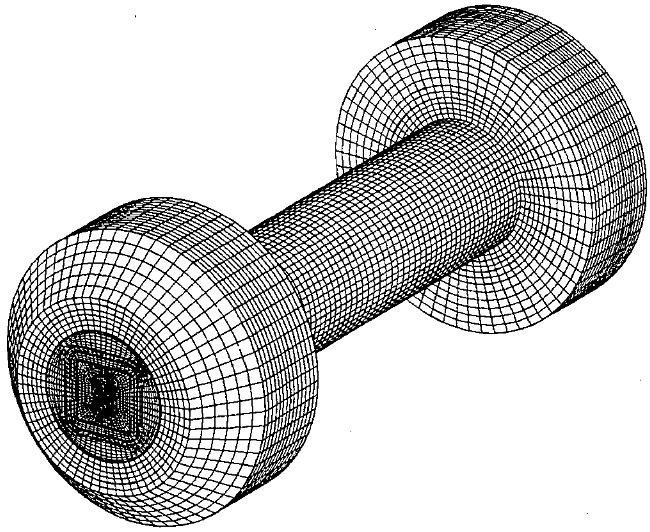


Figure 4-35. Finite element model of the Truck-DU cask

4.4.2. Simulation of the spent fuel region

As with the rail casks, the fuel region comprising the fuel basket and the fuel 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 fuel 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. 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 4-38 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.

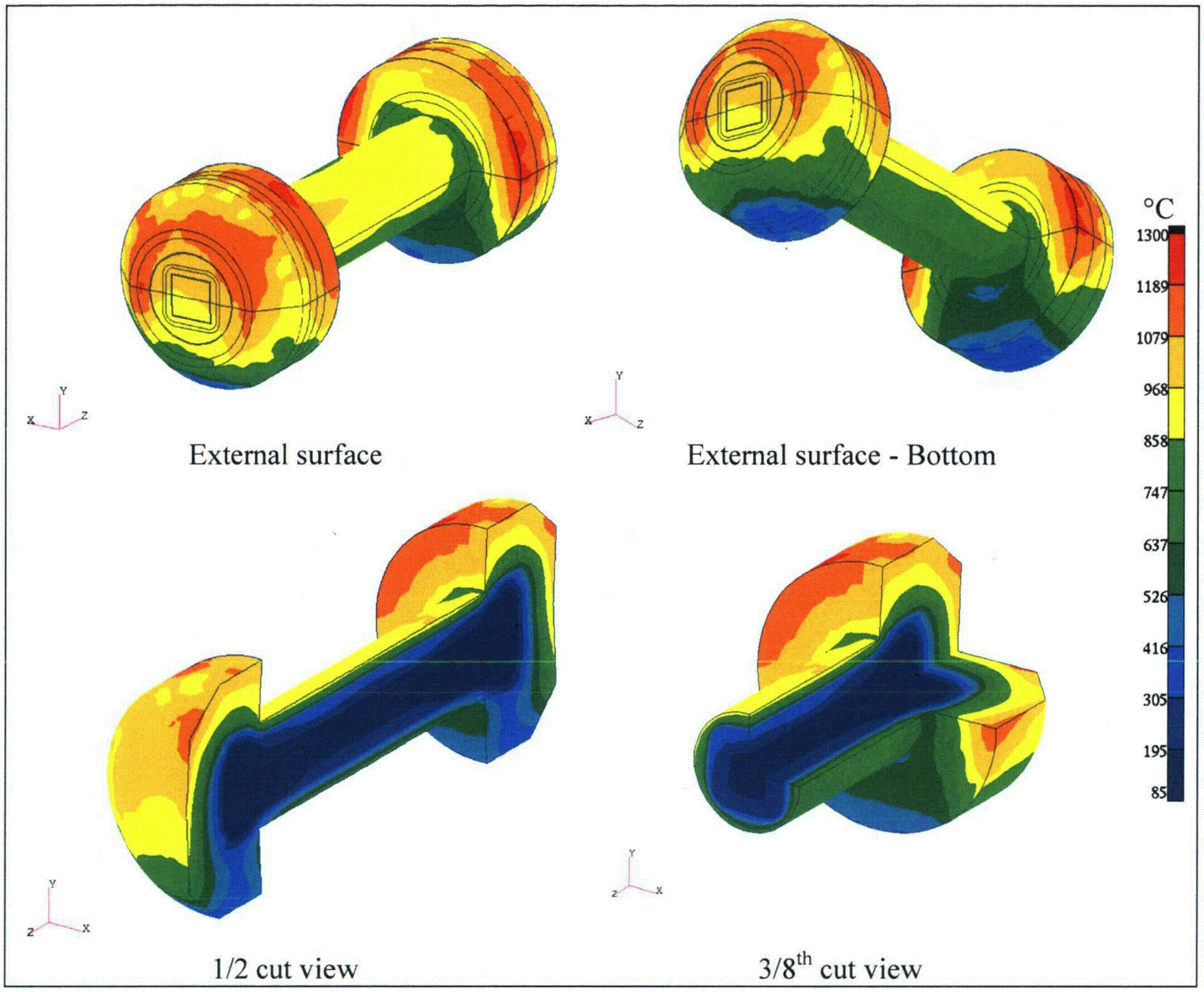


Figure 4-36. Temperature distribution of the Truck-DU cask at the end of the 1-hour concentric CAFE fire - cask on ground

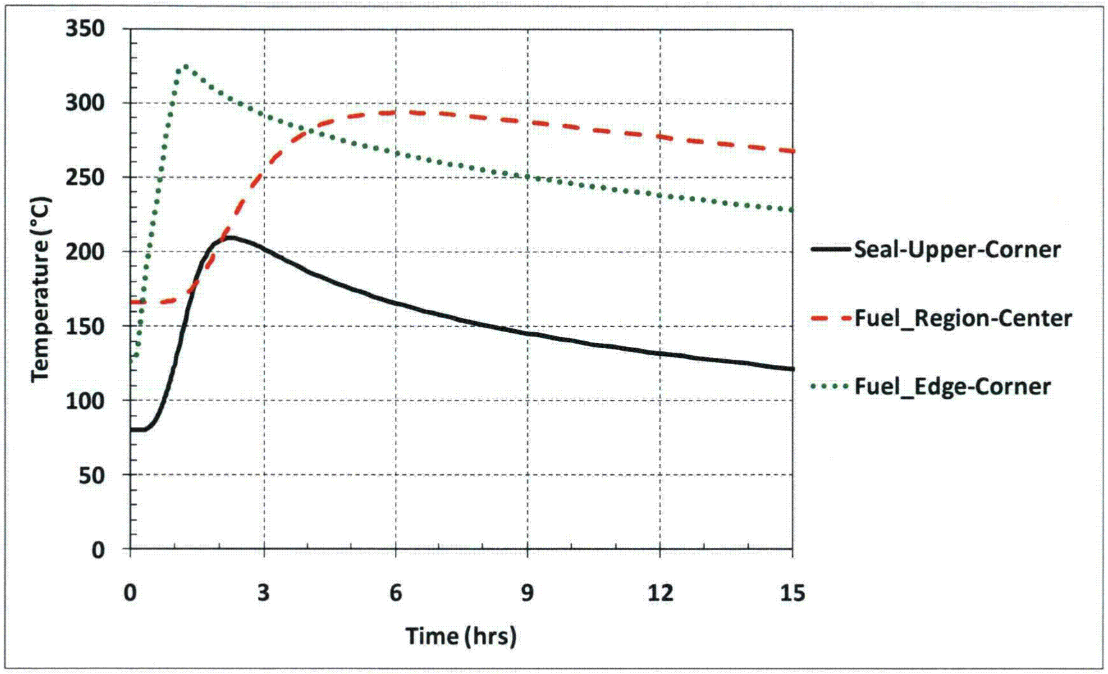
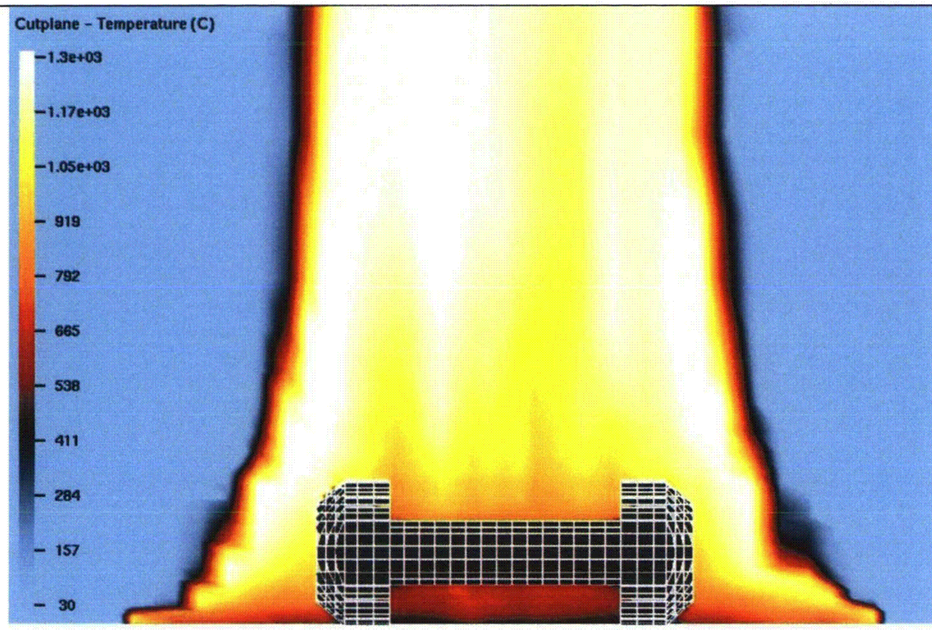
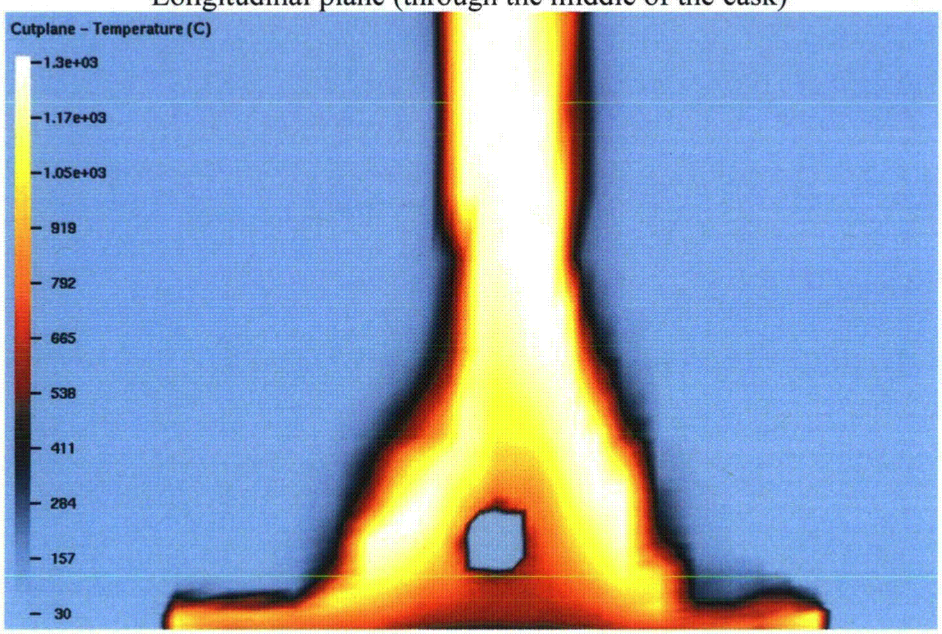


Figure 4-37. Temperature of key cask regions, Truck-DU cask – Cask on ground, concentric fire

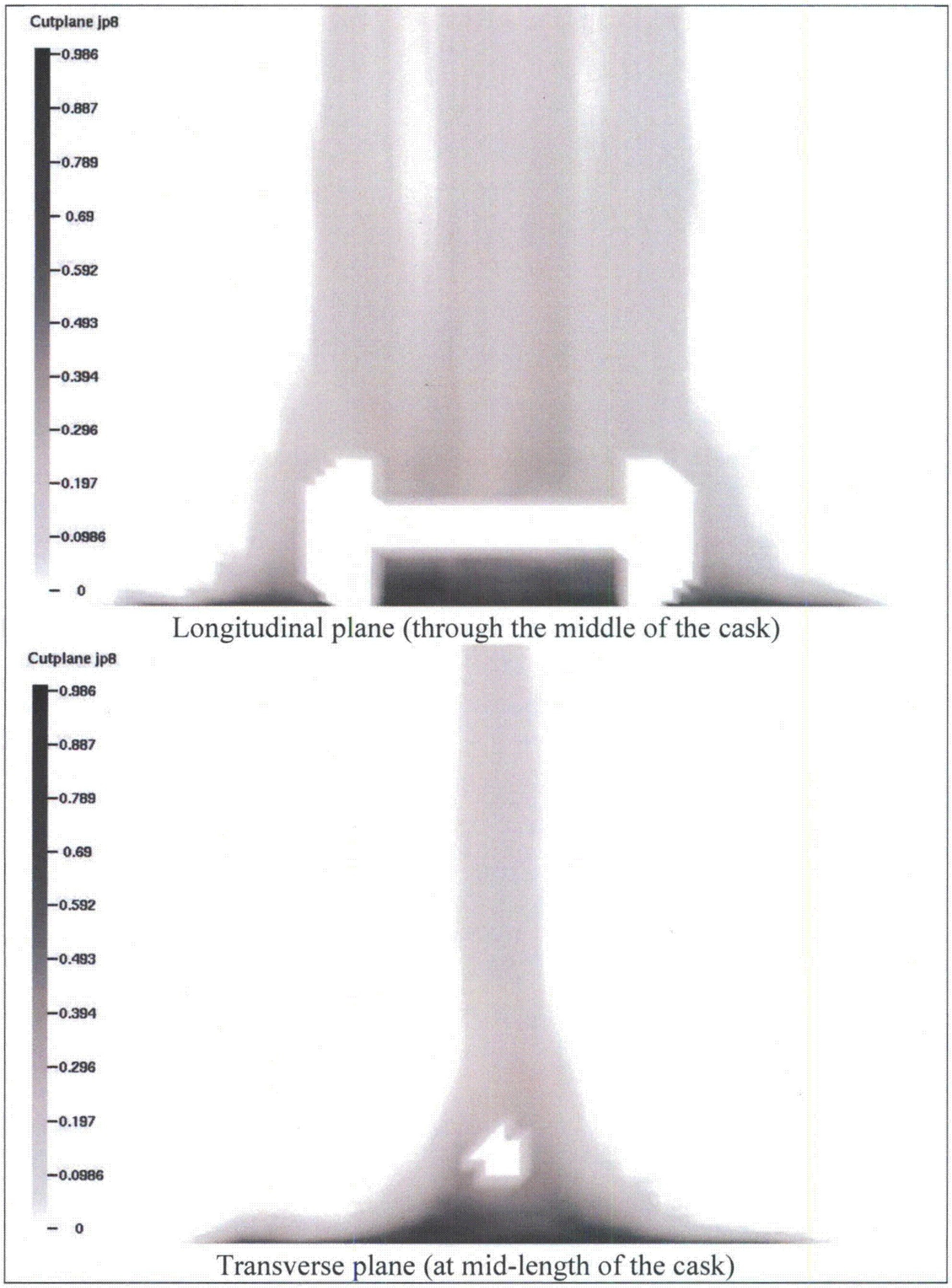


Longitudinal plane (through the middle of the cask)



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 fuel rods from burst rupture and is also capable of maintaining containment when exposed to the severe fire environment analyzed in this study. That is, the fuel region stayed 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 hypothetical fire accident scenarios. These are the regulatory fire described in 10 CFR 71.73, a cask on the ground concentric with a fuel pool sufficiently large to engulf the cask, a cask on the ground with a pool fire offset by the width of a rail car (3 meters), and a cask on the ground with a pool fire offset by the length of a rail car (18 meters). These analyses are performed for the Rail-Steel and the Rail-Lead casks. 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 fuel 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 addition, the Truck-DU cask is able to maintain containment if it were to be exposed to a realistically maximum truck accident fire duration of about an hour. These results demonstrate the adequacy of current regulations to ensure the safe transport of spent nuclear fuel. Furthermore, the results demonstrate that SNF casks designed to meet the current regulations will prevent the loss of radioactive material in all realistic fire accidents.

CHAPTER 5

TRANSPORTATION ACCIDENTS

5.1 Types of Accidents and Incidents

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

- Accidents in which the spent fuel cask is not damaged or affected.
  - Minor traffic accidents (fender-benders, flat tires) that result in minor damage to the vehicle. These are usually called "incidents."[16]
  - Accidents which damage the vehicle and/or trailer enough that the vehicle cannot move from the scene of the accident under its own power, but which 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 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 dose 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. Whe n 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 could be affected. Accidents damaging the fuel but not damaging the cask, and potential consequence to workers are not included in this study because it is assumed a cask involved in an accident will be handled as a special case and the workers will be afforded special protection when opening the cask.

5.2 Accident probabilities

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

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

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[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 result in non-routine transportation are "incidents." This document uses the term "accident" for both accidents and incidents.

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

The net probability of a particular accident scenario 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*0.003 = 0.0025	0.82*0.003*0.0165 = 0.000041
0.00138	Rail/truck 80 kph collision at grade crossing	0.7355*0.985*0.0604*0.0113 = 0.00049	0.7355*0.985*0.0604*0.0113*0.00138 = 0.0000067
0.00087	Railcar falling off bridge at 48 kph	0.7355*0.2665*0.9887 = 0.194	0.7355*0.2665*0.9887*0.00087 = 0.00017

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

Accident probability is calculated from the number of accidents per 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 large truck 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 so sharply between 1971 and the 1990s that, for this analysis, rates from 1996 through 2007 are used:0.0019 accidents per thousand large truck-km and 0.00011accidents 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.

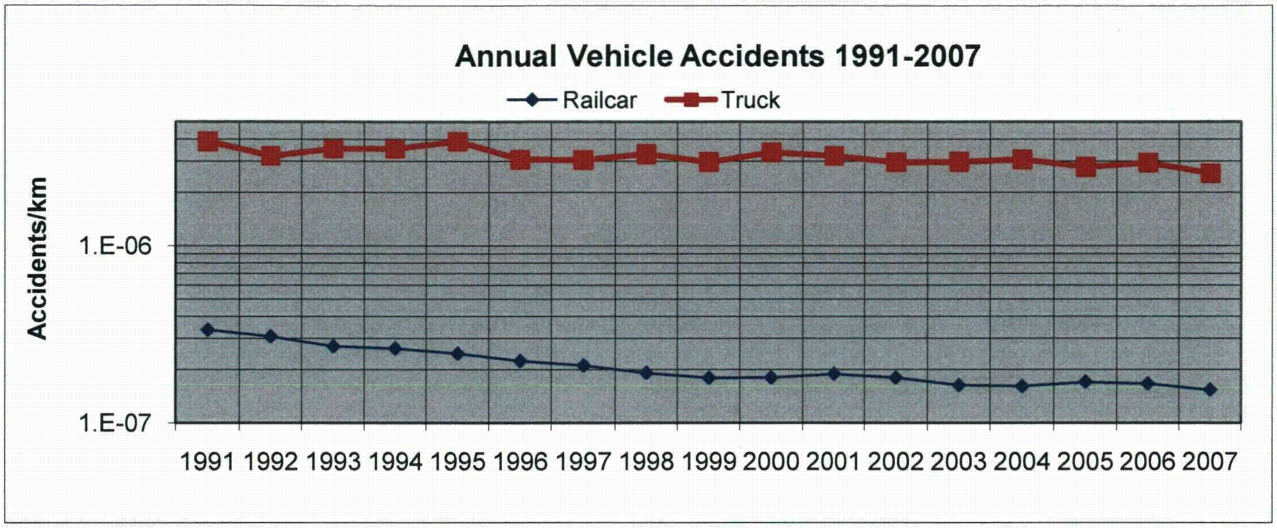


Figure 5-1. Accident frequencies in the U.S. from 1991 until 2007

As Chapters 3 and 4 show, 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:

- Some collisions with hard rock or equivalent at impact speeds greater than 97 km/hour (60 mph) that result in some loss of lead gamma radiation shielding or damage to the cask seals.
- Fires of long enough duration to cause lead melt and subsequent reduction in lead gamma shielding.

Whether or not these accidents happen depends on the likelihood (conditional probability) of the accident scenario as well as on the accident frequency. The event trees for truck and rail, Figures V-1 and V-2 of Appendix V, show some of the elements of accident scenarios in each branch of the respective event tree. The dependence on probability is illustrated by the example of Figure V-5, which shows the sequence of events needed for a pool fire that can burn long enough to compromise 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 the conditional probability of an accident 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, direct loaded cask	3.6×10^{-6}
No loss of lead shielding and no release of radioactive material, direct loaded cask	0.999991
No loss of lead shielding and no release of radioactive material, canistered fuel	0.999995

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 for the direct loaded Rail-Lead cask and 99.9995 percent for the canistered fuel Rail-Lead cask (and 100 percent for the Rail-Steel and Truck-DU casks). The doses to the public and to emergency responders from an accident in which no material is released and there is no loss of lead gamma shield are shown in Tables 5-3 and 5-4. These doses depend only on the external dose rate from the cask in the accident. The radiation dose depends on:

- The external dose rate from 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.
- 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 from 10 CFR 71.51[a]
Truck-DU	1.0 E-03	0.10
Rail-Lead	9.2E-04	0.10
Rail-Steel	6.9E-04	0.10

[a] Calculated by multiplying the allowed dose rate from 10 CFR 71.51 by the 10 hour stop duration.

Table 5-4 and Table 5-5 show collective dose risks in person-sieverts (person-Sv) for the ten-hour stop that follows the accident. The conditional probability of this type of accident (this accident scenario) is 0.99999 (Table 5-2), so that the results could be called "collective doses" instead of "collective dose risks." Collective dose risks 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 collective dose risk is thus the collective dose risk that residents on that route segment could receive if the accident happened at a spot on that particular route segment.

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

FROM	TO	Rail-Lead			Rail-Steel		
		Rural	**Suburban**	**Urban**	**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 dose risks to the public from a no-shielding loss, no-release accident involving a truck cask (person-Sv)

FROM	TO	Truck-DU		
		Rural	Suburban	Urban
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

These collective dose risks may be compared to the background dose that the exposed population would sustain. The average individual U.S. background dose for ten hours is 4.1×10^{-6} Sv. Average background doses for the 16 routes analyzed, for ten hours, are therefore:

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

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

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

The urban collective doses from this type of accident would be indistinguishable from the urban collective background dose. The rural and suburban collective doses from the accident add four percent and five percent, respectively, to background dose on this particular route. Any dose to an individual is well below the doses allowed by 10 CFR 71.51.

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

Spent fuel transportation packages are designed to carry highly radioactive material and need shielding in addition to that provided by the package shell. 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 cannot exceed 0. 1 mSv per hour at two meters from the cask, by regulation (10 CFR 71.47(b)(3)). The three cask types analyzed in this assessment meet this criterion.

Each spent fuel transportation cask analyzed uses a different gamma shield. The Rail-Steel cask has a steel wall thick enough to attenuate gamma radiation to acceptable levels instead of the relatively thinner walls of the other two casks. The Truck-DU cask uses metallic depleted uranium (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. Thus, the only type of cask that could lose gamma shielding is a lead shielded cask like the Rail-Lead cask.

In a hard impact, the lead shield could slump, and a small section of the spent fuel in the cask would then be shielded only by the steel shells. Figure 5-2 and Figure 5-3 show the maximum individual radiation dose to a receptor exposed for one hour at various distances from the damaged cask, for a range of gaps. The gaps are equivalent to slumped fractions of the lead shield. Figure 5-2 shows that one-hour doses larger than the external dose rate in 10 CFR 71.51 occur when the lead shielding gap is more than two percent of the shield.

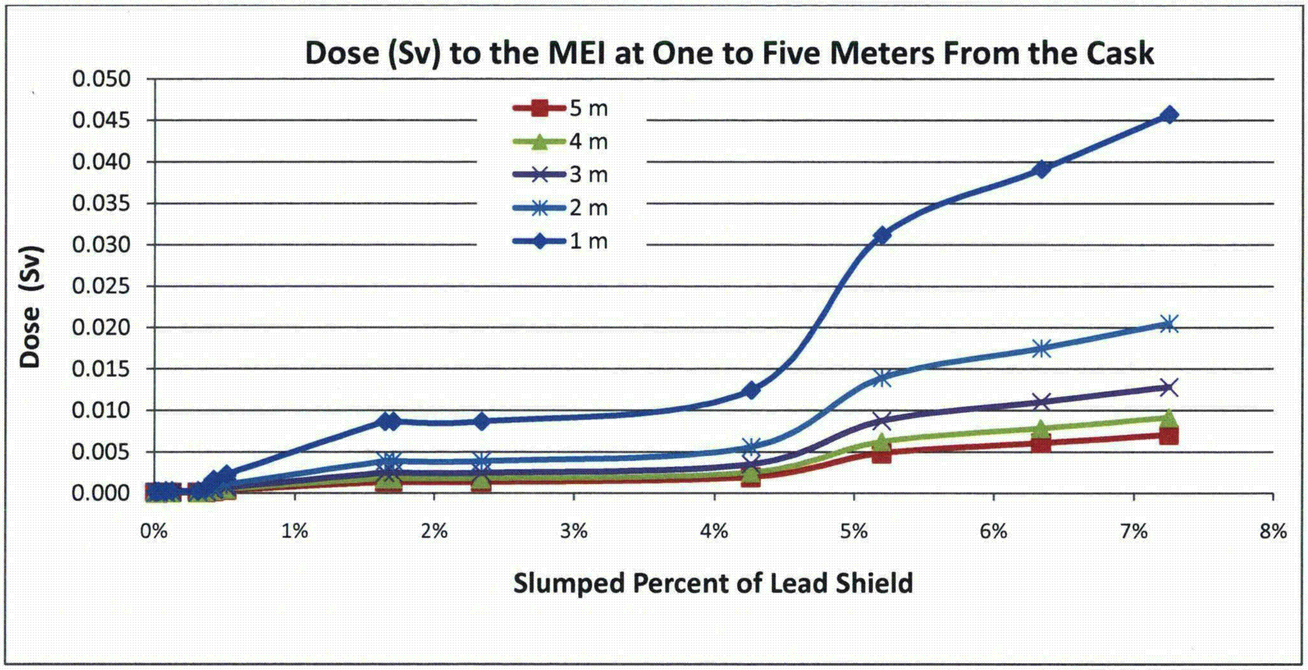


Figure 5-2. Radiation dose for one hour to the maximally exposed individual (MEI) from loss of lead gamma shielding at distances from one to five meters from the cask carrying spent fuel.

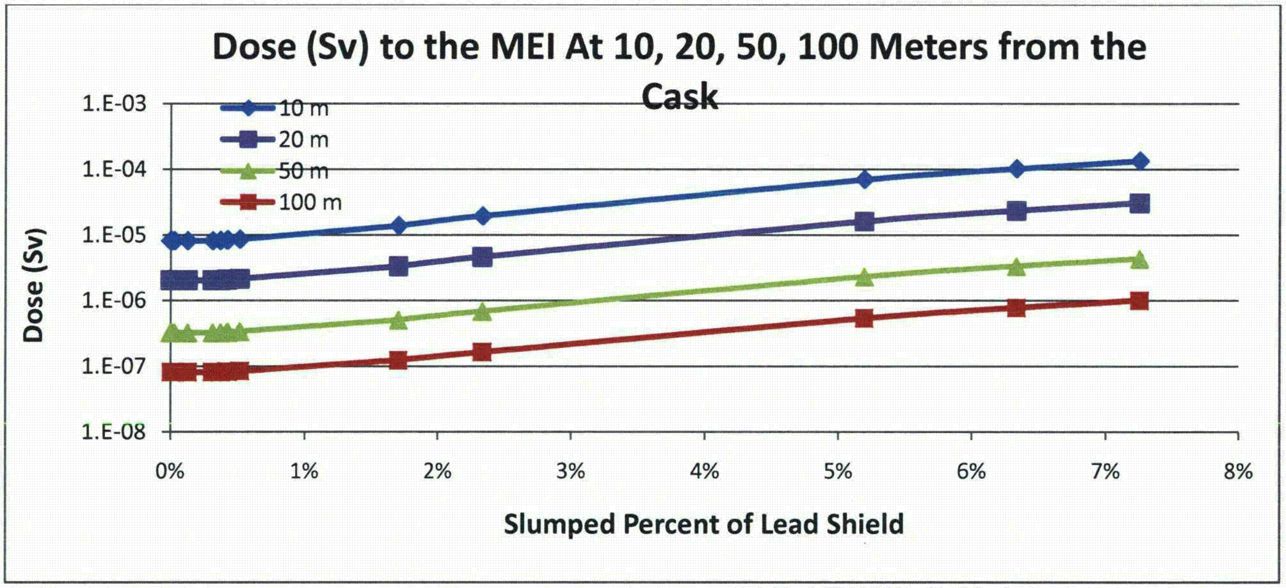


Figure 5-3. Radiation dose to the maximally exposed individual from loss of lead gamma shielding at distances from 20 to 100 meters from the cask carrying spent fuel. The vertical axis is logarithmic, so that all of the doses can be shown on the same graph.

Table 5-2 shows that the probability of an impact accident causing loss of lead shielding is five per million (5×10^{-6}), or one in 200,000 accidents. "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 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. The probability that the lead shielding loss is significant is:

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

or about one in 10 billion per Main Yankee to Hanford shipment.

These very small probabilities indicate that such severe accidents, which are more traumatic to the cask than the tests shown in Figure 1-1, are not likely to happen. The conditions which can cause significant loss of lead shielding 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 PER SHIPMENT
MAINE YANKEE	ORNL	6.5×10^{-7}	0.00328
	DEAF SMITH	5.8×10^{-7}	0.00195
	HANFORD	4.2×10^{-7}	0.00178
	SKULL VALLEY	5.1×10^{-7}	0.00108
KEWAUNEE	ORNL	4.3×10^{-7}	0.00328
	DEAF SMITH	3.3×10^{-7}	0.00130
	HANFORD	2.4×10^{-7}	0.00062
	SKULL VALLEY	3.7×10^{-7}	0.00066
INDIAN POINT	ORNL	8.8×10^{-6}	0.00052
	DEAF SMITH	6.2×10^{-7}	0.04206
	HANFORD	5.1×10^{-7}	0.00190
	SKULL VALLEY	5.5×10^{-7}	0.00203
INL	ORNL	3.6×10^{-7}	0.00069
	DEAF SMITH	3.5×10^{-7}	0.00038
	HANFORD	3.2×10^{-7}	0.00067
	SKULL VALLEY	2.8×10^{-7}	0.00015

The overall collective dose risks to the resident population from a lead shielding loss accident on the sixteen routes studied, are shown in Table 5-7. These include 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 lead shielding will be melted and redistributed in a fire involving the cask is about 10^{-10}. 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 because 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 width.

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

5.4.2 Loss of neutron shielding

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

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(a)(2)
Truck-DU	0.0073	0.10
Rail-Lead	0.0076	0.10
Rail-Steel	0.0076	0.10

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 overall probability depends on the accident rate on the particular route segment traveled by the shipment. 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 fail the bolts that hold the cask lid, dislodge the lid, fail the seals, damage the fuel rods, and release the pressure in the rods. There must be positive pressure to sweep material from the cask to the environment. Even if the bolts and seals fail, if the fuel is in a closed canister in the transportation cask, no radioactive material will be released. As discussed earlier, the only cask of the three studied that could release radioactive material in an accident is the direct loaded Rail-Lead cask. The potential releases discussed in this section would be from this cask. The potential accidents that could result in such a release are discussed in Chapter 3. This chapter discusses the probability of such accidents and the consequences of releasing a fraction of the radionuclide inventory.

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 GWD/MTU and has been cooled for nine years. The Rail-Lead cask is certified to carry 26 PWR assemblies in its direct loaded configuration.

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	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	4.02E+01
^{154}Eu	9.01E+02
^{134}Cs	4.03E+02
^{85}Kr	2.26E+03
^{241}Am	1.58E-01
^{242}Cm	1.00E+00
^{155}Eu	2.63E+02
^{231}Pa	3.12E-02
^{106}Ru	7.50E+00
^{236}U	1.92E-01
^{233}U	8.99E+02
^{241}Pu	5.75E-01
^{113m}Cd	6.13E-01

The ^{60}Co inventory listed is not part of the nuclear fuel. It is the main constituent of CRUD, a corrosion product that accumulates on the outside of the rods, and is formed by corrosion of hardware in the fuel pool. It is listed here with the inventory because it is released to the environment under the same conditions that spent fuel particles are released. A discussion of CRUD release in a hypothetical accident is in Appendix V, Section V.5.4.1.

5.5.2 Conditional probabilities and release fractions

Seven accident scenarios involving the direct loaded 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. The total probabilities of accidental release of radioactive material, the products of accident probabilities and conditional probabilities of each type of accident, are shown for each route in Table 5-11. 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 Impact Speed (kph)	End 193	Corner 193	Side 193	Side 193	Side 145	Side 145	Corner 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[a]	**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

[a] The rod-to-cask release fraction is for the maximum burn-up fuel allowed in the Certificate of Compliance. Appendix VI discusses the effect of high burn-up fuel on this release fraction.

Table 5-11. Total probability of accidental release of radioactive material for each route (for uncanistered fuel in the Rail-Lead cask)

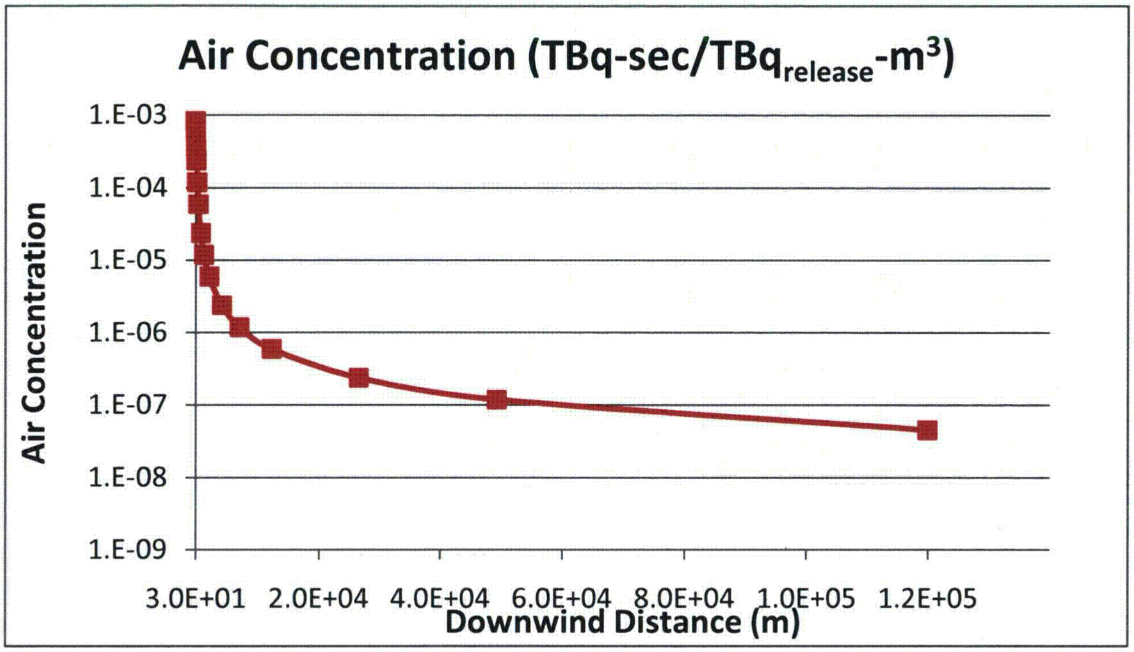
FROM	TO	CONDITIONAL PROBABILITIES 2.68E-08	1.61E-07	8.02E-08	8.02E-08	1.52E-06	1.52E-06	5.81E-05	TOTAL PROBABILITY (metal)
MAINE YANKEE	ORNL	8.79E-11	5.28E-10	2.63E-10	2.63E-10	4.99E-09	4.99E-09	1.91E-07	1.96E-07
	DEAF SMITH	5.23E-11	3.14E-10	1.56E-10	1.56E-10	2.96E-09	2.96E-09	1.13E-07	1.17E-07
	HANFORD	4.77E-11	2.87E-10	1.43E-10	1.43E-10	2.71E-09	2.71E-09	1.03E-07	1.07E-07
	SKULL VALLEY	2.89E-11	1.74E-10	8.66E-11	8.66E-11	1.64E-09	1.64E-09	6.27E-08	6.47E-08
KEWAUNEE	ORNL	8.79E-11	5.28E-10	2.63E-10	2.63E-10	4.99E-09	4.99E-09	1.91E-07	1.96E-07
	DEAF SMITH	3.48E-11	2.09E-10	1.04E-10	1.04E-10	1.98E-09	1.98E-09	7.55E-08	7.79E-08
	HANFORD	1.66E-11	9.98E-11	4.97E-11	4.97E-11	9.42E-10	9.42E-10	3.60E-08	3.71E-08
	SKULL VALLEY	1.77E-11	1.06E-10	5.29E-11	5.29E-11	1.00E-09	1.00E-09	3.83E-08	3.95E-08
INDIAN POINT	ORNL	1.39E-11	8.37E-11	4.17E-11	4.17E-11	7.90E-10	7.90E-10	3.02E-08	3.11E-08
	DEAF SMITH	1.13E-09	6.77E-09	3.37E-09	3.37E-09	6.39E-08	6.39E-08	2.44E-06	2.52E-06
	HANFORD	5.09E-11	3.06E-10	1.52E-10	1.52E-10	2.89E-09	2.89E-09	1.10E-07	1.14E-07
	SKULL VALLEY	5.44E-11	3.27E-10	1.63E-10	1.63E-10	3.09E-09	3.09E-09	1.18E-07	1.22E-07
IDAHO NATIONAL LAB	ORNL	1.85E-11	1.11E-10	5.53E-11	5.53E-11	1.05E-09	1.05E-09	4.01E-08	4.13E-08
	DEAF SMITH	1.02E-11	6.12E-11	3.05E-11	3.05E-11	5.78E-10	5.78E-10	2.21E-08	2.28E-08
	HANFORD	1.80E-11	1.08E-10	5.37E-11	5.37E-11	1.02E-09	1.02E-09	3.89E-08	4.01E-08
	SKULL VALLEY	4.02E-12	2.42E-11	1.20E-11	1.20E-11	2.28E-10	2.28E-10	8.72E-09	8.98E-09

5.5.3 Dispersion

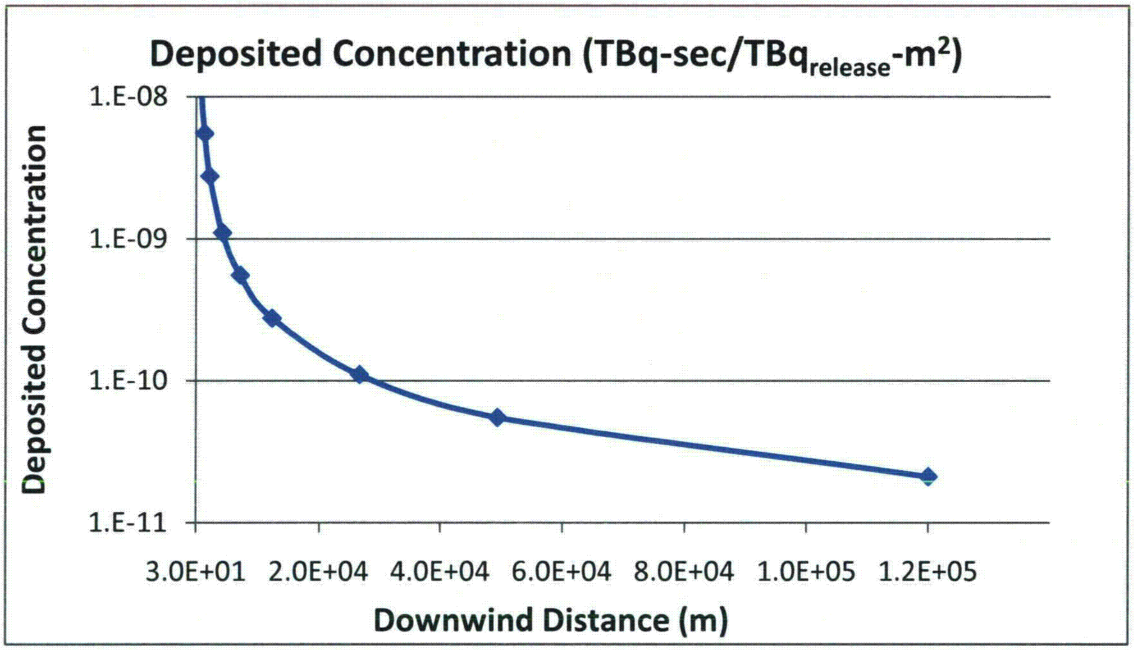
When material is swept from the cask and released into the environment, it is dispersed by wind and weather. The dispersion is modeled using the accident model in RADTRAN 6, which is a Gaussian dispersion model. The release would be at approximately 1.5 meters above ground level, since the cask is sitting 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. These concentrations are along the plume centerline and are the maximum concentrations in the plume. The figure shows the exponential decrease of airborne concentrations as the downwind distance increases. The ground (deposited) concentration also decreases in the downwind direction. The very rapid plume rise, compared to its decay, is responsible for the non-linear scale on the x-axis[18].

[18] Forcing the x-axis to a linear or logarithmic scale distorts the plume.



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 5-112.

Table 5-12. Doses (consequences) in Sv to the maximally exposed individual from accidents that involve a release

Cask Orientation	Impact Speed (kph)	Inhalation	Resuspension	Cloud-shine	Ground-shine	Total
End	193	1.6	1.4E-02	8.8E-05	9.4E-04	1.60
Corner	193	1.6	1.4E-02	8.8E-05	9.4E-04	1.60
Side	193	1.6	1.4E-02	8.8E-05	9.4E-04	1.60
Side	193	1.6	1.4E-02	8.8E-05	9.4E-04	1.60
Side	145	1.6	1.4E-02	4.5E-06	3.6E-05	1.59
Side	145	1.6	1.4E-02	8.8E-05	9.4E-04	1.60
Corner	145	0.73	6.3E-02	1.0E-04	1.0E-04	0.73

The doses listed in Table 5-112 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. 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 they can only receive an external dose for the duration they are in the vicinity of the accident.

Inhaled radioactive particles lodge in the body and are eliminated slowly through physiological processes that depend on the chemical form of the radionuclide. The inhaled dose is called a "committed" dose, because the exposure is for as long as the radionuclide is in the body, though the activity of the nuclide decreases exponentially as it decays. The NRC uses the total effective dose equivalent, the sum of the inhalation and external doses, as a measure of radiological impact.

A pool fire co-located with the cask and burning for a long enough period of 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}, which is not a credible accident. 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 5-113. Of the three casks in this study, only the Rail-Lead cask could result in a release in each kind of accident considered.

Table 5-13. Total collective dose risk (person-Sv) per shipment for release accidents 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 is shown in Table 5-114, and the sum of the two is shown in Table 5-115.

Table 5-14. Total collective dose risk (person-Sv) per shipment for each route from a loss of shielding accident

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	3.6E-10	2. 8E-10	2.5E-10	1.3E-10
KEWAUNEE	1.9E-10	9.1E-11	8.6E-11	7.7E-11
INDIAN POINT	7.4E-09	2.8E-10	3.4E-10	3.2E-10
IDAHO NATIONAL LAB	5.6E-11	9.5E-11	2.1E-11	1.3E-10

Table 5-15. Total collective dose risk (person-Sv) per shipment from release and loss of shielding accidents

	ORNL	DEAF SMITH	HANFORD	SKULL VALLEY
MAINE YANKEE	3.9E-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	6.3E-08	2.6E-09	2.7E-09	1.1E-09
IDAHO NATIONAL LAB	4.3E-10	6.9E-10	1.8E-10	1.2E-09

Table 5-116 shows the total collective dose risk for an accident involving the Rail-Lead 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-All 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 5-16. Total collective dose risk (person-Sv) per shipment 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 5-17 shows the collective accident risk for the 16 routes from loss of neutron shielding

Table 5-17. Total collective dose risk (person-Sv) per shipment 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 are:

- The sixteen routes selected for study are an adequate representation of U.S. routes for spent nuclear fuel. There was relatively little variation in the risks per km over these routes.
- The probability of a severe accident for either truck or rail is one in 100,000 (or less).
- The probability of a fire that would damage a cask on a railcar enough to cause loss of gamma shielding is negligible.
- The overall collective dose risks are extremely 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 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 (or evaluation) of the risks of transporting spent nuclear fuel, updating the assessment performed for NUREG-0170, the 1977 Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes. Both NUREG-0170 and this document provide a technical basis for the regulations of 10 CFR Part 71.

Regulations and regulatory compliance analyses are different from risk assessment. 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., over-estimate) of the damage to a cask in an accident and the radiation emitted from the cask during routine transportation. The original technical basis for 10 CFR Part 71, NUREG-0170, was also conservative, but for a different reason: only limited data were available to perform the required assessment, so NUREG-0170 deliberately used conservative parameter estimates. The NRC's conclusion was that NUREG-0170 showed transportation of radioactive materials to be safe enough, even with conservative assumptions, to support the regulation.

However, assessments are not regulations; they serve a different purpose. An assessment should be as realistic as possible so as to provide information needed to confirm, or to revise, regulations. 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 closer to a "real world" analysis than the previous analyses. Certified spent fuel cask types are analyzed, rather than generic designs. Recent (2005 or later) accident frequency data and population data are used in the analyses, and the modeling techniques have also been upgraded. This study, the Spent Fuel Transportation Risk Assessment, is another step in building a complete picture of spent nuclear fuel transportation safety, and an addition to the technical basis for 10 CFR Part 71. The results of this study are compared with preceding risk assessments in the figures that follow.

6.1 Routine Transportation

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

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

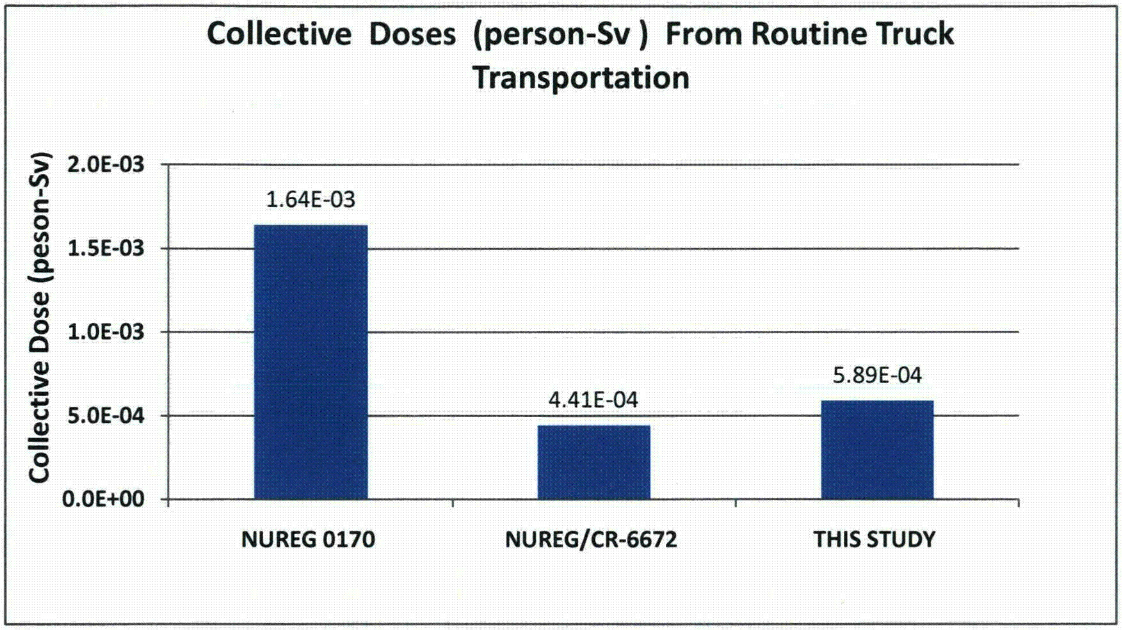


Figure 6-1. Collective doses (person-Sv) from routine truck transportation.

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 times and used the means of the distributions as parameters. As Figure 6-1 shows, the conservatism was decreased considerably.

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 to some extent by the 20 percent greater vehicle speeds used in the present study.

Figure 6-2 and 6-3 show the difference between the present study's calculation of average collective dose to the public and doses to rail and yard crew, and NUREG/CR-6672 for rail casks.

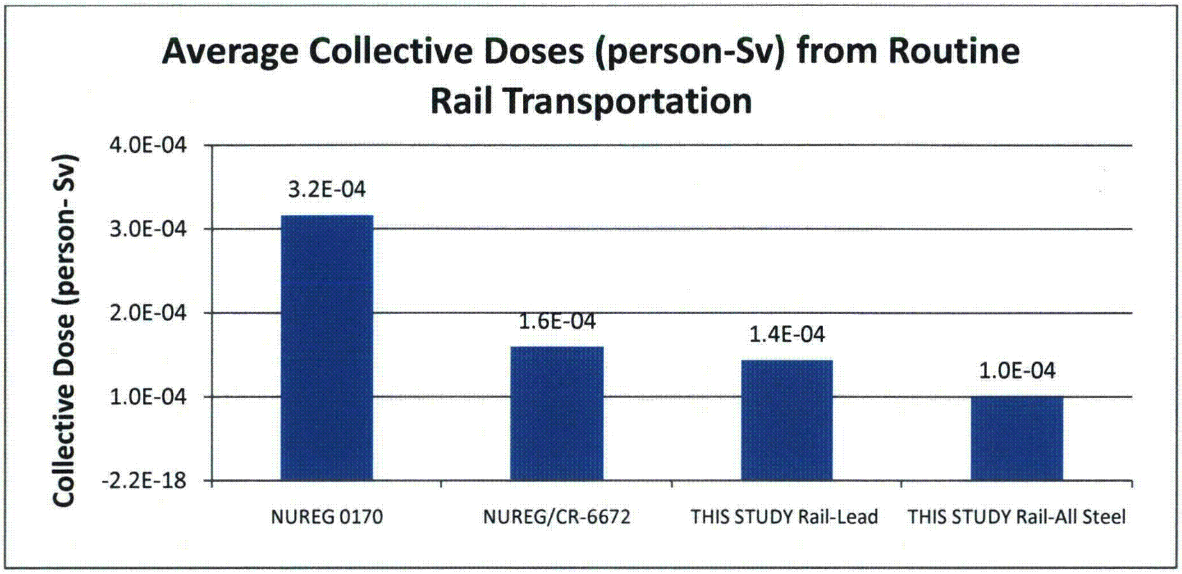


Figure 6-2. Collective doses (person-Sv) from routine rail transportation

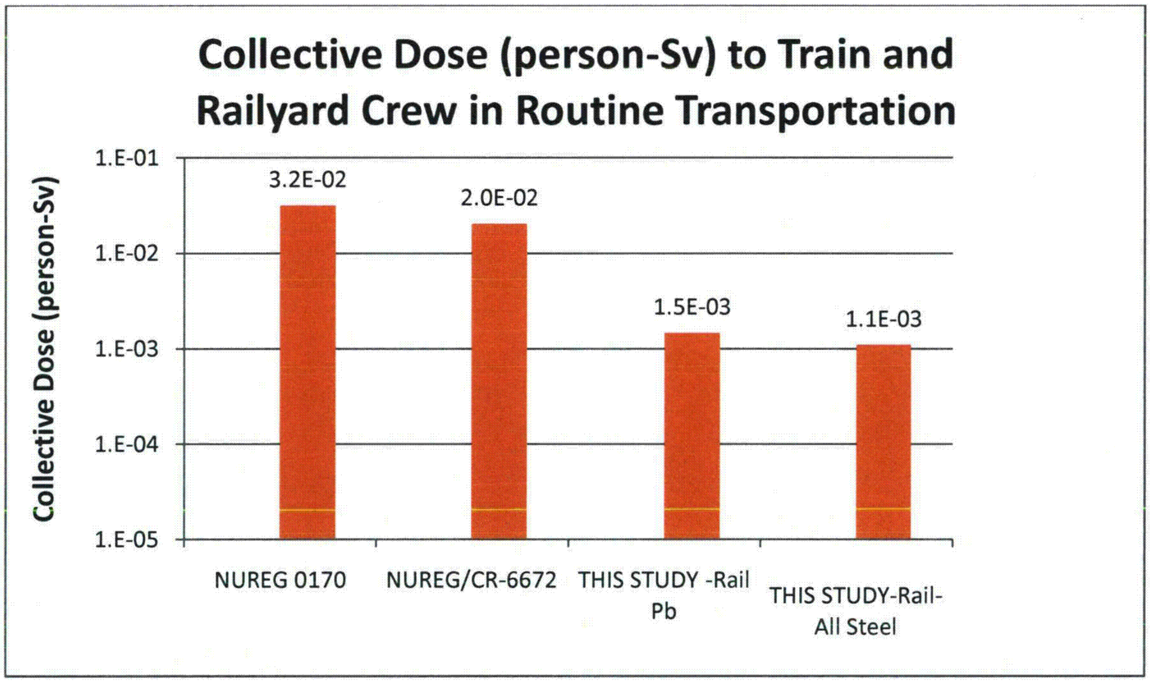


Figure 6-3. Collective dose (person-sv) to train and railyard crew in routine transportation

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

The differences in the collective doses from routine transportation between the cited studies are not the result of differences in external radiation from the spent fuel casks. The 1975 version of 10 CFR Part 71[19] 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 crew and yard 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 and24 kph on urban routes. Faster highway vehicle speeds results in lower individual doses from truck transportation. The present speeds are based upon reported speed distributions for trucks and trains 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 code WebTRAGIS and used the mean values of Gaussian distributions of population densities on 200 routes in the United States. This study uses 2000 census data, updated to 2009, for the rural, suburban, and urban truck and rail route segments in each state traversed in each of the sixteen routes studied.
- 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 very large difference in vehicle density probably explains the differences in truck doses 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 (1986) calculation of doses to railyard workers, and did not calculate a dose to the crew on the train. This study calculated all doses using the formulations in RADTRAN 6, calculated an in-transit crew dose, used an updated value for the time of a classification stop (27 hours instead of 30 hours), and used in-transit stop times from WebTRAGIS rather than the stop dose formula, pegged to total trip length, used in NUREG/CR-6672. The in-transit crew dose calculated in this study was small enough that it contributed a negligible amount to these doses.

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

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

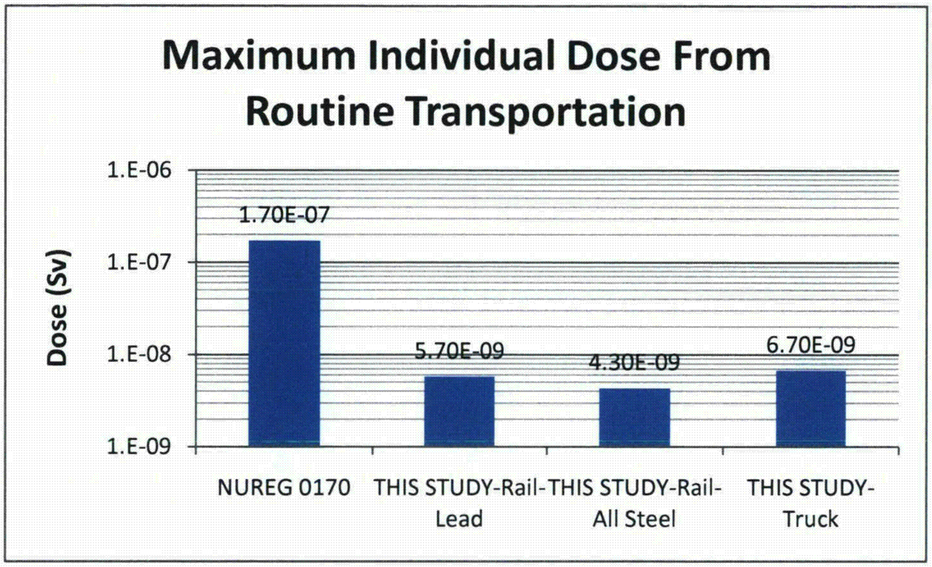

Figure 6-4. Maximum individual dose (Sv) from routine transportation.

6.2 Transportation Accidents

Radiological accident risk is expressed in units of "dose risk" that include the probability of an accident and the conditional probability of certain types of accidents. The units used are dose units (Sv) because probability is a unitless number. NUREG-0170, NUREG/CR-6672, and this study all used 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 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 accident.

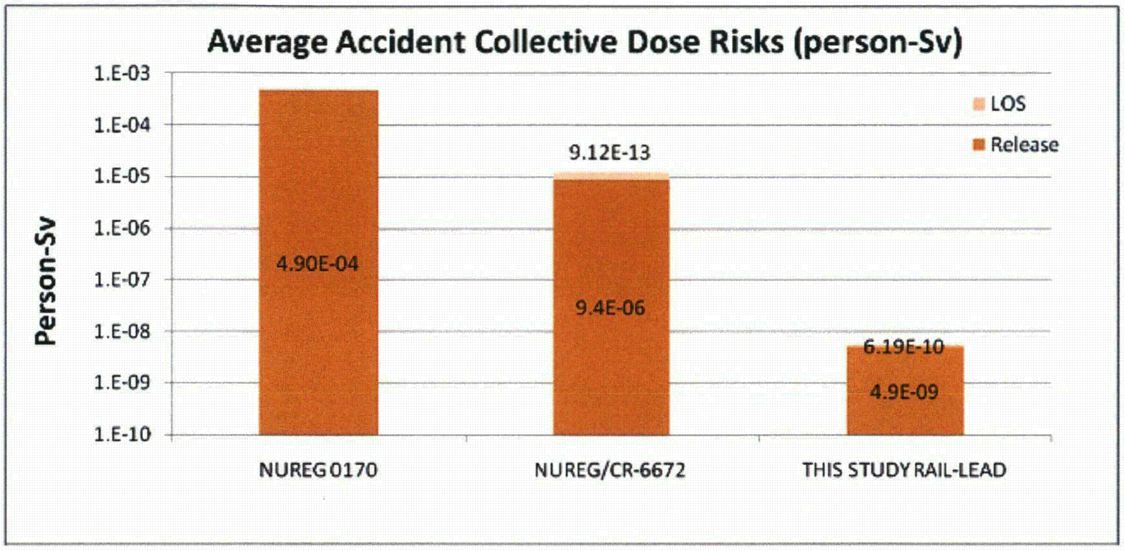

Figure 6-5. Accident collective dose risks from release and LOS accidents. The LOS bar representing the NUREG/CR-6672 collective dose is not to scale.

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

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

The present study followed the analytical outline of the NUREG/CR 6672 analysis, but analyzed the structural and thermal behavior of three certified cask designs loaded with the 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 accidental 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 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 in 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 6-5 for all three cask types studied. Even including this additional consequence type, the collective dose risk from this study is less than that reported in either NUREG-0170 or NUREG/CR-6672.

In conclusion, the three studies reviewed here show that the NRC regulation of transportation casks ensures safety and health. The use of data in place of engineering judgment shows that accidents severe enough to cause loss of shielding or release of radioactive material are improbable and the consequences of such unlikely accidents are 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 from a release and 1.1 Sv from loss of lead shielding, are chronic rather than acute doses.

The most significant consequence of an accident, in addition to any non-radiological consequence of the accident itself, is the external dose from a cask immobilized at the accident location. Average collective doses from this type of accident for the 16 routes studied are shown in Figure 6-9. 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.

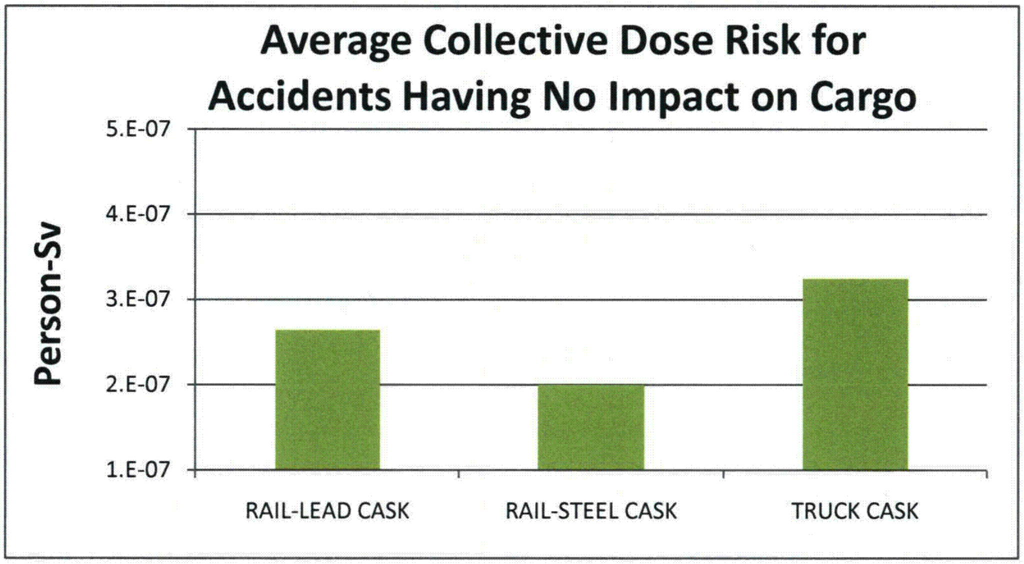

Figure 6-9. Average collective dose from accidents that have no impact on the cargo.

This study demonstrates that risks of transporting spent nuclear fuel are extremely small, and are essentially a tiny fraction of background radiation dose.

- Radiological risks to the public from routine transportation of spent fuel are very small for all shipments.
- When spent fuel is transported in a canister inside a rail cask, the cask is not expected to release radioactive material in an accident, even if the accident involves a fire.
- When spent fuel is transported in a truck cask, the cask is not expected to release radioactive material in an accident, even if the accident involves a fire.

- When spent fuel is transported in a rail cask without an inner canister, only extremely severe and rare accidents can lead to any release of radioactive material—approximately one accident in 100,000.

- Even this extremely rare accident results in a relatively low dose consequence.

- An accidental fire that burns hot enough for long enough, and is close enough to either a truck or rail cask to cause damage is highly unlikely – the probability that this situation occurs is less than one in ten billion, and the accident would not result in any release of radioactive material, only a loss of shielding.

- The regulation of spent fuel transportation guarantees that such transportation will have no adverse impact on people or the environment. The regulation is effective and accomplishes its purpose.

Public perception of radiological risk of transportation may have been distorted by an emphasis on the number of people exposed to a shipment and by failing to recognize that those same people are exposed to background radiation, which is continuous and which delivers a much larger dose. Transportation risk depends more on artifacts of calculation, parameter selection (like the number of people along a route) and assumptions than on the amount of radiation emitted. The conservative estimates of NUREG-0170 may have inadvertently contributed to this misperception The more realistic and less conservative the analysis, the greater the likelihood of redirecting public perception to the more realistic result that spent fuel transportation is so well regulated that it carries almost no risk.

In the nearly 40 years since the NRC published NUREG-0170, there have three reconfirmations of the results—that regulations for spent fuel transport adequately protect the public. Each of these subsequent studies has utilized more sophisticated analysis techniques and improved data to obtain an improved estimate of the risks of transporting spent fuel, and each time the reported risks have been less. While this study has used improved analysis techniques in place of conservative estimates, it still retains conservatism. Some of these conservatisms are:

- Assuming the casks will transport the worst-case fuel they are certified for.
- Neglecting the protection provided by the trailer or railcar that is carrying the cask.
- Assuming the fire accidents happen after the cask has been exposed to extreme normal conditions of transportation temperature.
- Not taking credit for operational controls, such as speed restrictions for trains transporting spent fuel.
- Hard rock surfaces are assumed to be unyielding, but would have to be able to withstand a force of more than 146 MN (33,000,000 pounds) in order to cause a release from any of the casks studied.

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

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

NAC-LWT

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

I.1.2 Rail Casks

NAC-STC

- Steel-lead-steel design – relatively flexible – should have plastic deformation of body before seal failure
- Certified for both direct loaded fuel and for fuel in a welded canister
- Contains either 26 directly loaded PWR assemblies or 1 Transportable Storage Container (3 configurations, all for PWR fuel)
- Can have either elastomeric or metallic seals – must choose a configuration for analysis
- Lead shielding – could melt during severe fires (leads to loss of shielding)
- Polymer neutron shielding
- Wood impact limiter (redwood and balsa)
- Cask is similar to the steel-lead-steel rail cask from 6672
- Two casks have been built and are being used outside of the US

NAC-UMS

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

HI-STAR 100

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

TN-68

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

MP-187

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

MP-197

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

TS125

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

I.2 Certificates of Compliance

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

PREAMBLE

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

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

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.

NRC FORM

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

9 The personnel barrier shall be installed at all times while transporting a loaded overpack.

10 The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71 17

11 Transport by air of fissile material is not authorized

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

13 Expiration Date: March 31, 2014

Attachment Appendix A

REFERENCES:

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

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

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: May 8, 2009

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 7

MODEL NO. HI-STAR 100 SYSTEM

INDEX TO APPENDIX A

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

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INDEX TO APPENDIX A

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

MPC MODEL: MPC-24

A Allowable Contents

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

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

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

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

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

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

II MPC MODEL: MPC-68

A Allowable Contents

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

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

II MPC MODEL MPC-68 (continued)

A Allowable Contents (continued)

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

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

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

II MPC MODEL: MPC-68 (continued)

A Allowable Contents (continued)

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

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

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

II MPC MODEL: MPC-68 (continued)

A Allowable Contents (continued)

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

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

II MPC MODEL MPC-68 (continued)

A. Allowable Contents (continued)

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

a Cladding type	ZR
b Composition:	98.2 wt % ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt % ^{235}U.
c Number of rods per Thoria Rod Canister:	$\leq$ 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 ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.45 wt% ^{235}U.
e. Fuel assembly length:	≤ 176.2 inches (nominal design)
f. Fuel assembly width:	≤ 5.85 inches (nominal design)
g. Fuel assembly weight:	≤ 400 lbs, including channels

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

III MPC MODEL: MPC-68F (continued)

A Allowable Contents (continued)

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

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

III MPC MODEL MPC-68F (continued)

A Allowable Contents (continued)

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

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 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.

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% 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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555-0001

SAFETY EVALUATION REPORT

Docket No. 71-9261
Model No. HI-STAR 100 Package
Certificate of Compliance No. 9261
Revision No. 7

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SAFETY EVALUATION REPORT

Docket No. 71-9261
Model No. HI-STAR HB
Certificate of Compliance No 9261
Revision No 7

SUMMARY

By application dated October 5, 2006, as supplemented 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, Holtec International (Holtec or the applicant) requested a revision to the 10 CFR Part 71 Certificate of Compliance (CoC) No. 9261 for the Model No HI-STAR 100 system The spent fuel cask that is the subject of this amendment request is contained in a site specific license granted by NRC to PG&E for storage of specific fuel at the Humboldt Bay Power Station (HB) site (Docket No. 72-27). Additional supporting changes also requested include incorporating Metamic as an approved neutron absorber and updating the cask identification to B(U)F-96 in accordance with 10 CFR 71.19(e). The staff informed Holtec of the acceptance of the transportation application for technical review by letter dated November 9, 2006.

Based on the statements and representations in the application, as supplemented, and Revision 12 of the SAR, the staff concludes, per its evaluation described in this Safety Evaluation Report (SER), that the requested changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

The following sections summarize the applicant's change requests with respect to the packaging and its contents.

1.1 Packaging

With respect to the packaging, the applicant is proposing to:

- add the HI-STAR 100 Version HB (HI-STAR HB) for use at HB,
- change the Safety Analysis Report (SAR) and licensing drawings to add Metamic as a neutron absorber for use in the HI-STAR HB, and
- change the Package Identification Number of the HI-STAR 100 System from B(U)F-85 to B(U)F-96 in the CoC and Chapter 1 of the SAR.

A new shorter version of HI-STAR 100 was designed specifically for use at HB because the fuel assemblies are shorter than typical Boiling Water Reactor (BWR) fuel assemblies. The design includes a HI-STAR HB overpack, the multiple-purpose canister for HB (MPC-HB), and the impact limiters. The shorter design results in a reduced gross weight. Additionally, a HB specific Damaged Fuel Container (DFC) has been designed.

For the addition of Metamic as a neutron absorber for the HI-STAR HB, the drawings specified Metamic and the minimum B-10 loading, Metamic was added to SAR text, a description of the material was added. an updated criticality analysis was provided and the acceptance testing requirements were provided in SAR Chapter 8

To support the request to change the Package Identification Number of the HI-STAR 100 System from B(U)F-85 to B(U)F-96 in the CoC. staff reviewed the nineteen issues considered in the rulemaking process that resulted in the revised 10 CFR Part 71 dated January 26, 2004 The staff evaluated the applicant's request, as described below.

- Issue 1. Changing Part 71 to the International Systems of Units (SI) Only

  This proposal was not adopted in the final rule, and therefore no changes are needed in the package application or the CoC to conform to the new rule

- Issue 2. Radionuclide Exemption Values

  The final rule adopted radionuclide activity concentration values and consignments activity limits in TS-R-1 for the exemption from regulatory requirements for the shipment or carriage of certain radioactive low-level materials. In addition, the final rule adopted an exemption from regulatory requirements for certain natural material and ores containing naturally occurring radionuclides. Based on the design purpose of the package and the allowed contents specified in the certificate, this change is not applicable to the HI-STAR 100 package. Thus, no changes are needed to conform to the new rule.

- Issue 3, Revision of A_1 and A_2.

  The final rule adopted changes in the A_1 and A_2 values from TS-R-1, with the exception of two radionuclides. The A_1 and A_2 values were modified in TS-R-1 based on refined modeling of possible doses from radionuclides, and the NRC agreed that incorporating the latest in dosimetric modeling would improve transportation regulations. The applicant provided an updated containment analysis in Chapter 4 of the application incorporating the revised A_2 values, which are for radioactive material in normal form. In general, the A_2 values for radionuclides important to the containment requirements for spent fuel shipments were increased. Although the calculated maximum allowable leakage rates were increased, the applicant retained the maximum allowable leakage rate that is demonstrated for the cask through leak testing (4.3 E-6 atm cm^3/sec He). The staff agrees that the package meets the containment requirements of 10 CFR 71.51 considering the changes in the A_1 and A_2 values in Appendix A, Table A-1, of the revised 10 CFR Part 71.

- Issue 4, Uranium Hexafluoride (UF_6) Package Requirements.

  These changes are not applicable, since the package is not authorized for the transport of uranium hexafluoride. Therefore, no changes are needed to conform to the new rule.

- Issue 5, Criticality Safety Index (CSI).

  The final rule adopted the CSI requirement from TS-R-1. The applicant revised Chapters 1 and 6 to clearly distinguish between the CSI and the Transport Index (TI).

- Issue 6. Type C Packages and Low Dispersible Material

  This proposal was not adopted for the final rule. Thus, no changes are necessary.

- Issue 7. Deep Immersion Test

  The final rule adopted an extension of the previous version of 10 CFR 71.61 from packages for irradiated fuel to any Type B package containing activity greater than 10^5 A_2. Because the Model No. HI-STAR 100 package is designed to transport irradiated fuel, the applicant already complies with the deep immersion requirements of 10 CFR 71.61. Thus, no changes are necessary to conform to the new rule.

- Issue 8. Grandfathering Previously Approved Packages

  The final rule adopted a process for allowing continued use, for specific periods of time, of a previously approved packaging design without demonstrating compliance to the final rule. The applicant has decided in accordance with 10 CFR 71.19(e) to submit information demonstrating compliance with the final rule. Thus, grandfathering the design of the package is not necessary.

- Issue 9, Changes to Various Definitions.

  The final rule adopted several revised and new definitions. These changes were adopted to provide clarity to Part 71. No change is necessary to conform to the new rule.

- Issue 10, Crush Test for Fissile Material Packages.

  The revised 10 CFR 71.73 expanded the applicability of the crush test to fissile material packages. The crush test is required for packages with a mass not greater than 500 kilograms (1100 pounds). Since the Model No. HI-STAR 100 package has a mass greater than this, the crush test is not applicable. Therefore, no change is necessary to conform to the new rule.

- Issue 11, Fissile Material Package Design for Transport by Aircraft.

  The final rule adopted a new section, Section 71.55(f), which addresses packaging design requirements for packages transporting fissile material by air. The package is not authorized for shipment by air, and this requirement is not applicable to the Model No. HI-STAR 100 package.

- Issue 12, Special Package Authorizations.

  The final rule adopted provisions for special package authorization that will apply only in limited circumstances and only to one-time shipments of large components. This provision is not applicable to the Model No. HI-STAR 100 package. Thus, no change is necessary to conform to the new rule.

- Issue 13, Expansion of Part 71 Quality Assurance (QA) Requirements to Certificate Holders.

The final rule expanded the scope of Part 71 QA requirements to apply to any person holding or applying for a CoC. QA requirements apply to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging that are important to safety. The applicant must meet the QA program requirements of 10 CFR 71 101(a), (b), and (c). No change is needed to conform to the new rule.

- Issue 14, Adoption of the American Society of Mechanical Engineers (ASME) code

  This proposal was not adopted in the final rule. Thus, no change is needed to conform to the new rule

- Issue 15, Change Authority for Dual-Purpose Package Certificate Holders

  This proposal was not adopted for the final rule. Thus, no change is necessary to conform to the new rule.

- Issue 16, Fissile Material Exemptions and General License Provisions.

  The final rule adopted various revisions to the fissile material exemptions and the general license provisions in Part 71 to facilitate effective and efficient regulation of the transport of small quantities of fissile material. The criticality safety of the Model No. HI-STAR 100 package does not rely on limiting fissile materials to exempt or generally licensed quantities. Chapter 6 of the package application demonstrates criticality safety of the package with the authorized fissile contents. Therefore, no change is necessary to conform to the new rule.

- Issue 17, Double Containment of Plutonium.

  The final rule removed the requirement that packages with plutonium in excess of 0.74 terabecquerel (20 curies) have a second separate inner container. Holtec revised the package application to remove references and discussions related to the second inner container requirement. Additionally, the requirement for helium leak testing has been removed, since the MPC is no longer a containment boundary. Further, the CoC has been revised to delete the limits based on previous double containment requirement in the following condition:

  Condition 5(a)(2), deleted the words "BWR fuel debris may be shipped only in the MPC-68F;" "PWR spent fuel assemblies classified as fuel debris may be loaded only in MPC-24-EF;" and "For the HI-STAR 100 System transporting fuel debris in a MPC-68F or MPC-24EF, the MPC provides the second inner container, in accordance with 10 CFR 71.63. The MPC pressure boundary is a welded enclosure constructed entirely of a stainless steel alloy."

- Issue 18, Contamination Limits as Applied to Spent Fuel and High Level Waste Packages.

  This proposal was not adopted for the final rule. Thus, no change is needed to conform to the new rule.

- Issue 19, Modification of Events Reporting Requirements

  The final rule adopted modified reporting requirements. While the final rule is applicable to the package, no change is needed to either the CoC or the package application to conform to the new rule

"-96" Conclusion

Based on the statements and representations in the application, the staff concludes that the design has been adequately described and meets the requirements of the revised 10 CFR Part 71 Thus, the staff agrees that including the designation "-96" in the identification number is warranted To allow time to modify the packaging markings to include the "-96" designation in the package identification number, the certificate has been conditioned to allow use of packagings marked with the "-85" designation for a period of approximately one year. After May 31, 2009, the packaging must be marked with the package identification number including the "-96" designation.

1.2 Contents

The applicant has requested the following additions or changes to the contents to:

- add the HB fuel as authorized contents,
- change the definition of Damaged Fuel Assembly, and
- add a definition of Undamaged Fuel Assembly.

Changes were made to incorporate the fuel specific to the HB plant into the CoC.

The damaged fuel definition is changed to:

> 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.

The undamaged fuel definition added is:

> 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.

These changes were made for consistency with the corresponding definitions in the HI-STORM Storage CoC (72-1014).

1 3 Generic Changes

The applicant has proposed the following generic changes that are not specific to the HI-STAR HB:

- delete the contents of CoC Section 6.(a) and replace with a direct reference to SAR Chapter 7.
- in the revised Chapter 7, a change was made to the closure plate bolt torque-from 2895 ft.-lbs to 2000 ft.-lbs.
- delete the contents of CoC Section 6.(b) and replace with a direct reference to SAR Chapter 8
- in the revised Chapter 8, dimensional and B-10 loading requirements that are already specified on the licensing drawings are not repeated.
- change the minimum enrichment from 1.8 wt%^{235}U to 1.45 wt%^{235}U in CoC Appendix A, Sections II.A.1.d.i, II.A.2.d, III.A.1.d, III.A.2.d, and III.A.3.d,
- add two assembly cooling times, burnups, and initial enrichments in CoC Appendix A, Table A.7,
- change drawing to allow the use of SA 350 LF3 as an alternate material to SA203A for the HI-STAR Inner Containment Shell and Port Cover.

The modification of CoC Conditions 6.(a) and 6.(b) to reference SAR Chapters 7 and 8 were made to remove unnecessary duplication and details from the CoC and SAR and provide consistency with other 10 CFR Part 71 CoCs. The structural evaluations in SAR Chapter 2 have been revised to support the closure plate bolt torque.

The changes to cooling times, burnups, and initial enrichments are in response to client needs. SAR Chapter 5 was updated to reflect these changes.

The structural evaluations were updated to show that the Safety Factors remain acceptable for both Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) for the allowance of SA 350 LF3 as an alternate material to SA203A for the HI-STAR Inner Containment Shell and Port Cover.

There were other SAR changes that reflected a need to update the SAR and did not require a CoC change.

1.4 Drawings

The applicant has requested approval of changes to Drawing Nos. 3913, 3923, 3930, and 5014-C1765 and added Drawing Nos. 4082, 4102, 4103, and 4113. Drawing 3927 was updated to change its title.

The changes to Drawing No 3913 consist of specifying all necessary details of the buttress plate attachment holes in one callout on sheet 5 rather than spread throughout the drawing. The tolerance on the attachment bolt circle was corrected from +/- 1/8" on Sheet 5 to +/-0.06". The drawing changes corrected an editorial error in detailing the closure plate bolt threads as UNC. The drawing now reflects the actual threads (UN) that are used. The drawing also now reflects the allowance of the use of SA 350 LF3 as an alternate material to SA203A for the HI-STAR Inner Containment Shell and Port Cover.

The changes to Drawing No 3923 include replacing the thru hole in the upper fuel spacer plate with a threaded hole; making the lower plate optional for PWR fuel with control components: changes to eliminate the potential for an undersized condition in the shell, baseplate, and lid assembly; change to the optional lid diameter for the MPC-68 as justified in the structural evaluation; delete the requirements for a secondary containment on plutonium shipments (as allowed by the 10 CFR Part 71 rule change in 2004); and eliminate some redundancy

The changes to Drawing No. 5014-C1765 include changes to update the current licensing drawing to allow for the use of the HI-STAR HB impact limiters as supported in the structural evaluation and editorial changes.

Drawing Nos. 4082, 4102, 4103, and 4113 were added to include the needed packaging specifications for HB fuel.

The staff reviewed the revised set of licensing drawings and finds that the information on the drawings provides an adequate basis for its evaluation against 10 CFR Part 71 requirements. The information on the drawings is consistent with the package as described and evaluated in the SAR.

2.0 STRUCTURAL

Supplement 2.I to the application provides a structural evaluation of the HI-STAR HB package, a shortened version of HI-STAR 100. The organization of the supplement mirrors the format and content of Chapter 2 of the application for the approved HI-STAR 100 package, except it only contains material directly pertinent to the HI-STAR HB.

The staff reviewed the application to revise the Model No. HI-STAR 100 package structural design and evaluation to assess whether the package will remain within the allowable values or criteria for normal conditions of transport (NCT) and hypothetical accident conditions (HAC) as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 2 (Structural Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

2.1 Structural Design

2.1.1 Design Feature Changes

The HI-STAR HB package consists of three principal structural components: (1) the HI-STAR HB overpack, (2) the MPC-HB, and (3) the impact limiters.

The HI-STAR HB overpack, including materials, configurations of inner and intermediate shells, top and bottom forgings, and closure lids, is structurally identical to those of the HI-STAR 100 except for the shorter overall length, lower package weight, reduced strength of impact limiter crush materials, and smaller-diameter threads on lifting trunnions. Other exceptions, which are determined to have insignificant adverse effects on the structural performance, include the optional use of the stacked SA350 LF3 ring forgings to replace the 2.5-inch thick inner shell and the decreased number and length of enclosure shell radial gussets, which connect the outer and intermediate shells to form the cavity for the placement of Holtite-A neutron shield material

The MPC-HB, which consists of an enclosure vessel (EV) and a fuel basket, is configured to hold up to 80 Humboldt Bay fuel assemblies Except for shorter length, the structural features of the EV are identical to those with the other HI-STAR 100 MPCs, including materials, shell diameters, and dimensions of base plate, lid, and penetrations. The MPC-HB fuel basket, as a honeycombed structural weldment, is similar in construction to other approved HI-STAR 100 MPC basket assemblies. It is held inside the EV cavity against the angle and bar spacers welded to the EV shell

The impact limiters of the HI-STAR HB package maintain identical design in form and dimensions to that of the HI-STAR 100 demonstrated structurally adequate for the 60-g design basis deceleration limits. The crush strengths of the aluminum honeycomb material are reduced, however, to accommodate lighter weight of the HI-STAR HB to ensure that the maximum cask decelerations remain to be bounded by the 60-g design basis limits for the 30-ft, free-drop, hypothetical accident conditions.

2.1.2 Design Criteria and Performance Overview

Section 2.I.0 of the supplement notes that the applicable codes, standards, and design criteria, including the design basis maximum cask decelerations of 60 g for the HI-STAR HB, are identical to those for the HI-STAR 100.

For overpack structural components other than the enclosure shell, the application argued that the reduced length and weight of the HI-STAR HB ensure that all stress-based evaluations performed on the HI-STAR 100 produced lower bound safety factors compared to those that would have been calculated for the HI-STAR HB. Sections 2.I.6.1 and 2.I.6.3 evaluate the modified enclosure shell for the heat and reduced external pressure conditions, respectively. The evaluation concluded and the staff agrees that there exist large stress safety factors against an internal pressure developing from off-gassing of the neutron shield material. The staff also concurs that the change associated with the radial gussets will have minor effects on the global response of the overpack subject to a lateral drop.

Recognizing that the MPC-HB has been demonstrated capable of withstanding a side impact deceleration of 60 g for the Part 72 license for the Humboldt Bay ISFSI, the application concluded, and the staff agrees that no new analyses of the MPC-HB are required as long as other design and test conditions, such as the heat and cold pressure/temperature design bases, remain unchanged from those for the HI-STAR 100 enclosure vessel.

On the basis of the above, the staff focuses its review primarily on the impact limiter crush strengths modification for ensuring that maximum cask decelerations remain to be bounded by the design bases.

2.2 Weights and Centers of Gravity

Table 2.1.2.1 of the supplement lists weights of the impact limiters at 26,000 lbs, the loaded MPC-HB at 59,000 lbs, the combined weight at 161,200 lbs for the overpack plus loaded MPC-HB, and the total package weight at 187.200 lbs. The center of gravity of the loaded package is 61 4 inches above the base of the cask.

2.3 Mechanical Properties of Materials

This Materials Evaluation Report is part of an NRC certification review of Holtec's Model No. HI-STAR 100 as a spent nuclear fuel transportation package under requirements specified in accordance with 10 CFR Part 71

The changes proposed are changes that introduce the HB version of the HI-STAR system (HI-STAR HB and MPC-HB), generic changes that are necessary to support the HI-STAR HB (Metamic neutron absorber), but also apply to the other HI-STAR versions and minor changes not directly related to the HI-STAR HB. These minor changes do not affect the materials currently approved in the HI-STAR 100 System. Fuel assemblies for HB are shorter than typical BWR fuel assemblies; therefore, a shorter version of HI-STAR 100 was designed specifically for use at HB (i.e., one time transportation for use at HB only). The shorter design results in a reduced gross weight. Additionally, a HB specific DFC has been designed and fuel class arrays 6x6D and 7x7C have been characterized and analyzed for the HI-STAR HB.

The staff's materials evaluation of the proposed FSAR revisions is based on whether the applicant meets the applicable requirements of 10 CFR Part 71 for packaging and transportation of radioactive material. The evaluation focused on a brief review of the previously approved HI-STAR 100 System and the specific material modifications requested in the application. The objectives of this material review are to ensure adequate material properties exist.

2.3.1 Description

The following is a discussion of the HI-STAR 100 System (generic), the HI-STAR HB design, materials, and the changes proposed that introduce the HB version of the HI-STAR 100 system (HI-STAR HB and MPC-HB). Table 1.3.3 of the application, "Materials and Components of the HI-STAR 100 System," lists the specific material specifications. The HI-STAR 100 System was briefly reviewed for material properties generic to all components followed by a material review of specific material changes proposed that introduce the HB version HI-STAR HB and MPC-HB. No new determination on the adequacy of material properties was made unless it was used as the basis for the proposed revisions.

The HI-STAR 100 System in general consists of three primary components as follows: the multi-purpose canister (MPC), the overpack (HI-STAR) assembly and a set of impact limiters. The overpack confines the MPC and provides the containment boundary for transport conditions. The MPC is a hermetically sealed, welded cylinder with flat ends and an internal honeycomb fuel basket for storing and shipping spent nuclear fuel (SNF) within the overpack containment boundary. A set of impact limiters attached at both ends of the overpack provide energy absorption capability for NCT and HAC.

There are seven MPC models, all are designed to have similar exterior dimensions, one exception is custom designed for the HB plant (MPC-HB), which is approximately 6.3 feet shorter than the generic Holtec MPC designs Each corresponding fuel basket design varies based on the MPC model

2.3.2 HI-STAR 100 Overpack (generic):

The HI-STAR 100 overpack is a heavy-walled steel cylindrical vessel

The overpack containment boundary is formed by an inner shell welded to a bottom plate and to a heavy top flange with a bolted closure plate (all SA 350-LF3, 3½% nickel alloy steel with good impact toughness for use at low temperatures). The closure plate is machined with two concentric grooves for seals (Alloy X750 – NiCrFe).

The outer surface of the overpack inner shell is reinforced with intermediate shells (SA 516-70 carbon steel for moderate and low temperature service with improved notch toughness) of gamma shielding that are installed to ensure a permanent state of contact between adjacent layers. These layers provide additional strength to the overpack to resist puncture or penetration. Radial channels (SA 515-70, carbon steel intended for intermediate or high temperature service) are vertically welded to the outside surface of the outermost intermediate shell at equal intervals around the circumference acting as fins improving heat conduction to the overpack outer enclosure shell surface and as cavities for retaining and protecting the neutron shielding (Holtite-A). An enclosure shell (SA 515-70) is formed by welding panels between each of the radial channels forming additional cavities. These panels together with the exterior flats of the radial channels form the overpack outer enclosure shell. The Holtite-A is placed into each of the radial cavity segments formed, the outermost intermediate shell, and the enclosure shell panels. Pressure relief devices (rupture disks) are positioned in a recessed area on top of the outer enclosure to relieve internal pressure that may develop as a result of a fire accident and subsequent off-gassing of the neutron shield material. A layer of silicone sponge is positioned within each radial channel acting as thermal expansion foam to compress as the neutron shield expands in the axial direction.

The exposed steel surfaces, except seal seating surfaces, of the overpack and the intermediate shell layers are coated with Thermaline 450 and Carboline 890 to prevent corrosion based on expected service conditions. The inner cavity of the overpack is coated with a material appropriate to its high temperatures and the exterior of the overpack is coated with a material appropriate for fuel pool operations and environmental exposure. The coating applied to the intermediate shells acts as a surface preservative and is not exposed to the fuel pool or ambient environment.

Lifting trunnions are manufactured from a high strength alloy (SB 637 - NiCrFe) and installed in threaded openings to the overpack top flange for lifting and rotating the cask body between vertical and horizontal positions. Pocket trunnions were eliminated from the original design and are no longer considered qualified tie-down devices. For transportation, the HI-STAR 100 System is engineered to be mounted on a transport frame secured to the transporter bed.

2.3.3 Multi-Purpose Canisters (generic)

The HI-STAR 100 MPCs are welded cylindrical structures with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, a canister shell, a lid with vent/drain ports and cover plates, and a closure ring (all from Alloy X) Generic MPCs are interchangeable, which have identical exterior dimensions. MPC baskets are formed from an array of plates (Alloy X) welded to each other, such that a honeycomb structure is created.

A series of basket supports (Alloy X) are welded to the inside of the shell position and support the MPC fuel basket Optional aluminum (Alloy 1100) heat conduction elements are installed in some early production models in the peripheral area formed by the basket, the MPC shell, and the basket supports A refined thermal analysis has allowed making this aluminum design feature "optional"

The MPC lid (Alloy X) is a circular plate (fabricated from one piece, or two pieces - split top and bottom) that is edge-welded to the MPC shell. Only the top piece is analyzed as part of the enclosure vessel pressure boundary if the two piece lid design is employed. The bottom piece acts primarily as a radiation shield and is attached to the top piece with a non-structural, non-pressure retaining weld. The MPC lid is equipped with vent and drain ports that are used to remove moisture and gas from the MPC and backfill the MPC with a specified pressure of inert helium gas.

Holtec states that the free volume of the MPC and the annulus between the external surface of the MPC and the inside surface of the overpack containment boundary are filled with 99.995% pure helium gas during fuel loading operations. Following MPC drying operations, the MPC is backfilled with a pre-determined amount of helium gas. The helium backfill ensures adequate heat transfer and provides an inert atmosphere for fuel cladding integrity. The helium gas also provides conductive heat transfer across any gaps between the metal surfaces inside the MPC and in the annulus between the MPC and the overpack containment boundary. Metal conduction transfers the heat throughout the MPC fuel basket, through the MPC aluminum heat conduction elements (if installed) and shell, through the overpack inner steel shell, intermediate steel shells, steel radial connectors, and finally, to the outer neutron shield enclosure steel shell.

The closure ring (Alloy X) is a circular ring edge-welded to the MPC shell and MPC lid. The lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by the threaded holes in the MPC lid during transfer from the storage only HI-STORM 100 System to the HI-STAR 100 overpack for transportation. Threaded insert plugs (Alloy X) are installed to provide shielding when the threaded holes are not in use.

MPCs are designed to store and transport intact fuel assemblies, damaged fuel assemblies, and fuel classified as fuel debris. Intact SNF can be placed directly into the MPC. Damaged SNF and fuel debris must be placed into a Holtec DFC for transportation inside the MPC and the HI-STAR 100 overpack.

HB damaged fuel and fuel debris will be transported in the MPC-HB.

All MPCs are constructed entirely from stainless steel alloy materials except for the neutron absorber and aluminum vent and drain cap seal washers with no carbon steel parts. All structural components in a HI-STAR 100 MPC will be fabricated of Alloy X. For the MPC design and analysis, any steel part in an MPC may be fabricated from any of the acceptable Alloy X materials listed as follows, except that all steel pieces comprising the MPC shell must be fabricated from the same Alloy X stainless steel type: Type 316, Type 316LN, Type 304 and Type 304LN. Holtec states that the Alloy X approach is accomplished by qualifying the MPC for all mechanical, structural, neutronic, radiological, and thermal conditions using material thermophysical properties that are the least favorable and bounding for the entire group for the analysis in question

2.3.4 Impact Limiters (generic)

Once the HI-STAR 100 overpack is positioned and secured in the transport frame the overpack is fitted with (ALSTAR) aluminum honeycomb impact limiters, one at each end Impact limiters ensure the inertia loadings during NCT and HAC are maintained below design levels

2 3 5 Shielding (generic)

To minimize personnel exposure the HI-STAR 100 System is provided with shielding. Initial attenuation of gamma and neutron radiation emitted by the radioactive SNF is provided by the MPC fuel basket structure built from inter-welded plates (Alloy X) and Boral neutron poison panels with sheathing (Alloy X) attached to the fuel cell walls. The MPC canister shell, baseplate, and lid provide additional thicknesses of steel to further reduce gamma radiation and to a lesser extent, neutron radiation at the outer MPC surfaces.

Primary HI-STAR 100 shielding is located in the overpack and consists of neutron shielding (Holtite-A) and additional layers of steel for gamma shielding. Gamma shielding is provided by the overpack inner, intermediate and enclosure steel shells with additional axial shielding provided by both the bottom steel plate and top closure steel plate. Impact limiters provide an increase in gamma shielding and provide additional distance from the radiation source at the ends of the package during transport. A circular segment of neutron shielding is contained within each impact limiter to provide neutron attenuation.

Both Boral and Metamic are neutron absorber materials made of B_4C and Aluminum. Metamic and Boral are both approved materials for the HI-STAR HB only.

2.3.6 Holtite-A Neutron Shielding (Overpack, generic):

Holtec International states Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A may be specified with a B_4C content of up to 6.5 wt%, however, Holtite-A is specified with a nominal B_4C loading (finely dispersed powder form) of 1 wt% percent for the HI-STAR 100 System. The nominal specific gravity of Holtite-A is 1.68 g/cm^3 and is reduced for shielding analysis by 4% to 1.61 g/cm^3 to conservatively bound and account for any potential weight loss at the design temperature and any inability to reach theoretical density. The nominal weight concentration of hydrogen is 6.0%. However, all shielding analyses will conservatively assume 5.9% hydrogen by weight in the calculations.

2.3.7 Gamma Shielding Material (Overpack, generic)

Carbon steel is used in successive layers of plate stock form with each layer of the intermediate shells constructed from two halves. Both halves of the shell are sheared, beveled, and rolled to the required radii. The two halves of the second layer are wrapped around the first shell. Each shell half is positioned in its location and while applying pressure using a uniquely engineered fixture, the halves are tack welded. The second layer is made by joining the two halves using two longitudinal welds. Successive layers are installed by repeating this process. The welding of every successive shell provides a certain inter-layer contact

2.3.8 Coolants (generic).

No coolants are used. however. helium is sealed within the MPC internal cavity The annulus between the MPC outer surface and overpack containment boundary is purged and filled with helium gas. The overpack annulus is backfilled with helium gas for heat transfer and seal testing. Concentric (Alloy X750 – NiCrFe) metallic seals in the overpack closure plate prevent leakage of the helium gas from the annulus and provide the containment boundary for the release of radioactive materials.

2.3.9 Chemical and Galvanic Reactions (generic):

Holtec states that no plausible mechanism exists for significant chemical or galvanic reactions in the HI-STAR 100 System during loading operations. The MPC, filled with helium, provides a non-aqueous and inert environment. Corrosion is a long-term time-dependent occurrence. The inert gas environment in the MPC precludes the incidence of corrosion during transportation. Additionally, the only dissimilar material groups in the MPC are (1) the neutron absorber material and stainless steel and (2) aluminum and stainless steel. Neutron absorber materials and stainless steel have been used in close proximity in wet storage for over 30 years. Many spent fuel pools at nuclear plants contain fuel racks, which are fabricated from neutron absorber materials and stainless steel materials, with geometries similar to the HI-STAR 100 MPC. Not one case of chemical or galvanic degradation has been found in fuel racks built by Holtec. This service experience provides a basis to conclude that corrosion will not likely occur in these materials. Furthermore, aluminum rapidly passivates in an aqueous environment, leading to a thin ceramic (Al_2O_3) barrier, which renders the material essentially inert and corrosion-free over long periods of application.

The HI-STAR 100 overpack combines low-alloy and nickel alloy steels, carbon steels, neutron and gamma shielding, thermal expansion foam, and bolting materials. All of these materials have a history of non-galvanic behavior within close proximity of each other. The internal and external carbon steel surfaces of the overpack and closure plates are sandblasted and coated to preclude surface oxidation. The coating does not chemically react with borated water. Therefore, chemical or galvanic reactions involving the overpack materials are unlikely and are not expected. Furthermore, the interfacing seating surfaces of the closure plate metallic seals are clad with stainless steel to assure long-term sealing performance and to eliminate the potential for localized corrosion.

2.3.10 Design Code Applicability (generic)

The ASME Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997, is the governing code for the construction of the HI-STAR 100 System. The ASME Code is applied to each component consistent with the function of the structure, system, and components (SSCs) of the HI-STAR 100 System that are labeled Important to Safety (ITS). Some components perform multiple functions and in those cases, the most restrictive code is applied.

The HI-STAR overpack top flange, closure plate, inner shell, and bottom plate are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NB, to the maximum extent practical. The remainder of the HI-STAR overpack steel structure is designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF, to the maximum extent practical.

2.3.11 Acceptance Tests and Maintenance Program (generic).

Weld examinations shall be performed in accordance with written and approved procedures, by qualified personnel. All results, including relevant indications, shall be made a permanent part of the quality records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

ASME Code Section III and Regulatory Guides 7.11 and 7.12 require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures. Charpy V-notch testing shall be performed on each plate or forging for the HI-STAR 100 Package containment boundary (overpack inner shell, bottom plate, top flange, and closure plate) in accordance with ASME Code Section III, Subsection NB, Article NB-2300. Weld material used in fabricating the containment boundary shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NB, Articles NB-2300 and NB-2430.

Non-containment portions of the overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NF, Articles NF-2300, and NE-2430. The non-containment materials to be tested include the intermediate shells, overpack port cover plates, and applicable weld materials.

2.3.12 HI-STAR 100 System, Version for HB:

Holtec states that the HI-STAR 100 System has been expanded to include options specific for use at PG&E's HB plant for dry storage and future transportation of SNF.

HB fuel has a cooling time of more than 25 years and relatively low burnup. Heat load and nuclear source terms of this fuel are substantially lower than the design basis fuel. Peak cladding temperatures and dose rates are below the regulatory limits with a significant margin. All major dimensions and features, such as diameter, wall thickness, flange design, top and bottom thicknesses, are maintained identical to the standard (generic) design.

The HI-STAR HB overpack is a heavy-walled, steel cylindrical vessel identical to the standard (generic) HI-STAR 100 overpack, except that the outer height is approximately 128 inches and the inner height is approximately 115 inches. The HI-STAR HB overpack does not contain radial channels vertically welded to the outside surface of the outermost intermediate shell as installed on the HI-STAR 100 overpack (generic) The HI-STAR HB overpack utilizes neutron shielding (Holtite-A) placed in the annulus region between the multi-layered shells and enclosure shell without connecting ribs. This feature is unique to the "HB" version. The annular shield, a thick layer of a low conductivity material, Holtite-A, retards the lateral transmission of fire heat during hypothetical accidents, which minimizes the heating of the HI-STAR HB package internals and the stored fuel during fires.

MPC-HB is similar to the generic MPC-68F, except it is approximately 114 inches high The MPC-HB is designed to transport up to 80 HB BWR SNF assemblies Damaged HB fuel and fuel debris must be transported in the Holtec custom designed HB DFC

Holtec considers that almost all of the HB fuel assemblies not classified as damaged are intact. however. the inspection records of the HB fuel assemblies precludes classifying the assemblies as intact fuel since the interior rods of the assembly are in an unknown condition These rods are classified as undamaged and can perform all fuel specific and system related functions. even with possible breaches or defects.

Applicable design codes, standards and criteria for the HI-STAR HB are identical to HI-STAR 100 except that the internal surfaces of the intermediate shells will not be coated with a silicone encapsulate due to its lower heat loads.

Differences between the HI STAR HB and HI-STAR 100 are limited to: shorter overall length, lower package weight, reduced strength of impact limiter crush materials, smaller diameter threads on lifting trunnions, and MPC-HB neutron absorber material as follows:

Holtec states that Metamic is a neutron absorber material developed by the Reynolds Àluminum Company for spent fuel reactivity control in dry and wet storage applications. Metamic is requested to be used in the HI-STAR HB. Metallurgically, Metamic is a metal matrix composite (MMC) consisting of a matrix of 6061 aluminum alloy (precipitation hardening, with magnesium and silicon as its major alloying elements) containing Type 1 ASTM C-750 boron carbide. Metamic is characterized by extremely fine aluminum (325 mesh or better) and boron carbide powder. Typically, the average B_4C particle size is between 10 and 15 microns. High performance and reliability of Metamic is derived from the particle size distribution of its constituents, rendered into a metal matrix composite by the powder metallurgy process. This yields uniform homogeneity. For the Metamic sheets used in the MPCs, the extruded form is rolled down into the required thickness.

Metamic has been subjected to an extensive array of qualification tests sponsored by the Electric Power Research Institute (EPRI) which evaluated the functional performance of the material at elevated temperatures (900°F) and radiation levels. Test results, documented in an EPRI report, indicate that Metamic maintains its physical and neutron absorption properties with little variation in its properties from the unirradiated state.

Time required to dehydrate a Metamic equipped MPC is expected to be less when compared to an MPC (generic) containing Boral, due to the absence of interconnected porosities. Analyses performed by Holtec show that the streaming due to particle size is virtually non-existent in Metamic. Metamic is a solid material, therefore, no capillary path through which spent fuel pool water can penetrate Metamic panels can chemically react with aluminum in the interior of the material to generate hydrogen. Chemical reaction of the outer surfaces of the Metamic neutron absorber panels that may occur with water to produce hydrogen transpires rapidly and reduces to an insignificant amount in a short period of time Nevertheless, combustible gas monitoring for Metamic equipped MPCs and purging or exhausting the space under the MPC lid during welding and cutting operations is required until sufficient field experience is gained that confirms that little or no hydrogen is released by Metamic during these operations.

Holtec states that each manufactured lot of neutron shield material shall be tested to verify that the material composition (aluminum and hydrogen), boron concentration, and neutron shield density (or specific gravity) meet specified requirements Testing and installation shall be performed in accordance with written and approved procedures and/or standards and shall become part of the QA record. The material manufacturer's QA program and its implementation shall be subject to review and ongoing assessment, including audits and surveillances. Procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps or voids from occurring in the material Neutron shield integrity shall be verified via measurements either at first use or with a check source over the entire surface of the neutron shield, including the impact limiters

2.3.13 Conclusion

For the purposes of the this review, the staff did revisit previously approved material properties used in the original HI-STAR 100 Cask System application and no new determination on the adequacy of those material properties was made unless it was used as the basis for the proposed revision requested in this application.

Furthermore, staff finds that the HI-STAR HB is composed of materials with a service proven history of use. The staff concludes that the materials and manufacture of Metamic as stated in the application for use in the HI-STAR HB are sufficient. The staff's conclusion regarding the manufacture, qualification, and use of Metamic, for the purposes of this review, is applicable to the HI-STAR HB. Definitions for Fuel Debris, Damaged Fuel Assemblies and Undamaged Fuel Assemblies, added to address the HB fuel assembly limited inspection records for packaging and transportation, are acceptable based on structural, containment, and criticality reviews.

The staff determined that, based on review of the application and supplements, that all material properties used by the applicant for the Model No. HI-STAR 100 for use at HB continue to meet the requirements of 10 CFR Part 71.

2.4 General Standards for All Packages

Section 2.I.4 of the supplement notes that the HI-STAR HB is a shorter and lighter version of the HI-STAR 100, and the design features presented in Section 2.4 of the application continue to apply to the HI-STAR HB. Therefore, the staff finds that the HI-STAR HB package meets the 10 CFR 71.43 requirements for the general standards for all packages.

2.5 Lifting and Tie-Down Standards

Section 2.I.5.1 of the supplement considers a bounding lifting weight of 161,200 lbs of the HI-STAR HB to recalculate the governing section moment and stresses for the trunnions with a slightly smaller diameter at the threaded portion of the trunnion than that for the HI-STAR 100. Considering the NUREG-0612 load multipliers of 6 and 10 for the yield and ultimate section load capacities, respectively, the application determines that all safety margins are greater than 1.0 This meets the 10 CFR 71.45(a) provision, which requires that lifting devices be sized to resist three times the design load without reaching material yield strength

Section 2.I.5.2 notes that the tie-down devices and the reaction loads in Section 2 5 of the application bound those for the HI-STAR HB, which is shorter and lighter than the HI-STAR 100 On this basis, the staff agrees with the applicant's conclusion that no new analysis needs to be performed for the tie-down devices for satisfying the requirements of 10 CFR 71 45 (b)(1)

Section 2.I.5.3 determines that the ultimate bearing capacity at the trunnion-to-top forging interface is greater than the trunnion load limit. The ultimate moment capacity at and beyond the interface is also greater than the trunnion moment limit. This demonstrates that the trunnion shank reaches ultimate structural capacities prior to the top forging reaching its corresponding ultimate load capacities. Therefore, the staff agrees with the applicant's conclusion that failure of the external shank of the lifting trunnion will not cause loss of any other structural or shielding function of the overpack. This satisfies the excessive load requirements of 10 CFR 71.45(a) and 10 CFR 71.45(b)(3).

2.6 Normal Conditions of Transportation

Heat and Free-Drop. Section 2.I.6.1 of the supplement notes that the operating temperatures for the Humboldt Bay fuel, which are at or below comparable temperatures for the HI-STAR 100 analyses, give relatively higher at-temperature stress allowables for the HI-STAR HB. It includes a bounding stress evaluation of the cask enclosure shell. On this basis, the applicant argued and the staff agrees that, for the same cask free-drop deceleration limits and design pressures/temperatures, all other stress analyses of the HI-STAR HB would have resulted in additional margins compared to those for the HI-STAR 100. Section 2.I.6.7 notes that, as part of the Section 2.I.7 impact limiter drop analyses, the cask decelerations for the 1-ft free drops for the HI-STAR HB are shown to be less than the HI-STAR 100 design basis limits. Hence, the applicant's evaluations satisfy the requirements of 10 CFR 71(c)(1) and 10 CFR 71(c)(7) for the heat condition and free drop tests, respectively.

Cold, Reduced External Pressure, Increased Internal Pressure, and Vibration. Section 2.I.6.3 of the supplement refers to the Section 2.I.6.1, "Bounding Evaluation of the Enclosure Shell," which is also applicable to the reduced external pressure condition. As a result, Sections 2.I.6.2 through 2.I.6.5 note and the staff agrees that no new analyses or other calculations need to be performed for the normal conditions of transport cold, reduced external pressure, increased external pressure, and vibration, respectively, to satisfy the requirements of 10 CFR 71(c)(2), (3), (4), and (5).

Water Spray, Corner Drop, and Compression. The HI-STAR HB is identical to the HI-STAR 100 in all respects except for the length of the overpack. As such, Sections 2.I.6.6, 2.I.6.8, and 2.I.6.9 note and the staff agrees that the respective 10 CFR 71(c)(6), (8), and (9) conditions of water spray, corner drop, and compression are not applicable to the HI-STAR HB package.

2.7 Hypothetical Accident Conditions

30-Ft Free Drop. Section 2.7 of the application provides an evaluation of 10 CFR 71.73 hypothetical accident conditions for the HI-STAR 100, which is being modified with limited design feature changes for the HI-STAR HB. Since the applicable design criteria and safety analyses for the HI-STAR 100 remain to be bounding for the HI-STAR HB, the staff focuses its review primarily on the Holtec approach of using analysis alone to establish the design basis cask decelerations associated with the impact limiter modifications.

Drawing No. C1765, sheet 2 of 7, of the application lists reduced nominal crush strengths of aluminum honeycomb materials for the five section types, which range from about 50% to 60% of the corresponding strengths of the HI-STAR 100. The configuration, including overall geometry and attachment to the overpack of the HI-STAR HB impact limiter, is identical to that of the HI-STAR 100, with the sole difference being the impact limiter crush material strengths.

There is no scale model drop testing of the HI-STAR HB. Contrary to the common practice of qualifying impact limiters of spent fuel transportation packages by drop tests, the applicant adapted the HI-STAR 100 differential equation method for evaluating the HI-STAR HB free-drop accidents only by analysis. Section 2.I.7.1 of the supplement provides a summary description of the method as used for the HI-STAR 100. Essentially, the method entails a combined analysis and testing approach in three steps: (1) the 1/8-scale quasi-static tests of the impact limiter to establish its load-deflection characteristics; (2) the ¼-scale cask drop tests to demonstrate that the experimentally obtained cask rigid body decelerations were below the design bases; and (3) the development of analytical models capable of correlating the observed and calculated results.

The HI-STAR 100 analytical models involve either a single second-order differential equation for end drops or a set of three differential equations for side and slap-down drops. There exist, in each equation, one or two resistive force terms each formulated as product of the impact limiter static deformation and dynamic multiplier, also called dynamic correlation function. To implement the method for the HI-STAR HB, given that all other relevant physical attributes of the cask system can be incorporated into the differential equations directly, the dynamic correlation function remains the only parameter that is updated, at each time integration step, as a linear function of the concomitant crush velocity of the impact limiter. In a September 30, 2008, letter the applicant addressed the staff RAI on appropriate selection of the dynamic multipliers for the HI-STAR HB without relying on scale model drop tests. As discussed in Section 2.I.7.1, Revision 13c, of the supplement, the applicant further clarified it by noting that, based on manufacturer's catalog, no information suggesting that the dynamic multipliers in the differential equation method are a function of crush material strength. Therefore, the staff has reasonable assurance to agree with the applicant that the dynamic multipliers originally determined for the HI-STAR 100 remain valid for the HI-STAR HB analytical method, as discussed below.

The numerical simulation analysis of the HI-STAR HB impact limiters uses the same dynamic multipliers as those for the HI-STAR 100, for which, analysis results correlate adequately with the drop test results. Compared to the HI-STAR 100 impact limiters, the staff notes that the HI-STAR HB aluminum honeycomb sections are identically configured, constrained, and supported by an essentially rigid backbone structure. This ensures development of similar load paths for the resistive forces in both the HI-STAR 100 and HI-STAR HB impact limiters for same dynamic multipliers Thus, the staff has reasonable assurance that the HI-STAR 100 differential equation method is an acceptable predictive tool for determining HI-STAR HB cask decelerations and impact limiter crush depths. The staff also agrees with the applicant's assessment that the HI-STAR HB impact limiters will continue to remain attached to the cask if the calculated maximum cask decelerations are below 60 g, given that the HI-STAR 100 impact limiters remained attached to the overpack during the scale model drop tests

Table 2.I 7.1 of the supplement lists calculated maximum cask decelerations and impact limiter crush depths for the 30-ft free drop accidents. With nominal strengths of the crush material, the maximum decelerations are 56.5 g, 45.6 g, 34.8 g, 33.8 g, and 45.9 g for the top-end, bottom-end, side, C.G.-Over-Corner, and slapdown drops, respectively. Table 2.I.7.3 lists the crush strength sensitivity analysis results for ten drop cases. For a crush strength increase of 15% over the nominal, the maximum decelerations are 59 g, 38.5 g, and 49.2 g for the respective top-end, side, and slapdown drops, which are all below the design basis limit of 60 g For the 30-ft side drop, for a decrease of crush strength by 15%, the calculated crush depth of 15.2 inch indicates that the impact limiters experience some material lockup. This results in a cask side-drop deceleration incrrease from the nominal 34.8 g rise to 43 g, as it would be. However, the impact limiters still provide acceptable protection for the cask subject to a maximum side-drop deceleration far less than the 60 g design basis limits.

On the basis of the review above, the staff concludes that the HI-STAR HB design basis decelerations are bounded by the 60 g used also for the HI-STAR 100 package evaluation. As such, all relevant Section 2.7.1 evaluations for the HI-STAR 100 remain to be applicable to the HI-STAR HB in demonstrating its structural capabilities for meeting the 10 CFR 71.73 (c)(1) free-drop requirements.

Puncture. Section 2.I.7.2 of the supplement notes and the staff agree that the structure at the puncture locations is unchanged from the HI-STAR 100. Hence, no new or modified calculations need be performed for the HI-STAR HB for meeting the puncture test requirements of 10 CFR 71.73(c)(3).

Thermal. Section 2.I.7.3 of the supplement notes the thermal evaluation of fire accident. The staff agrees with the applicant's assessment that no new or modified structural calculations need be performed to qualify the HI-STAR HB for the 10 CFR 71.73(c)(4) fire test requirements.

Immersion - Fissile Material and Immersion – All Packages. The staff evaluated the structural feature differences between the HI-STAR HB and HI-STAR 100 packages and concurs that no new or modified calculations need be performed to qualify the HI-STAR HB for the subject immersion requirements of 10 CFR 71.73(c)(5) and (6).

2.8 Fuel Rods

The Humboldt Bay fuel is shorter than the design basis fuel carried by the HI-STAR 100 and will, therefore, exhibit larger structural margins against the side- and end-drop accidents. Thus, the staff concurs with the applicant's conclusion that the performance of the HI-STAR HB fuel rods is bounded by that of the HI-STAR 100, and is, therefore, acceptable. Table 1.0.1 of the CoC defines undamaged fuel assemblies as those where all exterior rods in the assembly are visually inspected and shown to be intact even if the cladding of interior rods are of unknown condition. Section 2.I.9 of the supplement notes that the exterior fuel rods serve to confine the interior fuel rods, thereby preventing interior fuel rods from dislocating and falling to the bottom of the fuel basket This potential, but limited, reconfiguration of interior fuel rods of undamaged fuel assembly will result in negligible change of fuel mass and center of gravity height, which has insignificant effects on the structural response of the HI-STAR HB. Furthermore, as reviewed in Section 6.4 of this safety evaluation, reconfigured interior fuel rods are acceptable based on the conditions analyzed for criticality control This justifies loading the "undamaged fuel assemblies" into the MPC-HB directly without placing them in the HI-STAR HB damaged fuel containers.

2.9 Review Findings

The staff reviewed the statements and representations in the application by considering the regulations, appropriate Regulatory Guides, applicable codes and standards, and acceptable engineering practices The staff concludes that the structural design has been adequately described and evaluated for meeting the requirements of 10 CFR Part 71

3.0 THERMAL

The staff reviewed the application to revise the Model No. HI-STAR 100 package thermal design and evaluation to assess whether the package temperatures will remain within their allowable values or criteria for NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 3 (Thermal Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

3.1 Thermal Review Description

The purpose of this Revision is to facilitate the transport of HB reactor spent fuel; however, some of the changes are generic in nature and apply to the entire family of contents of the HI-STAR 100. Besides the fuel type change, other changes impacting the thermal evaluation are: the addition of Metamic as a neutron absorber material, the addition of a higher emissivity value for stainless steel plates, and the request to remove the thermal acceptance test along with the thermal periodic test.

3.2 Thermal Evaluation

The fuel from HB is low burnup fuel that has been sitting in the spent fuel pool and/or dry cask storage for many years, and as a consequence, the decay heat limit for this physically smaller package is only 2 kW for its associated 80 fuel assemblies. The current approved HI-STAR 100 thermal design limit for BWR fuel is 18.5 kW for 68 fuel assemblies.

Additionally, the applicant has chosen to add Metamic as an additional neutron absorber in the SAR. Since Metamic is made of the same materials as Boral, it exhibits the same thermal conductivity, even though the manufacturing process is different. As a consequence, the same values of thermal conductivity for Boral are used.

A higher value emissivity (0.587) for stainless steel plates was provided, but it was stated, for conservatism, that a lower value of 0.36 continues to be used in the thermal calculations. The staff was concerned that this higher emissivity value may not be conservative for the HAC fire. However, after reviewing the Holtec response, the staff agrees that the starting point for the temperature distribution within the cask would be lower, and as a consequence, the impact of the 30 minute HAC fire would be alleviated by this and the thermal inertia of the massive transportation cask.

From the thermal analysis for HB, the NCT maximum temperature for the fuel cladding is 419°F and the overpack top plate is 129°F These values are considerably less than the design basis HI-STAR 100 BWR fuel of 713°F for the fuel cladding and 162°F for the overpack closure plate Also, since the decay heat load of the fuel for HB is significantly lower than the design basis heat load of the HI-STAR 100 and the HB overpack utilizes a heat shield, the HB thermal loading is bounded by the current design basis of the HI-STAR 100 for NCT and HAC

At the request of the staff, a description of the heat shield (referred to in Table 3 I.5) was added to SAR supplement Section 3.I.1. The heat shield is another term for the neutron shield where the previous radial support channels for the outer enclosure shell of the neutron shield were replaced with gussets at the top and bottom of the shield - so that only neutron shield material would be at the radial centerline of the cask to further inhibit heat input to the fuel during the HAC fire

Additionally, the applicant initially requested that the thermal acceptance test and the thermal periodic test be removed for all package configurations. Since the thermal acceptance test was limited to the first fabricated HI-STAR overpack, and that test was completed satisfactorily, the staff has no objection in removing this commitment. For the periodic thermal test, the staff is confident in the time dependent thermal performance of the HI-STAR 100 packaging materials except for the shielding material Holtite-A, and requested that its time dependent thermal characteristics be evaluated since Holtite-A is a polymer and, as such, is typically susceptible to heat and radiation degradation. In response to the staff's request HOLTEC submitted two reports entitled:

1) "Holtite-A: Results of Pre- and Post- Irradiation Tests and Measurements," HI-2002420, Rev. 1, dated 4/8/03, and

2) "Holtite-A Development History and Thermal Performance Data," HI-2002396, Rev. 3, dated 4/10/03.

The former report documents irradiation aging and performed physical observations; weight and density determinations; dimensional measurements; and neutron attenuation testing or chemical analyses on the Holtite-A samples. The latter report documents that the thermally aged samples of Holtite-A were free of large voids and gaps; visual examination confirmed material was stable (no warping, swelling, or cracking); and that weight loss results were less than 4%. Also, these tests were performed independently and did not evaluate the synergistic effects of dose and temperature. The staff did not find a direct correlation between these reports measurements/conclusions and providing assurance that the thermal conductivity of Holtite-A does not change with time. As a result, the staff requested Holtec to provide additional data of Holtite-A to determine the time dependent effect of radiation and heat on Holtite-A, or continue performing the periodic thermal test. Holtec chose to continue performing the periodic thermal test. This commitment is included in the CoC since the CoC requires that all procedures for acceptance testing and maintenance be developed from the provisions of SAR Chapter 8.

From a review of the proposed application of HB, the staff concludes that adequate justification has been presented to conclude that the material temperature limits of the HI-STAR HB transportation package have been satisfied considering the large design margins stemming from the relatively lower decay heat load and the approximate 300°F difference between the HAC cladding temperature and its limit of 1058°F

3.3 Conclusion

Based on the review of the application, the staff found reasonable assurance that the applicant has demonstrated that the HI-STAR HB package meets the thermal requirements for NCT and HAC as required by 10 CFR Part 71

4.0 CONTAINMENT

The staff reviewed the application to revise the Model No HI-STAR 100 package to verify that the package containment design has been described and evaluated under NCT and HAC as required in 10 CFR Part 71 This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 4 (Containment Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

4.1 Containment System Design

The purpose of this Revision request is to facilitate the transport of HB reactor spent fuel. Besides the fuel type change, other changes that could impact the containment evaluation are: the revised A_2 values from 10 CFR Part 71, the removal of the MPC as a secondary containment (including the commitment to leak test it), updated procedures to change the closure bolt torque to 2000 ft-lbs from 2895 ft-lbs, and the changed definition of "damaged fuel assembly."

4.2 Containment Evaluation

The HB spent fuel is bounded by the design basis fuel for source term, and consequently, its reference leak rates are more than that of the design basis fuel and result in no impact on the previously approved HI-STAR 100 package. Furthermore, the reference leak rate calculations, presented in Table 4.1.1, "Summary of Containment Boundary Design Specifications," of the SAR continue to justify a leakage rate acceptance criterion of 4.3 E-6 atm cm^3/sec, He. This leakage rate acceptance criterion has been verified to be valid for the HB fuel, as well as for the changes in the A_2 values.

Since the double containment requirement for plutonium shipments has been removed from the regulations (ref. 10 CFR 71.63 dated 1/26/2004), the requirement to have the MPC serve as a second containment boundary and be leak tested is no longer required and that information has been removed from the SAR.

The lessening of the closure bolt torque values has been reviewed in the structural section and staff agrees that these new torque values still provide an adequate closure force to ensure integrity of the containment boundary.

The change in the definition of a damaged fuel assembly has no effect upon the containment evaluation because it has no effect on the source term.

Also, the latest version of ANSI N14.5 (i.e . 1997) was referenced by the applicant at the request of staff

4 3 Conclusion

Based on the statements and representations in the application, staff agrees that the applicant has shown that the use of the Model No. HI-STAR 100 for use at HB continues to meet the containment requirements of 10 CFR Part 71

5.0 SHIELDING

The staff reviewed the application to revise the Model No HI-STAR 100 package to verify that the shielding design has been described and evaluated under NCT and HAC. as required in 10 CFR Part 71 This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 5 (Shielding Review) of NUREG-1617. "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel "

5 1 Description of Shielding Design

5.1.1 Packaging Design Features

Staff reviewed the changes to the design features proposed in the amendment request. The following changes were proposed that affect the shielding design:

Authorization of Metamic as a neutron absorber material was added to the application for use in the HI-STAR HB. The shielding analysis takes credit for the presence of the absorber plates. The currently approved design uses Boral as the neutron absorber.

Addition of the HB overpack, MPC, and basket (and HB-specific DFC). The design for the HB system differs from the standard HI-STAR 100 system in a few parameters. A significant difference is the reduced axial height of the overpack and the MPC. The shielding materials are unchanged. An additional difference is the lack of radial channels welded to the outermost intermediate steel shell of the HB overpack; thus, the neutron shield is not penetrated by steel channels that will result in neutron streaming paths through the neutron shield. Additionally, the HB overpack neutron shield thickness is a minimum of 4 inches, while the minimum thickness is 4.3 inches for the standard overpack. The MPC-HB basket holds 80 HB fuel assemblies (versus the 68 assemblies of the standard BWR MPC baskets).

No other proposed changes affect the shielding design. The staff reviewed the licensing drawings and descriptions of the HI-STAR 100 system as modified in the proposed amendment and finds there is sufficient detail to perform a shielding evaluation. The staff reviewed the proposed changes and finds them to be described in sufficient detail to perform an evaluation of the shielding design.

5.1.2 Codes and Standards

The applicant continues to use the flux-to-dose conversion factors from ANSI 6.1.1-1977. The staff finds use of these conversion factors to be acceptable.

5.1.3 Summary Table of Maximum Radiation Levels

The summary dose rate tables for the MPC-24, MPC-32, and MPC-68 are provided in Section 5.5 of the amended SAR. Some of the dose rates for all three MPCs were modified due to a modification of the impact limiter model used in the analyses. Dose rates for the MPC-68 were modified to account for the proposed additional contents at dose locations where the proposed additional contents result in higher dose rates. The tables show maximum dose rates for the side, top, and base of the HI-STAR 100 to be below the 10 CFR Part 71 regulatory dose rate limits. Staff reviewed the dose rates presented in Section 5.5 of the SAR and finds they are consistent with those reported in the rest of the analysis and are the maximum dose rates

5.2 Source Specification

The staff reviewed the specifications for the contents proposed in the amendment request. No changes were proposed to the allowable PWR contents. Proposed changes to the BWR contents include a decreased minimum allowable enrichment for intact and damaged 6x6A, 6x6C, 7x7A, and 8x8A assembly arrays/classes and fuel debris from these assembly classes (the proposed minimum enrichment is 1.45 wt.%) and two new maximum burnup, minimum cooling time, and minimum enrichment combinations for the remaining ZR-clad BWR assembly arrays/classes. One new combination is a maximum burnup of 10,000 MWd/MTU and a minimum of 5 years cooling for assemblies with a minimum enrichment of 0.7 wt.%, and the other combination is a maximum burnup of 20,000 MWd/MTU and a minimum of 7 years cooling for assemblies with a minimum enrichment of 1.35 wt.%.

The amendment also proposes to remove the HB assembly arrays/classes from the allowable contents of the MPC-68 and MPC-68F and allow transportation of these assemblies in the MPC-HB. The amendment proposes to allow transportation of the HB assemblies, 6x6D and 7x7C assembly arrays/classes, as intact, undamaged and damaged assemblies and fuel debris. Damaged fuel assemblies and fuel debris must be loaded in HB Damaged Fuel Canisters (DFC). The applicant proposes to load either 28 damaged assemblies in the peripheral basket locations or 40 damaged assemblies in a checkerboard pattern. These two configurations are illustrated in SAR Figures 6.I.3 and 6.I.4.

Based upon HB fuel records, there are no more than 337 linear inches of fuel fragments remaining at the HB reactor site. This fuel debris may include loose zirconium-clad pellets, stainless steel-clad pellets, unclad pellets and rod segments. Assuming all the fragments are clad with stainless steel and considering the cladding dimensions, the amount of stainless steel cladding that could be loaded is 1.25 kilograms. Therefore, the applicant proposed a limit for fuel debris of 1.5 kilograms of stainless steel cladding per cask. This limit is included in Section VII of Table A.1 in Appendix A to the CoC. The shielding analysis also includes the contribution of the steel cladding to the source terms used to demonstrate compliance with the regulatory dose rate limits. The design-basis HB assembly is the 6x6D class assembly with the characteristics provided in Table 5.I.1 of the SAR.

A definition for undamaged fuel was proposed to be added to the CoC. This definition is strictly limited in application to HB fuel assembly classes/arrays. Undamaged assemblies are those HB assemblies for which the condition of the assembly could not be completely verified to meet the CoC definition of intact fuel; however, visual examinations of these assemblies supported the determination that the outer rods of the assemblies are intact. These examinations also confirmed that the assembly interior rods are in place but could not confirm the condition of the cladding of the interior rods Thus, any changes to the assembly configuration due to accident conditions would only take place in the assembly interior and be confined by the outer rods to the assembly envelope (see Section 2.3.12 of this SER). The shielding analysis considers assemblies of this condition (see Section 5.4 of this SER).

5.2 1 Gamma Source

The applicant used the SAS2H and ORIGEN-S modules of the SCALE suite of codes to calculate the source terms, both neutron and gamma, for the proposed contents. SAR Table 5.2.5 lists the gamma source strength for design-basis ZR-clad BWR assemblies for the different maximum burnup, minimum cooling time, and minimum enrichment combinations analyzed. Table 5.2.6 lists the gamma source strength for the Dresden 1 assembly arrays/classes Table 5.1.3 lists the gamma source strength for the HB assembly arrays/classes Table 5.2.10 lists the Cobalt-60 source strength per assembly from assembly hardware, with Table 5.1.4 as the equivalent table for the HB assembly hardware. The staff reviewed the gamma source strengths provided in these tables and also performed confirmatory calculations of the gamma sources. Based upon its review and calculations, the staff finds the calculated source strengths to be acceptable for the proposed contents.

5.2.2 Neutron Source

The neutron source strengths for the design-basis ZR-clad BWR assemblies, the Dresden 1 assemblies, and the HB assemblies are provided in SAR Tables 5.2.13, 5.2.14, and 5.1.2, respectively. The staff reviewed these neutron source strengths and performed confirmatory calculations of the neutron sources. Based upon its review and calculations, the staff finds the calculated source strengths to be acceptable for the proposed contents.

5.3 Model Specification

The applicant uses the same shielding models as for analyses performed in the previously approved amendment, with the exception of a correction to the thickness of the impact limiter ribs. Sections 2 and 3 of this SER describe staff's evaluations of the structural and thermal performance of the HI-STAR 100 system as proposed in the amendment. None of the proposed changes were found to exceed the bounding conditions affecting shielding as evaluated in the previously approved amendment. Additionally, the shielding configuration of the HI-STAR HB is generally bounded by the shielding configuration of the design-basis HI-STAR 100. The neutron shield is 0.3 inches thinner for the HB overpack than it is for the standard HI-STAR 100 overpack, a condition which increases the neutron dose rates by about 30%. The applicant addressed this difference in the neutron shield in its evaluation of the HB system. Also, while the amendment proposes to allow use of Metamic as a neutron absorber, in addition to Boral, the models continue to use Boral. The modeled Boron-10 areal densities are the same and the plate thicknesses are essentially the same for the two absorbers; thus, the amount of aluminum and B_4C are essentially the same, resulting in no distinction between the two materials from a shielding perspective. Based on its review of these proposed changes, the staff finds that the shielding models, with the correction to the impact limiters, remain appropriate.

5.4 Evaluation

The applicant performed the shielding analysis with MCNP-4A, the same code the applicant used for the shielding analyses in the previously approved amendment. The applicant calculated the dose rates for design-basis BWR fuel with the proposed maximum burnup and minimum cooling time and enrichment combinations. These dose rates are presented in SAR Tables 5.4.9, 5.4.11, and 5.4.13 for the surface dose rates and two-meter dose rates for normal conditions and one-meter dose rates for accident conditions. The correction of the impact limiter thickness in the model affected the normal conditions dose rates at the cask base and top (surface and two-meter distance); therefore, updated dose rates for normal conditions are provided for all the MPCs in SAR Tables 5.4.8-11, 19, 20, 22-24, 26, 27, 29, 30, 32, and 33. These changes also necessitated updates to the summary tables, SAR Tables 5.5.1-3, used by the applicant to show compliance with the 10 CFR Part 71 regulatory dose limits. The maximum dose rates, however, continue to remain below the regulatory limits

Dose rates were not calculated for the Dresden 1 assemblies with the proposed minimum enrichment. Instead, the assembly neutron and gamma source strengths were compared on a source strength per inch basis with the source strengths of the design-basis BWR fuel assemblies (39,500 MWd/MTU and 14 years cooling). The comparison was made for damaged Dresden 1 assemblies that had reconfigured under accident conditions and was initially done using only the total neutron and gamma source strengths. This comparison showed that the design-basis BWR fuel source bounds the Dresden 1 fuel source. The applicant then also compared the source strengths on an energy-group basis. This second comparison showed that the design-basis neutron and gamma source strengths bound those of the damaged Dresden 1 assemblies over all energy groups except the 1.0 to 1.5 MeV gamma source. However, dose rate calculations showed that the dose rates (on the overpack side) from the damaged Dresden 1 assemblies are significantly lower than those from the design-basis intact BWR assemblies (by about 20%). The applicant concludes, therefore, that the dose rates from the Dresden 1 fuel assemblies will always be bounded by the dose rates from the design basis BWR fuel assemblies. The staff reviewed this information and performed a confirmatory analysis and finds that the dose rates from the design-basis intact BWR assemblies bound the dose rates from the Dresden 1 fuel assembly arrays/classes.

The applicant used a similar source term comparison (on an energy-group basis) between the design-basis intact BWR assemblies and the HB assemblies. In this case, however, since the MPC-HB and MPC-68 hold different numbers of assemblies, the source per inch was multiplied by the number of assemblies in the respective MPC. For accident conditions, the MPC-HB source per inch accounted for reconfiguration of the damaged HB assemblies. The comparison was made with design-basis intact BWR assemblies having a burnup of 24,500 MWd/MTU and 8 years cooling. The applicant's comparison indicated that the MPC-HB source strength over all energy groups (both neutron and gamma) is bounded by the source strength of the design-basis intact BWR assemblies, meaning the HI-STAR HB dose rates are bounded by the HI-STAR 100 containing a MPC-68 loaded with design-basis intact fuel.

This initial comparison looked solely at comparisons for the entire cask loading. However, staff questioned whether this comparison was sufficient given the different loading schemes for damaged fuel in the HB overpack and the relative importance of different MPC basket zones on the dose rates (interior versus exterior basket locations). For example, for side dose rates, fuel assemblies in the outermost rings of basket locations tend to dominate the gamma dose rates. This configuration will provide a better indication of the impact on dose rates of the 28 damaged assembly pattern in the HB overpack. In response to staff's questions, the applicant modified its evaluation to account for the relative importance of the different MPC basket zones to dose rates. The modified evaluation continued to indicate that the HI-STAR HB dose rates are bounded by the HI-STAR 100 containing the MPC-68 The modified evaluation also accounts for the difference in neutron shield thickness, increasing the neutron source by 30%, the assembly hardware, and the stainless steel cladding of fuel debris in the DFCs For accident conditions, the applicant performed further comparisons with the assumption that all assemblies in the HI-STAR HB are damaged. This latter comparison was performed as a bounding approach for addressing the condition of assemblies where the condition of the cladding could not be verified for fuel rods interior to the assembly lattice These assemblies are classified as undamaged, which classification applies only to the HB fuel arrays/classes. This comparison also indicates the dose rates from the HI-STAR HB are bounded by the dose rates from the HI-STAR 100 containing the MPC-68

The staff reviewed this information and also performed its own comparisons of the source terms. The comparisons included accident conditions for both allowable loading configurations of damaged fuel and the presence of undamaged fuel as well as a comparison of the source terms for assemblies loaded in equivalent outer basket cell regions, under both normal and accident conditions. Based on its review of the applicant's analysis and its own comparisons, the staff finds that the dose rates from the HI-STAR HB will be bounded by the dose rates from a HI-STAR 100 loaded with design-basis intact BWR fuel since the HI-STAR HB source strength, on a per inch basis, is bounded by that of the HI-STAR 100 containing design-basis intact BWR fuel in all the evaluated comparison scenarios.

5.5 Conclusion

Based on its review of the information and representations provided by the applicant in the amendment request and the SAR and independent analyses, the staff has reasonable assurance that the changes to the package design and contents satisfy the shielding requirements and dose limits in 10 CFR Part 71.

6.0 CRITICALITY

The purpose of this review is to verify that the proposed amendment meets the criticality safety requirements under normal conditions of transport and hypothetical accident conditions. The objectives include a review of the criticality design features and fuel specifications; review of the configuration and material properties for the HI-STAR HB Overpack; and a review of the methodology and results found in the criticality evaluation.

The staff reviewed the criticality safety analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that the HI-STAR HB with the MPC-HB basket configuration meets the following regulatory requirements: 10 CFR 71.31, 71.33, 71.35, and 71.59. The staff's review also involved a determination on whether the cask system fulfills the acceptance criteria listed in Section 6 of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

6.1 Description of Criticality Design

6.1.1 Packaging Design Features

The MPC-HB is a fuel basket with an increased capacity to accommodate up to 80 assemblies while maintaining the same MPC outer diameter. The criticality safety design continues to rely on the geometry of the fuel basket, fuel enrichment limits, and poison plates for criticality control, but has added Metamic as an alternative to the Boral poison plates

Results of the structural and thermal analyses show that the packaging design features important to criticality safety are not adversely affected by the tests specified in 10 CFR 71 71 and 71.73. The staff reviewed the description of the package design and found that the important features were appropriately identified and adequately described The engineering drawings and other information are sufficient to permit an independent evaluation

6.1.2 Summary Table of Criticality Evaluations

A summary of the criticality evaluation results for the HB fuel is reported in Table 6.I.1 of the SAR. The table includes results for a single package and arrays of undamaged and damaged packages including damaged fuel and fuel debris being transported in the MPC-HB.

The results show that the package design meets the requirements of 10 CFR Part 71 for criticality safety. All values of k_{eff}, after being adjusted for uncertainty and biases, fall below the acceptance limit of 0.95 given in the Standard Review Plan (SRP).

6.1.3 Criticality Safety Index

The applicant's analyses considered an infinite array of packages under both normal conditions of transport and hypothetical accident conditions and showed that they were below the acceptable limit. Therefore, the Criticality Safety Index is 0.0 for the package.

6.2 Spent Nuclear Fuel Contents

The applicant requested the addition of the 6x6D and 7x7C fuel assembly types with or without channels in the MPC-HB for transport in a modified (shorter) HI-STAR HB. The maximum assembly average enrichment is 2.6% for intact fuel, undamaged fuel, damaged fuel, and fuel debris. Undamaged fuel assemblies are assemblies where all of the exterior rods are shown to be intact; while the interior rods of the assembly are in place the condition of the cladding on these rods cannot be verified. Damaged fuel and fuel debris must be canned in a Holtec designed Damaged Fuel Container (DFC). Damaged fuel and fuel debris may be loaded into the 28 peripheral cell locations in the basket or into 40 cell locations in a checkerboard pattern in the basket.

The staff has reviewed the description of the spent fuel contents and concludes that it provides an adequate basis for the criticality evaluation.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The applicant used the same general assumptions and modeling methods as previously reviewed for intact fuel, damaged fuel, and fuel debris. Notable modeling features used for this application are: (1) modeling the MPC-HB with 80 fuel assemblies and a shorter length than the current canister and transportation overpack, (2) analyzing variations of assembly positioning within the fuel basket cells where the fuel assemblies are shifted toward the basket center as well as centered in the fuel cells, (3) analyzing the damaged fuel and fuel debris as bare rods of fuel, (4) modeling undamaged fuel assemblies in place of intact assemblies to analyze the effects on reactivity for both approved basket configurations (peripheral and checkerboard), (5) modeling cases with Metamic and Boral plates separately to assess their analytic similarity, and (6) simulating fabrication damage to the poison plates as a hole up to 1-inch diameter

6.3.2 Material Properties

The material properties remained the same as previously reviewed except for the addition of Metamic where 90% credit was taken for the minimum boron content in this type of absorber plate.

6.3.3 Computer Codes and Cross Section Libraries

The applicant used the three-dimensional continuous energy code Monte Carlo N-Particle (MCNP4a) for the criticality analysis. The staff agrees that the codes and cross-section sets used in the analysis are appropriate for this application and fuel system.

6.3.4 Demonstration of Maximum reactivity

In the safety analysis, the applicant used fuel and basket dimensions within the tolerance limits which maximize k_{eff} as were found in previous analyses. These optimum conditions are: maximum active fuel length, maximum fuel pellet diameter, minimum cladding outside diameter, maximum cladding inside diameter, minimum guide type thickness, maximum channel thickness, minimum cell pitch, minimum cell inner dimensions, and nominal cell wall thickness.

The applicant performed a sensitivity analysis for the HB fuel and found that a package was more reactive when all fuel assemblies are shifted toward the center of the basket versus being centered in each cell but found no statistically significant difference between the cases of each poison plate being damaged with a 1 inch diameter hole in its middle versus an undamaged plate. In the final safety analyses, the applicant assumed all fuel assemblies were shifted toward the basket center and that all poison plates were damaged with a 1 inch diameter hole in the center.

Staff found the methods used to identify the parameters values which maximize k_{eff} to be appropriate and found the set of parameters used in the analysis to be acceptable.

6.3.5 Analysis Approach

The applicant performed comparative calculations for the two different HB fuel types and determined that the 6x6D was more reactive than the 7x7C assembly type. Subsequently, the

6x6D was used as the intact fuel type when analyzing a transportation package loaded with bounding combinations of intact, undamaged, and damaged fuel assemblies

In addition to analyses to support the inclusion of HB fuel in the HI-STAR 100, the applicant also analyzed the acceptability of the alternative poison plate material. Analyses were performed with Metamic poison plates and then compared to a cask fabricated with Boral poison plates In the criticality analysis, 75% of the minimum B-10 content is credited for Boral while 90% of the B-10 content is credited for the Metamic. The B-10 content in Metamic is chosen to be lower, and is chosen as the B-10 content so that both materials are the same in the analysis The difference in k_{eff} was not statistically significant

6 4 Single Package Evaluation

The applicant performed calculations for the MPC-HB which show the design is more reactive when internally flooded with full density water versus flooding with lower density water. Thus, a fully flooded package was used in the subsequent single package analyses. The applicant then performed a series of calculations for single HI-STAR HB packages containing the MPC-HB with 6x6D and 7X7C fuel for each case of an unreflected package, full reflection by water, and full reflection of the containment only. This was performed for two different cases: one with either all intact or undamaged assemblies; and one with damaged assemblies loaded with intact or undamaged assemblies. The bounding case was for the model with the damaged fuel (7x7 rod array) and the 6x6D undamaged fuel (in intact assembly positions) in an eccentric assembly positioning with assumed poison plate damage. The subsequent single package calculations used the results for a single unreflected package with full internal moderation but without the assumed poison damage. All of the applicant's results were below the acceptance level of 0.95 for k_{eff}.

6.5 Evaluation of Package Arrays Under Normal Conditions of Transport

The applicant performed calculations for an infinite array of undamaged packages containing the MPC-HB with intact and undamaged assemblies. All of the applicant's results were below the acceptance level of 0.95 for k_{eff} In these calculations, the packages were internally dry and had no moderator between packages.

6.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

The applicant performed calculations for an infinite array of damaged packages containing the MPC-HB with the limiting case of full internal moderation as found for the single package. All of the applicant's results were below the acceptance level of 0.95 for k_{eff}.

6.7 Benchmark Evaluations

The applicant's benchmarking procedures and methods have not changed and staff's evaluation is provided in a previous SER.

6.8 Evaluation Findings

The staff has reviewed the criticality description and evaluation of the package and concludes that it addressed the criticality safety requirements of 10 CFR Part 71.

The applicant analyzed the reactivity involving approved loading scenarios. The evaluation analyzed the effects of intact fuel, undamaged fuel, and damaged fuel loaded into peripheral and checkerboard loading configurations within the MPC-HB. A number of conservatives were used when determining the bounding condition. Reactivity results were evaluated for both approved loading configurations (peripheral and checkerboard) having bounding combinations of intact, undamaged, and/or damaged fuel.

Section 6.I.4.1 of the application states that assemblies with defects are considered damaged. and need to be placed into damaged fuel containers (DFCs). Calculations were performed in the application to ensure that assemblies with intact rods in the outermost rods but which may also have defects in the inner rods could be loaded without being placed into DFCs As part of the evaluation the rod pitch was varied inside the assembly to analyze the effects of reflection by water. The application shows that loading the MPC-HB with undamaged assemblies (in place of intact assemblies) along with damaged assemblies still yields results which are below the acceptance level of 0.95 for k_{eff}. Staff agree that the undamaged fuel assemblies with parameters described in the SAR report are acceptable to be loaded in the MPC-HB without being placed in a DFC as long as the overall integrity of the assembly (outer intact rod positioning, guide plates, etc.) remains structurally intact (See Section 2.3.12 of this SER).

Staff identified a minor error in the criticality evaluation regarding a supplemental report provided by the applicant (Holtec Report No: HI-2033010). Table C-1 stated that the bounding condition used for the 6x6 array with full water reflection came from C-55, which corresponds to a case where the model is centered. This is not consistent with the criticality evaluation found in Chapter 6 of the SAR, which states that the bounding case included the assembly being modeled in an eccentric manner. A telephone conference was conducted in which the applicant verified the assumption of NRC staff that it was merely a typographical error, and the applicant plans to correct this in the next revision of the report.

It should be noted that staff finds the 6x6D and 7x7C fuel assemblies, having parameters listed in the SAR, acceptable for loading into the MPC-HB. This is based on the conservatisms used in the analysis as well as the bounding reactivity in relation to the acceptance level of 0.95.

6.9 Conclusion

Based on the review of the presentations and information supplied by the applicant, staff finds reasonable assurance that the proposed amendment meets the criticality safety requirements of 10 CFR Part 71.

7.0 Package Operations

As part of the amendment, the applicant proposed a significant revision of the package operations. Instead of including the package operations explicitly in the CoC, the applicant proposed to place this information in Chapter 7 of the SAR and incorporate this chapter into the CoC by reference. Several items in Chapter 7 were modified as part of the proposed change in addition to inclusion of operations for the HI-STAR HB. Initially, the chapter included operations descriptions for dry loading as well as wet loading operations. However, due to concerns such as how dry loading operations would be performed and/or limited to prevent fuel oxidation, the applicant removed these operations.

The operations chapter also included a statement that indicated that HI-STAR 100 users could modify the sequence of, add or remove operations as necessary This statement introduces uncertainty into what constitutes the essential package operations, the delineation of which is the purpose of Chapter 7. In response to staff's questions, the applicant removed the statement Also, for operations for which the sequence does not impact the package preparation, the applicant modified Chapter 7 to explicitly indicate the operations which may be performed in any sequence in relation to each other. Staff also noted several important operations and the completion/acceptance criteria for other operations were removed from the descriptions in Chapter 7 as part of the proposed amendment. The applicant restored these descriptions in response to staff's questions

Based upon its review of the descriptions in the application, the staff finds that the package operations meet the requirements of 10 CFR Part 71 and that the operations are adequate to assure the package will be operated in a manner consistent with its evaluation for approval

8.0 Acceptance Tests and Maintenance Programs

As part of the amendment, the applicant proposed a significant revision of the acceptance test and maintenance program. Instead of including this information explicitly in the CoC, the applicant proposed to place this information in Chapter 8 of the SAR and incorporate this chapter into the CoC by reference. A number of items in Chapter 8 were modified as part of the proposed change. The acceptance tests and maintenance program is fully applicable to the HI-STAR HB without modification.

In its review of the Chapter 8 descriptions, the staff found that certain important acceptance criteria and tests related to shielding materials and shielding effectiveness had been removed or modified. In particular, the requirements for gamma shielding materials were not included in the proposed Chapter 8. However, staff considers that acceptance tests and criteria for all shielding materials should be included in the Chapter 8 program descriptions. The applicant modified the neutron shield acceptance test and periodic neutron shield integrity verification test to consist only of the radiation surveys performed prior to transport (and upon package receipt, for the periodic test) as described in Chapter 7 of the SAR. Staff considers that these surveys do not meet the purposes of the acceptance test and periodic integrity verification test of the neutron shield. The pre- and post-transport surveys only serve to ensure the 10 CFR Part 71 dose rate limits are not exceeded for a particular shipment. The acceptance and periodic verification tests ensure that the as fabricated neutron shield performs as designed, by comparing dose rates for given contents with the dose rates estimated by analysis for the same contents. Survey results that differ from estimated results, accounting for uncertainties in the measurements and the calculations, would indicate a problem with the as fabricated neutron shield. The tests in the currently approved CoC, fulfill these conditions. Based upon staff's considerations, the applicant included the acceptance tests and criteria for all shielding materials and modified the proposed acceptance and periodic neutron shield tests to retain the currently approved tests.

For the neutron shielding acceptance tests, the applicant describes two tests. The first test verifies shield integrity and is performed with either a check source (prior to first use) or the loaded contents at first use. This test examines the entire surface of the neutron shield, including the impact limiters. The measurements are compared with calculated values that are representative of either the check source or the loaded contents at first use. The second test, described as a shielding effectiveness test, is performed after the first fuel loading in a similar manner to the first test, except that the test does not cover the entire shield surface. This second test is performed when a check source is used for the shield integrity test Staff has reviewed these acceptance tests and finds that these tests are acceptable. This finding is based upon the verification of the performance of the cask's entire neutron shield, with the test measurements being compared versus calculated values for the respective test source

The applicant, in response to staff's questions, also modified the proposed visual inspections in the acceptance tests in a manner that clarified the tests and their acceptance criteria

Based upon its review of the descriptions in the application, the staff finds that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71 and that the maintenance program is adequate to assure packaging performance during its service life.

CONDITIONS

The CoC has been revised as follows

Condition No. 5(a)(2):

A packaging description HI-STAR HB was added.

Condition Nos. 5(a)(3):

Nine drawings were revised.

Condition No. 5(b)(1):

The definitions of damaged fuel, damaged fuel containers, and fuel debris were modified. The definition of undamaged fuel was added to account for the fuel specific to HB.

Condition Nos. 6.(a) and 6.(b):

Revisions were made to reference SAR Chapters 7 and 8 in the CoC.

Condition No. 7:

Revision to add the maximum gross weight of the HI-STAR HB to the CoC.

Appendix A:

Revisions were made to add some cooling times, burnups, and initial enrichments, add the fuel specifications for the HB fuel, and provide some clarifications.

Condition No. 20:

Allows the use of Revision 6 of this certificate for one year.

CONCLUSION

The staff has reviewed the requested amendment to Certificate of Compliance No. 9261 Based on the statements and representations in the application, as supplemented, and Revision 12 of the SAR, the staff concludes that the requested changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71. Certificate of Compliance No. 9261 for the HI-STAR 100 transport package has been amended as requested by Holtec International

Issued with Certificate of Compliance No 9261, Revision No 7
on May 8, 2009

NRC FORM 618

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

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

4. CONDITIONS

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

5. (a) Packaging

(1) Model No.: NAC-STC

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5.(a)(3) Drawings

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5.(b) Contents

(1) Type and form of material

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

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

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

1 a CERTIFICATE NUMBER	b REVISION NUMBER	c DOCKET NUMBER	d PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	11	71-9235	USA/9235/B(U)F-96	7	OF	12

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

FUEL COOL TIME TABLE
Minimum Fuel Cool Time in Years

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

Notes:
(1) - Framatome-Cogema 17x17 fuel only.

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

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

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

Notes:

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

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

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

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

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

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

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

Notes:

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

5.(b) Contents (Continued)

(2) Maximum quantity of material per package

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

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

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

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

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

5.(c) Criticality Safety Index: 0.0

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

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

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

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

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

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

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

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

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

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

12 Transport by air is not authorized.

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

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

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

16 Expiration date May 31, 2014

REFERENCES

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

As supplemented June 3, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

B. [signature] for

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

Date: June 12, 2009

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555-0001

SAFETY EVALUATION REPORT

Docket No. 71-9235
Model No. NAC-STC Package
Certificate of Compliance No. 9235
Revision No. 11

SUMMARY

On May 29, 2009, the U.S. Nuclear Regulatory Commission, issued Revision 10 to Certificate of Compliance (CoC) No. 9235 for the Model No. NAC-STC package. After issuing Revision 10 of CoC No. 9235, four typographical errors were identified in the title headings by the applicant in an e-mail dated June 3, 2009. On pages 3, 4, and 9, the Revision number in block b. was incorrectly left as 9. It should have been 10. Also, page 4 of 12 is incorrectly identified as page 4 of 122. The errors have been corrected. Changes made to the enclosed CoC are indicated by vertical lines in the margin.

CONDITIONS

No Conditions were changed. The typographical errors in the title headings have been corrected. The references were changed to add the June 3, 2009, supplement.

CONCLUSION

These changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9235, Revision No. 11, on June 12, 2009.

NRC FORM 618
(8-2000)
10 CFR 71

U.S NUCLEAR REGULATORY COMM SSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

2 PREAMBLE

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

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

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

a ISSUED TO *(Name and Address)*

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

b TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics application dated
January 6, 2009

4 CONDITIONS

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

5

a. Packaging

(1) Model No.: GA-4

(2) Description

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

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

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

5.a. (2) (continued)

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

Cask

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

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

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

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

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

5.a. (2) (continued)

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

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

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

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

5.(b) Contents

(1) Type and Form of Material

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

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

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

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

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

5.b. (1) (continued)

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

(2) Maximum Quantity of Material per Package

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

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

Table 1 - PWR Fuel Assembly Characteristics

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

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

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

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

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

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

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

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

(2) That before shipment the licensee shall:

(a) Perform a measured radiation survey to assure compliance with 49 CFR 173.441 and 10 CFR 71.47 and assure that the neutron and gamma measurement instruments are calibrated for the energy spectrums being emitted from the package.

(b) Verify that measured dose rates meet the following correlation to demonstrate compliance with the design bases calculated hypothetical accident dose rates: 3.4 x (peak neutron dose rate at any point on cask surface at its midlength) + 1.0 x (gamma dose rate at that location) ≤ 1000 mR/hr.

(c) Verify that the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87.

7.a.(2) (continued)

(d) Inspect all containment seals and closure sealing surfaces for damage. Leak test all containment seals with a gas pressure rise test after final closure of the package. The leak test shall have a test sensitivity of at least 1 x10^{-3} standard cubic centimeters per second of air (std-cm^3/sec) and there shall be no detectable pressure rise. A higher sensitivity acceptance and maintenance test may be required as discussed in Condition 7.b.(5), below

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

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

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

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

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

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

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

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

7.b.(continued)

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

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

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

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

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

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

Table 2

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

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

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

10. Expiration Date: October 31, 2013.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

REFERENCES

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

FOR THE U.S NUCLEAR REGULATORY COMMISSION

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

Date 2/5/09