6. FINDINGS AND CONCLUSIONS

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

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

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

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

6.1 Routine Transportation

Figure 6-1 and Figure 6-2 show results of routine truck and rail transportation of a single shipment of SNF using the single example route from NUREG-0170, the average of the 200 routes from NUREG/CR-6672, and the average of the 16 truck or rail routes from this study.

Figure 6-1 plots average collective radiation dose (person-Sv) from truck transportation, and Figure 6-2 plots average collective radiation dose from rail transportation. These average doses include doses to the population along the route, doses to occupants of vehicles sharing the route, doses at stops, and doses to vehicle crew and other workers. Doses without the crew and worker dose (labeled public only) are also shown.

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

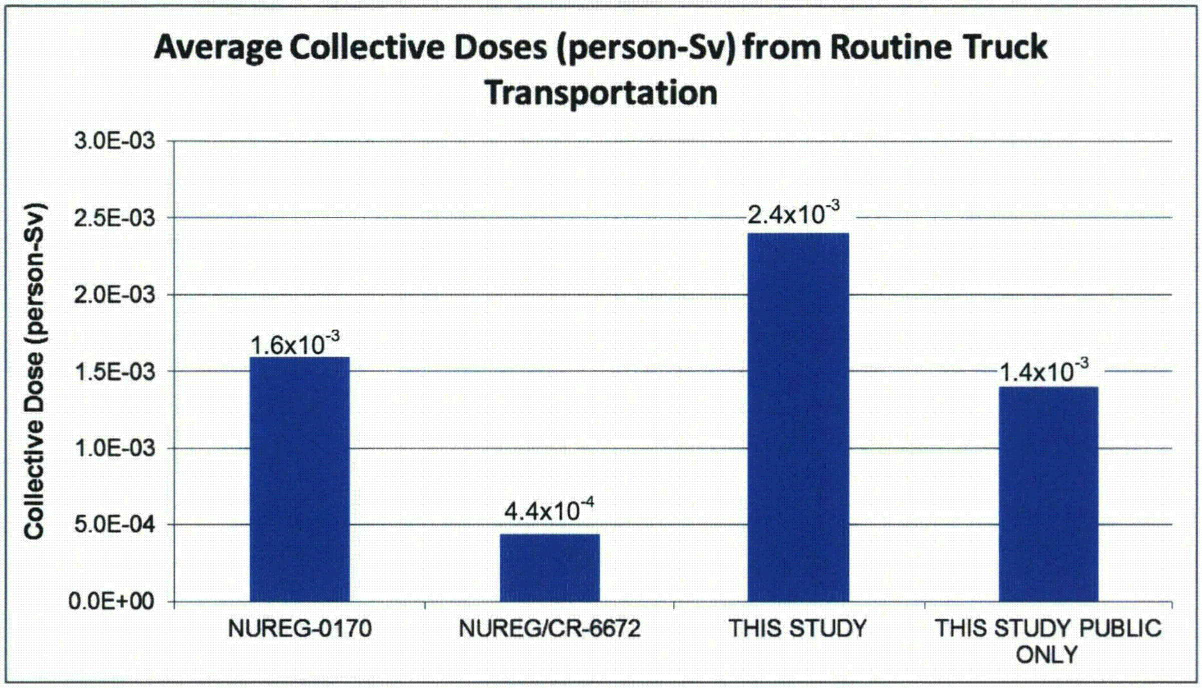

Figure 6-1 Collective doses (person-Sv) from routine truck transportation

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

Figure 6-1 shows that the conservatism was decreased by more than a factor of three.

The collective average dose in the present study is larger than the NUREG/CR-6672 result because present populations are generally larger, particularly along rural routes, and vehicle densities are much greater (see Chapter 2). The higher vehicle speeds used in the present study offset these increases. The largest contributor to higher doses in this study is the parameters used for stops. In this study, stops were assumed to occur every 845 kilometers versus 1,290 kilometers and last for 50 minutes versus 30 minutes. The combination of these two factors results in a 2.5 times increase in the stop dose. This is especially significant because the greatest contributor to the public collective dose is from people sharing truck stops with the cask (56 percent of the collective dose). The second largest contributor is from people sharing the highway with the cask (38 percent of the collective dose). Residents along the route only receive 6 percent of the collective dose and residents near truck stops only receive 1 percent.

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

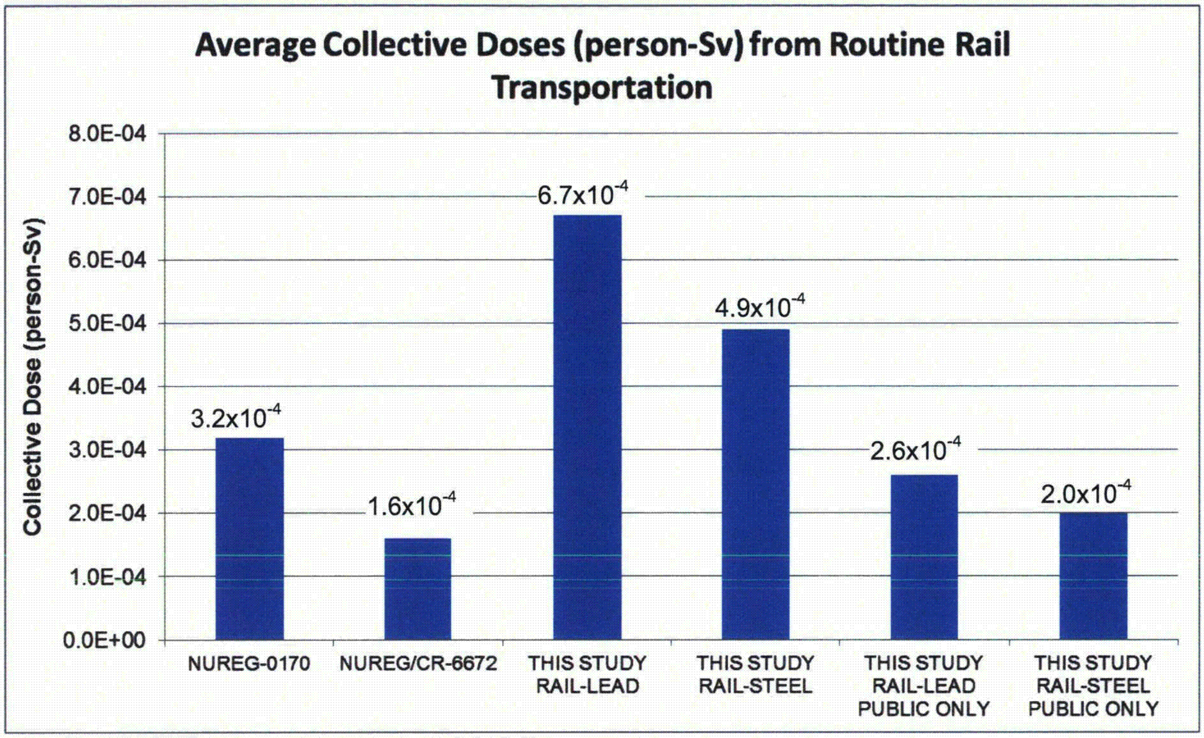

Figure 6-2 Collective doses (person-Sv) from routine rail transportation

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

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[21] specified the same limit on external radiation (the TI) as Part 71 specifies today. Instead, these differences reflect improvements to modeling methods and the increase in population and traffic levels. Also the groups of people exposed that various studies considered has changed. For example, this study includes inspector doses not included in the other two studies.

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

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

- *Differences in vehicle speed.* The faster the cask moves past a receptor, the less that receptor is exposed. NUREG-0170 and NUREG/CR 6672 used 80 kph (50 mph) for all truck routes and 64 kph (40 mph) on rural rail routes, 40 kph (25 mph) on suburban rail routes, and 24 kph (15 mph) on urban rail routes. The truck speeds used in this study are 108 kph (67 mph) on rural routes, 102 kph (63 mph) on suburban routes, and 97 kph (60 mph) on urban routes. The rail speeds are 40 kph (25 mph) on rural and suburban routes and 24 kph (15 mph) on urban routes. The present speeds are based on data instead of the estimated values previous studies used.

- *Differences in populations along the routes.* NUREG-0170 used 6 persons per km^2 (15.5 persons per mi^2) for rural populations, 719 per km^2 (1862 per mi^2) for suburban routes, and 3,861 per km^2 (10,000 per mi^2) for urban routes. NUREG/CR-6672 used 1990 census data provided by the codes HIGHWAY and INTERLINE and used the mean values of Gaussian distributions of population densities on 200 routes in the United States. This study uses 2000 census data provided by WebTRAGIS (Johnson and Michelhaugh, 2002), with some updates based on 2008 Bureau of Census data (U.S. Bureau of the Census, 2008), for the rural, suburban, and urban truck and rail route segments in each State traversed for each of the 16 origin/destination pairs studied. The variation from the NUREG-0170 values is considerable.

- *Differences in vehicles per hour on highways.* NUREG-0170 and NUREG/CR-6672 both used the 1975 values of 470 vehicles per hour on rural routes, 780 on suburban routes, and 2,800 on urban routes. This study used 2002 state vehicle density data for each State traversed. The national average vehicle density is 1,119 vehicles per hour on rural routes, 2,464 on suburban routes, and 5,384 on urban routes. This large difference in vehicle density contributes to the difference in collective doses for routine truck transportation between NUREG/CR-6672 and this study.

- *Differences in calculating doses to rail crew.* NUREG-0170 estimated the distance between the container carrying radioactive material and the crew member to calculate doses to rail and railyard crew. NUREG/CR-6672 used the Wooden (1980) calculation of doses to railyard workers and did not calculate a dose to the train crew. 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 instead of the stop dose formula, which is pegged to total trip length and 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 MEI is a better indication than collective dose of the radiological effect of routine transportation. The same event results in different collective doses depending on the population affected, which varies by location and the consideration of rush hour. The MEI dose is shown in Figure 6-3 for NUREG-0170 and for the three cask types of this study. NUREG/CR-6672 did not calculate this dose for routine transportation. The reduction is because of the higher speeds this study used.

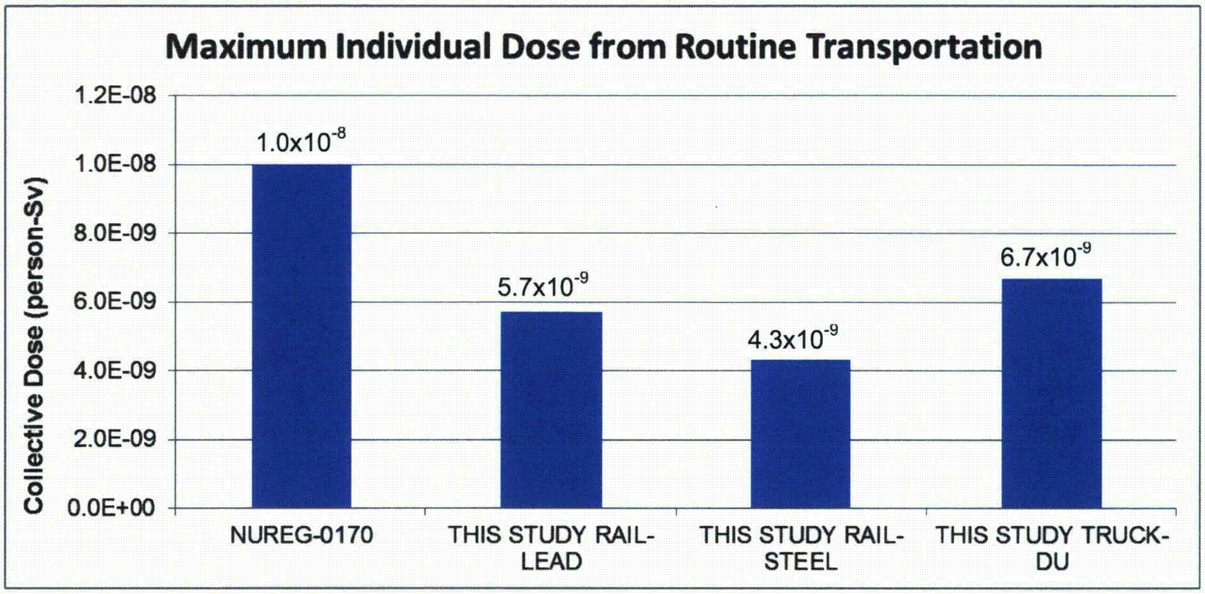

Figure 6-3 Maximum individual dose (Sv) from routine transportation

6.2 Transportation Accidents

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

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

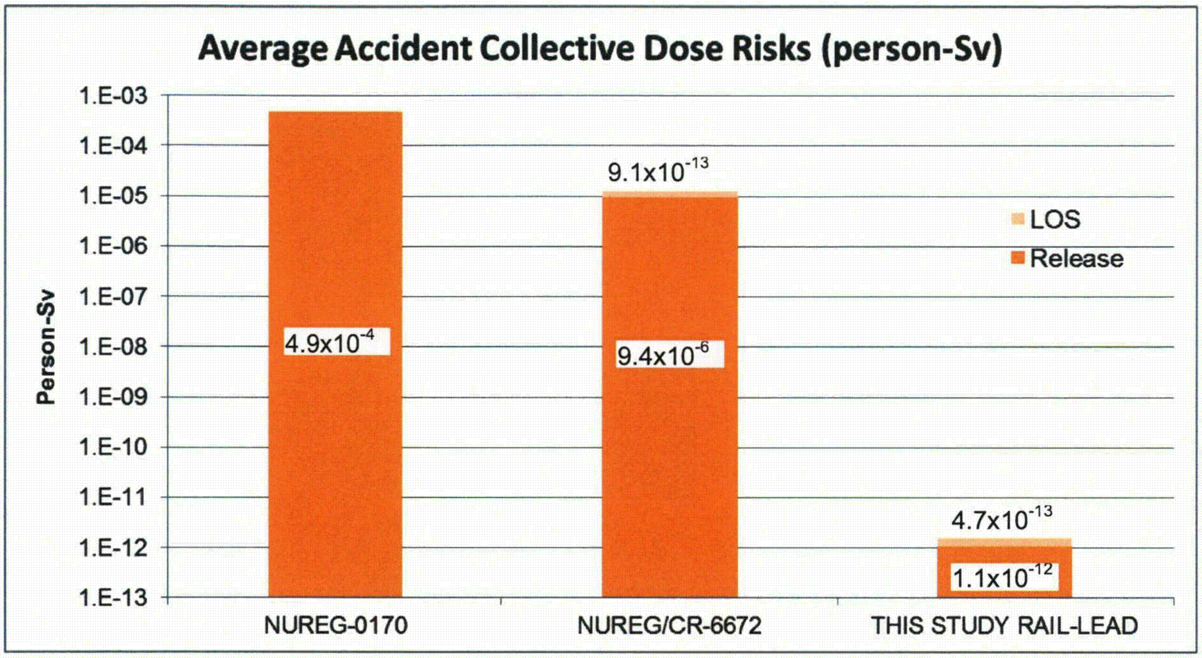

Figure 6-4 Accident collective dose risks from release and loss of lead shielding (LOS) accidents. The LOS bars are not to scale.

The results in Figure 6-4 reflect the different amounts of radioactive material released and the different amounts of lead shielding lost as estimated in the respective studies. NUREG-0170 used a scheme of 8 different accident scenarios; 4 postulated release of the entire releasable contents of the cask, 2 postulated no release, 1 postulated a 10 percent release, and 1 postulated a 1 percent release. The range of conditional probabilities ranged from 1×10^{-5} for the most severe (100 percent release) accident to 80 percent for the 2 no-release scenarios. The NUREG-0170 “universe” of accidents and their consequences was primarily based on engineering judgment, which 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 SNF through the casks. These five groups are particulate matter, semi-volatile substances, ruthenium, gas, and CRUD. The spent fuels considered were high burnup and low burnup PWR and BWR fuel. This analysis resulted in 19 truck accident scenarios and 21 rail accident scenarios, each with an attendant possibility, including a no-release scenario, which had better than 99.99 percent probability.

The present study followed the analytical outline of the NUREG/CR 6672 analysis, but analyzed the structural and thermal behavior of a certified lead-shielded cask design loaded with fuel that the cask is certified to transport. Instead of the 19 truck scenarios and 21 rail scenarios that included potential releases of radioactive material, the current study resulted in only 7 rail scenarios that included releases, as described in Chapters 3 and 5. The seals are the only parts of the cask structure that could be damaged enough to allow a release. Release could take place through the seals only if the seals fail and if the cask is carrying uncanistered fuel. No potential truck accident scenario resulted in seal failure, nor did any fire scenario. In the present study, only the Rail-Lead cask response to extremely severe accident conditions resulted in a release. A comparison of the collective dose risks from potential releases in this study to both NUREG-0170 and NUREG/CR-6672 is appropriate, since the latter two studies considered only potential releases. The collective dose risks decrease with each succeeding study as expected, since the overall conditional probability of release and the quantity of material potentially released decreases with each successive study. The decrease in release is primarily because of the replacement of conservative estimates of cask performance in an accident with FE analyses of cask performance in an accident. Basically, in succeeding studies, the calculated performance of the cask is better (it releases less) than estimated previously.

The collective dose risk from a release depends on dispersion of the released material, which 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 RADTRAN model, although the RADTRAN 6 model is much more flexible than the previous versions and can model elevated releases. NUREG-0170 only calculated 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 dispersion modeling has resulted in the decrease in collective dose risk from releases shown in Figure 6-4.

Frequently, public interest in the transportation of SNF focuses solely on the consequences of possible accidents without regard to the likelihood that an accident will occur. The maximum estimated consequence, based on average population density, from the accident with the largest release is 2.18 person-Sv (218 person-rem). This consequence is orders of magnitude less than the 110 person-Sv (11,000 person-rem) in NUREG-0170 and the 9,000 person-Sv (900,000 person-rem) estimated in NUREG/CR-6672 Figure 8.27. The reduction in consequence is the result of using the actual spent fuel being shipped, a smaller release fraction, and improvements in the RADTRAN model. The maximum estimated dose to any person from this accident is 1.6 Sv (160 rem), and would be non-fatal.

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

The present study followed the same general calculation scheme, but with a more sophisticated model of gamma radiation from the cask due to the damaged shield and using 18 potential accident scenarios instead of 10. Most of the difference between the NUREG/CR-6672 dose risks from shielding loss and this study is the inclusion of accident scenarios that have a higher conditional probability (i.e., accidents that are more likely to happen) than any scenarios in NUREG/CR-6672. The consequence of a loss of lead shielding estimated in NUREG/CR-6672 Table 8.13 is 41,200 person-Sv (4,120,000 person-rem), about 100 times the 690 person-Sv (6,900 person-rem) estimated in this study because of the more conservative loss of lead shielding model used in NUREG/CR-6672 and the overestimation of the amount of lead slump in that study. Loss of lead shielding clearly affects only casks with a lead gamma shield; casks using DU or thicker steel shielding would not be affected.

More than 99.999999 percent of potential accident scenarios do not affect the cask at all and would not result in a release of radioactive material or an increased dose from loss of lead shielding. However, these accidents would result in an increased external radiation dose from the cask to the population near the accident because the cask would remain at the accident location until it could be moved. A nominal 10-hour delay in moving the cask was assumed for this study. The resulting collective dose risk is shown in Figure 6-5 for all three cask types studied. Even including this additional consequence type, the accident collective dose risk from this study is less than that reported in either NUREG-0170 or NUREG/CR-6672.

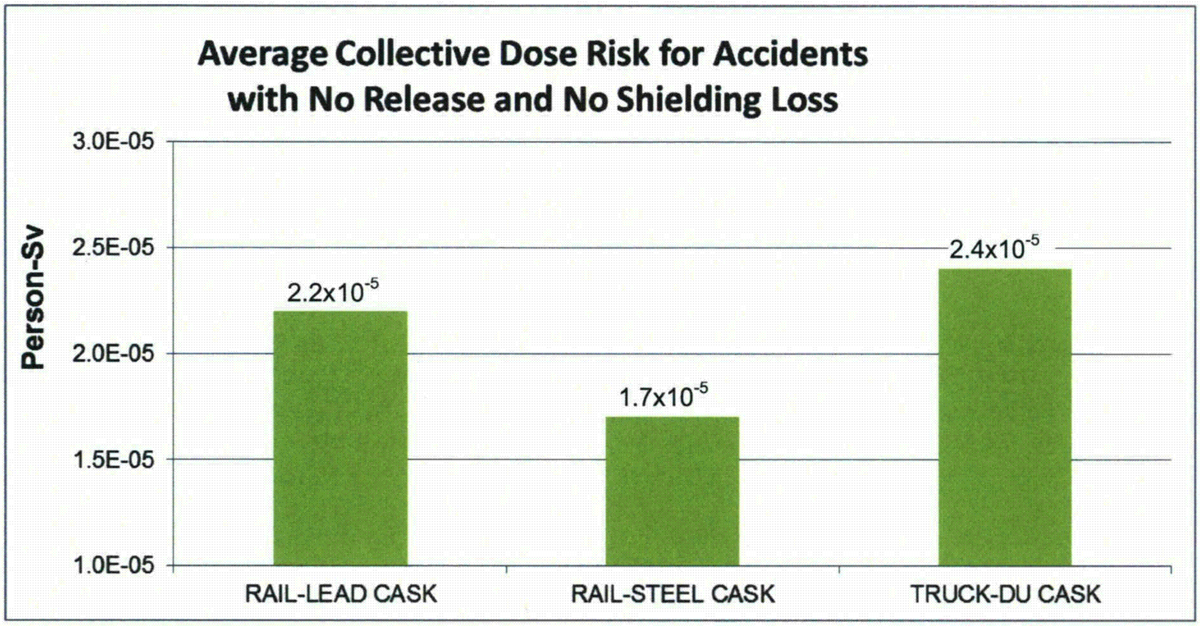

Figure 6-5 Average collective dose from accidents that have no impact on the cargo

For the most probable accident, one that does not involve either loss of shielding or release of radioactive material, the most significant consequence, in addition to any nonradiological consequence of the accident itself, is the external dose from a cask immobilized at the accident site.

Figure 6-5 shows the average collective doses from this type of accident for the 16 truck routes and 16 rail routes studied. 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.

Each of the three transportation risk assessments conducted for the NRC show that the NRC regulation of transportation casks ensures safety and health. The use of data in place of engineering judgment shows that accidents severe enough to cause a loss of shielding or release of radioactive material are improbable and the consequences of such unlikely accidents would require mitigation, but would not result in large radiation doses to even the maximally exposed individual. Moreover, these consequences depend on the size of the population exposed rather than on the radiation or radioactive material released.

6.3 Effect of Transportation of Higher Burnup Spent Nuclear Fuel

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

The regulatory external dose rates must still be met, so there is no effect on incident-free transport or on the results from accidents that do not result in cask damage. The higher burnup fuel will have to be cooled longer before it is transported to meet the cask's decay heat and dose rate limits and the expected radiation emanating from the fuel should not change substantially (it cannot increase above the regulatory surface dose rates, and the casks studied here are either at that limit or very near to it). Therefore, results from loss of shielding accidents will not change significantly. In all of the accidents that are severe enough to have a release path from the cask, the acceleration level is high enough to fail the cladding of all of the fuel, whether it is high burnup or not. Higher burnup fuel has a rim layer with a higher concentration of radionuclides. This will lead to the rod-to-cask release fraction being higher but will not affect the cask-to-environment release fraction. (Table 5-10 gives the release fractions used in this study.) In addition, the isotopic mixture of the higher burnup fuel cooled for a longer period of time will have more transuranic isotopes and less fission product. For example, the inventory of ^{241}Am goes up from 193 TBq at 45 GWD burnup to 1,980 TBq at 60 GWD burnup (5,210 Ci to 53,400 Ci) and the inventory of ^{90}Sr drops from 40,400 TBq to 30,600 TBq (1,090,000 Ci to 826,000 Ci). Insufficient data exists to accurately estimate the rod-to-cask release fractions for higher burnup fuel. If the release fractions remain the same, the effect of the change in radionuclide inventory increases the number of A_2s released by a factor of 5.9. This increase does not alter the conclusions of this study.

6.4 Findings and Conclusion

The following findings are reached from this study:

- The collective dose risks from routine transportation are vanishingly small. Theses doses are about four to five orders of magnitude less than collective background radiation doses.
- The routes selected for this study adequately represent the routes for SNF transport, and there was relatively little variation in the risks per kilometer over these routes.
- Radioactive material would not be released in an accident if the fuel is contained in an inner welded canister inside the cask.
- Only rail casks without inner welded canisters would release radioactive material and only then in exceptionally severe accidents.
- If there were an accident during a spent fuel shipment, there is only about a one in a billion chance that the accident would result in a release of radioactive material.
- If there were a release of radioactive material in a spent fuel shipment accident, the dose to the MEI would be less than 2 Sv (200 rem), and would be neither acute nor lethal.

- The collective dose risks for the two types of extra-regulatory accidents (accidents involving a release of radioactive material and loss of lead shielding accidents) are negligible compared to the risk from a no-release, no-loss of shielding accident.
- The risk of loss of lead shielding from a fire is negligible.
- None of the fire accidents investigated in this study resulted in a release of radioactive material.

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

APPENDIX I

CASK DETAILS AND CERTIFICATES OF COMPLIANCE

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. It also provides the certificates of compliance for those casks selected for evaluation.

I.1.1 Truck Casks

GA-4	The Steel-DU-Steel cask design is stiffer than lead casks and has smaller deformations.
	The 4 PWR assembly capacity of this cask makes it the likely workhorse truck cask for any large transportation campaign.
	Elastomeric seals (ethylene propylene) allow larger closure deformations before leakage.
	Truck casks have hydrogenous neutron shielding.
	Larger capacity allows for larger radioactive material inventory and possible larger consequences from an accident.
	The design is from the late 1980s; General Atomics used finite element analyses and model test results in certification.
	The depleted uranium (DU) shielding is made from five segments, which have been shown to not result in gaps during the regulatory accident sequence, but which could possibly result in gaps during extra-regulatory accidents.
	The cask body has a square cross-section, which provides more possible orientations.
	The cask has an aluminum honeycomb impact limiter.
NAC-LWT	The steel-lead-steel design is relatively flexible, which should result in plastic deformation of the body before seal failure.
	The NAC-LWT cask contains either a single pressurized-water reactor (PWR) assembly or two boiling-water reactor (BWR) assemblies.
	The cask has both elastomeric and metallic seals. The low compression of the elastomeric seal (metallic is primary) allows little closure movement before leakage but may perform better in a fire.
	The lead shielding could melt during severe fires, leading to loss of shielding.
	With liquid neutron shielding, the tank is likely to fail in extra-regulatory impacts.
	The bottom end impact limiter is attached to the neutron shielding tank, making side drop analysis more difficult.
	The NAC-LWT has an aluminum honeycomb impact limiter.
	The cask is very similar to the generic steel-lead-steel cask from NUREG/CR-6672, "Reexamination of Spent Fuel Shipment Risk Estimates."
	The cask is being used for foreign research reactor shipments.

I.1.2 Rail Casks

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

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

I.2 Certificates of Compliance

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

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)*

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

5.(a)(2) Description (continued)

shorter than the generic Holtec MPC designs. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 series is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies; the MPC-32 is designed to contain up to 32 intact PWR assemblies; and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. The MPC-HB is designed to contain up to 80 Humboldt Bay BWR fuel assemblies.

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

Overpack

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

Impact Limiters

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

(3) Drawings

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

5.(a)(3) Drawings (continued)

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

5.(b) Contents

(1) Type, Form, and Quantity of Material

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

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

(b) The following definitions apply:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

13. Expiration Date: March 31, 2014

Attachment: Appendix A

REFERENCES:

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

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

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

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

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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.
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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.

A-10	Table A. 1 (Cont'd)	MPC-68F: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-11		MPC-68F: Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-12		MPC-68F: Mixed Oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
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A-24 to A-27	Table A.2	PWR Fuel Assembly Characteristics
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INDEX TO APPENDIX A

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

I. 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.

Table A.1 (Page 2 of 23)
Fuel Assembly Limits

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

Table A.1 (Page 3 of 23)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

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

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

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

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

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

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 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.

Table A.1 (Page 6 of 23)
Fuel Assembly Limits

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.

Table A.1 (Page 7 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

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

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

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

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

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

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

Table A.1 (Page 9 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

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

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

Table A.1 (Page 10 of 23)
Fuel Assembly Limits

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

Table A.1 (Page 14 of 23)
Fuel Assembly Limits

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.

Table A.1 (Page 15 of 23)
Fuel Assembly Limits

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

Table A.1 (Page 16 of 23)
Fuel Assembly Limits

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.

Table A.1 (Page 17 of 23)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents

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

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

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

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

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

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

Table A.1 (Page 19 of 23)
Fuel Assembly Limits

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.

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

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

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

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

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

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

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

Notes:

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

2. ZR Designates cladding material made of Zirconium or Zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. Each guide tube replaces four fuel rods.

5. Minimum burnup and maximum initial enrichment as specified in Table A.12.

6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.

7. Trojan plant-specific fuel is governed by the limits specified for array/class 17x17B and will be transported in the custom-designed Trojan MPC-24E/EF canisters. The Trojan MPC-24E/EF design is authorized to transport only Trojan plant fuel with a maximum initial enrichment of 3.7 wt.% ^{235}U.

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

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

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

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

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

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

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

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

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

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

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

Notes:

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

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

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

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

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

6. Square, replacing nine fuel rods.

7. Variable

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

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

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

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

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

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

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

Table A.4

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

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

Table A.5

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

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

Table A.6

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

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 19	≤ 30,000	≥ 3.1
≥ 24	≤ 40,000	≥ 3.1

Table A.7

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

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

Table A.8

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

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

NOTES:

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

Table A.9

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

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

Table A.10

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

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

Table A.11

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

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

Table A.12

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

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

NOTES:

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

Table A.13

LO ADING CONFIGURATIONS FOR THE MPC-32

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

Appendix A - Certificate of Compliance 9261, Revision 7

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.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
NAC International
3930 East Jones Bridge Road, Suite 200
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
NAC International, Inc., application dated February 19, 2009.

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

5. (a) Packaging
   (1) Model No.: NAC-STC
   (2) Description: For descriptive purposes, all dimensions are approximate nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5.(a)(3) Drawings

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5.(b) Contents

(1) Type and form of material

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

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

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

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

FUEL COOL TIME TABLE
Minimum Fuel Cool Time in Years

	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.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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	12	OF	12

12. Transport by air is not authorized.

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

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

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

16. Expiration date: May 31, 2014.

REFERENCES

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

As supplemented June 3, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

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

Date: June 12, 2009

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics application dated
January 6, 2009

4. CONDITIONS

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

5.

a. Packaging

(1) Model No.: GA-4

(2) Description

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

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

Cask Assembly

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

5.a. (2) (continued)

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

<u>Cask</u>
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 x10^{-7} std-cm^3/sec. The leak tests shall have a test sensitivity of at least 5 x 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.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

/RA/

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

Date February 5, 2009

APPENDIX II

DETAILS OF RISK ANALYSIS OF ROUTINE, INCIDENT-FREE TRANSPORTATION

II.1 Introduction

In NUREG-0170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes," issued December 1977 (NRC, 1977), the U.S. Nuclear Regulatory Commission (NRC) documented estimates of the radiological consequences and risks associated with shipping by truck, train, plane, or barge. This report covered about 25 different radioactive materials, including power reactor spent fuel. These estimates were calculated using Version 1 of the RADTRAN code (Taylor and Daniel, 1977), which was developed for the NRC by Sandia National Laboratories specifically to support the NUREG-0170 study. In this new updated study, researchers used the computational tool RADTRAN Version 6.0, integrated with the input file generator RadCat (Neuhauser et al.,[1] 2000; Weiner et al., 2009).

Researchers widely accept the basic risk-assessment method employed in the RADTRAN code.[2] A software quality assurance plan, consistent with American National Standards Institute guidelines, tracks changes to the code. The incident-free module of an earlier version of RADTRAN—RADTRAN 5.25—was validated by measurement (Steinman et al., 2002); RADTRAN 6.0, the version used in the current study, employs this same module. Dennis et al. (2008) documents the verification and validation of RADTRAN 6.0.

II.2 The RADTRAN Model of Routine Transportation

II.2.1 Description of the RADTRAN Program

RADTRAN calculates the radiological consequences and risks associated with the shipment of a specific radioactive material in a specific package along a specific route. Shipments that take place without the occurrence of accidents are routine, incident-free shipments, and the radiation doses to various receptors (exposed persons) are called "incident-free doses." Since the probability of routine, incident-free shipment is essentially equal to one, RADTRAN calculates a dose rather than a risk for such shipments.[3] The dose from a routine shipment is based on the external dose from the part of the vehicle carrying the radioactive cargo, referred to as the "vehicle" in this discussion of RADTRAN. Doses to receptors from the external radiation from the vehicle depend on the distance between the receptor and the radioactive cargo being transported and the exposure time. Exposure time is the length of time the receptor is exposed to external emissions from the radioactive cargo. The doses in routine transportation depend only on the external dose rate from the cargo and not on the radioactive inventory of the cargo.

RADTRAN models the vehicle as a spherical radiation source traveling along the route. The source strength is the transport index (TI), 100 times the dose rate in millisievert per hour

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

2 RADTRAN was used to calculate risks for NUREG-0170 (NRC, 1977), the Yucca Mountain Final Environmental Impact Statement (U.S. Department of Energy, 2002), the recertification of the Waste Isolation Pilot Plant, and other studies. RADTRAN today has 600 registered users, about 25 precent of whom are U.S. Government contractors and about 25 percent of whom are international users. The list of users is proprietary.

3 The probability of a transportation incident or accident depends on the trip length and is about 10^{-3} for a cross-country trip. The probability of routine transportation on such a trip is 1 − 0.001, or 0.999, or essentially one. For a shorter trip, the probability of routine transportation is even closer to one.

(mSv/h)[4] at 1 meter (40 inches) from the cask, which is treated as an isotropically radiating virtual source at the center of the sphere, as shown in Figure II-1 (see Neuhauser et al. (2000) for a detailed explanation).

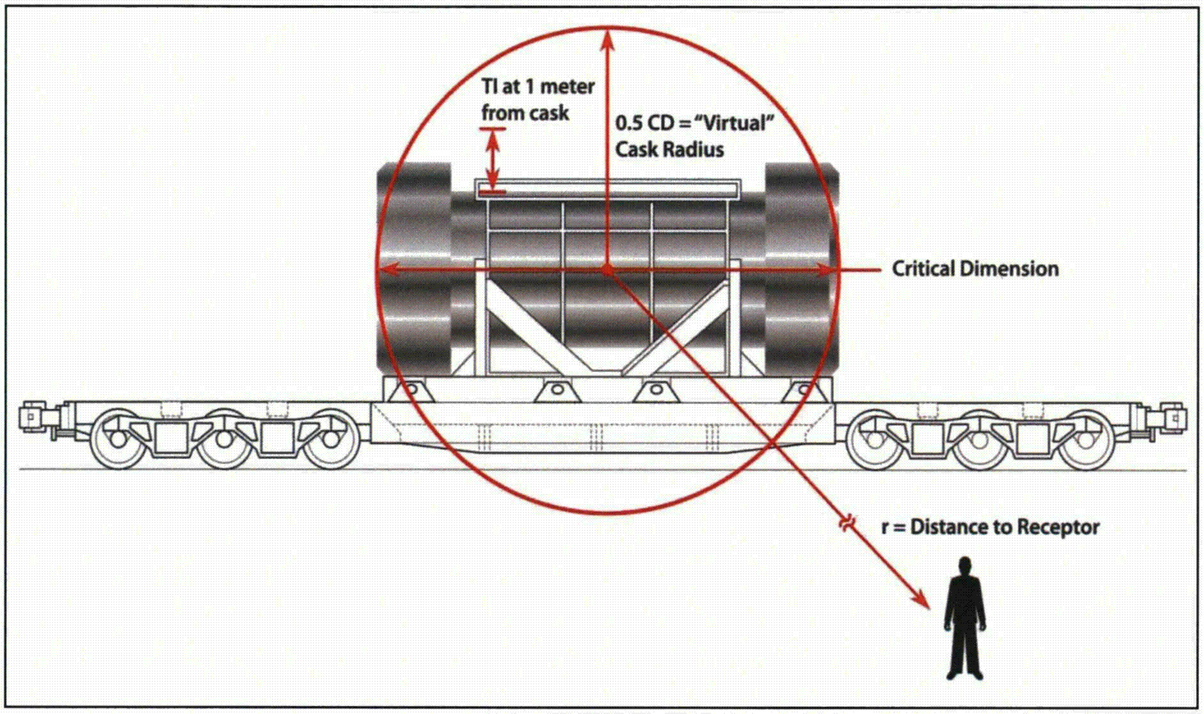

Figure II-1 RADTRAN model of the vehicle in routine, incident-free transportation

When the distance to the receptor *r* is much larger than the critical dimension, RADTRAN models the dose to the receptor as proportional to $1/r^2$. When the distance to the receptor *r* is similar to or less than the critical dimension, as for crew or first responders, RADTRAN models the dose to the receptor as proportional to $1/r$. The TIs for the Rail-Lead and the Rail-Steel casks were calculated from the dose rates at 2 meters, as reported in the safety analysis reports of these casks (Holtec International, 2004, NAC international, 2004) and are shown in Table II-1.

Equation II-1 serves as the basic equation for calculating incident-free doses to a population along a transportation route:

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

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4 One mSv = 100 millirem (mrem). Thus, 100 times the dose rate in mSv/h at 1 meter (40 inches) from the package is equivalent to the dose rate in mrem/h.

Where:

- x is the distance between the receptor and the source, perpendicular to the route
- Q includes factors that correct for unit differences
- k_0 is the package shape factor[5]
- DR_v is the vehicle external dose rate: the TI
- V is the vehicle speed
- μ is the radiation attenuation factor
- B is the radiation buildup factor
- r is the distance between the receptor and the source along the route

Neuhauser et al. (2000) provides additional details of the application of this and similar equations.

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

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

For calculating doses from neutron radiation, on the other hand, RADTRAN uses Equation II-3 where the coefficients are characteristics of the material.

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

Equation II-2 can be rewritten (Neuhauser et al., 2000) as Equation II-4.

(II-4) $$D(x) = \frac{Qk_0 DR_v}{V}\left[f_\gamma * I_\gamma + f_n * I_n\right]$$

Where:

- f_γ is the gamma fraction of the external radiation
- f_n is the neutron fraction of the external radiation
- I_γ is the double integral in Equation II-1 using Equation II-2
- I_n is the double integral in Equation II-1 using Equation II-3

Collective (population) doses are calculated by integrating over the band along the route where the population resides (the x integration in Equation II-1) and then integrating along the route from minus to plus infinity (the r integration in Equation II-1). Figure II-2 illustrates this calculation method for a truck route. The x integration limits in Figure II-2 are not to scale: *xmin* is usually 30 meters (98 feet) (200 meters (656 feet) near a rail classification stop) and *xmax* is usually 800 meters (1/2 mile). Integration of x to distances greater than 800 meters (1/2 mile) results in risks not significantly different from integration to 800 meters (1/2 mile), since the decrease in dose with distance is exponential.

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5 For details on the package shape factor, please see Equations II-4 and II-5 and accompanying text of Neuhauser et al., (2000).

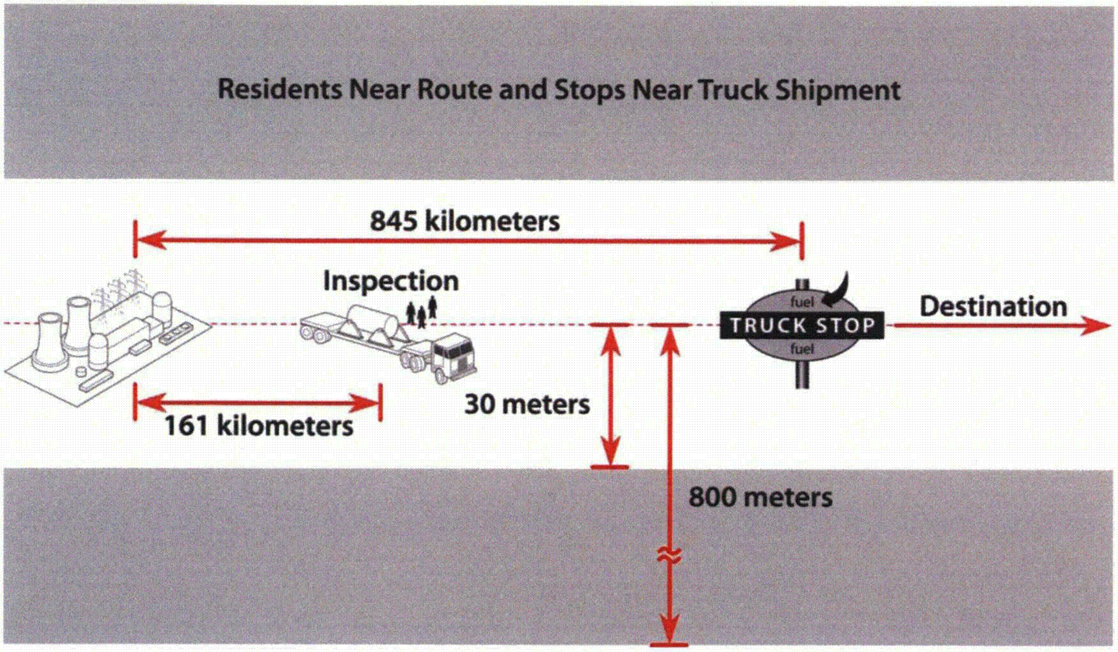

Figure II-2 Diagram of a truck route as modeled in RADTRAN. The 845-km value is the average distance a very large truck travels on half of its fuel capacity. The 161-km (100-mile) value is the distance between spent fuel shipment inspections required by regulation. (from U.S. Department of Energy, 2002)

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

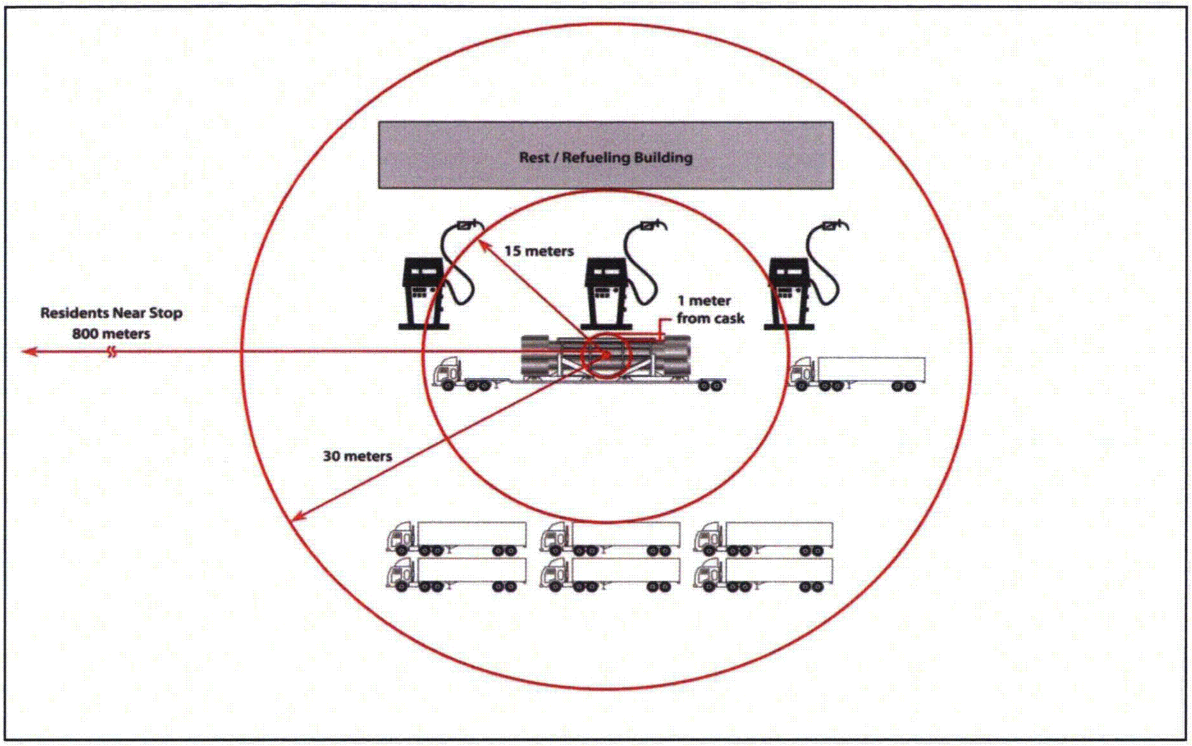

Figure II-3 Diagram of truck stop model

II.2.2 The RADTRAN Software

This section briefly describes the RADTRAN software program. The RadCat 3.0 User Guide (Weiner et al., 2009) provides a full description of the software and how to use it. The equations that RADTRAN uses, variants of Equation II-1, are programmed in FORTRAN 95. RADTRAN uses the following information:

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

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

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

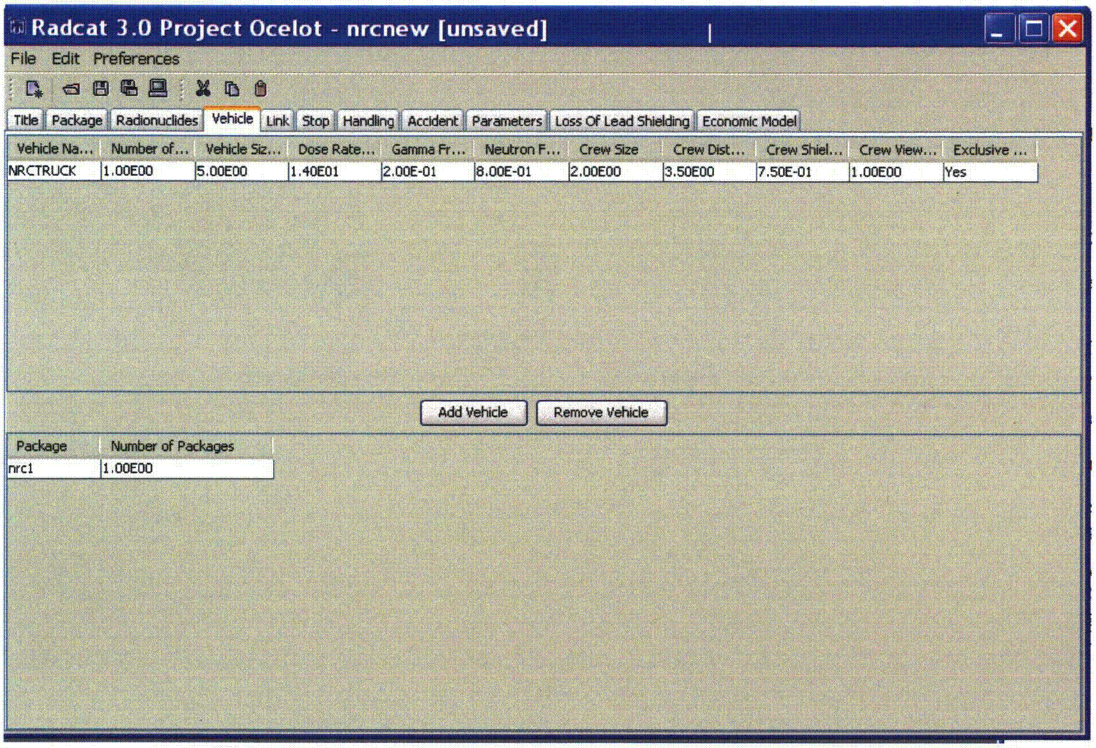

Figure II-4 RadCat vehicle screen

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

II.3 RADTRAN Input Parameters

II.3.1 Vehicle-Specific Input Parameters

RADTRAN does not allow for the offset of the package from the trailer edge, so the physical dimensions of the package are considered the physical dimensions of the vehicle. Table II-1 shows the vehicle-specific input parameters to RADTRAN and includes the parameter values used in this analysis. The Rail-Steel model is calculated as if transporting canistered fuel; the Rail-Lead model is based on transporting uncanistered fuel. The analysis includes a third model—the Truck-DU model—a truck cask with depleted uranium (DU) gamma shielding. While the Truck-DU is a truck cask, the other two are rail casks. This analysis assumes that the Truck-DU cask is transported by truck and the Rail-Lead and Rail-Steel casks are transported by rail.

Table II-1 Vehicle-Specific Parameters

	Truck-DU	Rail-Lead	Rail-Steel
Transportation mode	Highway	Rail	Rail
Length (critical dimension)	5.94 m	4.90 m	5.08 m
Diameter ("crew view")	2.29 m	2.5 m	3.2 m
Distance from cargo to crew cab	3.5 m	150 m minimum	150 m minimum
TI	14	14.02	10.34
Gamma fraction	0.77	0.89	0.90
Neutron fraction	0.23	0.11	0.10
Number of packages per vehicle	1 per truck	1 per railcar	1 per railcar
Number of crew	2	3	3
Exclusive use?	yes	NA	NA
Dedicated rail	NA	yes	yes
17 × 17 PWR assemblies	4	26	24

II.3.2 Route-Specific Input Parameters

Table II-2 shows the route parameters for a unit risk calculation. These route parameters are the common input parameters for the 16 specific rail routes and 16 specific truck routes analyzed.

Table II-2 Route Parameters for Unit Risk Calculation (from U.S. Department of Transportation, 2004, 2006

Parameter	Interstate Highway	Freight Rail
Rural vehicle speed (U.S. average kph)	108	40.4
Suburban vehicle speed (U.S. average kph)	102	40.4
Urban vehicle speed (U.S. average kph)	97	24
Rural vehicle density (U.S. average vehicles/h)	1,119	17[a]
Suburban vehicle density (U.S. average vehicles/h)	2,464	17
Urban vehicle density (U.S. average vehicles/h)	5,384	17
Persons per vehicle	1.5	2
Farm fraction	0.5	0.5
Minimum distance of stop from nearby residents (m)	30	200
Maximum distance of stop from nearby residents (m)	800	800
Stop time for classification (hours)	NA	27
Stop time in transit for railroad change (hours)	NA	variable
Stop time for truck inspections (hours)	0.75	NA
Stop time at truck stops (hours)	0.83	NA
Average number of people sharing the stop	6.9[b]	NA
Minimum distance to people sharing the stop (m)	1[b]	NA
Maximum distance to people sharing the stop (m)	15[b]	NA
Truck stop worker distance from cask (m)	15	NA
Truck stop worker shielding factor	0.018	NA
Truck crew shielding factor	0.377	NA
Escort distance from cask (m)	4	16

a Railcars per hour
b Griego et al., 1996

II.3.3 Other Parameters

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

Table II-3 Parameter Values in the RADTRAN 6 Analysis

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

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

II.3.4 RADTRAN Input and Output Files

Figure II-5 shows the incident-free unit risk input file for the Truck-DU cask. Figure II-6 shows the incident-free unit risk input file for the Rail-Lead and Rail-Steel casks. In the interests of space, only the portion of the input files relevant to routine incident-free transportation are shown.

```
OUTPUT CI_REM
FORM UNIT
DIMEN 1 0 18
PARM 0 1 3 0
PACKAGE GA4 14.0 0.77 0.22999999999999998 5.94
VEHICLE -1 GA_4 1.400000E01 0.77 0.23 5.94 1.0 2.0 3.5 0.38 2.29
GA4 1.0
FLAGS
IACC 2
IUOPT 2
REGCHECK 0
MODSTD
DISTOFF FREEWAY 3.000000E01 3.000000E01 8.000000E02
DISTOFF SECONDARY 2.700000E01 3.000000E01 8.000000E02
DISTOFF STREET 5.000000E00 8.000000E00 8.000000E02
DISTON
FREEWAY 1.500000E01
SECONDARY 3.000000E00
STREET 3.000000E00
ADJACENT 4.000000E00
MITDDIST 3.000000E01
MITDVEL 2.400000E01
RR 1.000000E00
RU 1.800000E-02
RS 8.700000E-01
LINK R GA_4 1.0 108.0 1.5 1.0 1119.0 1.0 0.0 R 1 0.5
LINK S GA_4 1.0 108.0 1.5 1.0 2464.0 1.0 0.0 S 1 0.0
LINK U GA_4 0.9 102.0 1.5 1.0 5384.0 1.0 0.0 U 1 0.0
LINK U_RUSH GA_4 0.1 51.0 1.5 1.0 10760.0 1.0 0.0 U 1 0.0
STOP STOP_1 GA_4 9180.0 1.0 15.0 1.0 0.83
STOP RURAL GA_4 1.0 30.0 800.0 1.0 0.83
STOP SUBURBAN GA_4 1.0 30.0 800.0 0.87 0.83
STOP URBAN GA_4 1.0 30.0 800.0 0.018 0.83
EOF
```

Figure II-5 RADTRAN unit risk input file for the Truck-DU cask

```
RADTRAN 6      July 2008
 && SEE APPENDIX A.2 FOR DETAILS
 && UNIT RISK FACTOR
 && INCIDENT-FREE TRANSPORT
 && AVERGE TI
 && PWR 35GHWD/MTHM BURNUP   10 YEAR COOLED 24 ASSEMBLIES
 && REMARK
 TITLE NAC-STC
 OUTPUT CI_REM
 FORM UNIT
 DIMEN 1 0 18
 PARM 0 1 3 0
 PACKAGE NAC-STC 14.02 0.89 0.10999999999999999 4.9
 PACKAGE HI-STAR_100 10.034 0.9 0.09999999999999998 5.08
VEHICLE -2 NAC-STC 1.400000E01 0.89 0.11 4.9 1.0 3.0 150.0 1.0 2.5
     NAC-STC 1.0
     HI-STAR 0.0
 VEHICLE -2 HI-STAR 1.030000E01 0.9 0.1 5.08 1.0 3.0 150.0 1.0 3.2
     NAC-STC 0.0
     HI-STAR 1.0
 FLAGS
   IACC 2
   ITRAIN 2
   IUOPT 2
   REGCHECK 0
 MODSTD
   DDRWEF 1.800000E-03
   FMINCL 1.000000E00
   DISTOFF RAIL 3.000000E01 3.000000E01 8.000000E02
   DISTON
       RAIL 3.000000E00
   MITDDIST 3.000000E01
   MITDVEL 2.400000E01
   RPD 6.000000E00
   RR 1.000000E00
   RU 1.800000E-02
   RS 8.700000E-01
 LINK NAC_R NAC-STC 1.0 40.4 3.0 1.0 8.0 1.0 0.0 R 3 0.5
 LINK NAC_S NAC-STC 1.0 40.4 3.0 1.0 8.0 1.0 0.0 S 3 0.0
 LINK NAC_U NAC-STC 1.0 24.0 3.0 1.0 17.0 1.0 0.0 U 3 0.0
 LINK HISTAR_R HI-STAR 1.0 40.4 3.0 1.0 8.0 1.0 0.0 R 3 0.5
 LINK HISTAR_S HI-STAR 1.0 40.4 3.0 1.0 8.0 1.0 0.0 S 3 0.0
 LINK HISTAR_U HI-STAR 1.0 24.0 3.0 1.0 17.0 1.0 0.0 U 3 0.0
 STOP CLASSIFICATION NAC-STC 1.0 200.0 800.0 1.0 27.0
 STOP CLASSIFICATION HI-STAR 1.0 200.0 800.0 1.0 27.0
 EOF
```

Figure II-6 RADTRAN unit risk input file for the Rail-Lead and Rail-Steel casks

II.4 Routes

This study analyzes both the per-kilometer doses from a single shipment on rural, suburban, and urban route segments and doses to receptors from a single shipment between 16 representative pairs of origins and destinations, chosen to represent a range of route lengths and a variety of populations. The actual truck and rail routes were selected for a number of reasons. The combination of four origins and four destinations represent a variety of route lengths and population densities and both private and government facilities, as well as a large number of States. The selected routes also include the origins and destinations analyzed in of NUREG/CR 6672, "Reexamination of Spent Fuel Shipment Risk Estimates," issued March 2000, thereby permitting the results of the studies to be compared.

Power reactor spent nuclear fuel and high-level radioactive waste are currently stored at 77 locations in the United States (67 nuclear generating plants, five storage facilities at sites of decommissioned nuclear plants, and five U.S Department of Energy defense facilities). The origin sites (Table II-4) include two nuclear generating plants (Indian Point and Kewaunee), a storage site (Maine Yankee), and a national laboratory (Idaho National Laboratory). The destination sites include the two proposed repository sites not characterized (Deaf Smith County, TX, and Hanford, WA) (U.S. Department of Energy, 1986), the site of the proposed private fuel storage facility (Skull Valley, UT), and a national laboratory site (Oak Ridge, TN). Table II-4 shows the routes modeled. The populations within 800 meters (1/2 mile) of the route were determined from output of the WebTRAGIS (Johnson and Michelhaugh, 2003) routing code, modified by the increase in population between 2000 and 2006 (see Table II-5). Both truck and rail versions of each route are analyzed. These routes are used for illustrative purposes. No actual spent fuel shipments on these routes are occurring or planned.

Table II-4 Specific Routes Modeled
(Urban Kilometers Are Included in Total Kilometers)

Origin	Destination	Population within 800 m (1/2 mile)		Total Kilometers		Urban Kilometers	
		Rail	Truck	Rail	Truck	Rail	Truck
Maine Yankee Site, ME	Hanford, WA	1,647,190	1,129,685	5,084	5,013	355	116
	Deaf Smith County, TX	1,321,024	1,427,973	3,362	3,596	211	165
	Skull Valley, UT	1,451,325	1,068,032	4,068	4,174	207	115
	Oak Ridge, TN	1,146,478	1,137,834	2,125	1,748	161	135
Kewaunee NP, WI	Hanford, WA	476,914	423,163	3,028	3,453	60	52
	Deaf Smith County, TX	677,072	494,920	1,882	2,146	110	60
	Skull Valley, UT	806,115	505,226	2,755	2,620	126	58
	Oak Ridge, TN	779,613	646,034	1,395	1,273	126	92
Indian Point NP, NY	Hanford, WA	961,026	869,763	4,781	4,515	229	97
	Deaf Smith County, TX	1,027,974	968,282	3,088	3,074	204	109
	Skull Valley, UT	1,517,758	808,107	3,977	3,672	229	97
	Oak Ridge, TN	1,146,245	561,723	1,264	1,254	207	60
Idaho National Lab, ID	Hanford, WA	164,399	132,662	1,062	959	20	15
	Deaf Smith County, TX	298,590	384,912	1,913	2,291	40	52
	Skull Valley, UT	169,707	132,939	455	466	26	19
	Oak Ridge, TN	593,680	569,240	3,306	3,287	75	63

WebTRAGIS, which uses census data from the 2000 census, provided the route segments and population densities. The 2008 Statistical Abstract (U.S. Census Bureau, 2008) provided updated population data through 2006. Table 13 of the 2008 Statistical Abstract shows the percent increase in population for each of the 50 U.S. States, as well as for the United States as a whole. Table 20 of the Abstract shows the percent increase in population for all metropolitan areas within the United States with more than 250,000 people. Data from these two tables were combined to give population multipliers for States along routes for which the collective dose and the population increase were significant enough to make a correction.

Table II-5 shows the population multipliers used. "Significant" was defined as a population difference of more than 1 percent (i.e., multipliers between 0.99 and 1.01 were not considered significant). The State-specific multiplier was applied to rural and suburban routes through the State (even though some of these routes would be within the largest metropolitan area), and the multiplier for the largest metropolitan area in that State was applied to the urban route segments (even though some of the urban segments may not be within the largest metropolitan area). For States without a metropolitan area with more than 250,000 people (Delaware, Montana, North Dakota, South Dakota, Vermont, and Wyoming), the Statewide increase was used.[6]

6 For the final version of this report, the routes will be rerun using the 2010 census data. This will eliminate the need for population multipliers.

Table II-5 Population Multipliers

State	Rural, Suburban, Urban Designation	Population Multiplier	State	Rural, Suburban, Urban Designation	Population Multiplier
Arkansas	Rural, Suburban	1.051	New Hampshire	Rural, Suburban	1.064
	Urban	1.069		Urban	1.058
Colorado	Rural, Suburban	1.105	New Jersey	Rural, Suburban	1.037
	Urban	1.105		Urban	1.027
Connecticut	Rural, Suburban	1.029	New Mexico	Rural, Suburban	1.075
	Urban	1.020		Urban	1.119
Delaware	Rural, Suburban	1.089	New York	Rural, Suburban	1.017
	Urban	1.089		Urban	1.027
District of Columbia	Rural, Suburban	1.017	Ohio	Rural, Suburban	1.011
	Urban[a]	1.017		Urban	0.984
Idaho	Rural, Suburban	1.133	Oklahoma	Rural, Suburban	1.037
	Urban	1.221		Urban	1.070
Illinois	Rural, Suburban	1.033	Oregon	Rural, Suburban	1.082
	Urban	1.045		Urban	1.109
Indiana	Rural, Suburban	1.038	Pennsylvania	Rural, Suburban	1.013
	Urban	1.092		ban	1.025
Iowa	Rural, Suburban	1.019	South Dakota	Rural, Suburban	1.036
	Urban	1.110		Urban	1.036
Kansas	Rural, Suburban	1.028	Tennessee	Rural, Suburban	1.061
	Urban	1.037		Urban	1.109
Kentucky	Rural, Suburban	1.041	Texas	Rural, Suburban	1.127
	Urban	1.051		Urban	1.163
Maine	Rural, Suburban	1.037	Utah	Rural, Suburban	1.142
	Urban	1.054		Urban	1.102
Maryland	Rural, Suburban	1.060	Vermont	Rural, Suburban	1.025
	Urban	1.103		Urban	1.025
Massachusetts	Rural, Suburban	1.014	Virginia	Rural, Suburban	1.080
	Urban	1.014		Urban	1.103
Minnesota	Rural, Suburban	1.050	Washington	Rural, Suburban	1.085
	Urban	1.069		Urban	1.072
Missouri	Rural, Suburban	1.044	Wisconsin	Rural, Suburban	1.036
	Urban	1.036		Urban	1.006
Montana	Rural, Suburban	1.047	Wyoming	Rural, Suburban	1.043
	Urban	1.047		Urban	1.043
Nebraska	Rural, Suburban	1.033			
	Urban	1.072			

[a] For urban areas within the District of Columbia, the growth rate of the District was used rather than the growth rate of metropolitan Washington. The growth rate of metropolitan Washington was used for urban areas within Maryland and Virginia.

Parameters like population density and route segment lengths, which are specific to each route, were developed using WebTRAGIS. Figures II-7 through II-10 are WebTRAGIS maps showing the routes.

Maine Yankee NP Routes

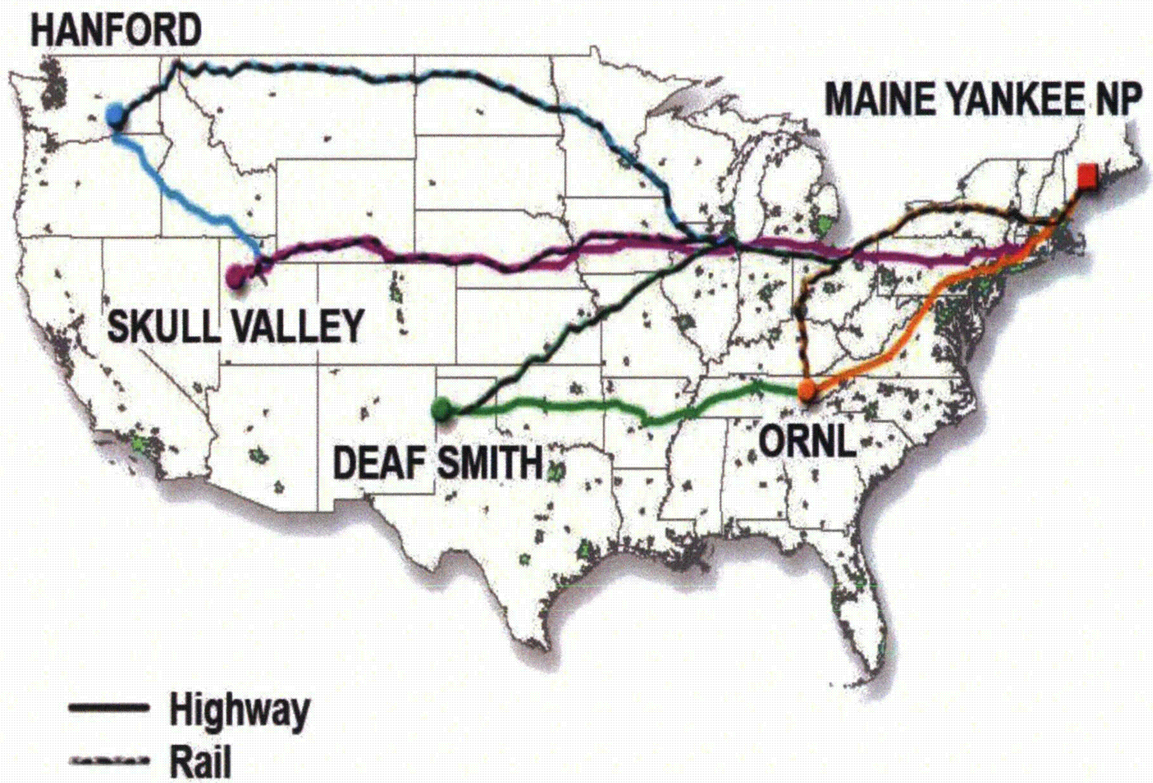

Figure II-6 Highway and rail routes from the Maine Yankee Nuclear Plant (NP) site

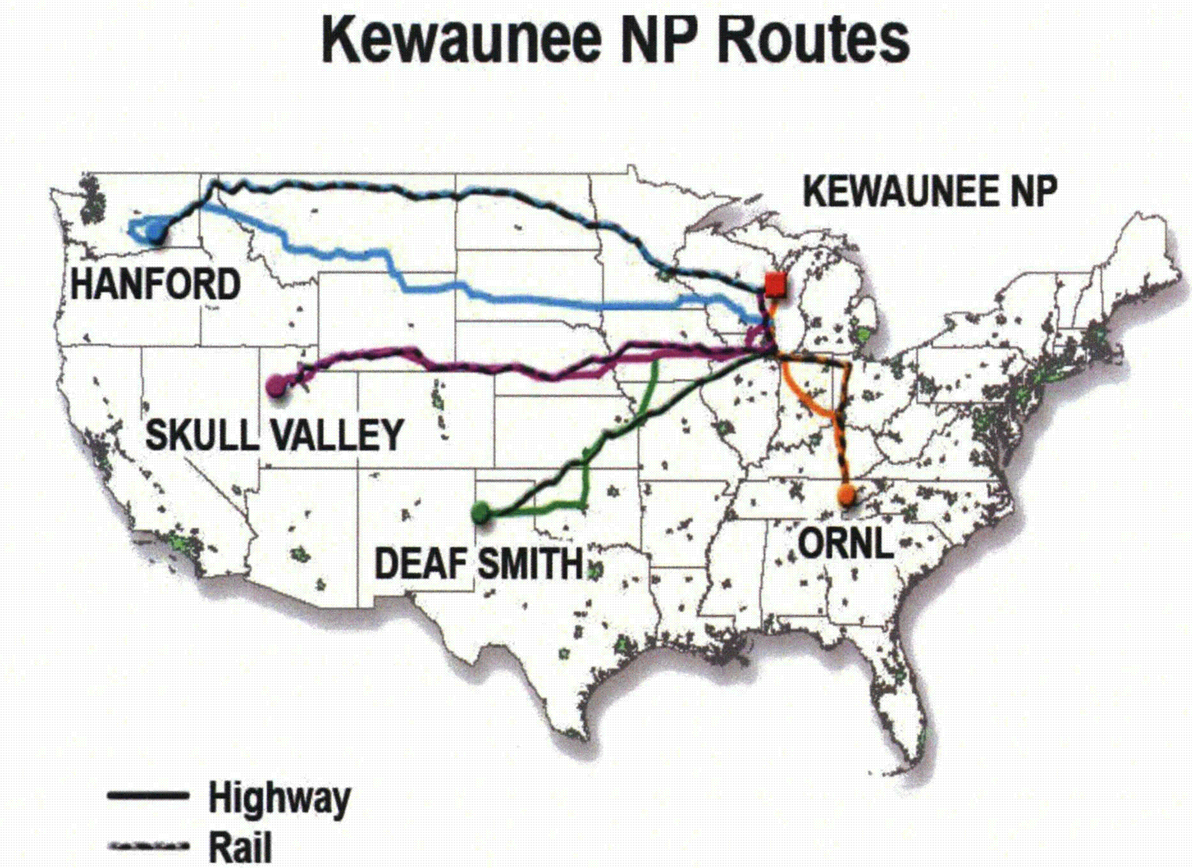

Figure II-7 Highway and rail routes from the Kewaunee NP site

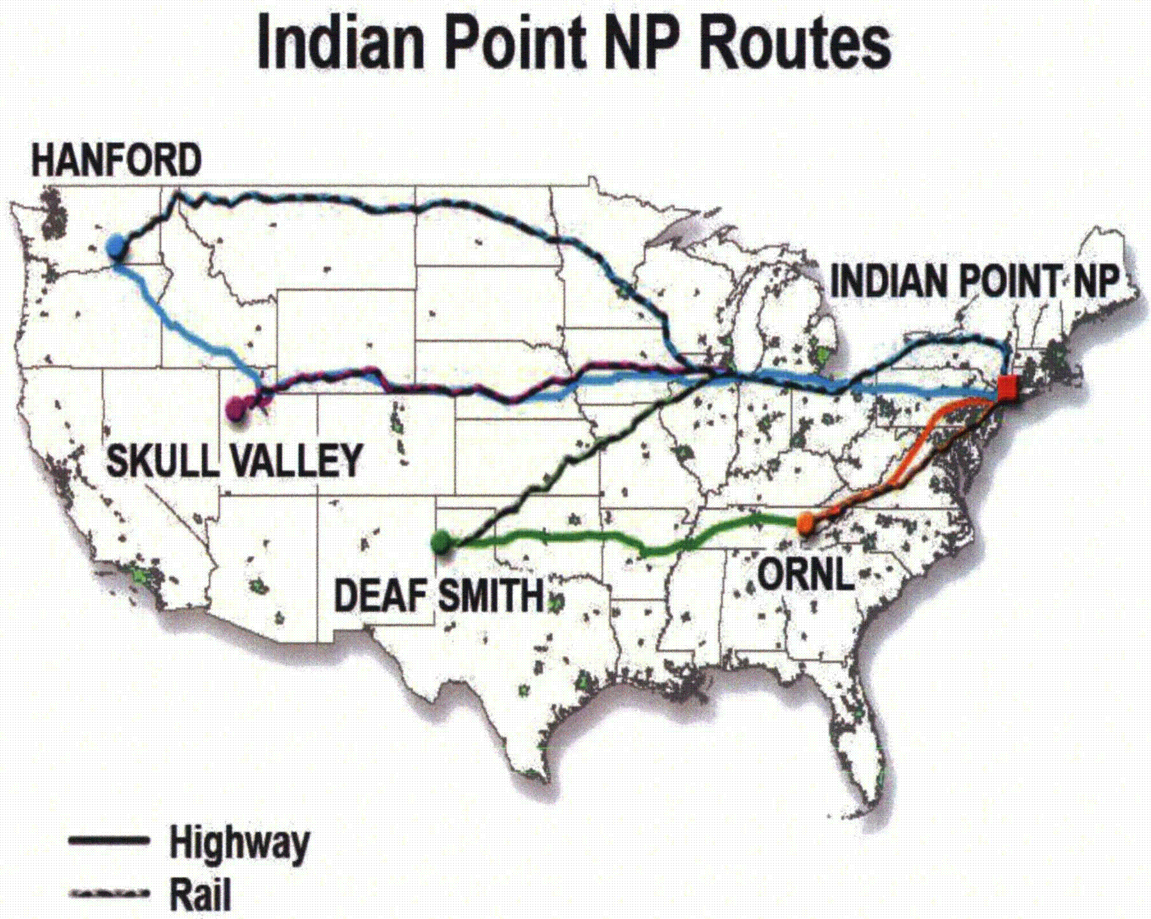

Figure II-8 Highway and rail routes from Indian Point NP site

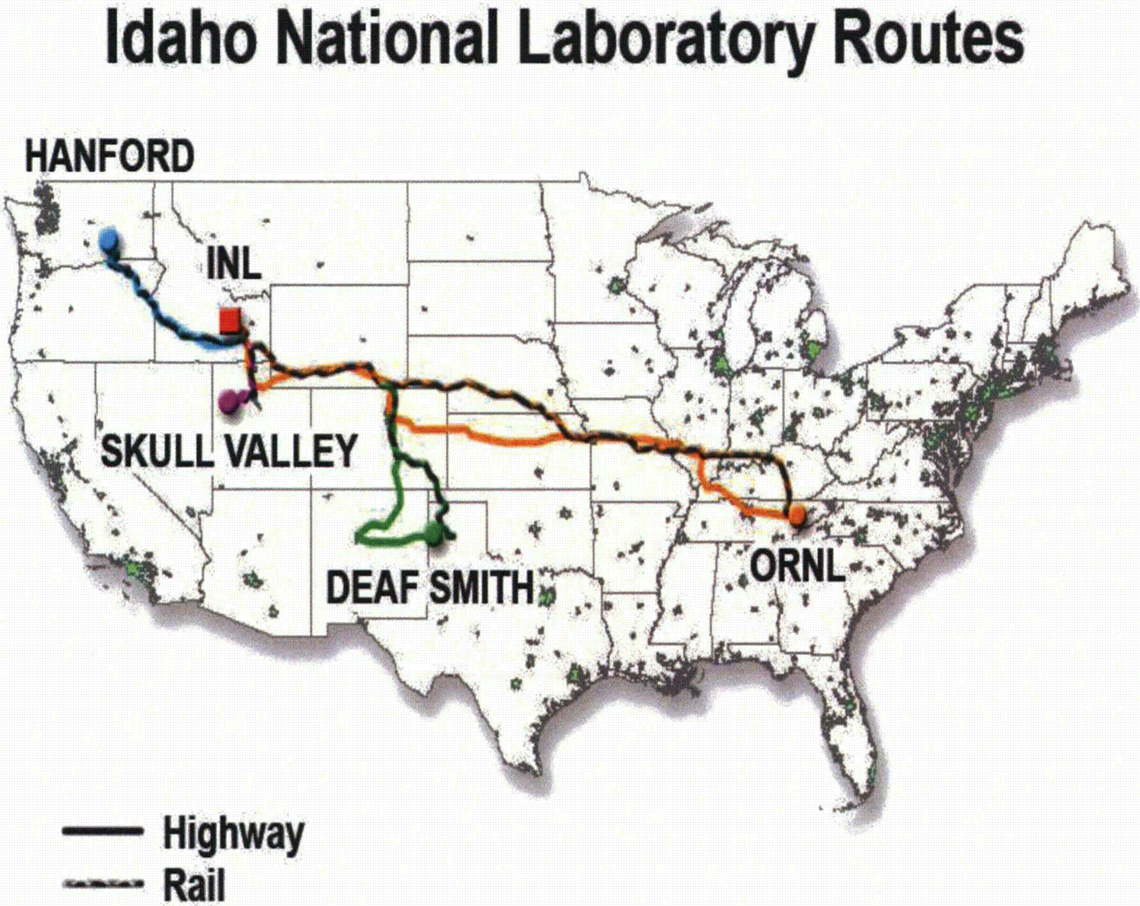

Figure II-9 Highway and rail routes from Idaho National Laboratory (INL)

II.5 Results

II.5.1 Maximally Exposed Resident In-Transit Dose

The largest dose from a moving vehicle to an individual member of the public is sustained when that individual is 30 meters (a conservative estimate of the Interstate right-of-way) from the moving vehicle and the vehicle is moving at the slowest speed it would be likely to maintain. This speed is 24 kilometers per hour (kph) (16 miles per hour (mph)) for both rail and truck. Table II-6 shows the maximum individual dose, in Sv, for each package. These doses are directly proportional to the external dose rate (TI) of each package. For comparison, a single dental x-ray delivers a dose of 4×10^{-5} Sv (Stabin, 2009), about 7,000 times the doses shown in Table II-6.

Table II-6 Maximum Individual Doses

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

Figure II-11 and Figure II-12 show the portion of the RADTRAN output file that reflects these doses.

```
RUN DATE: [ 03-02-2010   AT 18:07 ]          PAGE  11
             TRUCK URF -- PUBLIC

        MAXIMUM INDIVIDUAL IN-TRANSIT DOSE
              GA_4       6.70x10^-7 REM
```

Figure II-10 RADTRAN output for maximum individual truck doses

```
RUN DATE: [ 02-19-2010   AT 10:55 ]          PAGE  10
                NAC-STC
        MAXIMUM INDIVIDUAL IN-TRANSIT DOSE

            NAC-STC    5.67x10^-7 REM
            HI-STAR    4.30x10^-7 REM
```

Figure II-11 RADTRAN output for maximum individual rail doses

II.5.2 Unit Risk—Rail Routes

Table II-7 shows the doses to railyard workers along the route, to residents and others along the route, and to occupants of vehicles that share the route from a single shipment (one rail cask) traveling 1 kilometer past a population density of one person per square kilometer (km^2). The dose units are person-Sv. The doses are calculated assuming one cask on a train because railcar-km is the unit usually used to describe freight rail transport. The data in this table may be used to calculate collective doses along routes as follows:

- This is a conservatively calculated dose that assumes that the railyard crew receives a fraction of the classification yard dose when the train stops. The railyard crew dose is multiplied by the length of each type of route traveled. The classification yard occupational collective dose (Wooden, 1986), assuming a 30-hour classification stop, is integrated into RADTRAN. This integrated dose was adjusted to reflect the 27-hour stop (Table II-3) (U.S. Department of Transportation, 2006).

- The area of the band occupied by the population along the route is equal to the product of the kilometers traveled and the band width (usually 800 meters (1/2 mile) on each side of the route). RADTRAN calculates the doses to residents along the route by integrating over this area. This "unit dose to a resident along the route" is multiplied by the area of the band and the appropriate population density (obtained from a routing code like WebTRAGIS).

Table II-7 Individual Doses ("Unit Doses" or "Unit Risks") to Various Receptors for Rail Routes
(The units of the dose to the residents near a railyard where a train has stopped, Sv-km²/h, reflect the output of the RADTRAN stop model, which incorporates the area occupied.)

Cask and route type	Resident along route, Sv-km[a]	Resident near railyard Sv-km²/h[b]	Occupants of vehicles sharing the route
Rail-Lead rural	7.3×10^{-10}	3.5×10^{-7}	1.6×10^{-8}
Rail-Lead suburban	6.3×10^{-10}	3.5×10^{-7}	1.6×10^{-8}
Rail-Lead urban	2.2×10^{-11}	3.5×10^{-7}	4.6×10^{-8}
Rail-Steel rural	5.6×10^{-10}	2.7×10^{-7}	1.2×10^{-8}
Rail-Steel suburban	4.8×10^{-10}	2.7×10^{-7}	1.2×10^{-8}
Rail-Steel urban	1.7×10^{-11}	2.7×10^{-7}	3.5×10^{-8}

a To obtain the collective dose to residents along the route, multiply this number by the route length and the population density.
b To obtain the collective dose to residents near a railyard, multiply this number by the population density and the stop duration.
c To obtain the collective dose to occupants of vehicles sharing the route, multiply this number by the route length (the vehicle density and occupants/vehicle used are the national average).

Figure II-13 shows the RADTRAN output for Table II-7 (in rem). The relevant data in the output are in bold print.

IN-TRANSIT POPULATION EXPOSURE IN PERSON-REM

LINK	CREW	OFF LINK	ON LINK
NAC_R	4.32×10^{-5}	**7.29×10^{-8}**	**1.63×10^{-6}**
NAC_S	4.32×10^{-5}	**6.34×10^{-8}**	**1.63×10^{-6}**
NAC_U	7.28×10^{-5}	**2.21×10^{-9}**	**4.63×10^{-6}**
HISTAR_R	3.27×10^{-5}	**5.55×10^{-8}**	**1.24×10^{-6}**
HISTAR_S	3.27×10^{-5}	**4.83×10^{-8}**	**1.24×10^{-6}**
HISTAR_U	5.51×10^{-5}	**1.68×10^{-9}**	**3.50×10^{-6}**

STOP EXPOSURE IN PERSON-REM

ANNULAR AREA CLASSIFICA **3.51×10^{-5}**
ANNULAR AREA CLASSIFICA **2.66×10^{-5}**

Figure II-12 RADTRAN output for Table II-7

II.5.3 Unit Risk—Truck Routes

Table II-8 shows the doses to truck crew, residents and others along the route, and to occupants of vehicles that share the route from a single shipment (one truck cask) traveling 1 kilometer past a population density of one person/km². The dose units are person-Sv. Rural, suburban, and urban doses to residents living near stops are calculated by multiplying the appropriate stop dose (truck stops are not typically located in urban areas) by the appropriate population density (obtained from a routing code like WebTRAGIS). The number of stops on each route segment is calculated by dividing the length of the route segment by 845 kilometers (average distance between refueling stops for a large semidetached trailer truck (U.S. Department of Energy, 2002, Appendix J). The area of the band occupied by the population along the route is equal to the kilometers traveled multiplied by, for example, 1.6 for a bandwidth of 800 meters on each side of the route.

Table II-7 Individual Dose ("Unit Risk") to Various Receptors along Truck Routes Unit Risks are Shown for Each EPA Region (See Table II-18)

	Resident along route[a] Sv-km	Resident near stops[b] Sv-km²/h	Occupants of vehicles sharing route[c], person-Sv/km									
			EPA Regions									
			1	2	3	4	5	6	7	8	9	10
Truck-DU rural	$3.1x10^{-10}$	$3.26x10^{-8}$	4.	1.	2.	1.	1.	9.	1.	8.	1.	1.
Truck-DU suburban	$2.7x10^{-10}$	$2.84x10^{-8}$	8.	2.	4.	3.	2.	1.	1.	2.	4.	2.
Truck-DU urban	$5.2x10^{-12}$		2.	4.	6.	6.	4.	3.	2.	4.	8.	6.
Truck-DU urban rush hour[d]	$1.2x10^{-12}$		2.	4.	5.	5.	4.	3.	2.	3.	7.	5.
6.9 people sharing stop (person-Sv)		$2.3x10^{-4}$										

[a] To obtain the collective dose to residents along the route, multiply this number by the route length and the population density.

[b] To obtain the collective dose to residents near a truck stop, multiply this number by the population density and the stop duration.

[c] To obtain the collective dose to occupants of vehicles sharing the route, multiply this number by the route length (the vehicle density and occupants/vehicle used are the regional average).

[d] One-tenth of the urban route segment is considered "rush-hour km," which is equivalent to the truck spending 10 percent of the urban transit distance during rush hour. RADTRAN has historically assumed that the vehicle density doubles during rush hour and the vehicle speed is halved. The slower vehicle speed impacts the dose to urban residents along the route, but the vehicle density does not. Both factors influence the dose to occupants of vehicles sharing the route. The unit risk factors in the table incorporate 9/10 of the distance during nonrush hour and 1/10 of the distance during rush hour, so both factors should be multiplied by the total urban distance.

II.5.4 Doses along Selected Routes

Doses to receptors along the routes shown in Table II-4 are presented below.

II.5.4.1 Collective Doses to Receptors along the Route

Using route data from WebTRAGIS, researchers calculated collective doses from incident-free transportation. For rural and suburban route segments, collective doses calculated were doses sustained by the resident population. Nonresident populations were included with residents as receptors along the urban segments of the routes. Tables II-9 to II-12 show collective doses along rail routes; Table II-13 to Table II-16 show collective doses along highway routes. Blank cells in the tables indicate that no route miles for that population type were present along that route (e.g., not all routes transit urban areas in all States).

Table II-8 Collective Doses to Residents along the Route (Person-Sv) from Rail Transportation; Shipment Origin—INL

DEST. AND ROUTE	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
ORNL								
Colorado	$1.3x10^{-7}$	$5.8x10^{-7}$	0	$7.1x10^{-7}$	$1.0x10^{-7}$	$4.4x10^{-7}$	0	$5.4x10^{-7}$
Idaho	$1.7x10^{-6}$	$7.6x10^{-6}$	$3.4x10^{-7}$	$9.7x10^{-6}$	$1.3x10^{-6}$	$5.8x10^{-6}$	$2.6x10^{-7}$	$7.4x10^{-6}$
Illinois	$1.8x10^{-6}$	$1.7x10^{-5}$	$4.5x10^{-7}$	$1.9x10^{-5}$	$1.3x10^{-6}$	$1.3x10^{-5}$	$3.4x10^{-7}$	$1.4x10^{-5}$
Indiana	$1.7x10^{-6}$	$8.2x10^{-6}$	$1.8x10^{-7}$	$1.0x10^{-5}$	$1.3x10^{-6}$	$6.3x10^{-6}$	$1.4x10^{-7}$	$7.7x10^{-6}$
Kansas	$1.3x10^{-6}$	$6.6x10^{-6}$	$1.6x10^{-7}$	$8.1x10^{-6}$	$9.7x10^{-7}$	$5.0x10^{-6}$	$1.2x10^{-7}$	$6.1x10^{-6}$
Kentucky	$2.6x10^{-6}$	$2.1x10^{-5}$	$8.6x10^{-7}$	$2.5x10^{-5}$	$2.0x10^{-6}$	$1.6x10^{-5}$	$6.6x10^{-7}$	$1.9x10^{-5}$
Missouri	$2.4x10^{-6}$	$2.2x10^{-5}$	$1.1x10^{-6}$	$2.6x10^{-5}$	$1.8x10^{-6}$	$1.7x10^{-5}$	$8.7x10^{-7}$	$2.0x10^{-5}$
Nebraska	$3.5x10^{-6}$	$1.2x10^{-5}$	$3.5x10^{-7}$	$1.6x10^{-5}$	$2.7x10^{-6}$	$9.5x10^{-6}$	$2.6x10^{-7}$	$1.2x10^{-5}$
Tennessee	$1.2x10^{-6}$	$7.8x10^{-6}$	$4.2x10^{-8}$	$9.1x10^{-6}$	$9.4x10^{-7}$	$6.0x10^{-6}$	$3.2x10^{-8}$	$6.9x10^{-6}$
Wyoming	$1.4x10^{-6}$	$8.8x10^{-6}$	$2.1x10^{-7}$	$1.0x10^{-5}$	$1.1x10^{-6}$	$6.7x10^{-6}$	$1.6x10^{-7}$	$7.9x10^{-6}$
DEAF SMITH								
Colorado	$3.3x10^{-6}$	$4.1x10^{-5}$	$1.7x10^{-6}$	$4.6x10^{-5}$	$2.5x10^{-6}$	$3.2x10^{-5}$	$1.3x10^{-6}$	$3.5x10^{-5}$
Idaho	$1.7x10^{-6}$	$7.6x10^{-6}$	$3.4x10^{-7}$	$9.7x10^{-6}$	$1.3x10^{-6}$	$5.8x10^{-6}$	$2.6x10^{-7}$	$7.4x10^{-6}$
Oklahoma	$1.1x10^{-7}$	$1.8x10^{-7}$	0	$2.9x10^{-7}$	$8.3x10^{-8}$	$1.4x10^{-7}$	0	$2.2x10^{-7}$
Texas	$4.1x10^{-7}$	$3.4x10^{-6}$	$5.9x10^{-8}$	$3.8x10^{-6}$	$3.1x10^{-7}$	$2.6x10^{-6}$	$4.4x10^{-8}$	$2.9x10^{-6}$
Wyoming	$1.1x10^{-6}$	$6.0x10^{-6}$	$1.5x10^{-7}$	$7.3x10^{-6}$	$8.5x10^{-7}$	$4.6x10^{-6}$	$1.2x10^{-7}$	$5.6x10^{-6}$
HANFORD								
Idaho	$3.7x10^{-6}$	$1.6x10^{-5}$	**$6.0x10^{-7}$**	$2.0x10^{-5}$	$2.8x10^{-6}$	$1.2x10^{-5}$	$4.6x10^{-7}$	$1.5x10^{-5}$
Oregon	$1.4x10^{-6}$	$9.2x10^{-6}$	$2.2x10^{-7}$	$1.1x10^{-5}$	$1.1x10^{-6}$	$7.0x10^{-6}$	$1.7x10^{-7}$	$8.3x10^{-6}$
Washington	$1.2x10^{-7}$	$4.4x10^{-6}$	$2.6x10^{-7}$	$4.7x10^{-6}$	$8.9x10^{-8}$	$3.3x10^{-6}$	$2.0x10^{-7}$	$3.6x10^{-6}$
SKULL VALLEY								
Idaho	$1.4x10^{-6}$	$6.6x10^{-6}$	$3.3x10^{-7}$	$8.3x10^{-6}$	$1.1x10^{-6}$	$5.0x10^{-6}$	$2.5x10^{-7}$	$6.3x10^{-6}$
Utah	$1.6x10^{-6}$	$1.9x10^{-5}$	$1.1x10^{-6}$	$2.2x10^{-5}$	$1.2x10^{-6}$	$1.4x10^{-5}$	$8.5x10^{-7}$	$1.6x10^{-5}$

Sample calculation: Urban route from INL to Hanford through Idaho, Rail-Lead cask
RADTRAN output (unit risk): $2.21x10^{-9}$ person-rem (from Figure II-13)
Population density: 2,281 persons/km^2
Population multiplier: 1.133
Route segment length: 10.5 km
Population (collective) dose = $2.21x10^{-9}$ * 2,281 * 1.133 * 10.5 = $6.00x10^{-5}$ person-rem
Convert to SI units: $6.00x10^{-5}$ person-rem * 0.01person-Sv/person-rem = $6.00x10^{-7}$ person-Sv
Blank cell indicates no route miles of this population density.

Table II-9 Collective Doses to Residents along the Route (Person-Sv) from Rail Transportation; Shipment Origin—Indian Point

DEST. AND ROUTES	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
ORNL								
Delaware	1.2x10^{-8}	7.3x10^{-6}	8.2x10^{-7}	8.2x10^{-6}	9.5x10^{-9}	5.6x10^{-6}	6.3x10^{-7}	6.2x10^{-6}
DC	3.2x10^{-9}	8.9x10^{-7}	4.5x10^{-7}	1.3x10^{-6}	2.4x10^{-9}	6.8x10^{-7}	3.5x10^{-7}	1.0x10^{-6}
Maryland	6.9x10^{-7}	2.2x10^{-5}	2.0x10^{-6}	2.5x10^{-5}	5.3x10^{-7}	1.7x10^{-5}	1.5x10^{-6}	1.9x10^{-5}
New Jersey	4.0x10^{-7}	1.2x10^{-5}	1.4x10^{-6}	1.4x10^{-5}	3.1x10^{-7}	9.1x10^{-6}	1.1x10^{-6}	1.0x10^{-5}
New York	3.0x10^{-8}	1.6x10^{-6}	3.4x10^{-6}	5.0x10^{-6}	2.3x10^{-8}	1.3x10^{-6}	2.5x10^{-6}	3.8x10^{-6}
Pennsylvania	4.9x10^{-8}	8.6x10^{-6}	3.2x10^{-6}	1.2x10^{-5}	3.8x10^{-8}	6.5x10^{-6}	2.4x10^{-6}	9.0x10^{-6}
Tennessee	2.2x10^{-6}	3.0x10^{-5}	6.6x10^{-7}	3.3x10^{-5}	1.7x10^{-6}	2.3x10^{-5}	5.0x10^{-7}	2.5x10^{-5}
Virginia	4.1x10^{-6}	5.9x10^{-5}	2.3x10^{-6}	6.5x10^{-5}	3.1x10^{-6}	4.5x10^{-5}	1.7x10^{-6}	5.0x10^{-5}
DEAF SMITH								
Illinois	1.5x10^{-6}	2.7x10^{-5}	2.4x10^{-6}	3.1x10^{-5}	1.1x10^{-6}	2.0x10^{-5}	1.8x10^{-6}	2.3x10^{-5}
Indiana	2.1x10^{-6}	1.1x10^{-5}	**5.4x10^{-7}**	1.4x10^{-5}	1.6x10^{-6}	8.6x10^{-6}	4.1x10^{-7}	1.1x10^{-5}
Iowa	3.0x10^{-7}	6.2x10^{-7}	3.1x10^{-8}	9.5x10^{-7}	2.3x10^{-7}	4.8x10^{-7}	2.4x10^{-8}	7.2x10^{-7}
Kansas	2.0x10^{-6}	1.8x10^{-5}	7.9x10^{-7}	2.1x10^{-5}	1.5x10^{-6}	1.4x10^{-5}	6.0x10^{-7}	1.6x10^{-5}
Missouri	1.2x10^{-6}	7.0x10^{-6}	2.4x10^{-7}	8.5x10^{-6}	9.2x10^{-7}	5.4x10^{-6}	1.8x10^{-7}	6.5x10^{-6}
New York	5.5x10^{-6}	6.1x10^{-5}	4.9x10^{-6}	7.1x10^{-5}	4.2x10^{-6}	4.6x10^{-5}	3.7x10^{-6}	5.4x10^{-5}
Ohio	2.5x10^{-6}	3.2x10^{-5}	2.3x10^{-6}	3.6x10^{-5}	1.9x10^{-6}	2.4x10^{-5}	1.7x10^{-6}	2.8x10^{-5}
Oklahoma	4.5x10^{-7}	4.0x10^{-6}	5.2x10^{-8}	4.5x10^{-6}	3.4x10^{-7}	3.1x10^{-6}	3.9x10^{-8}	3.4x10^{-6}
Pennsylvania	4.1x10^{-7}	9.3x10^{-6}	4.9x10^{-7}	1.0x10^{-5}	3.1x10^{-7}	7.1x10^{-6}	3.7x10^{-7}	7.8x10^{-6}
Texas	7.3x10^{-7}	5.1x10^{-6}	1.2x10^{-7}	6.0x10^{-6}	5.6x10^{-7}	3.9x10^{-6}	9.4x10^{-8}	4.5x10^{-6}
HANFORD								
Idaho	9.8x10^{-7}	6.7x10^{-6}	9.6x10^{-8}	7.8x10^{-6}	7.5x10^{-7}	5.1x10^{-6}	7.3x10^{-8}	5.9x10^{-6}
Illinois	1.4x10^{-6}	2.1x10^{-5}	2.2x10^{-6}	2.4x10^{-5}	1.0x10^{-6}	1.6x10^{-5}	1.7x10^{-6}	1.9x10^{-5}
Indiana	2.1x10^{-6}	1.1x10^{-5}	5.4x10^{-7}	1.4x10^{-5}	1.6x10^{-6}	8.6x10^{-6}	4.1x10^{-7}	1.1x10^{-5}
Minnesota	3.2x10^{-6}	2.9x10^{-5}	1.2x10^{-6}	3.4x10^{-5}	2.4x10^{-6}	2.2x10^{-5}	8.9x10^{-7}	2.6x10^{-5}
Montana	2.2x10^{-6}	1.3x10^{-5}	1.4x10^{-7}	1.6x10^{-5}	1.7x10^{-6}	1.0x10^{-5}	1.1x10^{-7}	1.2x10^{-5}
New York	5.5x10^{-6}	6.1x10^{-5}	4.9x10^{-6}	7.1x10^{-5}	4.2x10^{-6}	4.6x10^{-5}	3.7x10^{-6}	5.4x10^{-5}
North Dakota	1.0x10^{-6}	8.2x10^{-6}	2.6x10^{-7}	9.5x10^{-6}	7.7x10^{-7}	6.3x10^{-6}	2.0x10^{-7}	7.2x10^{-6}
Ohio	2.5x10^{-6}	3.2x10^{-5}	2.3x10^{-6}	3.6x10^{-5}	1.9x10^{-6}	2.4x10^{-5}	1.7x10^{-6}	2.8x10^{-5}
Pennsylvania	4.1x10^{-7}	9.3x10^{-6}	4.9x10^{-7}	1.0x10^{-5}	3.1x10^{-7}	7.1x10^{-6}	3.7x10^{-7}	7.8x10^{-6}
Washington	1.1x10^{-6}	1.3x10^{-5}	6.5x10^{-7}	1.5x10^{-5}	8.5x10^{-7}	1.0x10^{-5}	5.0x10^{-7}	1.2x10^{-5}
Wisconsin	1.7x10^{-6}	8.2x10^{-6}	3.7x10^{-7}	1.0x10^{-5}	1.3x10^{-6}	6.2x10^{-6}	2.8x10^{-7}	7.8x10^{-6}
SKULL VALLEY								
Colorado	1.3x10^{-7}	5.8x10^{-7}	0	7.1x10^{-7}	1.0x10^{-7}	4.4x10^{-7}	0	5.4x10^{-7}
Illinois	1.3x10^{-6}	2.1x10^{-5}	2.7x10^{-6}	2.5x10^{-5}	9.9x10^{-7}	1.6x10^{-5}	2.1x10^{-6}	1.9x10^{-5}
Indiana	2.1x10^{-6}	1.1x10^{-5}	5.4x10^{-7}	1.4x10^{-5}	1.6x10^{-6}	8.6x10^{-6}	4.1x10^{-7}	1.1x10^{-5}
Iowa	4.0x10^{-6}	1.8x10^{-5}	3.4x10^{-7}	2.2x10^{-5}	3.1x10^{-6}	1.4x10^{-5}	2.6x10^{-7}	1.7x10^{-5}
Nebraska	4.2x10^{-6}	2.0x10^{-5}	6.2x10^{-7}	2.5x10^{-5}	3.2x10^{-6}	1.5x10^{-5}	4.7x10^{-7}	1.9x10^{-5}
New York	5.5x10^{-6}	6.1x10^{-5}	4.9x10^{-6}	7.1x10^{-5}	4.2x10^{-6}	4.6x10^{-5}	3.7x10^{-6}	5.4x10^{-5}
Ohio	2.5x10^{-6}	3.2x10^{-5}	2.3x10^{-6}	3.6x10^{-5}	1.9x10^{-6}	2.4x10^{-5}	1.7x10^{-6}	2.8x10^{-5}
Pennsylvania	4.1x10^{-7}	9.3x10^{-6}	4.9x10^{-7}	1.0x10^{-5}	3.1x10^{-7}	7.1x10^{-6}	3.7x10^{-7}	7.8x10^{-6}
Utah	1.3x10^{-6}	1.8x10^{-5}	1.1x10^{-6}	2.0x10^{-5}	9.7x10^{-7}	1.4x10^{-5}	8.6x10^{-7}	1.6x10^{-5}
Wyoming	1.4x10^{-6}	9.5x10^{-6}	2.3x10^{-7}	1.1x10^{-5}	1.0x10^{-6}	7.2x10^{-6}	1.8x10^{-7}	8.4x10^{-6}

Sample calculation: Urban route from Indian Point to Deaf Smith through Indiana, Rail-Lead cask
RADTRAN output (unit risk): 2.21×10^{-9} person-rem (from Figure II-13)
Population density: 2,305.9 persons/km^2
Population multiplier: 1.0
Route segment length: 10.6 km
Population (collective) dose = 2.21×10^{-9} * 2,305.9 * 1.0 * 10.6 = 5.40×10^{-5} person-rem
Convert to SI units: 5.40×10^{-5} person-rem * 0.01person-Sv/person-rem = 5.40×10^{-7} person-Sv

Table II-10 Collective Doses to Residents along the Route (Person-Sv) from Rail Transportation; Shipment Origin—Kewaunee

DEST. AND ROUTES	Rail-Lead				Rail-Steel			
	Rural	**Suburban**	**Urban**	**Total**	Rural	Suburban	Urban	Total
ORNL								
Illinois	$2.4x10^{-7}$	$2.1x10^{-5}$	$2.5x10^{-6}$	$2.3x10^{-5}$	$1.8x10^{-7}$	$1.6x10^{-5}$	$1.9x10^{-6}$	$1.8x10^{-5}$
Indiana	$2.1x10^{-6}$	$1.1x10^{-5}$	$5.4x10^{-7}$	$1.4x10^{-5}$	$1.6x10^{-6}$	$8.6x10^{-6}$	$4.1x10^{-7}$	$1.1x10^{-5}$
Kentucky	$3.2x10^{-6}$	$1.6x10^{-5}$	$7.0x10^{-7}$	$2.0x10^{-5}$	$2.4x10^{-6}$	$1.3x10^{-5}$	$5.3x10^{-7}$	$1.5x10^{-5}$
Ohio	$2.2x10^{-6}$	$3.0x10^{-5}$	$1.4x10^{-6}$	$3.3x10^{-5}$	**$1.6x10^{-6}$**	$2.3x10^{-5}$	$1.1x10^{-6}$	$2.5x10^{-5}$
Tennessee	$7.4x10^{-7}$	$5.0x10^{-6}$	$4.1x10^{-8}$	$5.7x10^{-6}$	$5.6x10^{-7}$	$3.8x10^{-6}$	$3.1x10^{-8}$	$4.4x10^{-6}$
Wisconsin	$1.9x10^{-6}$	$2.5x10^{-5}$	$1.5x10^{-6}$	$2.9x10^{-5}$	$1.5x10^{-6}$	$1.9x10^{-5}$	$1.1x10^{-6}$	$2.2x10^{-5}$
DEAF SMITH								
Illinois	$1.6x10^{-6}$	$3.5x10^{-5}$	$3.0x10^{-6}$	$4.0x10^{-5}$	$1.2x10^{-6}$	$2.7x10^{-5}$	$2.3x10^{-6}$	$3.0x10^{-5}$
Iowa	$3.0x10^{-7}$	$6.2x10^{-7}$	$3.1x10^{-8}$	$9.5x10^{-7}$	$2.3x10^{-7}$	$4.8x10^{-7}$	$2.4x10^{-8}$	$7.2x10^{-7}$
Kansas	$2.0x10^{-6}$	$1.8x10^{-5}$	$7.9x10^{-7}$	$2.1x10^{-5}$	$1.5x10^{-6}$	$1.4x10^{-5}$	$6.0x10^{-7}$	$1.6x10^{-5}$
Missouri	$1.2x10^{-6}$	$7.0x10^{-6}$	$2.8x10^{-7}$	$8.5x10^{-6}$	$9.2x10^{-7}$	$5.4x10^{-6}$	$2.2x10^{-7}$	$6.5x10^{-6}$
Oklahoma	$4.5x10^{-7}$	$4.0x10^{-6}$	$5.2x10^{-8}$	$4.5x10^{-6}$	$3.4x10^{-7}$	$3.1x10^{-6}$	$3.9x10^{-8}$	$3.4x10^{-6}$
Texas	$7.3x10^{-7}$	$5.1x10^{-6}$	$1.2x10^{-7}$	$6.0x10^{-6}$	$5.6x10^{-7}$	$3.9x10^{-6}$	$9.4x10^{-8}$	$4.6x10^{-6}$
Wisconsin	$1.9x10^{-6}$	$2.5x10^{-5}$	$1.5x10^{-6}$	$2.9x10^{-5}$	$1.5x10^{-6}$	$1.9x10^{-5}$	$1.1x10^{-6}$	$2.2x10^{-5}$
HANFORD								
Idaho	$9.8x10^{-7}$	$6.7x10^{-6}$	$9.6x10^{-8}$	$7.8x10^{-6}$	$7.5x10^{-7}$	$5.1x10^{-6}$	$7.3x10^{-8}$	$5.9x10^{-6}$
Minnesota	$3.3x10^{-6}$	$3.0x10^{-5}$	$9.2x10^{-7}$	$3.5x10^{-5}$	$2.5x10^{-6}$	$2.3x10^{-5}$	$7.0x10^{-7}$	$2.6x10^{-5}$
Montana	$2.2x10^{-6}$	$1.3x10^{-5}$	$1.4x10^{-7}$	$1.3x10^{-5}$	$1.7x10^{-6}$	$1.0x10^{-5}$	$1.1x10^{-7}$	$1.0x10^{-5}$
North Dakota	$1.0x10^{-6}$	$8.2x10^{-6}$	$2.6x10^{-7}$	$9.5x10^{-6}$	$7.7x10^{-7}$	$6.3x10^{-6}$	$2.0x10^{-7}$	$7.2x10^{-6}$
Washington	$1.1x10^{-6}$	$1.3x10^{-5}$	$6.5x10^{-7}$	$1.5x10^{-5}$	$8.5x10^{-7}$	$1.0x10^{-5}$	$5.0x10^{-7}$	$1.2x10^{-5}$
Wisconsin	$3.5x10^{-6}$	$2.2x10^{-5}$	$8.9x10^{-7}$	$2.6x10^{-5}$	$2.7x10^{-6}$	$1.7x10^{-5}$	$6.8x10^{-7}$	$2.0x10^{-5}$
SKULL VALLEY								
Colorado	$1.3x10^{-7}$	$5.8x10^{-7}$	0	$7.1x10^{-7}$	$1.0x10^{-7}$	$4.4x10^{-7}$	0	$5.4x10^{-7}$
Illinois	$1.4x10^{-6}$	$2.7x10^{-5}$	$2.8x10^{-6}$	$3.1x10^{-5}$	$1.1x10^{-6}$	$2.1x10^{-5}$	$2.1x10^{-6}$	$2.4x10^{-5}$
Iowa	$4.0x10^{-6}$	$1.8x10^{-5}$	$3.4x10^{-7}$	$2.2x10^{-5}$	$3.1x10^{-6}$	$1.4x10^{-5}$	$2.6x10^{-7}$	$1.7x10^{-5}$
Nebraska	$4.2x10^{-6}$	$2.0x10^{-5}$	$6.2x10^{-7}$	$2.5x10^{-5}$	$3.2x10^{-6}$	$1.5x10^{-5}$	$4.7x10^{-7}$	$1.9x10^{-5}$
Utah	$1.3x10^{-6}$	$1.8x10^{-5}$	$1.1x10^{-6}$	$2.0x10^{-5}$	$9.7x10^{-7}$	$1.4x10^{-5}$	$8.6x10^{-7}$	$1.6x10^{-5}$
Wisconsin	$1.9x10^{-6}$	$2.5x10^{-5}$	$1.5x10^{-6}$	$2.9x10^{-5}$	$1.5x10^{-6}$	$1.9x10^{-5}$	$1.1x10^{-6}$	$2.2x10^{-5}$
Wyoming	$1.4x10^{-6}$	$9.5x10^{-6}$	$2.3x10^{-7}$	$1.1x10^{-5}$	$1.0x10^{-6}$	$7.2x10^{-6}$	$1.8x10^{-7}$	$8.4x10^{-6}$

Sample calculation: Rural route from Kewaunee to ORNL through Ohio, Rail-Steel cask
RADTRAN output (unit risk): $5.55x10^{-8}$ person-rem (from Figure II-13)
Population density: 14.8 persons/km^2
Population multiplier: 1.0
Route segment length: 200.6 km
Population (collective) dose = $5.55x10^{-8}$ * 14.8 * 1.0 * 200.6 = $1.65x10^{-4}$ person-rem
Convert to SI units: $1.65x10^{-4}$ person-rem * 0.01person-Sv/person-rem = $1.65x10^{-6}$ person-Sv

Table II-11 Collective Doses to Residents along the Route (Person-Sv) from Rail Transportation; Shipment Origin—Maine Yankee

DEST. AND ROUTES	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
ORNL								
Kentucky	3.2×10^{-6}	1.6×10^{-5}	7.0×10^{-7}	2.0×10^{-5}	2.4×10^{-6}	1.3×10^{-5}	5.3×10^{-7}	1.5×10^{-5}
Maine	9.3×10^{-7}	1.6×10^{-5}	6.2×10^{-7}	1.7×10^{-5}	7.1×10^{-7}	1.2×10^{-5}	4.7×10^{-7}	1.3×10^{-5}
Massachusetts	1.3×10^{-6}	2.9×10^{-5}	1.8×10^{-6}	3.2×10^{-5}	1.0×10^{-6}	2.2×10^{-5}	1.4×10^{-6}	2.4×10^{-5}
New Hampshire	3.8×10^{-7}	7.5×10^{-6}	2.5×10^{-7}	8.2×10^{-6}	2.9×10^{-7}	5.7×10^{-6}	1.9×10^{-7}	6.2×10^{-6}
New York	4.8×10^{-6}	5.2×10^{-5}	1.8×10^{-6}	5.9×10^{-5}	3.7×10^{-6}	4.0×10^{-5}	1.4×10^{-6}	4.5×10^{-5}
Ohio	3.6×10^{-6}	4.9×10^{-5}	3.2×10^{-6}	5.6×10^{-5}	2.7×10^{-6}	3.7×10^{-5}	2.5×10^{-6}	4.3×10^{-5}
Pennsylvania	4.2×10^{-7}	9.4×10^{-6}	5.1×10^{-7}	1.0×10^{-5}	3.2×10^{-7}	7.2×10^{-6}	3.9×10^{-7}	7.9×10^{-6}
Tennessee	7.4×10^{-7}	5.0×10^{-6}	4.1×10^{-8}	5.7×10^{-6}	5.6×10^{-7}	3.8×10^{-6}	3.1×10^{-8}	4.4×10^{-6}
Vermont	6.7×10^{-8}	5.2×10^{-7}	0	5.9×10^{-7}	5.1×10^{-8}	4.0×10^{-7}	0	4.5×10^{-7}
DEAF SMITH								
Illinois	1.5×10^{-6}	2.7×10^{-5}	2.4×10^{-6}	3.1×10^{-5}	1.1×10^{-6}	2.0×10^{-5}	1.8×10^{-6}	2.3×10^{-5}
Indiana	2.1×10^{-6}	1.1×10^{-5}	5.4×10^{-7}	1.4×10^{-5}	1.6×10^{-6}	8.6×10^{-6}	4.1×10^{-7}	1.1×10^{-5}
Iowa	3.0×10^{-7}	6.2×10^{-7}	3.1×10^{-8}	9.5×10^{-7}	2.3×10^{-7}	4.8×10^{-7}	2.4×10^{-8}	7.2×10^{-7}
Kansas	2.0×10^{-6}	1.8×10^{-5}	7.9×10^{-7}	2.1×10^{-5}	1.5×10^{-6}	1.4×10^{-5}	6.0×10^{-7}	1.6×10^{-5}
Maine	9.3×10^{-7}	1.6×10^{-5}	6.2×10^{-7}	1.7×10^{-5}	7.1×10^{-7}	1.2×10^{-5}	4.7×10^{-7}	1.3×10^{-5}
Massachusetts	1.3×10^{-6}	2.9×10^{-5}	1.8×10^{-6}	3.2×10^{-5}	1.0×10^{-6}	2.2×10^{-5}	1.4×10^{-6}	2.4×10^{-5}
Missouri	1.2×10^{-6}	7.0×10^{-6}	2.4×10^{-7}	8.5×10^{-6}	9.2×10^{-7}	5.4×10^{-6}	1.8×10^{-7}	6.5×10^{-6}
New Hampshire	3.8×10^{-7}	7.5×10^{-6}	2.5×10^{-7}	8.2×10^{-6}	2.9×10^{-7}	5.7×10^{-6}	1.9×10^{-7}	6.2×10^{-6}
New York	4.8×10^{-6}	5.2×10^{-5}	1.8×10^{-6}	5.9×10^{-5}	3.7×10^{-6}	4.0×10^{-5}	1.4×10^{-6}	4.5×10^{-5}
Ohio	2.5×10^{-6}	3.2×10^{-5}	2.3×10^{-6}	3.6×10^{-5}	1.9×10^{-6}	2.4×10^{-5}	1.7×10^{-6}	2.8×10^{-5}
Oklahoma	4.3×10^{-7}	3.9×10^{-6}	4.9×10^{-8}	4.5×10^{-6}	3.3×10^{-7}	3.0×10^{-6}	3.7×10^{-8}	3.4×10^{-6}
Pennsylvania	4.1×10^{-7}	9.3×10^{-6}	4.9×10^{-7}	1.0×10^{-5}	3.1×10^{-7}	7.1×10^{-6}	3.7×10^{-7}	7.8×10^{-6}
Texas	7.3×10^{-7}	5.1×10^{-6}	1.2×10^{-7}	6.0×10^{-6}	5.6×10^{-7}	3.9×10^{-6}	9.4×10^{-8}	4.5×10^{-6}
Vermont	6.7×10^{-8}	5.2×10^{-7}	0	5.9×10^{-7}	5.1×10^{-8}	4.0×10^{-7}	0	4.5×10^{-7}
HANFORD								
Idaho	9.8×10^{-7}	6.7×10^{-6}	9.6×10^{-8}	7.8×10^{-6}	7.5×10^{-7}	5.1×10^{-6}	7.3×10^{-8}	5.9×10^{-6}
Illinois	1.4×10^{-6}	2.1×10^{-5}	2.2×10^{-6}	2.4×10^{-5}	1.0×10^{-6}	1.6×10^{-5}	1.7×10^{-6}	1.9×10^{-5}
Indiana	2.1×10^{-6}	1.1×10^{-5}	5.4×10^{-7}	1.4×10^{-5}	1.6×10^{-6}	8.6×10^{-6}	4.1×10^{-7}	1.1×10^{-5}
Maine	9.3×10^{-7}	1.6×10^{-5}	6.2×10^{-7}	1.7×10^{-5}	7.1×10^{-7}	1.2×10^{-5}	4.7×10^{-7}	1.3×10^{-5}
Massachusetts	1.3×10^{-6}	2.9×10^{-5}	1.8×10^{-6}	3.2×10^{-5}	1.0×10^{-6}	2.2×10^{-5}	1.4×10^{-6}	2.4×10^{-5}
Minnesota	3.2×10^{-6}	2.9×10^{-5}	1.2×10^{-6}	3.4×10^{-5}	2.4×10^{-6}	2.2×10^{-5}	8.9×10^{-7}	2.6×10^{-5}
Montana	2.2×10^{-6}	1.3×10^{-5}	1.4×10^{-7}	1.6×10^{-5}	1.7×10^{-6}	1.0×10^{-5}	1.1×10^{-7}	1.2×10^{-5}
New Hampshire	3.8×10^{-7}	7.5×10^{-6}	2.5×10^{-7}	8.2×10^{-6}	2.9×10^{-7}	5.7×10^{-6}	1.9×10^{-7}	6.2×10^{-6}
New York	4.8×10^{-6}	5.2×10^{-5}	2.2×10^{-6}	5.9×10^{-5}	3.7×10^{-6}	4.0×10^{-5}	1.7×10^{-6}	4.5×10^{-5}
North Dakota	1.0×10^{-6}	8.2×10^{-6}	2.6×10^{-7}	9.5×10^{-6}	7.7×10^{-7}	6.3×10^{-6}	2.0×10^{-7}	7.2×10^{-6}
Ohio	2.5×10^{-6}	3.2×10^{-5}	2.3×10^{-6}	3.6×10^{-5}	1.9×10^{-6}	2.4×10^{-5}	1.7×10^{-6}	2.8×10^{-5}

Table II-11 Collective Doses to Residents along the Route (Person-Sv) from Rail Transportation; Shipment Origin Maine Yankee (continued)

DEST. AND ROUTES	Rail-Lead				Rail-Steel			
	Rural	**Suburban**	**Urban**	**Total**	**Rural**	**Suburban**	**Urban**	**Total**
HANFORD (cont.)								
Pennsylvania	$4.1x10^{-7}$	$9.3x10^{-6}$	$4.9x10^{-7}$	$1.0x10^{-5}$	$3.1x10^{-7}$	$7.1x10^{-6}$	$3.7x10^{-7}$	$7.8x10^{-6}$
Vermont	$6.7x10^{-8}$	$5.2x10^{-7}$	0	$5.9x10^{-7}$	$5.1x10^{-8}$	$4.0x10^{-7}$	0	$4.5x10^{-7}$
Washington	$1.1x10^{-6}$	$1.3x10^{-5}$	$6.5x10^{-7}$	$1.5x10^{-5}$	$8.5x10^{-7}$	$1.0x10^{-5}$	$5.0x10^{-7}$	$1.2x10^{-5}$
Wisconsin	$1.7x10^{-6}$	$8.2x10^{-6}$	$3.7x10^{-7}$	$1.0x10^{-5}$	$1.3x10^{-6}$	$6.2x10^{-6}$	$2.8x10^{-7}$	$7.8x10^{-6}$
SKULL VALLEY								
Colorado	$1.3x10^{-7}$	$5.8x10^{-7}$	0	$7.1x10^{-7}$	$1.0x10^{-7}$	$4.4x10^{-7}$	0	$5.4x10^{-7}$
Illinois	$1.3x10^{-6}$	$2.2x10^{-5}$	$2.7x10^{-6}$	$2.6x10^{-5}$	$1.0x10^{-6}$	$1.7x10^{-5}$	$2.0x10^{-6}$	$2.0x10^{-5}$
Indiana	$2.1x10^{-6}$	$1.1x10^{-5}$	$5.4x10^{-7}$	$1.4x10^{-5}$	$1.6x10^{-6}$	$8.6x10^{-6}$	$4.1x10^{-7}$	$1.1x10^{-5}$
Iowa	$4.0x10^{-6}$	$1.8x10^{-5}$	$3.4x10^{-7}$	$2.2x10^{-5}$	$3.1x10^{-6}$	$1.4x10^{-5}$	$2.6x10^{-7}$	$1.7x10^{-5}$
Maine	$9.6x10^{-7}$	$1.6x10^{-5}$	$1.6x10^{-7}$	$1.7x10^{-5}$	$7.3x10^{-7}$	$1.2x10^{-5}$	$1.2x10^{-7}$	$1.3x10^{-5}$
Massachusetts	$6.5x10^{-7}$	$2.8x10^{-5}$	$8.5x10^{-7}$	$3.0x10^{-5}$	$4.9x10^{-7}$	$2.1x10^{-5}$	$6.5x10^{-7}$	$2.3x10^{-5}$
Nebraska	$4.2x10^{-6}$	$2.0x10^{-5}$	$6.2x10^{-7}$	$2.5x10^{-5}$	**$3.2x10^{-6}$**	$1.5x10^{-5}$	$4.7x10^{-7}$	$1.9x10^{-5}$
New Hampshire	$1.1x10^{-7}$	$3.7x10^{-6}$	$4.9x10^{-8}$	$3.8x10^{-6}$	$8.6x10^{-8}$	$2.8x10^{-6}$	$3.7x10^{-8}$	$2.9x10^{-6}$
New York	$4.8x10^{-6}$	$5.2x10^{-5}$	$1.8x10^{-6}$	$5.9x10^{-5}$	$3.7x10^{-6}$	$4.0x10^{-5}$	$1.4x10^{-6}$	$4.5x10^{-5}$
Ohio	$2.5x10^{-6}$	$3.2x10^{-5}$	$2.3x10^{-6}$	$3.6x10^{-5}$	$1.9x10^{-6}$	$2.4x10^{-5}$	$1.7x10^{-6}$	$2.8x10^{-5}$
Pennsylvania	$4.1x10^{-7}$	$9.3x10^{-6}$	$4.9x10^{-7}$	$1.0x10^{-5}$	$3.1x10^{-7}$	$7.1x10^{-6}$	$3.7x10^{-7}$	$7.8x10^{-6}$
Utah	$1.3x10^{-6}$	$1.8x10^{-5}$	$1.1x10^{-6}$	$2.0x10^{-5}$	$9.7x10^{-7}$	$1.4x10^{-5}$	$8.6x10^{-7}$	$1.6x10^{-5}$
Vermont	$6.7x10^{-8}$	$5.2x10^{-7}$	0	$5.9x10^{-7}$	$5.1x10^{-8}$	$4.0x10^{-7}$	0	$4.5x10^{-7}$
Wyoming	$1.4x10^{-6}$	$9.5x10^{-6}$	$2.3x10^{-7}$	$1.1x10^{-5}$	$1.0x10^{-6}$	$7.2x10^{-6}$	$1.8x10^{-7}$	$8.4x10^{-6}$

Sample calculation: Rural route from Maine Yankee to Skull Valley through Nebraska, Rail-Steel cask
RADTRAN output (unit risk): $5.55x10^{-8}$ person-rem (from Figure II-13)
Population density: 9.3 persons/km^2
Population multiplier: 1.0
Route segment length: 621.7 km
Population (collective) dose = $5.55x10^{-8}$ * 9.3 * 1.0 * 621.7 = $3.21x10^{-4}$ person-rem
Convert to SI units: $3.17x10^{-4}$ person-rem * 0.01person-Sv/person-rem = $3.21x10^{-6}$ person-Sv

Table II-12 Collective Doses to Residents along the Route (Person-Sv) from Truck-DU; Shipment Origin—Maine Yankee

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Connecticut	2.8×10^{-7}	1.5×10^{-5}	5.6×10^{-7}	1.2×10^{-7}	1.58×10^{-5}
	Maine	4.0×10^{-7}	7.3×10^{-6}	6.6×10^{-8}	1.5×10^{-8}	7.74×10^{-6}
	Maryland	3.4×10^{-8}	1.3×10^{-6}	1.3×10^{-8}	3.0×10^{-9}	1.33×10^{-6}
	Massachusetts	2.7×10^{-7}	1.2×10^{-5}	2.0×10^{-7}	4.5×10^{-8}	1.23×10^{-5}
	New Hampshire	4.7×10^{-8}	1.5×10^{-6}	7.4×10^{-9}	1.6×10^{-9}	1.59×10^{-6}
	New Jersey	1.8×10^{-7}	6.4×10^{-6}	3.1×10^{-7}	6.9×10^{-8}	6.92×10^{-6}
	New York	**2.1×10^{-9}**	1.7×10^{-6}	5.1×10^{-7}	1.1×10^{-7}	2.28×10^{-6}
	Pennsylvania	1.1×10^{-6}	1.3×10^{-5}	1.6×10^{-7}	3.6×10^{-8}	1.42×10^{-5}
	Tennessee	7.6×10^{-7}	9.4×10^{-6}	7.9×10^{-8}	1.8×10^{-8}	1.02×10^{-5}
	Virginia	1.8×10^{-6}	1.9×10^{-5}	1.3×10^{-7}	2.8×10^{-8}	2.13×10^{-5}
	West Virginia	1.1×10^{-7}	2.6×10^{-6}	7.2×10^{-9}	1.6×10^{-9}	2.73×10^{-6}
DEAF SMITH	Connecticut	1.5×10^{-6}	9.9×10^{-6}	9.5×10^{-8}	2.1×10^{-8}	1.15×10^{-5}
	Maine	2.9×10^{-7}	1.5×10^{-5}	5.7×10^{-7}	1.3×10^{-7}	1.60×10^{-5}
	Maryland	4.0×10^{-7}	6.8×10^{-6}	3.7×10^{-8}	8.2×10^{-9}	7.28×10^{-6}
	Massachusetts	3.4×10^{-8}	1.3×10^{-6}	1.3×10^{-8}	3.0×10^{-9}	1.34×10^{-6}
	New Hampshire	2.7×10^{-7}	1.2×10^{-5}	2.0×10^{-7}	4.5×10^{-8}	1.23×10^{-5}
	New Jersey	4.7×10^{-8}	1.5×10^{-6}	7.4×10^{-9}	1.6×10^{-9}	1.59×10^{-6}
	New York	2.4×10^{-7}	8.6×10^{-6}	1.7×10^{-7}	3.9×10^{-8}	9.03×10^{-6}
	Oklahoma	2.4×10^{-8}	4.3×10^{-6}	2.0×10^{-7}	4.5×10^{-8}	4.53×10^{-6}
	Pennsylvania	1.6×10^{-6}	8.4×10^{-6}	1.1×10^{-7}	2.4×10^{-8}	1.02×10^{-5}
	Tennessee	9.4×10^{-7}	1.1×10^{-5}	1.3×10^{-7}	3.0×10^{-8}	1.19×10^{-5}
	Texas	2.9×10^{-6}	2.5×10^{-5}	3.9×10^{-7}	8.7×10^{-8}	2.83×10^{-5}
	Virginia	3.9×10^{-7}	2.2×10^{-6}	9.5×10^{-8}	2.1×10^{-8}	2.76×10^{-6}
	West Virginia	1.7×10^{-6}	1.8×10^{-5}	1.2×10^{-7}	2.7×10^{-8}	2.03×10^{-5}

Table II-12 Collective Doses to Residents along the Route (Person-Sv) from Truck-DU; Shipment Origin Maine Yankee (continued)

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
HANFORD	Connecticut	2.7×10^{-7}	1.1×10^{-5}	3.0×10^{-7}	6.6×10^{-9}	1.18×10^{-5}
	Idaho	1.4×10^{-6}	6.6×10^{-6}	9.6×10^{-8}	2.1×10^{-9}	8.15×10^{-6}
	Illinois	8.1×10^{-7}	6.5×10^{-6}	1.2×10^{-7}	2.7×10^{-9}	7.42×10^{-6}
	Indiana	8.3×10^{-7}	7.1×10^{-6}	1.2×10^{-7}	2.6×10^{-9}	8.10×10^{-6}
	Iowa	1.9×10^{-6}	6.8×10^{-6}	5.8×10^{-8}	1.3×10^{-9}	8.73×10^{-6}
	Maine	4.0×10^{-7}	6.8×10^{-6}	3.7×10^{-8}	8.2×10^{-10}	7.28×10^{-6}
	Massachusetts	2.8×10^{-7}	1.2×10^{-5}	2.1×10^{-7}	4.6×10^{-9}	1.28×10^{-5}
	Nebraska	2.0×10^{-6}	5.4×10^{-6}	8.8×10^{-8}	2.0×10^{-9}	7.48×10^{-6}
	New Hampshire	4.7×10^{-8}	1.5×10^{-6}	7.4×10^{-9}	1.6×10^{-10}	1.59×10^{-6}
	New York	2.8×10^{-7}	5.7×10^{-6}	5.0×10^{-8}	1.1×10^{-9}	6.05×10^{-6}
	Ohio	1.3×10^{-6}	1.2×10^{-5}	1.7×10^{-7}	3.8×10^{-9}	1.39×10^{-5}
	Oregon	8.1×10^{-7}	2.9×10^{-6}	2.6×10^{-8}	5.8×10^{-10}	3.74×10^{-6}
	Pennsylvania	2.0×10^{-6}	1.1×10^{-5}	8.2×10^{-8}	1.8×10^{-9}	1.33×10^{-5}
	Utah	6.3×10^{-7}	4.0×10^{-6}	1.8×10^{-8}	4.1×10^{-10}	4.68×10^{-6}
	Washington	8.9×10^{-8}	8.4×10^{-7}	5.0×10^{-8}	1.1×10^{-9}	9.92×10^{-7}
	Wyoming	9.1×10^{-7}	3.6×10^{-6}	3.5×10^{-8}	7.8×10^{-10}	4.54×10^{-6}
SKULL VALLEY	Connecticut	2.7×10^{-7}	1.1×10^{-5}	3.0×10^{-7}	6.6×10^{-9}	1.18×10^{-5}
	Illinois	8.1×10^{-7}	6.5×10^{-6}	1.2×10^{-7}	2.7×10^{-9}	7.42×10^{-6}
	Indiana	8.3×10^{-7}	7.1×10^{-6}	1.2×10^{-7}	2.6×10^{-9}	8.10×10^{-6}
	Iowa	1.9×10^{-6}	6.8×10^{-6}	5.8×10^{-8}	1.3×10^{-9}	8.73×10^{-6}
	Maine	4.0×10^{-7}	6.8×10^{-6}	3.7×10^{-8}	8.2×10^{-10}	7.28×10^{-6}
	Massachusetts	2.7×10^{-7}	1.2×10^{-5}	2.0×10^{-7}	4.5×10^{-9}	1.23×10^{-5}
	Nebraska	2.0×10^{-6}	5.4×10^{-6}	8.8×10^{-8}	2.0×10^{-9}	7.48×10^{-6}
	New Hampshire	4.7×10^{-8}	1.5×10^{-6}	7.4×10^{-9}	1.6×10^{-10}	1.59×10^{-6}
	New York	2.8×10^{-7}	5.7×10^{-6}	5.0×10^{-8}	1.1×10^{-9}	6.05×10^{-6}
	Ohio	1.3×10^{-6}	1.2×10^{-5}	1.7×10^{-7}	3.8×10^{-9}	1.39×10^{-5}
	Pennsylvania	2.0×10^{-6}	1.1×10^{-5}	8.2×10^{-8}	1.8×10^{-9}	1.33×10^{-5}
	Utah	5.2×10^{-7}	4.6×10^{-6}	2.0×10^{-7}	4.5×10^{-9}	5.40×10^{-6}
	Wyoming	9.1×10^{-7}	3.6×10^{-6}	3.5×10^{-8}	7.8×10^{-10}	4.54×10^{-6}

Sample calculation: Rural route from Maine Yankee to ORNL through New York, Truck-DU cask
RADTRAN output (unit risk): 3.05×10^{-8} person-rem (from Table II-8)
Population density: 4.4 persons/km^2
Population multiplier: 1.0
Route segment length: 1.6 km
Population (collective) dose = $3.05\times10^{-8} * 4.4 * 1.0 * 1.6 = 2.15\times10^{-7}$ person-rem
Convert to SI units: 2.15×10^{-7} person-rem * 0.01person-Sv/person-rem = 2.15×10^{-9} person-Sv

Table II-13 Collective Doses to Residents along the Route (Person-Sv) from Truck-DU; Shipment Origin—Indian Point

DESTINATION	ROUTES	Rural	Suburban	Urban	U. Rush Hour	Total
ORNL	Maryland	5.4×10^{-8}	2.1×10^{-6}	3.0×10^{-9}	3.0×10^{-9}	1.3×10^{-6}
	New Jersey	3.9×10^{-7}	1.4×10^{-5}	3.9×10^{-8}	3.9×10^{-8}	9.0×10^{-6}
	New York	7.5×10^{-8}	7.0×10^{-6}	4.9×10^{-8}	4.9×10^{-8}	4.7×10^{-6}
	Pennsylvania	9.4×10^{-7}	1.1×10^{-5}	3.0×10^{-8}	3.0×10^{-8}	1.2×10^{-5}
	Tennessee	7.9×10^{-7}	9.7×10^{-6}	1.5×10^{-8}	1.5×10^{-8}	1.1×10^{-5}
	Virginia	1.7×10^{-6}	1.8×10^{-5}	2.7×10^{-8}	2.7×10^{-8}	2.0×10^{-5}
	West Virginia	1.1×10^{-7}	2.6×10^{-6}	1.6×10^{-9}	1.6×10^{-9}	2.7×10^{-6}
DEAF SMITH	Arkansas	2.3×10^{-6}	1.6×10^{-5}	2.2×10^{-8}	2.2×10^{-8}	1.1×10^{-5}
	Maryland	5.4×10^{-8}	2.1×10^{-6}	3.0×10^{-9}	3.0×10^{-9}	1.3×10^{-6}
	New Jersey	3.9×10^{-7}	1.4×10^{-5}	3.9×10^{-8}	3.9×10^{-8}	9.0×10^{-6}
	New York	4.7×10^{-8}	4.3×10^{-6}	4.9×10^{-8}	4.9×10^{-8}	4.7×10^{-6}
	Oklahoma	1.7×10^{-6}	8.7×10^{-6}	2.6×10^{-8}	2.6×10^{-8}	1.0×10^{-5}
	Pennsylvania	9.4×10^{-7}	1.1×10^{-5}	3.0×10^{-8}	3.0×10^{-8}	1.2×10^{-5}
	Tennessee	2.9×10^{-6}	2.5×10^{-5}	3.9×10^{-7}	8.7×10^{-8}	2.8×10^{-5}
	Texas	2.9×10^{-6}	2.5×10^{-5}	8.7×10^{-8}	2.1×10^{-8}	2.8×10^{-6}
	Virginia	3.9×10^{-7}	2.2×10^{-6}	2.1×10^{-8}	2.7×10^{-8}	2.0×10^{-5}
	West Virginia	1.7×10^{-6}	1.8×10^{-5}	2.7×10^{-8}	1.6×10^{-9}	2.7×10^{-6}
HANFORD	Idaho	**1.4×10^{-6}**	6.6×10^{-6}	1.6×10^{-9}	2.1×10^{-8}	8.1×10^{-6}
	Illinois	7.8×10^{-7}	6.3×10^{-6}	2.1×10^{-8}	2.7×10^{-8}	7.4×10^{-6}
	Indiana	8.6×10^{-7}	7.1×10^{-6}	2.7×10^{-8}	2.6×10^{-8}	8.1×10^{-6}
	Iowa	1.9×10^{-6}	6.8×10^{-6}	2.6×10^{-8}	1.3×10^{-8}	8.7×10^{-6}
	Nebraska	2.0×10^{-6}	5.4×10^{-6}	1.3×10^{-8}	2.0×10^{-8}	7.5×10^{-6}
	New Jersey	2.6×10^{-7}	6.7×10^{-6}	2.0×10^{-8}	3.2×10^{-8}	7.1×10^{-6}
	New York	4.7×10^{-8}	4.3×10^{-6}	3.2×10^{-8}	4.9×10^{-8}	4.7×10^{-6}
	Ohio	1.3×10^{-6}	1.2×10^{-5}	4.9×10^{-8}	3.8×10^{-8}	1.4×10^{-5}
	Oregon	8.9×10^{-7}	3.2×10^{-6}	3.8×10^{-8}	5.8×10^{-9}	4.1×10^{-6}
	Pennsylvania	1.8×10^{-6}	8.7×10^{-6}	5.8×10^{-9}	8.1×10^{-9}	1.1×10^{-5}
	Utah	6.3×10^{-7}	4.0×10^{-6}	8.1×10^{-9}	4.1×10^{-9}	4.7×10^{-6}
	Washington	8.9×10^{-8}	8.4×10^{-7}	4.1×10^{-9}	1.1×10^{-8}	9.9×10^{-7}
	Wyoming	9.1×10^{-7}	3.6×10^{-6}	1.1×10^{-8}	7.8×10^{-9}	4.5×10^{-6}
SKULL VALLEY	Illinois	8.1×10^{-7}	6.5×10^{-6}	7.8×10^{-9}	2.7×10^{-8}	7.4×10^{-6}
	Indiana	8.3×10^{-7}	7.1×10^{-6}	2.7×10^{-8}	2.6×10^{-8}	8.1×10^{-6}
	Iowa	1.9×10^{-6}	6.8×10^{-6}	2.6×10^{-8}	1.3×10^{-8}	8.7×10^{-6}
	Nebraska	2.0×10^{-6}	5.4×10^{-6}	1.3×10^{-8}	2.0×10^{-8}	7.5×10^{-6}
	New Jersey	2.6×10^{-7}	6.7×10^{-6}	2.0×10^{-8}	3.2×10^{-8}	7.1×10^{-6}
	New York	4.7×10^{-8}	4.3×10^{-6}	3.2×10^{-8}	4.9×10^{-8}	4.7×10^{-6}
	Ohio	1.3×10^{-6}	1.2×10^{-5}	4.9×10^{-8}	3.8×10^{-8}	1.4×10^{-5}
	Pennsylvania	1.8×10^{-6}	8.7×10^{-6}	3.8×10^{-8}	8.1×10^{-9}	1.1×10^{-5}
	Utah	5.1×10^{-7}	4.6×10^{-6}	8.1×10^{-9}	4.5×10^{-8}	5.4×10^{-6}
	Wyoming	9.1×10^{-7}	3.6×10^{-6}	4.5×10^{-8}	7.8×10^{-9}	4.5×10^{-6}

Sample calculation: Rural route from Indian Point to Hanford through Idaho, truck cask
RADTRAN output (unit risk): 3.05×10^{-8} person-rem (from Table II-8)
Population density: 11.3 persons/km^2
Population multiplier: 1.133
Route segment length: 357 km
Population (collective) dose = 3.05×10^{-8}*11.3*1.133*357= 1.39×10^{-4} person-rem
Convert to SI units: 1.39×10^{-4} person-rem*0.01person-Sv/person-rem = 1.39×10^{-6} person-Sv

Table II-14 Collective Doses to Residents along the Route (Person-Sv) from Truck-DU; Shipment Origin INL

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Colorado	1.0×10^{-6}	4.7×10^{-6}	1.4×10^{-7}	3.1×10^{-8}	5.86×10^{-6}
	Idaho	6.3×10^{-7}	2.7×10^{-6}	2.4×10^{-8}	5.4×10^{-9}	3.32×10^{-6}
	Illinois	9.1×10^{-7}	6.0×10^{-6}	1.9×10^{-8}	4.3×10^{-9}	6.98×10^{-6}
	Kansas	1.6×10^{-6}	6.7×10^{-6}	1.1×10^{-7}	2.5×10^{-8}	8.50×10^{-6}
	Kentucky	5.8×10^{-7}	2.3×10^{-6}	1.8×10^{-9}	4.1×10^{-10}	2.86×10^{-6}
	Missouri	1.2×10^{-6}	1.5×10^{-5}	3.1×10^{-7}	6.9×10^{-8}	1.65×10^{-5}
	Tennessee	1.4×10^{-6}	8.8×10^{-6}	1.1×10^{-7}	2.5×10^{-8}	1.03×10^{-5}
	Utah	6.6×10^{-7}	4.2×10^{-6}	1.8×10^{-8}	4.1×10^{-9}	4.89×10^{-6}
	Wyoming	7.9×10^{-7}	2.5×10^{-6}	2.5×10^{-8}	5.5×10^{-9}	3.30×10^{-6}
DEAF SMITH	Colorado	1.3×10^{-6}	1.6×10^{-5}	4.4×10^{-7}	9.8×10^{-8}	1.74×10^{-5}
	Idaho	6.3×10^{-7}	2.7×10^{-6}	2.4×10^{-8}	5.4×10^{-9}	3.32×10^{-6}
	New Mexico	1.2×10^{-6}	5.6×10^{-6}	1.8×10^{-7}	4.1×10^{-8}	7.02×10^{-6}
	Texas	5.3×10^{-8}	9.4×10^{-8}	0	0	1.47×10^{-7}
	Utah	6.6×10^{-7}	**4.2×10^{-6}**	1.8×10^{-8}	4.1×10^{-9}	4.89×10^{-6}
	Wyoming	7.9×10^{-7}	2.5×10^{-6}	2.5×10^{-8}	5.5×10^{-9}	3.30×10^{-6}
HANFORD	Idaho	1.7×10^{-6}	8.7×10^{-6}	1.1×10^{-7}	2.5×10^{-8}	1.05×10^{-5}
	Oregon	3.0×10^{-6}	2.5×10^{-8}	5.6×10^{-10}	5.6×10^{-9}	3.83×10^{-6}
	Washington	8.4×10^{-7}	5.0×10^{-8}	1.1×10^{-9}	1.1×10^{-8}	9.92×10^{-7}
SKULL VALLEY	Idaho	6.3×10^{-7}	2.7×10^{-6}	2.4×10^{-8}	5.4×10^{-9}	3.32×10^{-6}
	Utah	6.0×10^{-7}	7.4×10^{-6}	2.4×10^{-7}	5.4×10^{-8}	8.33×10^{-6}

Sample calculation: Suburban route from INL to Deaf Smith through Utah, Truck-DU cask
RADTRAN output (unit risk): 2.73×10^{-8} person-rem (from Table II-8)
Population density: 260.1persons/km^2
Population multiplier: 1.102
Route segment length: 53.4 km
Population (collective) dose = 2.73×10^{-8} * 260.1 * 1.102 * 53.4= 4.18×10^{-4} person-rem
Convert to SI units: 4.18×10^{-4} person-rem * 0.01person-Sv/person-rem = 4.18×10^{-6} person-Sv

Table II-15 Collective Doses to Residents along the Route (Person-Sv) from Truck-DU; Shipment Origin—Kewaunee

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Illinois	2.1×10^{-7}	9.8×10^{-6}	3.6×10^{-7}	7.9×10^{-8}	1.05×10^{-5}
	Indiana	1.3×10^{-6}	1.2×10^{-5}	2.0×10^{-7}	4.5×10^{-8}	1.31×10^{-5}
	Kentucky	1.2×10^{-6}	1.0×10^{-5}	1.3×10^{-7}	2.9×10^{-8}	1.17×10^{-5}
	Ohio	5.9×10^{-8}	8.3×10^{-7}	1.1×10^{-8}	2.4×10^{-9}	9.04×10^{-7}
	Tennessee	3.6×10^{-7}	6.2×10^{-6}	7.8×10^{-8}	1.7×10^{-8}	6.67×10^{-6}
	Wisconsin	9.8×10^{-7}	7.6×10^{-6}	3.5×10^{-7}	7.8×10^{-8}	9.00×10^{-6}
DEAF SMITH	Illinois	7.5×10^{-7}	3.3×10^{-6}	1.2×10^{-8}	2.7×10^{-9}	4.02×10^{-6}
	Iowa	1.4×10^{-6}	7.4×10^{-6}	6.3×10^{-8}	1.4×10^{-8}	8.90×10^{-6}
	Kansas	1.0×10^{-6}	6.9×10^{-6}	1.6×10^{-7}	3.6×10^{-8}	8.16×10^{-6}
	Missouri	6.5×10^{-7}	6.1×10^{-6}	6.2×10^{-8}	1.2×10^{-8}	6.81×10^{-6}
	Oklahoma	1.1×10^{-6}	6.0×10^{-6}	1.8×10^{-9}	2.1×10^{-8}	7.18×10^{-6}
	Texas	3.9×10^{-7}	2.2×10^{-6}	9.5×10^{-8}	2.1×10^{-8}	2.76×10^{-6}
	Wisconsin	1.2×10^{-6}	7.6×10^{-6}	2.8×10^{-7}	6.1×10^{-8}	9.20×10^{-6}
HANFORD	Idaho	2.1×10^{-7}	4.1×10^{-6}	6.0×10^{-8}	1.1×10^{-8}	4.03×10^{-6}
	Minnesota	1.7×10^{-6}	2.7×10^{-6}	1.3×10^{-8}	2.9×10^{-9}	4.46×10^{-6}
	Montana	2.1×10^{-6}	8.2×10^{-6}	1.0×10^{-7}	2.2×10^{-8}	1.03×10^{-5}
	South Dakota	1.5×10^{-6}	3.9×10^{-6}	3.1×10^{-8}	6.9×10^{-9}	5.35×10^{-6}
	Washington	1.0×10^{-6}	9.2×10^{-6}	2.0×10^{-7}	4.5×10^{-8}	1.05×10^{-5}
	Wisconsin	2.1×10^{-6}	1.1×10^{-5}	2.7×10^{-7}	6.0×10^{-8}	1.38×10^{-5}
	Wyoming	5.6×10^{-7}	1.6×10^{-6}	2.2×10^{-8}	4.9×10^{-9}	2.21×10^{-6}
SKULL VALLEY	Illinois	7.5×10^{-7}	3.3×10^{-6}	1.2×10^{-8}	2.7×10^{-9}	4.02×10^{-6}
	Iowa	1.9×10^{-6}	6.8×10^{-6}	5.8×10^{-8}	1.3×10^{-8}	8.73×10^{-6}
	Nebraska	2.0×10^{-6}	5.4×10^{-6}	8.8×10^{-8}	2.0×10^{-8}	7.48×10^{-6}
	Utah	5.1×10^{-7}	4.6×10^{-6}	2.0×10^{-7}	4.5×10^{-8}	5.39×10^{-6}
	Wisconsin	1.2×10^{-6}	7.6×10^{-6}	**2.8×10^{-7}**	6.1×10^{-8}	9.20×10^{-6}
	Wyoming	9.1×10^{-7}	3.6×10^{-6}	3.5×10^{-8}	7.8×10^{-9}	4.54×10^{-6}

Sample calculation: Urban route from Kewaunee to Skull Valley through Wisconsin, Truck-DU cask, not during rush hour
RADTRAN output (unit risk): 5.22×10^{-10} person rem (from Table II-8)
Population density: 2,660 persons/km^2
Population multiplier: 1.0
Route segment length: 19.9 km
Population (collective) dose = 5.22×10^{-10} * 2,660 * 1.0 * 19.9 = 2.76×10^{-5} person-rem
Convert to SI units: 2.76×10^{-5} person-rem * 0.01person-Sv/person-rem = 2.76×10^{-7} person-Sv

Collective dose is best used in making comparisons, for example, in comparing the risks of routine transportation along different routes. All collective doses modeled are of the order of 10^{-5} person-Sv (1 person-mrem) or less. The tables show that, in general, urban residents sustain a slightly larger dose from rail transportation than from truck transportation on the same State route, even though urban population densities are similar. For example, for the Maine urban segment of the Maine Yankee-to-ORNL route, the collective dose differs depending on the transportation type used:

- The truck route urban population density is 2,706 persons/km^2 (7009 persons/mi^2) and the collective dose is 6.6×10^{-8} person-Sv (6.6×10^{-3} person-mrem).

- The rail route urban population density is 2,527 persons/km^2, (6545 persons/mi^2), but the collective dose is 6.2×10^{-7} person-Sv (6.2×10^{-2} person-mrem) from the Rail-Lead cask—almost 10 times larger than the dose from the Truck-DU cask, even though the external dose rates from the two casks are nearly the same.

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

II.5.4.2 Doses to Occupants of Vehicles Sharing the Route

The dose to occupants of vehicles sharing a highway route (typically referred to as the on-link dose) consists of the sum of three components:

(1) dose to persons in vehicles traveling in the opposite direction to the shipment
(2) dose to persons in vehicles traveling in the same direction as the shipment
(3) dose to persons in passing vehicles

In the case of rail, there is a dose only to occupants of railcars (the rail analog to highway vehicles) traveling in the opposite direction, since passing on parallel track is rarely the case. RADTRAN uses Equation II-4 to calculate the dose to occupants of vehicles traveling in the opposite direction. The result is as follows:

(II-5) $$D_{opp} = 2*\left(\frac{N*PPV}{V}\right)*\frac{Qk_0DR_v}{V}\left[f_\gamma*I_\gamma+f_n*I_n\right]$$

Where:

D_{opp} is the dose to occupants of railcars traveling in the opposite direction
N is the number of railcars sharing the route
PPV is the number of passengers per railcar

The other terms are defined as in preceding equations. The factor of 2 is included to account for the vehicle moving toward the radioactive cargo and then away from it. An additional factor of (*N*PPV/V*) accounts for the dose to people in the oncoming vehicle, which is assumed to be traveling at the same speed as the cargo. *N* is the number of oncoming vehicles per hour and *P* is the number of persons per vehicle.

<u>Rail</u>

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

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

The dose to occupants of other trains depends on train speed and the external dose rate from the spent fuel cask. Train speeds are available only for the entire United States, and not for each State. Therefore, Table II-17 shows the doses to occupants of trains that share the route with either a loaded Rail-Lead cask or a loaded Rail-Steel cask for the rural, suburban, and urban segments of each entire route, rather than State by State.

Table II-16 Collective Doses (Person-Sv) to Occupants of Trains Sharing the Route

SHIPMENT ORIGIN/ DESTINATION	Rail-Lead Cask				Rail-Steel Cask			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
MAINE YANKEE								
ORNL	$2.0x10^{-5}$	$1.2x10^{-5}$	$7.5x10^{-6}$	$4.0x10^{-5}$	$1.5x10^{-5}$	$9.3x10^{-6}$	$5.6x10^{-6}$	$3.0x10^{-5}$
DEAF SMITH	$3.8x10^{-5}$	$1.3x10^{-5}$	$9.7x10^{-6}$	$6.1x10^{-5}$	$2.9x10^{-5}$	$1.0x10^{-5}$	$7.4x10^{-6}$	$4.6x10^{-5}$
HANFORD	$6.2x10^{-5}$	$1.7x10^{-5}$	$1.6x10^{-5}$	$9.0x10^{-5}$	$4.7x10^{-5}$	$1.3x10^{-5}$	$1.2x10^{-5}$	$6.8x10^{-5}$
SKULL VALLEY	$4.8x10^{-5}$	$1.6x10^{-5}$	**$9.6x10^{-6}$**	$7.4x10^{-5}$	$3.6x10^{-5}$	$1.2x10^{-5}$	$7.3x10^{-6}$	$5.5x10^{-5}$
KEWAUNEE								
ORNL	$1.4x10^{-5}$	$7.0x10^{-6}$	$5.8x10^{-6}$	$2.7x10^{-5}$	$1.0x10^{-5}$	$5.3x10^{-6}$	$4.4x10^{-6}$	$2.0x10^{-5}$
DEAF SMITH	$2.4x10^{-5}$	$5.2x10^{-6}$	$5.1x10^{-6}$	$3.4x10^{-5}$	$1.8x10^{-5}$	$4.0x10^{-6}$	$3.9x10^{-6}$	$2.6x10^{-5}$
HANFORD	$4.2x10^{-5}$	$6.7x10^{-6}$	$2.8x10^{-6}$	$5.2x10^{-5}$	$3.2x10^{-5}$	$5.1x10^{-6}$	$2.1x10^{-6}$	$3.9x10^{-5}$
SKULL VALLEY	$3.5x10^{-5}$	$7.8x10^{-6}$	$5.8x10^{-6}$	$4.9x10^{-5}$	$2.7x10^{-5}$	$5.9x10^{-6}$	$4.4x10^{-6}$	$3.7x10^{-5}$
INDIAN POINT								
ORNL	$9.2x10^{-6}$	$8.1x10^{-6}$	$9.6x10^{-6}$	$2.7x10^{-5}$	$7.0x10^{-6}$	$6.1x10^{-6}$	$7.2x10^{-6}$	$2.0x10^{-5}$
DEAF SMITH	$3.6x10^{-5}$	$1.1x10^{-5}$	$9.4x10^{-6}$	$5.6x10^{-5}$	$2.8x10^{-5}$	$8.2x10^{-6}$	$7.1x10^{-6}$	$4.3x10^{-5}$
HANFORD	$6.0x10^{-5}$	$1.4x10^{-5}$	$1.1x10^{-5}$	$8.5x10^{-5}$	$4.6x10^{-5}$	$1.1x10^{-5}$	$8.0x10^{-6}$	$6.5x10^{-5}$
SKULL VALLEY	$4.8x10^{-5}$	$1.3x10^{-5}$	$1.1x10^{-5}$	$6.5x10^{-5}$	$3.6x10^{-5}$	$1.0x10^{-5}$	$8.0x10^{-6}$	$4.9x10^{-5}$
INL								
ORNL	$4.6x10^{-5}$	$7.1x10^{-6}$	$3.4x10^{-6}$	$5.7x10^{-5}$	$3.5x10^{-5}$	$5.4x10^{-6}$	$2.6x10^{-6}$	$4.3x10^{-5}$
DEAF SMITH	$2.7x10^{-5}$	$3.2x10^{-6}$	$1.9x10^{-6}$	$3.2x10^{-5}$	$2.1x10^{-5}$	$2.5x10^{-6}$	$1.4x10^{-6}$	$2.5x10^{-5}$
HANFORD	$1.5x10^{-5}$	$1.7x10^{-6}$	$9.3x10^{-7}$	$1.8x10^{-5}$	$1.2x10^{-5}$	$1.3x10^{-6}$	$7.0x10^{-7}$	$1.4x10^{-5}$
SKULL VALLEY	$5.5x10^{-6}$	$1.5x10^{-6}$	$1.2x10^{-6}$	$8.2x10^{-6}$	$4.2x10^{-6}$	$1.1x10^{-6}$	$9.0x10^{-7}$	$6.2x10^{-6}$

Sample calculation: Urban segment from Maine Yankee to Skull Valley, Rail-Lead cask
RADTRAN output (unit risk): $4.63 \text{ x}10^{-6}$ person-rem (from Figure II-13)
Route urban length: 207 km (from Table II-4)
Population (collective) dose = $4.63 \text{ x}10^{-6} * 207 = 9.58\text{x}10^{-4}$ person-rem
Convert to SI units: $9.58\text{x}10^{-4}$ person-rem * 0.01person-Sv/person-rem = $9.58\text{x}10^{-6}$ person-Sv

The RADTRAN calculation incorporates the number of occupants of other trains, the train speed, and the railcars per hour. This value is then multiplied by the total rural, suburban, and urban kilometers, respectively, of the route.

Truck

Vehicle density data for large semidetached trailer trucks traveling U.S. Interstates and primary highways is available and well qualified. Every State records traffic counts on major (and most minor) highways and publishes these routinely. Researchers have used average vehicle density data from each of the 10 U.S. Environmental Protection Agency (EPA) regions (Weiner et al., 2009, Appendix D). This study used the EPA regions because they include all of the "lower 48" U.S. States (Alaska and Hawaii are included in EPA Region 10 but are not considered in this risk assessment because no spent fuel will be shipped to or from those States). Table II-18 shows the 10 EPA regions and the average vehicle density data for the region, except for region 10, where the average vehicle density is from the three states listed.

Table II-17 States Comprising the 10 EPA Regions

Region	States Included in Region	Vehicles per Hour		
		Rural	Suburban	Urban
1	Connecticut, Massachusetts, Maine, New Hampshire, Rhode Island, Vermont	439	726	2,129
2	New Jersey, New York	1,015	2,094	4,163
3	Delaware. Maryland, Pennsylvania, Virginia, West Virginia	2,056	3,655	5,748
4	Alabama, Florida, Georgia, Kentucky, Mississippi, North Carolina, South Carolina, Tennessee	1,427	2,776	5,611
5	Illinois, Indiana, Michigan, Minnesota, Ohio, Wisconsin	1,200	2,466	4,408
6	Arkansas, Louisiana, New Mexico, Oklahoma, Texas	897	1,498	3,003
7	Iowa, Kansas, Missouri, Nebraska	926	1,610	2,463
8	Colorado, Montana, North Dakota, South Dakota, Utah, Wyoming	795	1,958	3,708
9	Arizona, California, Nevada	1,421	3,732	7,517
10	Idaho, Oregon, Washington	1,123	2,670	5,624

The calculation of doses to occupants sharing the highway route with the radioactive materials truck includes the dose to vehicles passing the radioactive cargo and vehicles in an adjoining lane, as well as vehicles traveling in the opposite direction. Equations 28 and 34 in Neuhauser et al. (2000) describe this calculation.

Figure II-14 is the diagram accompanying these equations and shows the parameters used in the calculation. Table II-1 gives the parameter values.

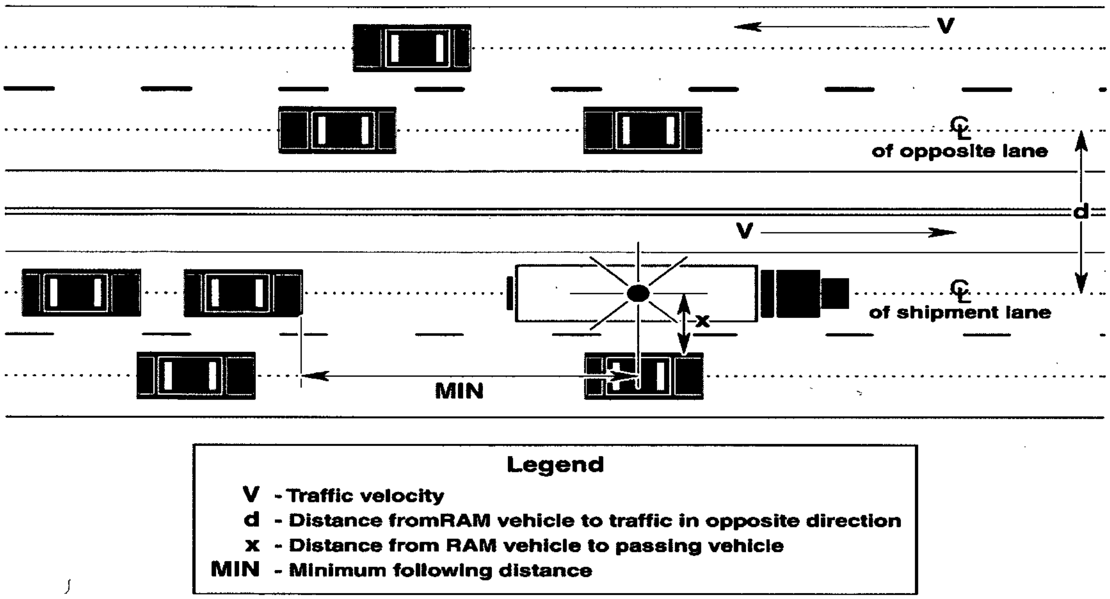

Figure II-13 Parameters for calculating doses to occupants of highway vehicles sharing the route with the radioactive shipment
(from Figure 3-2 of Neuhauser et al., 2000)

Tables II-18 to II-21 show the doses to individuals in vehicles sharing the highway route with the truck carrying a loaded Truck-DU cask. The RADTRAN calculation incorporates the number of occupants of other vehicles, the vehicle speed, and the vehicles per hour. This value is then multiplied by the rural, suburban, and urban kilometers, respectively, of each State transited.

Table II-18 Collective Doses to Persons Sharing the Route (Person-Sv) from Truck-DU; Shipment Origin—Maine Yankee

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Connecticut	**2.0×10^{-6}**	9.2×10^{-6}	9.2×10^{-6}	8.5×10^{-6}	2.9×10^{-5}
	Maine	2.9×10^{-6}	6.7×10^{-6}	1.1×10^{-6}	1.0×10^{-6}	1.2×10^{-5}
	Maryland	1.3×10^{-6}	4.9×10^{-6}	9.0×10^{-7}	8.3×10^{-7}	8.0×10^{-6}
	Massachusetts	1.7×10^{-6}	8.7×10^{-6}	3.4×10^{-6}	3.2×10^{-6}	1.7×10^{-5}
	New Hampshire	3.7×10^{-7}	1.4×10^{-6}	1.9×10^{-7}	1.8×10^{-7}	2.1×10^{-6}
	New Jersey	4.5×10^{-6}	1.6×10^{-5}	6.6×10^{-6}	6.1×10^{-6}	3.3×10^{-5}
	New York	7.5×10^{-7}	2.1×10^{-6}	1.3×10^{-5}	1.2×10^{-5}	2.7×10^{-5}
	Pennsylvania	3.0×10^{-5}	4.8×10^{-5}	7.0×10^{-6}	6.5×10^{-6}	9.2×10^{-5}
	Tennessee	1.7×10^{-5}	3.2×10^{-5}	4.2×10^{-6}	3.9×10^{-6}	5.6×10^{-5}
	Virginia	6.4×10^{-5}	9.3×10^{-5}	6.2×10^{-6}	5.7×10^{-6}	1.7×10^{-4}
	West Virginia	2.8×10^{-6}	1.2×10^{-5}	4.5×10^{-7}	4.1×10^{-7}	1.5×10^{-5}
DEAF SMITH	Arkansas	3.1×10^{-5}	2.1×10^{-5}	2.8×10^{-6}	2.6×10^{-6}	5.8×10^{-5}
	Connecticut	2.0×10^{-6}	9.2×10^{-6}	9.2×10^{-6}	8.5×10^{-6}	2.9×10^{-5}
	Maine	2.9×10^{-6}	6.8×10^{-6}	7.3×10^{-7}	6.8×10^{-7}	1.1×10^{-5}
	Maryland	1.3×10^{-6}	4.9×10^{-6}	9.0×10^{-7}	8.3×10^{-7}	8.0×10^{-6}
	Massachusetts	1.7×10^{-6}	8.7×10^{-6}	3.4×10^{-6}	3.2×10^{-6}	1.7×10^{-5}
	New Hampshire	3.7×10^{-7}	1.4×10^{-6}	1.9×10^{-7}	1.8×10^{-7}	2.1×10^{-6}
	New Jersey	4.5×10^{-6}	1.6×10^{-5}	6.6×10^{-6}	6.1×10^{-6}	3.3×10^{-5}
	New York	7.5×10^{-7}	6.8×10^{-6}	6.9×10^{-6}	6.4×10^{-6}	2.1×10^{-5}
	Oklahoma	4.2×10^{-5}	1.6×10^{-5}	2.8×10^{-6}	2.6×10^{-6}	6.4×10^{-5}
	Pennsylvania	3.0×10^{-5}	4.8×10^{-5}	7.0×10^{-6}	6.5×10^{-6}	9.2×10^{-5}
	Tennessee	7.8×10^{-5}	8.6×10^{-5}	2.0×10^{-5}	1.8×10^{-5}	2.0×10^{-4}
	Texas	2.2×10^{-5}	3.1×10^{-6}	2.4×10^{-6}	2.2×10^{-6}	2.9×10^{-5}
	Virginia	6.4×10^{-5}	9.3×10^{-5}	6.2×10^{-6}	5.7×10^{-6}	1.7×10^{-4}
	West Virginia	2.8×10^{-6}	1.2×10^{-5}	4.5×10^{-7}	4.1×10^{-7}	1.5×10^{-5}

Table II-18 Collective Doses to Persons Sharing the Route (Person-Sv) from Truck-DU; Shipment Origin—Maine Yankee (continued)

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
HANFORD	Connecticut	$1.7x10^{-6}$	$8.0x10^{-6}$	$5.1x10^{-6}$	$4.7x10^{-6}$	$2.0x10^{-5}$
	Idaho	$4.4x10^{-5}$	$2.3x10^{-5}$	$4.6x10^{-6}$	$4.2x10^{-6}$	$7.6x10^{-5}$
	Illinois	$2.4x10^{-5}$	$2.0x10^{-5}$	$5.0x10^{-6}$	$4.6x10^{-6}$	$5.4x10^{-5}$
	Indiana	$1.8x10^{-5}$	$2.6x10^{-5}$	$4.6x10^{-6}$	$4.3x10^{-6}$	$5.3x10^{-5}$
	Iowa	$4.0x10^{-5}$	$1.7x10^{-5}$	$1.4x10^{-6}$	$1.3x10^{-6}$	$6.0x10^{-5}$
	Maine	$2.9x10^{-6}$	$6.8x10^{-6}$	$7.3x10^{-7}$	$6.8x10^{-7}$	$1.1x10^{-5}$
	Massachusetts	$1.7x10^{-6}$	$8.7x10^{-6}$	$3.4x10^{-6}$	$3.2x10^{-6}$	$1.7x10^{-5}$
	Nebraska	$6.7x10^{-5}$	$1.3x10^{-5}$	$1.9x10^{-6}$	$1.8x10^{-6}$	$8.4x10^{-5}$
	New Hampshire	$3.7x10^{-7}$	$1.4x10^{-6}$	$1.9x10^{-7}$	$1.8x10^{-7}$	$2.1x10^{-6}$
	New York	$2.5x10^{-6}$	$4.6x10^{-6}$	$1.1x10^{-6}$	$9.9x10^{-7}$	$9.2x10^{-6}$
	Ohio	$2.8x10^{-5}$	$6.9x10^{-5}$	$4.0x10^{-6}$	$3.7x10^{-6}$	$1.0x10^{-4}$
	Oregon	$3.7x10^{-5}$	$9.5x10^{-6}$	$1.4x10^{-6}$	$1.3x10^{-6}$	$4.9x10^{-5}$
	Pennsylvania	$8.7x10^{-5}$	$6.9x10^{-5}$	$4.0x10^{-6}$	$3.7x10^{-6}$	$1.6x10^{-4}$
	Utah	$1.6x10^{-5}$	$1.1x10^{-5}$	$6.2x10^{-7}$	$5.7x10^{-7}$	$2.8x10^{-5}$
	Washington	$7.6x10^{-6}$	$2.1x10^{-6}$	$2.6x10^{-6}$	$2.4x10^{-6}$	$1.5x10^{-5}$
	Wyoming	$7.5x10^{-5}$	$1.0x10^{-5}$	$2.1x10^{-6}$	$2.0x10^{-6}$	$8.9x10^{-5}$
SKULL VALLEY	Connecticut	$1.7x10^{-6}$	$8.0x10^{-6}$	$5.1x10^{-6}$	$4.7x10^{-6}$	$2.0x10^{-5}$
	Illinois	$2.4x10^{-5}$	$2.0x10^{-5}$	$5.0x10^{-6}$	$4.6x10^{-6}$	$5.4x10^{-5}$
	Indiana	$1.8x10^{-5}$	$2.6x10^{-5}$	$4.6x10^{-6}$	$4.3x10^{-6}$	$5.3x10^{-5}$
	Iowa	$4.0x10^{-5}$	$1.7x10^{-5}$	$1.4x10^{-6}$	$1.3x10^{-6}$	$6.0x10^{-5}$
	Maine	$2.9x10^{-6}$	$6.8x10^{-6}$	$7.3x10^{-7}$	$6.8x10^{-7}$	$1.1x10^{-5}$
	Massachusetts	$1.7x10^{-6}$	$8.7x10^{-6}$	$3.4x10^{-6}$	$3.2x10^{-6}$	$1.7x10^{-5}$
	Nebraska	$6.7x10^{-5}$	$1.3x10^{-5}$	$1.9x10^{-6}$	$1.8x10^{-6}$	$8.4x10^{-5}$
	New Hampshire	$3.7x10^{-7}$	$4.8x10^{-6}$	$4.8x10^{-7}$	$4.4x10^{-6}$	$1.0x10^{-5}$
	New York	$5.8x10^{-6}$	$1.3x10^{-5}$	$2.1x10^{-6}$	$1.9x10^{-6}$	$2.3x10^{-5}$
	Ohio	$2.8x10^{-5}$	$4.1x10^{-5}$	$7.3x10^{-6}$	$6.7x10^{-6}$	$8.3x10^{-5}$
	Pennsylvania	$8.7x10^{-5}$	$6.9x10^{-5}$	$4.0x10^{-6}$	$3.7x10^{-6}$	$1.6x10^{-4}$
	Utah	$1.8x10^{-5}$	$8.1x10^{-6}$	$6.1x10^{-6}$	$5.6x10^{-6}$	$3.8x10^{-5}$
	Wyoming	$7.5x10^{-5}$	$1.0x10^{-5}$	$2.1x10^{-6}$	$2.0x10^{-6}$	$8.9x10^{-5}$

Sample Calculation: Rural segment from Maine Yankee to ORNL through Connecticut (EPA Region 1)
Unit risk (From Table II-8): $4.80x10^{-8}$ Sv
Rural route segment length: 40.7 km
Dose to occupants of vehicles sharing the route: $4.80x10^{-8}$ * 40.7 = $1.95x10^{-6}$

Table II-19 Collective Doses to Persons Sharing the Route (Person-Sv) from Truck-DU; Shipment Origin—Indian Point

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban RH	Total
ORNL	Maryland	1.3×10^{-6}	4.9×10^{-6}	9.0×10^{-7}	8.3×10^{-7}	7.9×10^{-6}
	New Jersey	4.5×10^{-6}	1.6×10^{-5}	6.6×10^{-6}	6.1×10^{-6}	3.3×10^{-5}
	New York	1.3×10^{-6}	6.5×10^{-6}	7.6×10^{-6}	7.0×10^{-6}	2.2×10^{-5}
	Pennsylvania	3.0×10^{-5}	4.8×10^{-5}	7.0×10^{-6}	6.5×10^{-6}	9.2×10^{-5}
	Tennessee	1.7×10^{-5}	3.4×10^{-5}	3.8×10^{-6}	3.5×10^{-6}	5.8×10^{-5}
	Virginia	6.4×10^{-5}	9.3×10^{-5}	6.2×10^{-6}	5.7×10^{-6}	1.7×10^{-4}
	West Virginia	6.4×10^{-5}	1.2×10^{-5}	4.5×10^{-7}	4.1×10^{-7}	7.7×10^{-5}
DEAF SMITH	Arkansas	3.1×10^{-5}	2.1×10^{-5}	2.8×10^{-6}	2.6×10^{-6}	5.7×10^{-5}
	Maryland	1.3×10^{-6}	4.9×10^{-6}	9.0×10^{-7}	8.3×10^{-7}	7.9×10^{-6}
	New Jersey	4.5×10^{-6}	1.6×10^{-5}	6.6×10^{-6}	6.1×10^{-6}	3.3×10^{-5}
	New York	1.3×10^{-6}	6.5×10^{-6}	7.6×10^{-6}	7.0×10^{-6}	2.2×10^{-5}
	Oklahoma	4.2×10^{-5}	1.6×10^{-5}	2.8×10^{-6}	2.6×10^{-6}	6.3×10^{-5}
	Pennsylvania	3.0×10^{-5}	4.8×10^{-5}	7.0×10^{-6}	6.5×10^{-6}	9.2×10^{-5}
	Tennessee	7.8×10^{-5}	8.6×10^{-5}	2.0×10^{-5}	1.8×10^{-5}	2.0×10^{-4}
	Texas	2.2×10^{-5}	3.1×10^{-6}	2.4×10^{-6}	2.2×10^{-6}	3.0×10^{-5}
	Virginia	6.4×10^{-5}	9.3×10^{-5}	6.2×10^{-6}	5.7×10^{-6}	1.7×10^{-4}
	West Virginia	2.8×10^{-6}	1.2×10^{-5}	4.5×10^{-7}	4.1×10^{-7}	1.6×10^{-5}
HANFORD	Idaho	4.4×10^{-5}	2.3×10^{-5}	4.6×10^{-6}	**4.2×10^{-6}**	7.6×10^{-5}
	Illinois	2.4×10^{-5}	2.0×10^{-5}	5.0×10^{-6}	4.6×10^{-6}	5.4×10^{-5}
	Indiana	1.8×10^{-5}	2.6×10^{-5}	4.6×10^{-6}	4.3×10^{-6}	5.3×10^{-5}
	Iowa	4.0×10^{-5}	1.7×10^{-5}	1.4×10^{-6}	1.3×10^{-6}	6.0×10^{-5}
	Nebraska	6.7×10^{-5}	1.3×10^{-5}	1.9×10^{-6}	1.8×10^{-6}	8.4×10^{-5}
	New Jersey	4.8×10^{-6}	1.3×10^{-5}	5.6×10^{-6}	5.2×10^{-6}	2.9×10^{-5}
	New York	1.3×10^{-6}	6.5×10^{-6}	7.6×10^{-6}	7.0×10^{-6}	2.2×10^{-5}
	Ohio	1.5×10^{-6}	7.6×10^{-6}	8.1×10^{-6}	7.4×10^{-6}	2.5×10^{-5}
	Oregon	3.7×10^{-5}	9.5×10^{-6}	1.4×10^{-6}	1.3×10^{-6}	4.9×10^{-5}
	Pennsylvania	8.0×10^{-5}	5.7×10^{-5}	2.2×10^{-6}	2.0×10^{-6}	1.4×10^{-4}
	Utah	1.6×10^{-5}	1.1×10^{-5}	6.2×10^{-7}	5.7×10^{-7}	2.8×10^{-5}
	Washington	7.6×10^{-6}	2.1×10^{-6}	2.6×10^{-6}	2.4×10^{-6}	1.5×10^{-5}
	Wyoming	7.5×10^{-5}	1.0×10^{-5}	2.1×10^{-6}	2.0×10^{-6}	8.9×10^{-5}
SKULL VALLEY	Illinois	2.4×10^{-5}	2.0×10^{-5}	5.0×10^{-6}	4.6×10^{-6}	5.4×10^{-5}
	Indiana	1.8×10^{-5}	2.6×10^{-5}	4.6×10^{-6}	4.3×10^{-6}	5.3×10^{-5}
	Iowa	4.0×10^{-5}	1.7×10^{-5}	1.4×10^{-6}	1.3×10^{-6}	6.0×10^{-5}
	Nebraska	6.7×10^{-5}	1.3×10^{-5}	1.9×10^{-6}	1.8×10^{-6}	8.4×10^{-5}
	New Jersey	5.6×10^{-6}	1.5×10^{-5}	5.9×10^{-6}	5.5×10^{-6}	3.2×10^{-5}
	New York	1.5×10^{-6}	7.6×10^{-6}	8.1×10^{-6}	7.4×10^{-6}	2.5×10^{-5}
	Ohio	2.8×10^{-5}	4.1×10^{-5}	7.3×10^{-6}	6.7×10^{-6}	8.3×10^{-5}
	Pennsylvania	8.0×10^{-5}	5.7×10^{-5}	2.2×10^{-6}	2.0×10^{-6}	1.4×10^{-4}
	Utah	1.8×10^{-5}	8.1×10^{-6}	6.1×10^{-6}	5.6×10^{-6}	3.8×10^{-5}
	Wyoming	7.5×10^{-5}	1.0×10^{-5}	2.1×10^{-6}	2.0×10^{-6}	8.9×10^{-5}

Sample Calculation: Urban rush hour segment from Indian Point to Hanford through Idaho (EPA Region 10)
Unit risk (From Table II-8): 5.80×10^{-7} Sv
Urban route segment length: 7.3 km
Dose to occupants of vehicles sharing the route: $5.80 \times 10^{-7} * 7.3 = 4.23 \times 10^{-6}$

Table II-20 Collective Doses to Persons Sharing the Route (Person-Sv) from Truck-DU; Shipment Origin—INL

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Colorado	3.1×10^{-5}	1.1×10^{-5}	4.0×10^{-6}	3.7×10^{-6}	5.0×10^{-5}
	Idaho	2.2×10^{-5}	8.0×10^{-6}	1.3×10^{-6}	1.2×10^{-6}	3.3×10^{-5}
	Illinois	2.5×10^{-5}	2.4×10^{-5}	1.1×10^{-6}	1.0×10^{-6}	5.1×10^{-5}
	Kansas	6.2×10^{-5}	1.4×10^{-5}	2.7×10^{-6}	2.5×10^{-6}	8.1×10^{-5}
	Kentucky	1.8×10^{-5}	1.1×10^{-5}	1.2×10^{-7}	1.2×10^{-7}	2.9×10^{-5}
	Missouri	2.5×10^{-5}	2.3×10^{-5}	7.2×10^{-6}	6.7×10^{-6}	6.2×10^{-5}
	Tennessee	3.3×10^{-5}	3.5×10^{-5}	5.2×10^{-6}	4.8×10^{-6}	7.8×10^{-5}
	Utah	1.3×10^{-5}	1.1×10^{-5}	6.2×10^{-7}	5.7×10^{-7}	2.5×10^{-5}
	Wyoming	7.5×10^{-5}	1.0×10^{-5}	2.1×10^{-6}	2.0×10^{-6}	8.9×10^{-5}
DEAF SMITH	Colorado	3.9×10^{-5}	3.6×10^{-5}	1.9×10^{-5}	1.8×10^{-5}	1.1×10^{-4}
	Idaho	2.2×10^{-5}	8.0×10^{-6}	1.3×10^{-6}	1.2×10^{-6}	3.3×10^{-5}
	New Mexico	6.4×10^{-5}	9.8×10^{-6}	4.8×10^{-6}	4.4×10^{-6}	8.3×10^{-5}
	Texas	7.7×10^{-6}	1.7×10^{-7}	0	0	7.9×10^{-6}
	Utah	1.3×10^{-5}	**1.1×10^{-5}**	6.2×10^{-7}	5.7×10^{-7}	2.5×10^{-5}
	Wyoming	7.0×10^{-5}	7.6×10^{-6}	1.5×10^{-6}	1.4×10^{-6}	8.1×10^{-5}
HANFORD	Idaho	5.5×10^{-5}	6.3×10^{-5}	5.4×10^{-6}	5.0×10^{-6}	1.3×10^{-4}
	Oregon	3.7×10^{-5}	2.0×10^{-5}	1.4×10^{-6}	1.3×10^{-6}	6.0×10^{-5}
	Washington	7.6×10^{-6}	2.1×10^{-6}	2.6×10^{-6}	2.4×10^{-6}	1.5×10^{-5}
SKULL VALLEY	Idaho	2.2×10^{-5}	8.0×10^{-6}	1.3×10^{-6}	1.2×10^{-6}	4.2×10^{-5}
	Utah	1.5×10^{-5}	1.5×10^{-5}	7.2×10^{-6}	6.6×10^{-6}	4.4×10^{-5}

Sample Calculation: Suburban segment from INL to Deaf Smith through Utah (EPA Region 8)
Unit risk (From Table II-8): 2.20×10^{-7} Sv
Suburban route segment length: 53.4 km
Dose to occupants of vehicles sharing the route: $2.15 \times 10^{-7} * 53.4 = 1.148 \times 10^{-5}$

Table II-21 Collective Doses to Persons Sharing the Route (Person-Sv) from Truck-DU; Shipment Origin—Kewaunee

DESTINATION	ROUTES	1. Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Illinois	3.7x10^{-6}	2.0x10^{-5}	1.4x10^{-5}	1.3x10^{-5}	5.1x10^{-5}
	Indiana	3.3x10^{-5}	3.8x10^{-5}	8.3x10^{-6}	7.7x10^{-6}	8.7x10^{-5}
	Kentucky	2.7x10^{-5}	4.3x10^{-5}	7.2x10^{-6}	6.7x10^{-6}	8.4x10^{-5}
	Ohio	1.4x10^{-6}	2.5x10^{-6}	5.4x10^{-7}	5.0x10^{-7}	4.9x10^{-6}
	Tennessee	1.1x10^{-5}	1.8x10^{-5}	4.4x10^{-6}	4.1x10^{-6}	3.8x10^{-5}
	Wisconsin	2.0x10^{-5}	2.1x10^{-5}	1.3x10^{-5}	1.2x10^{-5}	6.6x10^{-5}
DEAF SMITH	Illinois	2.0x10^{-5}	1.2x10^{-5}	5.9x10^{-7}	5.4x10^{-7}	3.3x10^{-5}
	Iowa	3.2x10^{-5}	1.6x10^{-5}	1.6x10^{-6}	1.4x10^{-6}	5.1x10^{-5}
	Kansas	2.9x10^{-5}	1.2x10^{-5}	3.5x10^{-6}	3.2x10^{-6}	4.8x10^{-5}
	Missouri	1.4x10^{-5}	1.1x10^{-5}	1.3x10^{-6}	1.2x10^{-6}	2.8x10^{-5}
	Oklahoma	3.4x10^{-5}	1.1x10^{-5}	2.8x10^{-6}	2.6x10^{-6}	5.0x10^{-5}
	Texas	2.2x10^{-5}	3.1x10^{-6}	2.4x10^{-6}	2.2x10^{-6}	3.0x10^{-5}
	Wisconsin	2.5x10^{-5}	2.3x10^{-5}	9.8x10^{-6}	9.0x10^{-6}	6.7x10^{-5}
HANFORD	Idaho	9.3x10^{-6}	1.1x10^{-5}	3.0x10^{-6}	2.8x10^{-6}	2.6x10^{-5}
	Minnesota	5.2x10^{-5}	1.3x10^{-5}	5.4x10^{-7}	5.0x10^{-7}	6.6x10^{-5}
	Montana	9.6x10^{-5}	3.0x10^{-5}	5.4x10^{-6}	5.0x10^{-6}	1.4x10^{-4}
	South Dakota	5.3x10^{-5}	1.2x10^{-5}	1.0x10^{-6}	9.5x10^{-7}	6.7x10^{-5}
	Washington	4.6x10^{-5}	3.0x10^{-5}	1.1x10^{-5}	1.0x10^{-5}	9.7x10^{-5}
	Wisconsin	4.6x10^{-5}	4.0x10^{-5}	9.9x10^{-6}	9.2x10^{-6}	1.1x10^{-4}
	Wyoming	4.0x10^{-5}	4.1x10^{-6}	1.4x10^{-6}	1.3x10^{-6}	4.7x10^{-5}
SKULL VALLEY	Illinois	2.0x10^{-5}	1.2x10^{-5}	5.9x10^{-7}	5.4x10^{-7}	3.3x10^{-5}
	Iowa	4.0x10^{-5}	1.7x10^{-5}	1.4x10^{-6}	1.3x10^{-6}	6.0x10^{-5}
	Nebraska	6.7x10^{-5}	1.3x10^{-5}	1.9x10^{-6}	1.8x10^{-6}	8.4x10^{-5}
	Utah	2.4x10^{-5}	1.0x10^{-5}	8.8x10^{-6}	8.1x10^{-6}	4.4x10^{-5}
	Wisconsin	2.5x10^{-5}	2.3x10^{-5}	**9.8x10^{-6}**	9.0x10^{-6}	6.7x10^{-5}
	Wyoming	7.5x10^{-5}	1.0x10^{-5}	2.1x10^{-6}	2.0x10^{-6}	8.9x10^{-5}

Sample Calculation: Urban segment from Kewaunee to Skull Valley through Wisconsin (EPA Region 5), not during rush hour
Unit risk (From Table II-8): 4.90x10^{-7} Sv
Urban route segment length: 19.9 km
Dose to occupants of vehicles sharing the route: 4.90x10^{-7} * 19.9 = 9.75x10^{-6}

II.5.4.3 Doses from Stopped Vehicles

Rail

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

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

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

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

The collective dose to people living near a classification yard is calculated by multiplying the average dose from the rail cask to an individual living near a classification yard, as shown in Table II-7, by the population density between 200 and 800 meters (656 feet and ½ mile) from the railyard. The population density is obtained from WebTRAGIS, and the integration from 200 to 800 meters (656 feet and ½ mile) (Table II-2) is performed by RADTRAN.

Most train stops along any route are shown in the WebTRAGIS output for that route. Table II-22 shows the stops on the rail route from Maine Yankee to Hanford as an example.

Table II-22 Example of Rail Stops on the Maine Yankee-to-Hanford Rail Route

Stop	Reason	Route type (R, S, U)[a] and State	Time (hours)
Classification	Initial classification	S, ME	27
1	Railroad transfer (short line to ST)	S, ME	4.0
2	Railroad transfer (ST to CSXT)	R, NY	4.0
3	Railroad transfer (CSXT to IHB)	S, IL	2.0
4	Railroad transfer (IHB to BNSF)	S, IL	<<1
5	Railroad transfer (BNSF to UP)	S, WA	<<1
Classification	Final classification	S, WA	27

[a] Determined from the WebTRAGIS output

Railyard worker collective doses can then be calculated for Stops 1 and 2 in Table II-22. Parameter values are from Table II-22 and the classification yard dose above:

Dose: $(4/27)*(1.5\times10^{-5}) = 2.2\times10^{-6}$ person-Sv (0.22 person-mrem) for the Rail-Lead cask

Dose: $(4/27)*(1.1\times10^{-5}) = 1.6\times10^{-6}$ person-Sv (0.16 person-mrem) for the Rail-Steel cask

The doses for stop 3 would be ½ of these values.

The above equations include a factor of 4/27 because the classification stop doses are calculated by RADTRAN for activities lasting a total of 27 hours, and the in-transit stops are for only 4 hours.

The average dose to an individual living 200 to 800 meters (656 feet and ½ mile) from a classification yard, as calculated by RADTRAN, is as follows:

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

Collective doses to residents near a yard (a classification yard or railroad stop) are then calculated from the following general expression:

Dose (person-Sv) = (population density) * (dose/h to resident near yard) * (stop time) * (shielding factor)

Thus, for a suburban population density of 373.8 persons/km^2 (968 persons/mi^2) (the suburban population density through Maine along the Maine Yankee-to-Hanford route) living near Stop 1 in Table II-22, the dose can be calculated as follows:

$$\text{Dose} = (373.8\ \text{persons/km}^2) * (3.5 \times 10^{-7}\ \text{Sv-km}^2/\text{h}) * (4\ \text{h}) * 0.87 = 4.6\times10^{-4}\ \text{person-Sv}$$

Table II-23 gives results for the stops.

Table II-23 Doses at Rail Stops on the Maine Yankee-to-Hanford Rail Route

Stop	Route type (R, S, U)[a] and State	Time (hours)	Railyard worker dose (person-Sv)[b]		Residents near stop (person-Sv)	
			Rail-Lead	Rail-Steel	Rail-Lead	Rail-Steel
Classification, origin	S, ME	27	1.5×10^{-5}	1.1×10^{-5}	2.3×10^{-5}	1.8×10^{-5}
1	S, ME	4.0	2.16×10^{-6}	1.61×10^{-6}	4.6×10^{-4}	3.5×10^{-4}
2	R, NY	4.0	2.16×10^{-6}	1.61×10^{-6}	2.5×10^{-5}	1.9×10^{-5}
3	S, IL	2.0	1.08×10^{-6}	8.05×10^{-7}	2.9×10^{-4}	2.2×10^{-4}
Classification, destination	S, WA	27	1.5×10^{-5}	1.1×10^{-5}	1.9×10^{-5}	1.4×10^{-5}

[a] Determined from the WebTRAGIS output
[b] The yard worker dose depends only on the length of time the railcar is stopped in the yard, independent of population density and shielding factor.

Truck

Doses at truck stops are calculated differently. There are two types of receptors at a truck stop, in addition to the truck crew—residents who live near the stop and people who share the stop with the refueling truck. Griego et al. (1996) conducted some time and motion studies at a number of truck stops. They found that the average number of people at a stop between the gas pumps and the nearest building was 6.9, the average distance from the fuel pump to the nearest building was 15 meters, and the longest refueling time for a large semidetached trailer truck was 0.83 hour (50 minutes). With these parameters, the collective dose to the people sharing the stop would be 2.3×10^{-4} person-Sv (23 person-mrem) (Table II-8). The relationship between the collective dose and the number of receptors is not linear in this case.

The collective dose to residents near the stop is calculated in the same way as for rail transportation, using data in Table II-8, the population density of the region around the stop, and the stop time:

$$\text{Dose (person-Sv)} = (\text{population density}) * (\text{dose/h to resident near stop}) * (\text{stop time})$$

Thus, for a rural population density of 15.4 persons/km^2 (40 persons/mi^2) (the average along the Maine Yankee-to-Hanford truck route), the following dose can be calculated:

$$\text{Dose/stop} = (15.4\ \text{persons/km}^2) * (3.3\times10^{-8}\ \text{Sv-km}^2/\text{h}) * (0.83\ \text{h}) = 4.2\times10^{-5}\ \text{person-Sv}$$

The population density used in the calculation is the density around the truck stop; appropriate residential shielding factors are used in the calculation. Unlike a train, the truck will stop several times on any truck route to fill the fuel tanks. Very large trucks generally carry two 80-gallon tanks each and stop for fuel when the tanks are half empty. A semidetached trailer truck carrying a Truck-DU cask can travel an average of 845 kilometers (525 miles) (U.S. Department of Energy, 2002) before needing to refuel. The number of refueling (and rest) stops depends on the length of each type of route segment. This calculation uses the following equations:

Route segment length (km)/(845 km/stop) = stops/route segment

Dose (person-Sv) = (population/km^2) * (dose to resident near stop (Sv-km^2/h)) * (stops/route segment)*(hours/stop)

Table II-24 shows the collective doses to residents near stops for the rural and suburban segments of the 16 truck routes in Table II-4. Trucks carrying Truck-DU casks of spent fuel are unlikely to stop in urban areas.

The rural and suburban population densities in Table II-24 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 because the total distance from INL to the Skull Valley site is only 466.2 kilometers (289.7 miles). The route from Indian Point to ORNL illustrates another situation. This route is 1,028 kilometers (638.8 miles) long and would thus include one truck stop, which could be in either a rural or a suburban area.

Table II-24 Collective Doses to Residents near Truck Stops

Origin	Destination	Type	Persons/ km^2	Average number of stops	Person-Sv		
					Residents near Stops	Persons Sharing Stops	Total
Maine Yankee	ORNL	Rural	19.9	1.14	7.4×10^{-7}	3.9×10^{-4}	3.9×10^{-4}
		Suburban	395	0.93	1.0×10^{-5}	4.7×10^{-4}	4.8×10^{-4}
	Deaf Smith	Rural	18.6	2.47	1.5×10^{-6}	5.6×10^{-4}	5.6×10^{-4}
		Suburban	371	1.6	1.7×10^{-5}	3.6×10^{-4}	3.8×10^{-4}
	Hanford	Rural	15.4	4.33	2.2×10^{-6}	9.7×10^{-4}	9.8×10^{-4}
		Suburban	325	1.5	1.4×10^{-5}	3.4×10^{-4}	3.5×10^{-4}
	Skull Valley	Rural	16.9	3.5	1.9×10^{-6}	7.9×10^{-4}	7.9×10^{-4}
		Suburban	332.5	1.3	1.2×10^{-5}	2.9×10^{-4}	3.0×10^{-4}
Kewaunee	ORNL	Rural	19.8	0.81	5.2×10^{-7}	1.8×10^{-4}	1.8×10^{-4}
		Suburban	361	0.59	6.0×10^{-6}	1.3×10^{-4}	1.4×10^{-4}
	Deaf Smith	Rural	13.5	2.0	8.6×10^{-7}	4.5×10^{-4}	4.5×10^{-4}
		Suburban	339	0.52	5.0×10^{-6}	1.2×10^{-4}	1.2×10^{-4}
	Hanford	Rural	10.5	3.4	1.2×10^{-6}	7.7×10^{-4}	7.7×10^{-4}
		Suburban	316	0.60	5.4×10^{-6}	1.4×10^{-4}	1.4×10^{-4}
	Skull Valley	Rural	12.5	2.6	1.1×10^{-6}	5.9×10^{-4}	5.9×10^{-4}
		Suburban	324.5	0.44	4.1×10^{-6}	9.9×10^{-5}	1.0×10^{-4}
Indian Point	ORNL	Rural	20.5	0.71	4.7×10^{-7}	1.6×10^{-4}	1.6×10^{-4}
		Suburban	388	0.71	7.8×10^{-6}	1.6×10^{-4}	1.7×10^{-4}
	Deaf Smith	Rural	17.1	2.3	1.3×10^{-6}	5.2×10^{-4}	5.2×10^{-4}
		Suburban	370	1.2	1.3×10^{-5}	2.7×10^{-4}	2.8×10^{-4}
	Hanford	Rural	13.0	4.1	1.8×10^{-6}	9.2×10^{-4}	9.2×10^{-4}
		Suburban	338	1.1	1.1×10^{-5}	2.5×10^{-4}	2.6×10^{-4}
	Skull Valley	Rural	14.2	3.3	1.5×10^{-6}	7.4×10^{-4}	7.4×10^{-4}
		Suburban	351	0.93	9.3×10^{-6}	2.1×10^{-4}	2.2×10^{-4}
INL	ORNL	Rural	12.4	3.1	1.3×10^{-6}	7.0×10^{-4}	7.0×10^{-4}
		Suburban	304	0.72	6.3×10^{-6}	1.6×10^{-4}	1.7×10^{-4}
	Deaf Smith	Rural	7.8	2.3	5.8×10^{-7}	5.2×10^{-4}	5.2×10^{-4}
		Suburban	339	0.35	3.4×10^{-6}	7.9×10^{-5}	8.2×10^{-5}
	Hanford	Rural	6.5	0.43	2.0×10^{-7}	9.7×10^{-5}	9.7×10^{-5}
		Suburban	200	0.57	9.4×10^{-7}	1.3×10^{-4}	1.3×10^{-4}
	Skull Valley	Rural	10.1	0.42	1.4×10^{-7}	9.5×10^{-5}	9.5×10^{-5}
		Suburban	343	0.11	1.1×10^{-6}	2.5×10^{-5}	2.6×10^{-5}

[a] The number of stops is the kilometers of the route segment divided by 845 kilometers, the distance between stops, so that it may be a fraction. Retaining the fraction allows the calculation to be repeated.

Sample Calculation: Rural Stop from Maine Yankee to ORNL

Stop dose from RADTRAN output: 3.26×10^{-6} rem = 3.26×10^{-8} Sv. This takes into account the 30- to 800-meter bandwidth.

Average rural population density: 19.9 persons/km^2

Total rural km = 731
Distance between truck stops: 845 km (U.S. Department of Energy, 2002)
Number of truck stops: 731/845 = 0.865
Collective dose: 19.9 * 0.865 * $3.26 \times 10^{-8} = 5.6 \times 10^{-7}$

II.5.4.4 Occupational Doses

Occupational doses from routine, incident-free radioactive materials transportation include doses to truck and train crew, railyard workers, inspectors, and escorts. Not included are workers who handle spent fuel containers in storage, loading and unloading casks from vehicles or during intermodal transfer, and attendants who would refuel trucks, because truck refueling stops in the United States no longer have such attendants.[7]

Table II-25 summarizes the occupational doses.

Table II-25 Occupational Doses per Shipment from Routine, Incident-Free Transportation

Cask and route type	Train crew in transit[a]: 3 people; person-Sv	Truck crew in transit 2 people; person-Sv	Escort: Sv/hour[a]	Inspector: Sv per inspection	Truck stop worker: Sv per stop	Rail classification yard workers: person-Sv
Rail-Lead rural/suburban	5.4×10^{-9}		5.8×10^{-6}			1.5×10^{-5}
Rail-Lead urban	9.1×10^{-8}		5.8×10^{-6}			
Rail-Steel rural/suburban	4.1×10^{-9}		4.4×10^{-6}			1.1×10^{-5}
Rail-Steel urban	6.8×10^{-9}		4.4×10^{-6}			
Truck-DU rural/suburban		3.8×10^{-7}	4.9×10^{-9}	1.6×10^{-4}	6.7×10^{-6}	
Truck-DU urban		3.6×10^{-7}	4.9×10^{-9}			

[a] The truck crew is shielded while in transit to sustain a maximum dose of 0.02 mSv/h

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

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

Truck crew members are shielded so that they receive a maximum dose of 2.0×10^{-5} Sv/h (2 mrem/hr). This regulatory maximum was imposed in the RADTRAN calculation. Truck inspectors generally spend about 1 hour within 1 meter (40 inches) of the cargo (Weiner and Neuhauser, 1992), resulting in a relatively large dose. An upper bound to the duration of a truck refueling stop is about 50 minutes (0.83 hours) (Griego et al,, 1996). The truck stop worker whose dose is reflected in Table II-25 is assumed to be outside (unshielded) at 15 meters (49 feet) from the truck during the stop. Truck stop workers who are in concrete or brick buildings would be shielded from any radiation.

II.6 Interpretation of Collective Dose

Collective dose is essentially the product of an average radiation dose and the number of people who receive that average dose. The following example—a suburban segment on a particular route—is typical of all routes in all States; only the specific numbers change.

The following parameters characterize a representative segment of the Maine Yankee-to-Hanford truck route; the suburban segment through Illinois, shown below, is a representative example:

- Route segment length: 73 km (45 miles)
- Suburban population density: 324 persons/km^2 (839 persons/mi^2)
- Area occupied by that population: 0.800 km * 2 * 73 = 116.8 km^2 (45 mi^2)
- Total suburban population exposed to the shipment = 37,800 people
- From Table II-13, the collective radiation dose to that population, from routine, incident-free transportation = 6.5×10^{-6} person-Sv (0.65 person-mrem)
- U.S. background = 0.0036 Sv per year (4.1×10^{-7} Sv/h) (360 mrem/year = 0.041 mrem/hr)
- At an average speed of 108 kph (67 mph), time of population exposure = 0.675 h

The background dose sustained by each member of this population is 2.8×10^{-7} Sv (0.028 mrem), for a total collective dose of 0.0105 person-Sv (1,050 person-mrem). The total collective dose is thus 0.0105065 person-Sv (1,050.65 person-mrem) with the shipment, and 0.0105000 person-Sv (1,050 person-mrem) without the shipment. The collective dose from routine, incident-free transport is a very small increase in the collective dose the population continually receives from natural sources.

APPENDIX III

DETAILS OF CASK RESPONSE TO IMPACT ACCIDENTS

III.1 Introduction

For this study, the researchers performed explicit dynamic finite element calculations of the two spent fuel rail transportation casks described in Chapter 1 and shown in Figures 1-2 and 1-3 to assess their response to impact analyses. Information below provides the details of these analyses. In addition, the researchers summarized past explicit dynamic finite element analyses of the spent fuel truck transportation cask described in Chapter 1 and shown in Figure 1-4, and deduced the response to the same impact events and to other events based upon those analyses.

III.2 Finite Element Analysis of the Rail-Steel Cask

III.2.1 Problem Statement

Simulate the impact of a loaded Rail-Steel cask onto an unyielding surface. Consider the impact velocities of 48 kilometers per hour (kph) (30 miles per hour (mph)), 97 kph (60 mph), 145 kph (90 mph), and 193 kph (120 mph). Include end, side, and center-of-gravity (CG) over-corner impact orientations. Based on the results, assess the integrity of the containment boundary and estimate the extent of any possible breach. Although the deformation and failure of the lid closure bolts is of interest, the ultimate question of containment breach can be determined by assessing the integrity of the inner container. Predict the possible breach of the cask using plastic strains in the stainless steel inner container.

III.2.2 Geometric Assumptions and Mesh

A finite element model of the Rail-Steel cask was developed for use with the Sierra Mechanics code PRESTO. PRESTO is a nonlinear, transient dynamics finite element code developed at Sandia National Laboratories (Sandia) and is used extensively for weapons qualification work. The Rail-Steel cask model was developed and modified over several years to improve the initial limitations of PRESTO. Regulations required the model to include the most important geometric features without becoming so large that it could not be run on the available computational platforms. The final half-symmetric model consisted of 1.4 million solid hexahedral (hex) elements. The drop event lasted approximately 0.5 seconds. The simulation of this drop event required approximately 6 to 8 days of run time on 256 processors of a high-performance computer at Sandia.

An earlier version of the model used shell elements in areas of thin walled components. The code had difficulty with contact between hexes and embedded shells, and the boundary conditions between the shells and hexes required careful and complicated consideration. Ultimately, the shell elements were replaced by hex elements with two or three elements through the thickness. Although two elements through the thickness are considered insufficient to correctly predict bending response, these instances were limited to components for which bending responses were not considered important. For example, the outer shell of the impact limiters was modeled with two hex elements through the thickness. The purpose of this outer layer is to provide constraint to the aluminum honeycomb that comprises the impact limiter. The details of how it bends and folds away from the honeycomb are not important and not accurate with two elements through the thickness. Figure III-1 to Figure III-4 show the model details. To allow for internal impacts, gaps were included between the fuel region and the canister and between the canister and the cask interior. Figure III-5 shows the location and magnitude of these gaps.

Closure bolts were modeled with hex elements, with a minimum of four elements across the diameter of the bolt, as shown in Figure III-4. Any preload that would normally exist in these bolts because of tightening the bolts when they are installed was neglected. This assumption is conservative because it increases the amount of movement the closure lid would experience in any of the impact cases considered and maximizes any gaps that form between the closure lid and the cask body.

The total mass of the cask was 165,000 kilograms (kg) (weight of the cask was 364,700 pounds (lbs)). This weight is high because of an incorrect density value for the aluminum honeycomb that was not discovered until after the runs were completed. The overweight of the impact limiters results in a more severe loading environment because it increases the amount of kinetic energy that must be absorbed. The consequence of this increase is that all results are slightly conservative.

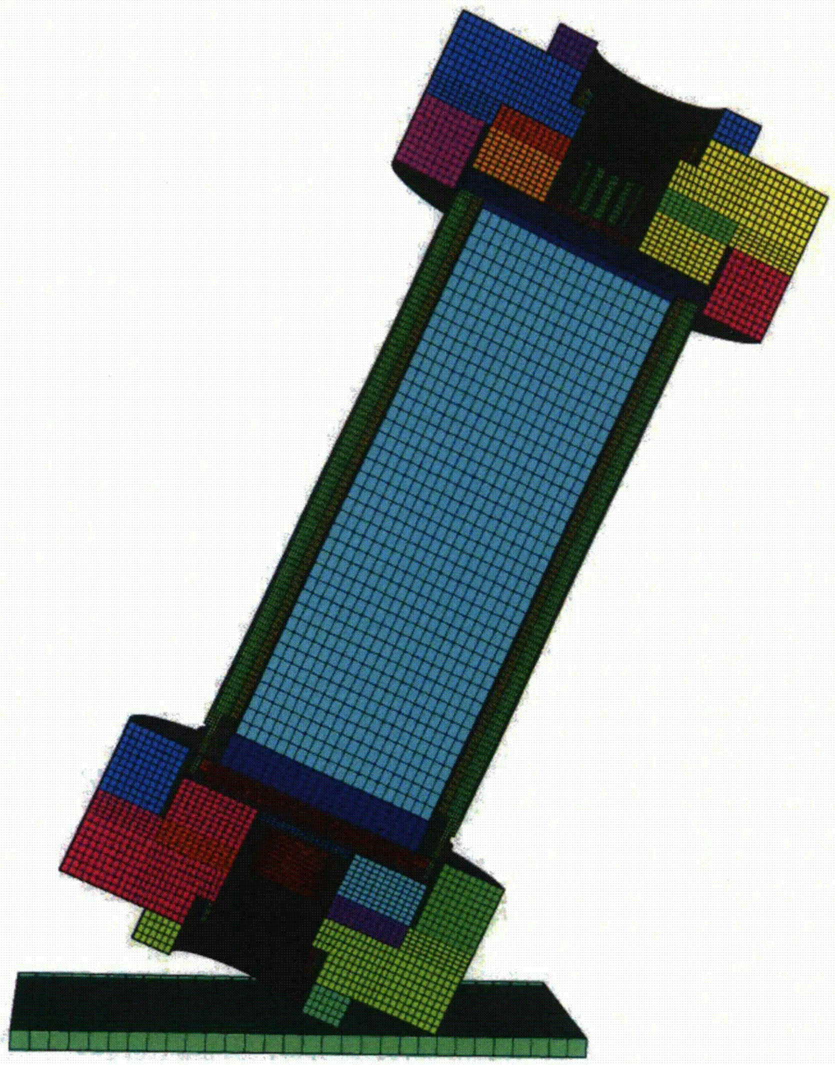


Figure III-1 Half-symmetric mesh of Rail-Steel cask

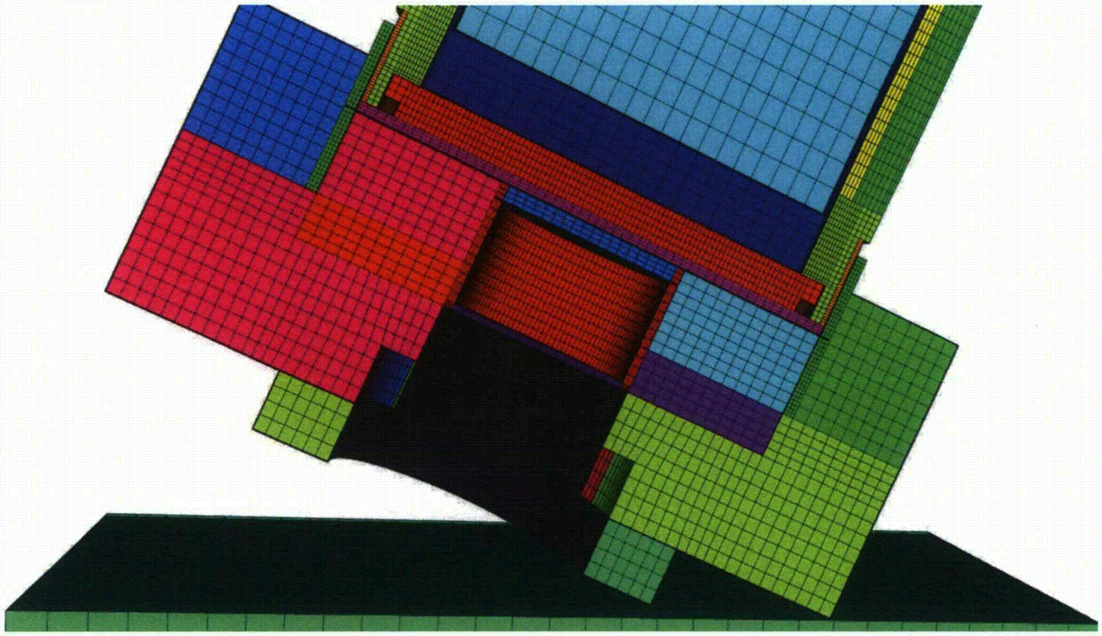


Figure III-2 Impact limiter mesh

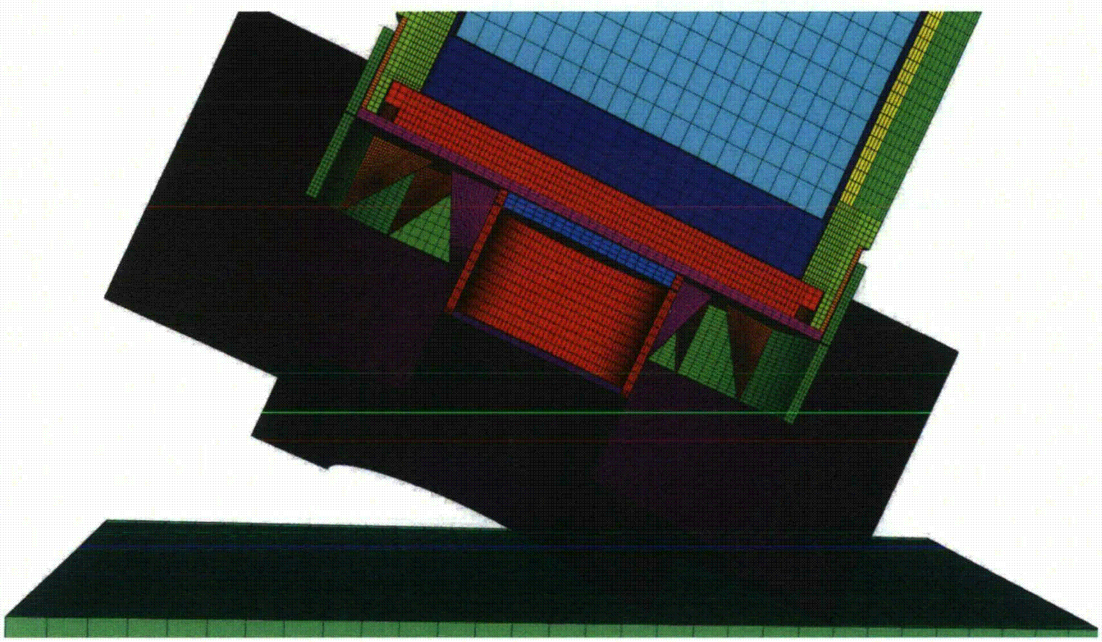


Figure III-3 Impact limiter mesh with honeycomb removed, showing the internal support structure

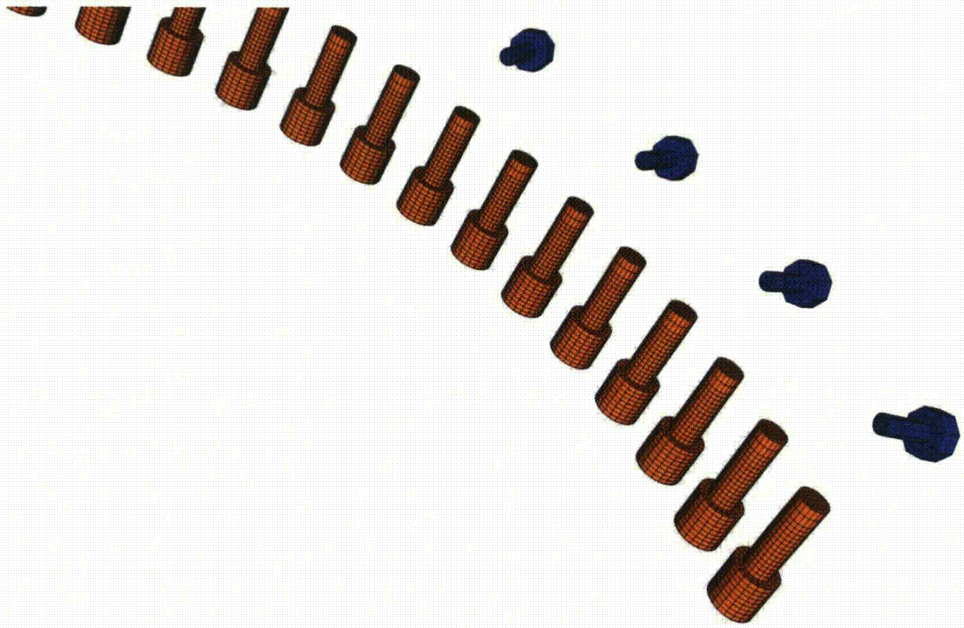


Figure III-4 Mesh of lid closure bolts and impact limiter attachment bolts

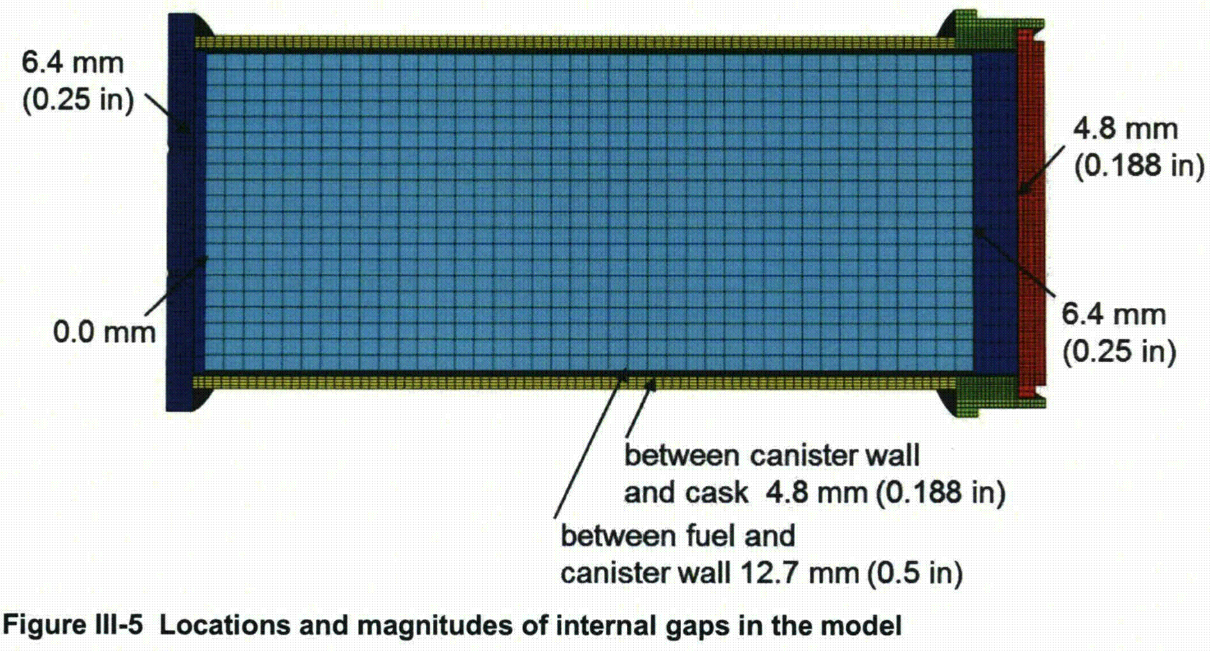


Figure III-5 Locations and magnitudes of internal gaps in the model

The orientation of the model is important to the definition of orthotropic material properties. The cask model is oriented as shown in Figure III-6, and the impact direction is changed for the three impact conditions. For an end drop, the initial velocity is in the +z direction. For a side drop, the initial velocity is in the –x direction. And for a CG over-corner drop, the initial velocity is in $0.38269x + 0.92388z$ direction.

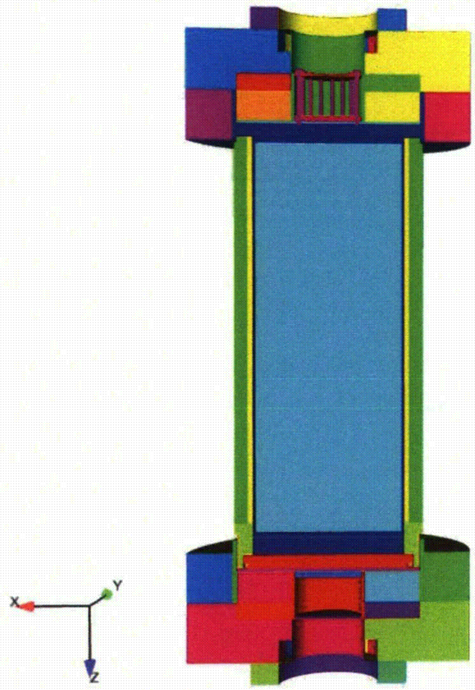


Figure III-6 Orientation of cask model for material property definitions

III.2.3 Material Properties

These analyses placed primary importance on the response of the closure bolts. The threaded ends of the bolts were modeled as fixed into their mating parts using equivalent nodes. The remainder of the bolt was allowed to slide into its through hole. Bolt failure was predicted by considering the equivalent plastic strain (EQPS) required for failure. Researchers assessed the value of EQPS that constitutes failure using tensile test data and references. Section III.2.4 provides details.

The analyses assumed that the aluminum honeycomb in the impact limiters was equally strong in the axial and radial directions and weaker in the circumferential direction. Properties were not varied at 15-degree increments, as specified by the design. Instead, properties were defined in the global x-y-z directions and aligned with the loading direction at the point of impact. The honeycomb was modeled with an orthotropic crush material model. The model has been used for many years in PRESTO and in the commercially available finite element method code LS-DYNA (LSTC, 1999). It is known to behave poorly at the transition to a fully compacted state when the material transitions from a unidirectional compaction to an isotropic compression with Poisson's expansion. For lower impact velocities (48 and 97 kph (30 and 60 mph)), this was not an issue. However, for the higher impact velocities, the model became unstable at material lockup. To allow the code to continue running, elements that were not correlating correctly were deleted. Since such elements had already absorbed the energy of the impact and were now just maintaining volume, their deletion was not considered important to the overall cask response.

Material properties are listed below, along with the parameters required by PRESTO (SIERRA Solids Mechanics Team, 2009): All inputs were in English units, so those are the values listed first with SI units in parentheses.

Material SA350-LF3

Material SA350-LF3 low-alloy steel (Holtec, 2004) is used for top lid and cask bottom.

Density = 0.00074 lb-s^2/in^4 (7.9 g/cm^3)
Material model ep_power_hard
Youngs Modulus = 28.0×10^6 psi (193×10^3 MPa)
Poissons Ratio = 0.27
Yield Stress = 37.0×10^3 psi (255 MPa)
Hardening Constant = 192746.0 psi (1,329 MPa)
Hardening Exponent = 0.748190
Luders Strain = 0.0

Material SA203E

Material SA203-E nickel alloy (Klamerus et al., 1996) is used for the overpack inner wall.

Density = 0.00074 lb-s^2/in^4 (7.9 g/cm^3)
Material model ep_power_hard
Youngs Modulus = 28.0×10^6 psi (193×10^3 MPa)
Poissons Ratio = 0.27
Yield Stress = 40.0×10^3 psi 276 MPa)
Hardening Constant = 192746 psi (1,329 MPa)
Hardening Exponent = 0.748190
Luders Strain = 0.0

Material SA-516, GR70

Material SA-516, Grade 70 (Klamerus et al., 1996) is used for overpack external wall, buttress plates, and impact limiter gusset plates.

Density = 0.00074 lb-s^2/in^4 (7.9 g/cm^3)
Material model ep_power_hard
Youngs Modulus = 29.0×10^6 psi (200×10^3 MPa)
Poissons Ratio = 0.3
Yield Stress = 53.097×10^3 psi (366 MPa)
Hardening Constant = 0.131331 $\times 10^6$ psi (90.55 MPa)
Hardening Exponent = 0.479290
Luders Strain = 0.00781

Material Testfoam

Material properties were taken from typical aluminum honeycomb data, as measured at Sandia (Hinnerichs et al., 2006). Properties used were for holtite and impact limiter aluminum cross-ply honeycomb.

Density = 0.0003002 lb-s^2/in^4 (3.2 g/cm^3)
Material model orthotropic_crush
Youngs Modulus = 4×10^6 psi (27.6×10^3 MPa)
Poissons Ratio = 0.3

Yield Stress = 40000 psi (27.6 MPa)
$Ex = 5.00x10^4$ psi (345 MPa)
$Ey = 5.00x10^4$ psi (345 MPa)
$Ez = 5.00x10^4$ psi (345 MPa)
$Gxy = 2.50x10^4$ psi (172 MPa)
$Gyz = 2.50x10^4$ psi (172 MPa)
$Gzx = 2.50x10^4$ psi (172 MPa)
Vmin = 0.70
Crush xx = 2300_T
Crush yy = 2300_T
Crush zz = 2300_L
Crush xy = 2300_T
Crush yz = 2300_T
Crush zx = 2300_T

Function 2300_L
0 1415.384615
0.05 2123.076923
0.1 2300
0.4 2300
0.5 1592.307692
0.6 3737.5
0.7 20000
0.9 20000

Function 2300_T
0 1415.384615
0.05 2123.076923
0.1 2300
0.4 2300
0.5 1592.307692
0.6 3737.5
0.7 20000
0.9 20000

Material Internals

This material is used for cask contents inside of inner container.
Density = 0.00029 $lb\text{-}s^2/in^4$ (3.1 g/cm^3)
Material model orthotropic_crush
Youngs Modulus = $0.5x10^6$ psi (3,450 MPa)
Poissons Ratio = 0.3
Yield Stress = 20,000.0 psi (138 MPa)
$Ex = 0.5x10^6$ psi (3,450 MPa)
$Ey = 0.5x10^6$ psi (3,450 MPa)
$Ez = 2.2x10^6$ psi ($15.2x10^3$ MPa)
$Gxy = 0.25x10^6$ psi (1,720 MPa)
$Gyz = 1.1x10^6$ psi (7,580 MPa
$Gzx = 1.1x10^6$ psi (7,580 MPa)
Vmin = 0.70
Crush xx = 2300_T
Crush yy = 700_W
Crush zz = 2300_L

Crush xy = foam_cross_1
Crush yz = foam_cross_2
Crush zx = foam_cross_1
Function foam_cross_1
0 1000
0.6 1000
0.7 10000
0.8 10000
Function foam_cross_2
0 500
0.6 500
0.7 5000
0.8 5000

Material SB637

Material SB637-N07718 (U.S. Department of Defense, 1993) is used for lid closure bolts.

Density = 0.00074 $lb\text{-}s^2/in^4$ (7.9 g/cm^3)
Material model ml_ep_fail
Youngs Modulus = 28.6×10^6 (197×10^3 MPa)
Poissons Ratio = 0.3
Yield Stress = 160000 psi (1,100 MPa)
Beta = 1.0
Hardening Function = MLEP_Hardening
Youngs Modulus Function = constant_one
Poissons Ratio Function = constant_one
Yield Stress Function = constant_one
Critical Tearing Parameter = 0.13
Critical Crack Opening Strain = 0.01

Material 304SS

Material 304SS is used for the inner welded container, bottom impact limiter bolts, top impact limiter bolts, and the shell surrounding impact limiters (Hucek, 1986).

Density = 0.00074 $lb\text{-}s^2/in^4$ (7.9 g/cm^3)
Material model ep_power_hard
Youngs Modulus = 53.3×10^6 psi[1] (367×10^3 MPa)
Poissons Ratio = 0.3
Yield Stress = 46.246×10^3 psi (319 MPa)
Hardening Constant = 319.05×10^3 psi (2,200 MPa)
Hardening Exponent = 0.68
Luders Strain = 0.0

1 The modulus of this material was artificially increased to resolve a contact chatter problem within the finite element model. This has very little effect on the response of the cask because this material is only used for thin shells that have relatively low strength and stiffness.

III.2.4 Criteria for Element Death and Bolt Failure

For all attachment bolts, element failure is defined according to PRESTO (SIERRA Solid Mechanics Team, 2009) convention.

Criterion is element value of EQPS > 1.12
Death on inversion = on

To account for instability in the orthotropic crush material model, elements are removed from the mesh if the following condition occurs, stated in the PRESTO element death convention:

- criterion is element value of solid_angle <= 0.05
- criterion is max nodal value of velocity(1) > 20,000
- criterion is max nodal value of velocity(2) > 20,000
- criterion is max nodal value of velocity(3) > 20,000
- criterion is max nodal value of velocity(1) < -20,000
- criterion is max nodal value of velocity(2) < -20,000
- criterion is max nodal value of velocity(3) < -20,000
- death on inversion = on

The impact limiter gusset plates and aluminum impact limiter honeycomb are in contact within the impact limiter. The honeycomb would likely fail before the gusset plates in an experiment. Because of the homogenized material modeling of the honeycomb and the relatively coarse mesh, the gusset plates are significantly deformed by the honeycomb. The failure of the gusset plates is defined according to PRESTO convention and includes the following conditions:

- criterion is element value of time-step < -0.01
- criterion is element value of volume <= 0.0
- death on inversion = on

III.2.5 Analysis Results

Figure III-7 through Figure III-11 depict the deformed shape of the cask following each impact analysis.

Time = 0.05280

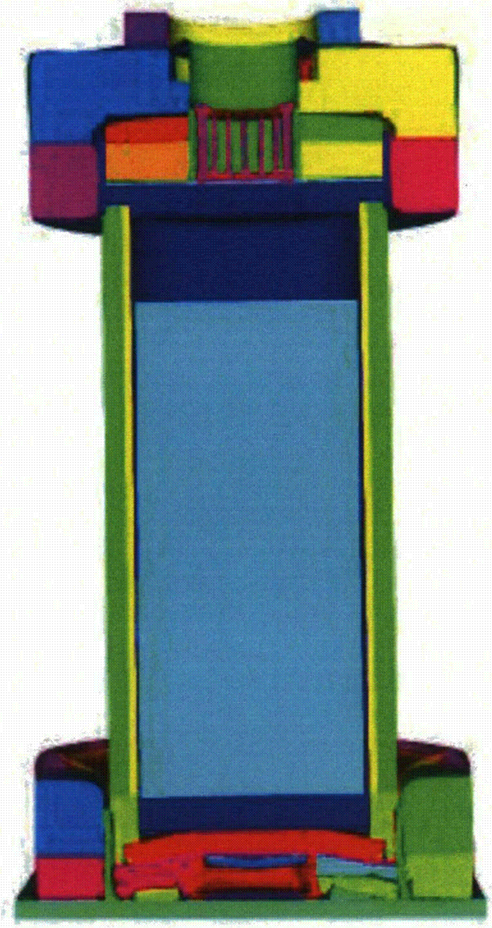


Figure III-7 Rail-Steel cask end impact at 193 kph (120 mph)

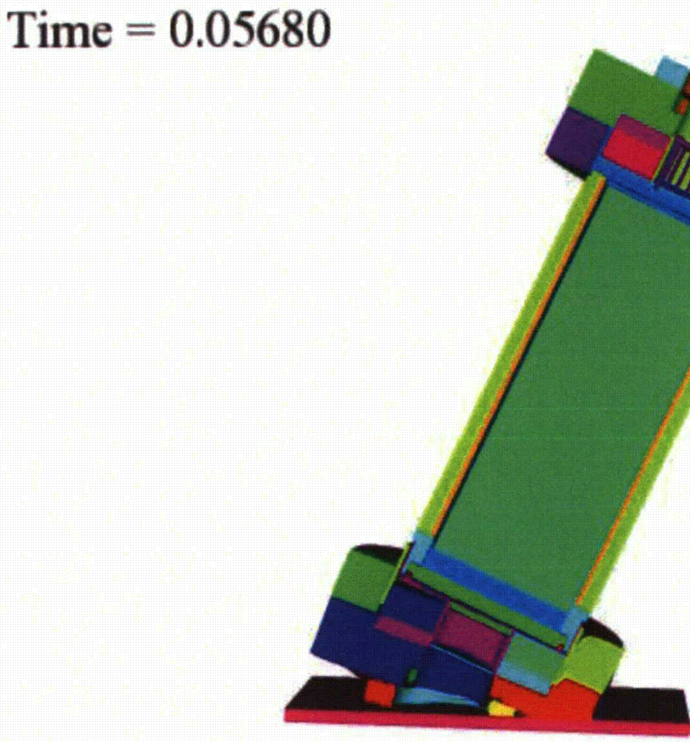


Figure III-8 Rail-Steel cask corner impact at 48 kph (30 mph)

Figure III-9 Rail-Steel cask corner impact at 97 kph (60 mph)

Figure III-10 Rail-Steel cask corner impact at 145 kph (90 mph)

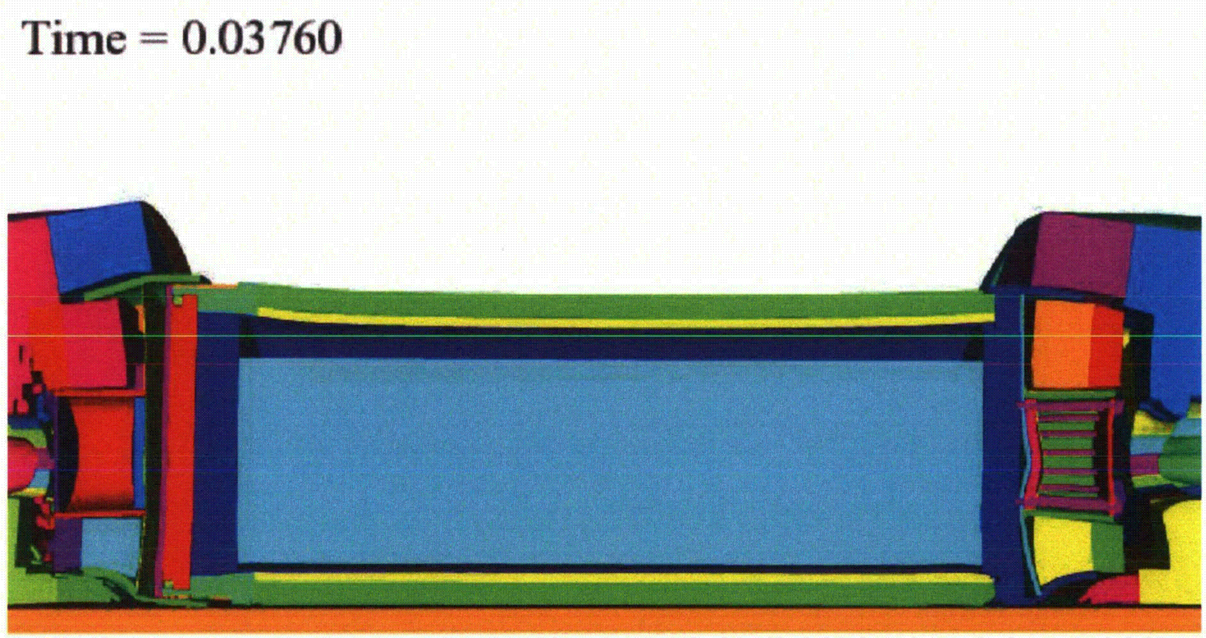


Figure III-11 Rail-Steel cask side impact at 193 kph (120 mph)

In Figure III-12 through Figure III-23, the EQPS in the welded inner canister is shown for each analysis case. The same contour interval is used for each figure and was chosen such that areas that were near failure would show up as red and could be clearly seen. All areas that are dark blue have plastic strains that are much lower than the failure strain and are not of concern.

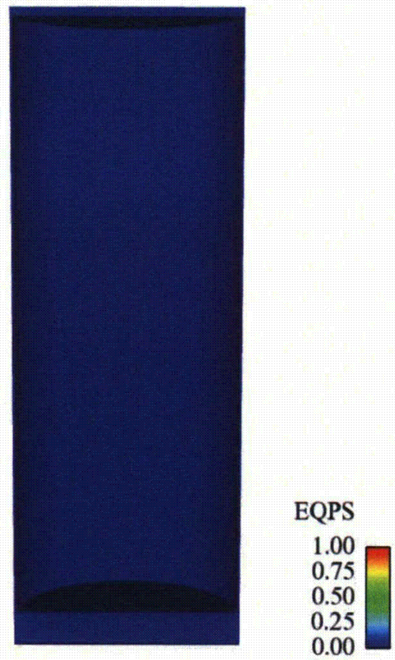


Figure III-12 Plastic strain in the interior welded canister of the Rail-Steel cask from the end impact at 48 kph (30 mph)

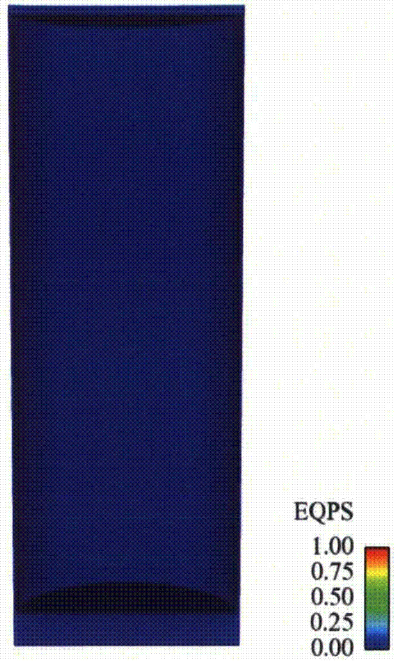


Figure III-13 Plastic strain in the interior welded canister of the Rail-Steel cask from the end impact at 97 kph (60 mph)

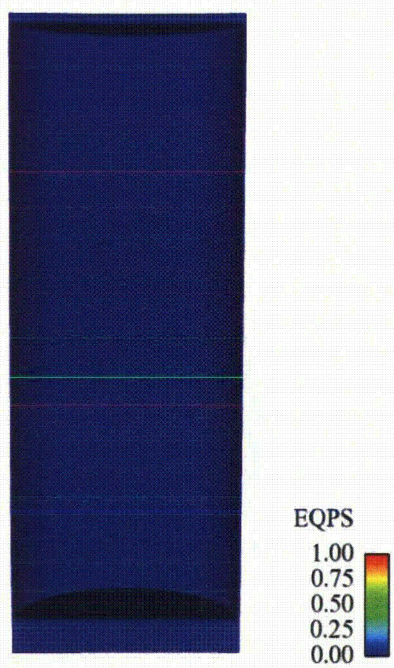


Figure III-14 Plastic strain in the interior welded canister of the Rail-Steel cask from the end impact at 145 kph (90 mph)

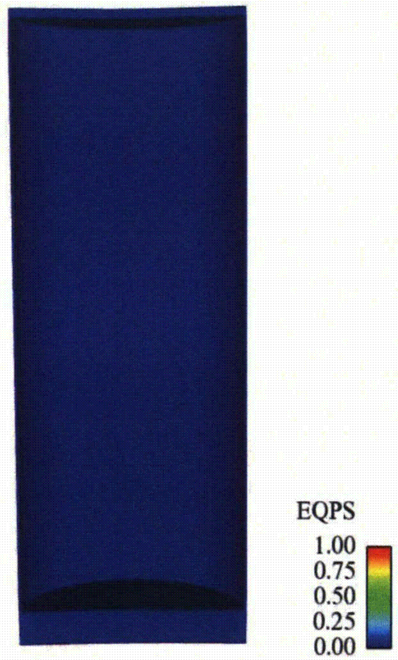


Figure III-15 Plastic strain in the interior welded canister of the Rail-Steel cask from the end impact at 193 kph (120 mph)

Figure III-16 Plastic strain in the interior welded canister of the Rail-Steel cask from the corner impact at 48 kph (30 mph)

Figure III-17 Plastic strain in the interior welded canister of the Rail-Steel cask from the corner impact at 97 kph (60 mph)

Figure III-18 Plastic strain in the interior welded canister of the Rail-Steel cask from the corner impact at 145 kph (90 mph)

Figure III-19 Plastic strain in the interior welded canister of the Rail-Steel cask from the corner impact at 193 kph (120 mph)

Figure III-20 Plastic strain in the interior welded canister of the Rail-Steel cask from the side impact at 48 kph (30 mph)

Figure III-21 Plastic strain in the interior welded canister of the Rail-Steel cask from the side impact at 97 kph (60 mph)

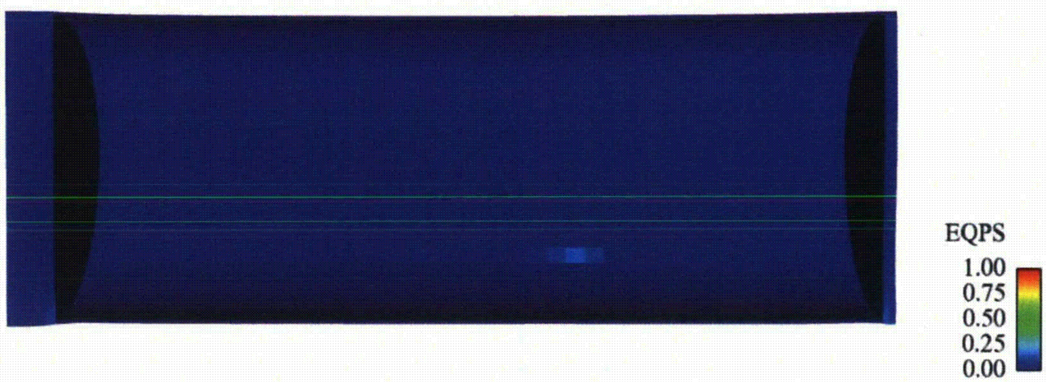


Figure III-22 Plastic strain in the interior welded canister of the Rail-Steel cask from the side impact at 145 kph (90 mph)

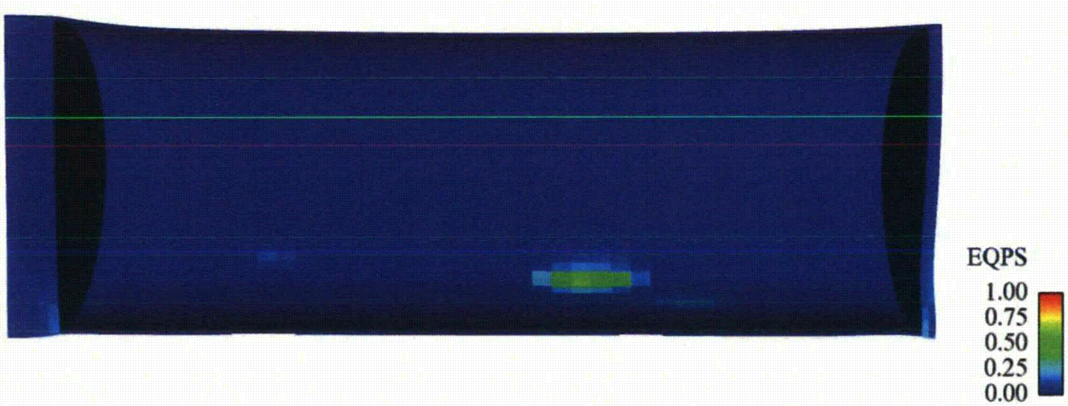


Figure III-23 Plastic strain in the interior welded canister of the Rail-Steel cask from the side impact at 193 kph (120 mph)

Analysis Summary

As expected, for all end, corner, and side impacts of the 48-kph (30-mph) impact analyses (the impact velocity from the regulatory hypothetical impact accident), the impact limiter absorbed almost all of the kinetic energy of the cask. No damage (permanent deformation) occurred to the cask body or canister. As the impact velocity increases, the first effect is additional damage to the impact limiter (for all orientations) because it is absorbing more kinetic energy (this shows the margin of safety in the impact limiter design). At 97 kph (60 mph), no significant damage to the cask body or canister occurred. At an impact speed of 145 kph (90 mph), damage to the cask and canister appears to begin. The impact limiter has absorbed all of the kinetic energy it can, and any additional kinetic energy is absorbed by plastic deformation in the cask body. **Error! Reference source not found.** gives the peak acceleration for each impact case. As expected, the accelerations for the side impacts are the highest and those for the corner impacts are the lowest.

Table III-1 Peak Acceleration from Each Analysis of the Rail-Steel Cask

Orientation	Speed, kph (mph)	Peak Accel. (g)
End	48 (30)	71
	97 (60)	115
	145 (90)	212
	193 (120)	276
Corner	48 (30)	66
	97 (60)	86
	145 (90)	*
	193 (120)	233
Side	48 (30)	*
	97 (60)	*
	145 (90)	355
	193 (120)	472

* Data from the finite element output file for these cases was not available.

For the side impact at 145 kph (90 mph), several of the lid bolts fail in shear (criteria for the failure model are included in Section III.2.4 above), but the lid remains attached. At this point, the metallic seal no longer maintains the leaktightness of the cask, but the spent fuel remains contained within the welded canister. Even at the highest impact speed, 193 kph (120 mph), the welded canister remains intact for all orientations, so the response of the closure is of less importance.

III.3 Finite Element Analysis of the Rail-Lead Cask

III.3.1 Problem Statement

Simulate impact of a loaded Rail-Lead cask onto an unyielding surface. Consider impact velocities of 48 kph (30 mph), 97 kph (60 mph), 145 kph (90 mph), and 193 kph (120 mph). Include end, side, and CG over-corner impact orientations. Based on the results, assess the integrity of the containment boundary and estimate the extent of any possible breach. Estimate the deformation and failure of the lid closure bolts and any resulting gap between the lids and the cask. Estimate the maximum lead slump distance.

III.3.2 Geometric Assumptions and Mesh

Researchers developed a finite element model of the Rail-Lead cask for use with the Sierra Mechanics code PRESTO (SIERRA Solid Mechanics Team, 2009). PRESTO is a nonlinear, transient dynamics finite element code developed at Sandia. The finite element model was built primarily of hex elements. Shell elements were used for the thin stainless steel skin that wraps around the impact limiters. The final half-symmetric model consisted of 750,000 elements. The drop event lasted approximately 0.5 seconds. The simulation of this drop event required approximately 36 to 60 hours of run time on 64 processors of the RedSky high-performance computer at Sandia.

The model details are shown in Figure III-24 through Figure III-27. Unlike the Rail-Steel cask, the basket in the Rail-Lead storage/transport cask completely fills the internal space of the cask. Gaps between the individual fuel elements and the cask lid are possible, but the probability of each of these fuel elements contacting the lid at the same time is very small. Thus, no gap was included in the model. Also, the presence of a gap could increase the force acting on the fuel elements, but for the severe impacts of concern in this study, such a scenario is unlikely to influence the overall deformation of the cask lid region because the fuel impact onto the lid would occur while the lid is being pushed onto the cask by the impact limiter, regardless of any initial gaps between the fuel and the lid.

Closure bolts were modeled with hex elements, with a minimum of four elements across the diameter of the bolt, as shown in Figure III-26. The model neglected any preload that would normally exist in these bolts as a result of the bolts being tightened during installation. This assumption is conservative because it increases the amount of movement the closure lid will experience in any of the impact cases considered and maximizes any gaps that form between the closure lid and the cask body.

The total mass of the cask was 112,000 kg (total weight of the cask was 247,300 lbs).

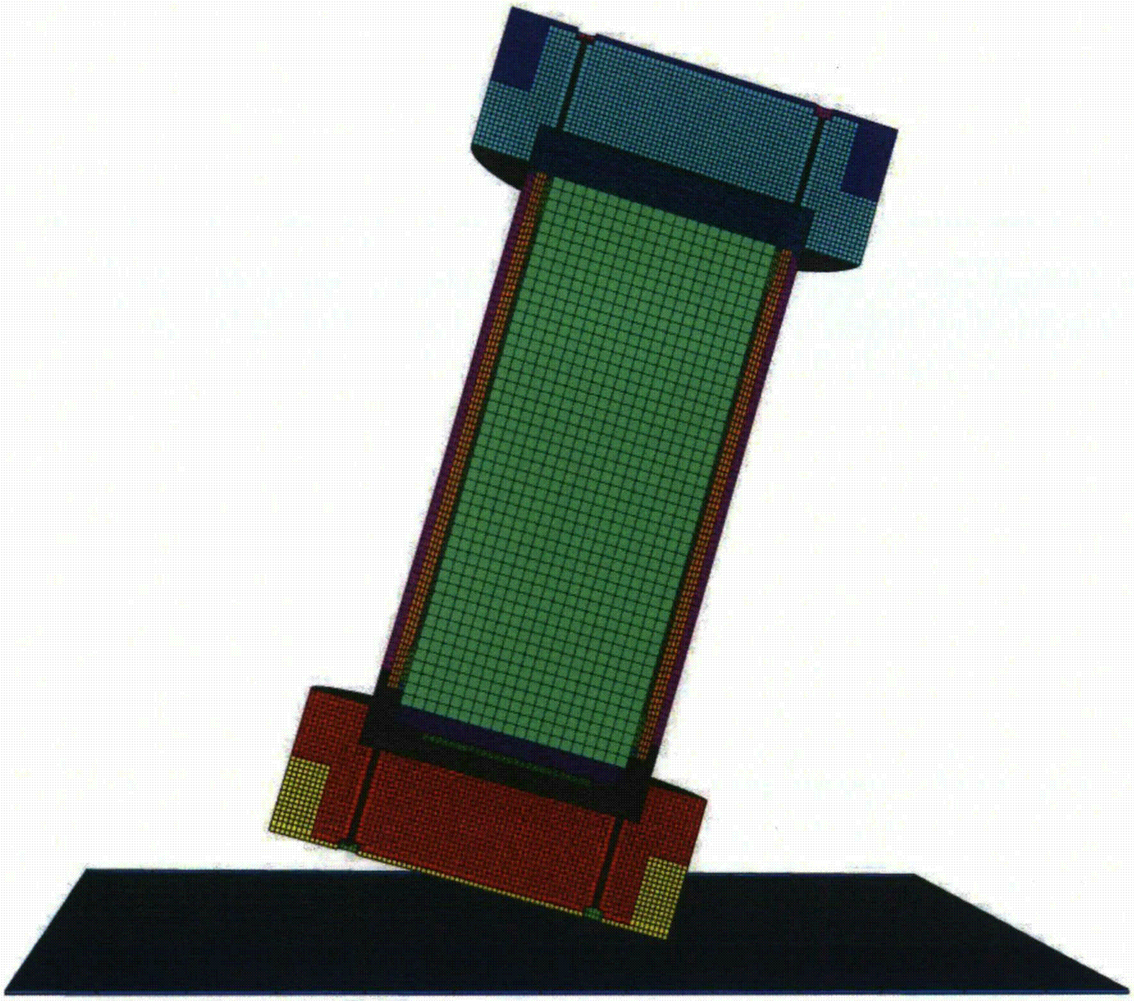


Figure III-24 Half-symmetric mesh of Rail-Lead cask

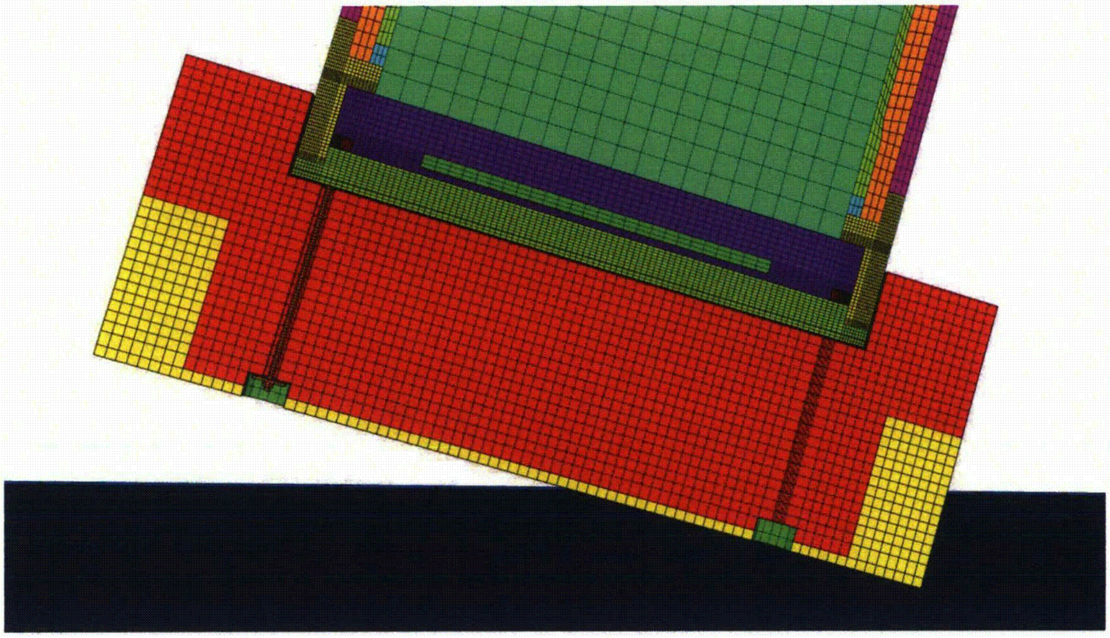


Figure III-25 Impact limiter mesh

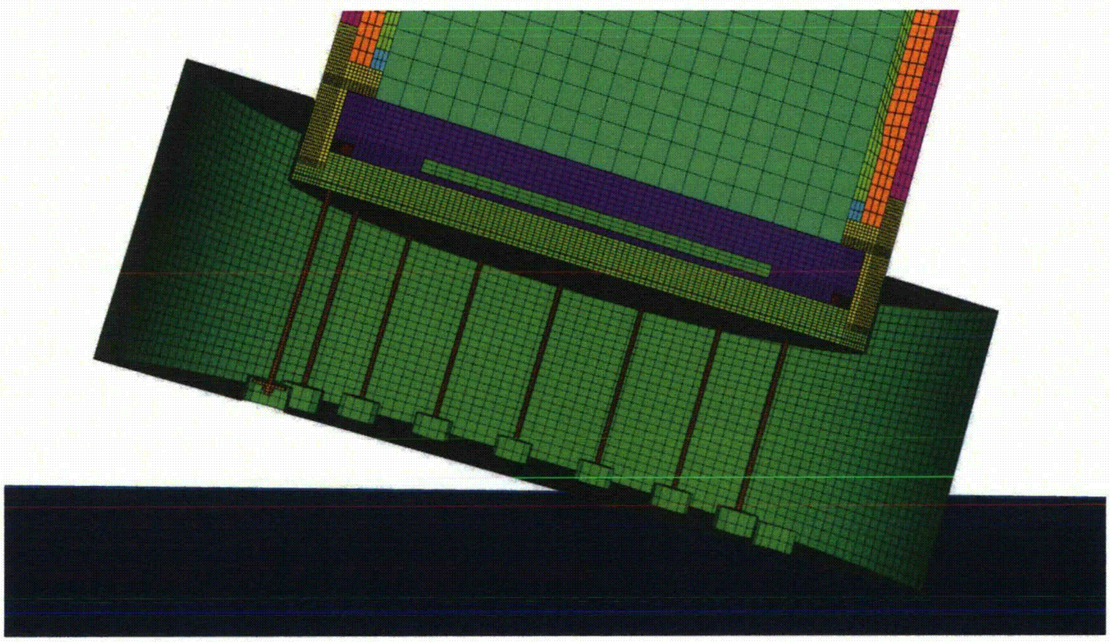


Figure III-26 Impact limiter mesh with wood removed

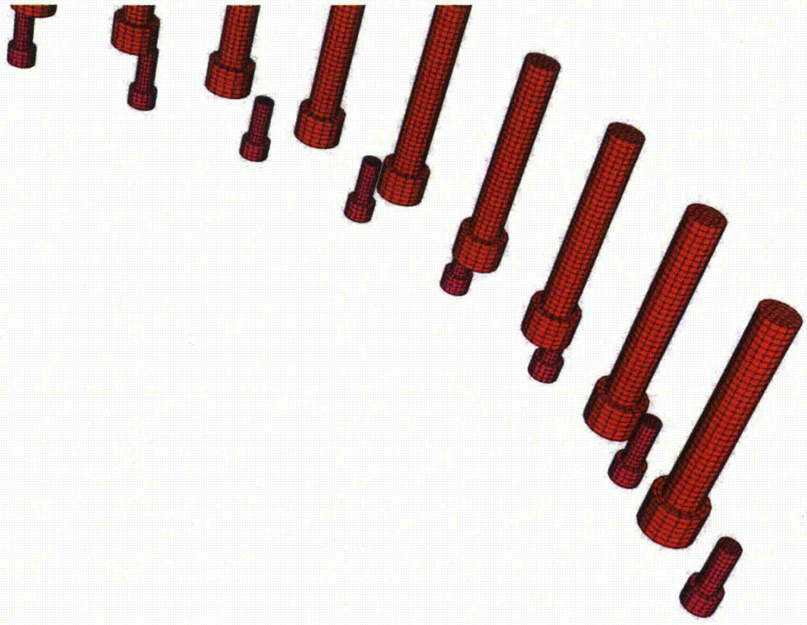


Figure III-27 Mesh of inner and outer lid closure bolts

The orientation of the model is important to the definition of orthotropic material properties. The cask model is oriented as shown in Figure III-28, and the impact direction is changed for the three impact conditions. For an end drop, the initial velocity is in the -y direction. For a side drop, the initial velocity is in the –x direction. And for a CG. over-corner drop, the initial velocity is in a 0.169912x - 0.98546y direction.

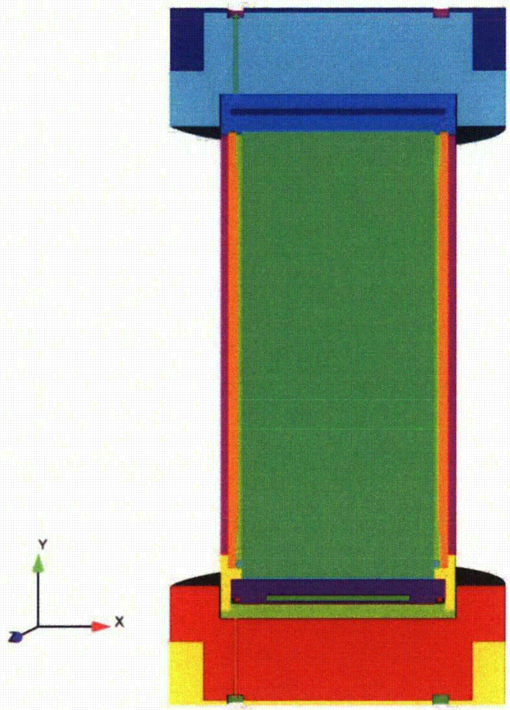


Figure III-28 Orientation of cask model for material property definitions

III.3.3 Material Properties

Material properties are listed below, along with the parameters required by PRESTO (SIERRA Solids Mechanics Team, 2009): All inputs were in English units, so those are the values listed first with SI units in parentheses.

Material Redwood

This material is used for top and bottom impact limiter.

Density = 5.682×10^{-5} lb-s^2/in^4 (0.61 g/cm^3)
Material model orthotropic_crush
Young's Modulus = 1.5×10^6 psi (10.3×10^3 MPa)
Poissons Ratio = 0.3
Yield Stress = 20000 psi (138 MPa)
Vmin=0.9
Ex = 1.5×10^6 psi (10.3×10^3 MPa)
Ey = 0.3×10^6 psi (2.1×10^3 MPa)
Ez = 1.5×10^6 psi (10.3×10^3 MPa)
Gxy = 0.2×10^6 psi (1.4×10^3 MPa)
Gyz = 0.2×10^6 psi (1.4×10^3 MPa)
Gzx = 0.2×10^6 psi (1.4×10^3 MPa)
Crush xx = redwood_strong
Crush yy = redwood_weak
Crush zz = redwood_strong
Crush xy = redwood_shear
Crush yz = redwood_shear

Crush zx = redwood_shear
Function redwood_strong
strain stress,psi (Mpa)
0. 2000 (13.8)
0.14 4200 (29.0)
0.28 5100 (35.2)
0.42 5430 (37.4)
0.57 6100 (42.1)
0.71 10100 (69.6)
0.80 15000 (103)
0.90 20000 (138)
Function redwood_weak
strain stress,psi (Mpa)
0. 400 (2.76)
0.14 986 (6.80)
0.28 1200 (8.27)
0.42 1275 (8.79)
0.57 1432 (9.87)
0.71 2371 (16.3)
0.80 3521 (24.3)
0.90 4690 (32.3)
Function redwood)_shear
strain stress,psi (Mpa)
0.0 1000 (6.9)
0.60 1000 (6.9)
0.70 10000 (69)
0.90 10000 (69)

Material Balsa

This material is used for outer corner of top and bottom impact limiters.
Density = $1.5x10^{-5}$ lb-s^2/in^4 (0.16 g/cm^3)
Material model orthotropic_crush
Young's Modulus = $1.5x10^6$ psi ($10.3x10^3$ MPa)
Poissons Ratio = 0.3
Yield Stress = 20000 psi
Vmin = 0.9
Ex = $1.5x10^6$ psi ($10.3x10^3$ MPa)
Ey = $0.3x10^6$ psi ($2.1x10^3$ MPa)
Ez = $1.5x10^6$ psi ($10.3x10^3$ MPa)
Gxy = $0.2x10^6$ psi ($1.4x10^3$ MPa)
Gyz = $0.2x10^6$ psi ($1.4x10^3$ MPa)
Gzx = $0.2x10^6$ psi ($1.4x10^3$ MPa)
Crush xx = balsa_strong
Crush yy = balsa_weak
Crush zz = balsa_strong
Crush xy = balsa_shear
Crush yz = balsa_shear
Crush zx = balsa_shear

Function balsa_strong
strain stress,psi (Mpa)
0. 2000 (13.8)
0.14 4200 (29.0)
0.28 5100 (35.2)
0.42 5430 (37.4)
0.57 6100 (42.1)
0.71 10100 (69.6)
0.80 15000 (103)
0.90 20000 (138)
Function balsa_weak
strain stress,psi (Mpa)
0. 400 (2.76)
0.14 986 (6.80)
0.28 1200 (8.27)
0.42 1275 (8.79)
0.57 1432 (9.87)
0.71 2371 (16.3)
0.80 3521 (24.3)
0.90 4690 (32.3)
Function balsa_shear
strain stress,psi (Mpa)
0.0 1000 (6.9)
0.60 1000 (6.9)
0.70 10000 (69)
0.90 10000 (69)

Material 304 SS

Properties for 304 stainless steel were obtained from tensile tests conducted at Sandia.

Elastic values match the Rail-Lead safety analysis report (SAR) (NAC, 2004), but complete response curve is used for placticity.

This material is used for inner and outer cask wall, shell surrounding impact limiters, and impact limiter attachment bolts.

Density = 7.48e-4 lb-s^2/in^4 (8.0 g/cm^3)
Material model ml_ep_fail
Youngs Modulus = 28.0×10^6 psi (193×10^3 MPa)
Poissons Ratio = 0.27
Yield Stress = 33.0×10^3 psi[2] (228 MPa)
Beta = 1.0
Youngs Modulus Function = 304_SS_YM
Poissons Ratio Function = 304_SS_PR
Yield stress Function = 304_SS_YS
Hardening Function = 304_SS_H
Critical Tearing Parameter = 7.779

2 The yield strength for this material is generally much higher than 33 kilopounds per square inch (ksi), but this value was used to be consistent with the value from the SAR. The actual yield strength for this material is generally closer to the 46 ksi used for the Rail-Steel cask analyses.

Critical Crack Opening Strain = 0.20

Function 304_SS_H

strain stress, psi (MPa)

0.0 0. (0)
0.0395 23.4×10^3 (161 MPa)
0.0782 34.9×10^3 (241 MPa)
0.1151 45.1×10^3 (311 MPa)
0.1509 54.0×10^3 (372 MPa)
0.1857 61.7×10^3 (425 MPa)
0.2197 68.5×10^3 (472 MPa)
0.2527 74.7×10^3 (515 MPa)
0.2848 80.5×10^3 (555 MPa)
0.3165 86.0×10^3 (593 MPa)
0.3470 91.2×10^3 (629 MPa)
0.3767 96.4×10^3 (665 MPa)
0.4077 101.5×10^3 (700 MPa)
0.4378 106.4×10^3 (734 MPa)
0.4690 111.4×10^3 (768 MPa)
0.5209 119.1×10^3 (821 MPa)
0.5797 128.4×10^3 (885 MPa)
0.6595 140.6×10^3 (969 MPa)
0.7520 156.5×10^3 (1,080 MPa)
0.8639 176.3×10^3 (1,220 MPa)
1.0129 204.2×10^3 (1,410 MPa)
1.2049 242.9×10^3 (1,680 MPa)
1.4476 298.5×10^3 (2,060 MPa)
1.7499 382.8×10^3 (2,640 MPa)
2.1246 519.1×10^3 (3,580 MPa)
2.5960 754.3×10^3 (5,200 MPa)
3.1689 1161.6×10^3 (8,010 MPa)
3.7371 1624.0×10^3 (11,200 MPa)
6.0 3465.5×10^3 (23,900 MPa)

<u>Material Filler</u>

This material is used for internals.

Density = 2.92×10^{-4} lb-s^2/in^4 (3.1 g/cm^3)
Material model elastic
Youngs Modulus = 122.0×10^3 psi (841 MPa)
Poissons Ratio = 0.30

<u>Material 17-4 SS</u>

Properties for 17-4 stainless steel were obtained from tensile tests conducted at Sandia. Elastic values match Rail-Lead SAR (NAC, 2004), but complete response curve is used for plasticity.

This material is used for outer lid and outer lid bolts.

Density = 7.48×10^{-4} lb-s^2/in^4 (8.0 g/cm^3)
Material model ml_ep_fail
Youngs Modulus = 28.0×10^6 psi (193×10^3 MPa)

Poissons Ratio = 0.28
Yield Stress = 100000. psi (689 MPa)
Beta = 1.0
Youngs Modulus Function = 304_SS_YM
Poissons Ratio Function = 304_SS_PR
Yield Stress Function = 304_SS_YS
Hardening Function = 17_4_SS_H
Critical Tearing Parameter = 10.0
Critical Crack Opening Strain = 0.20

Function 17_4_SS_H

strain stress	psi	(MPa)
0	100000.0	(689)
0.00407825	136477.69	(941)
0.00879119	153992.02	(1,060)
0.01402863	161193.41	(1,110)
0.01969711	164727.25	(1,140)
0.02677325	166808.60	(1,150)
0.03772328	168627.66	(1,160)
0.12541256	176332.05	(1,220)
0.24107482	183114.13	(1,260)
0.37338829	196318.29	(1.350)
0.51621765	212319.68	(1,460)
0.67105461	234527.78	(1,620)
0.84082846	261327.83	(1,800)
1.03088417	297249.64	(2,050)
1.24626188	344040.44	(2,370)
1.49347177	408459.72	(2,820)
1.78071924	499087.83	(3,440)
2.13871929	625460.64	(4,310)

Material SB-637

Material SB-637 Grade N07718 nickel alloy steel (NAC, 2004) is used for inner lid bolts.
Density = 7.324×10^{-4} lb-s^2/in^4 (7.8 g/cm^3)
Material model elastic_plastic
Youngs Modulus = 29.0×10^6 psi (200×10^3 MPa)
Poissons Ratio = 0.32
Yield Stress = 150.8×10^3 psi (1,040 MPa)
Hardening Modulus = 531.4×10^3 psi (3,664 MPa)
Beta = 1.0

Material Pb

Lead (Hoffman and Attaway, 1991) is used for midcask wall.
Density = 1.06×10^{-3} lb-s^2/in^4 (11.3 g/cm^3)
Material model elastic_plastic
Youngs Modulus = 2.0×10^6 psi 13.8×10^3 MPa)
Poissons Ratio = 0.3
Yield Stress = 1700. psi (11.7 MPa)
Hardening Modulus = 2000. psi (13.8 MPa)
Beta = 1

Material NS-4-FR

A solid synthetic polymer, NS-4-FR is used for neutron shielding inserts in top and bottom lids.

The neutron-shielding material was developed by BISCO Industries, Inc., and is now supplied by Genden Engineering Services & Construction Company.

NS-4-FR is an epoxy resin that contains boron.

Density = 1.571×10^{-4} lb-s^2/in^4 (1.7 g/cm^3)
Material model elastic
Youngs Modulus = 0.561×10^5 psi (387 MPa)
Poissons Ratio = 0.3

III.3.4 Criteria for Element Death and Bolt Failure

To account for instability in the orthotropic crush material model, elements are removed from the mesh if the following condition occurs, stated in the PRESTO (SIERRA Solid Mechanics Team, 2009) element death convention:

- criterion is max nodal value of velocity(1) > 20,000
- criterion is max nodal value of velocity(2) > 20,000
- criterion is max nodal value of velocity(3) > 20,000
- criterion is max nodal value of velocity(1) < -20,000
- criterion is max nodal value of velocity(2) < -20,000
- criterion is max nodal value of velocity(3) < -20,000
- death on inversion = on

For the impact limiter attachment bolts, elements failure is defined according to the PRESTO convention. This means that failure occurs when the critical tearing parameter (Wellman and Salzbrenner, 1992) is reached, as defined for 304 stainless steel:

Material criterion = ml_ep_fail

Failure of the outer lid and outer lid bolts was defined according to the PRESTO convention when a maximum value of EQPS was reached in 17-4 stainless steel. The PRESTO convention established this value of EQPS using an analysis of a tensile test specimen, and it defined failure at the true strain that corresponds to the true stress approximately midway between the true stress at maximum load and the final true stress. It chose the conservative value to compensate for the relatively coarse mesh in the bolt:

Criterion is element value of EQPS > 1.5

Failure of the inner lid bolts was defined according to the PRESTO convention when a maximum value of EQPS was reached in SB-637 Grade N07718 nickel alloy steel:

Criterion is element value of EQPS > 0.1

III.3.5 Analysis Results

Figure III-29 through Figure III-40 depict the deformed shape of the cask following each impact analysis.

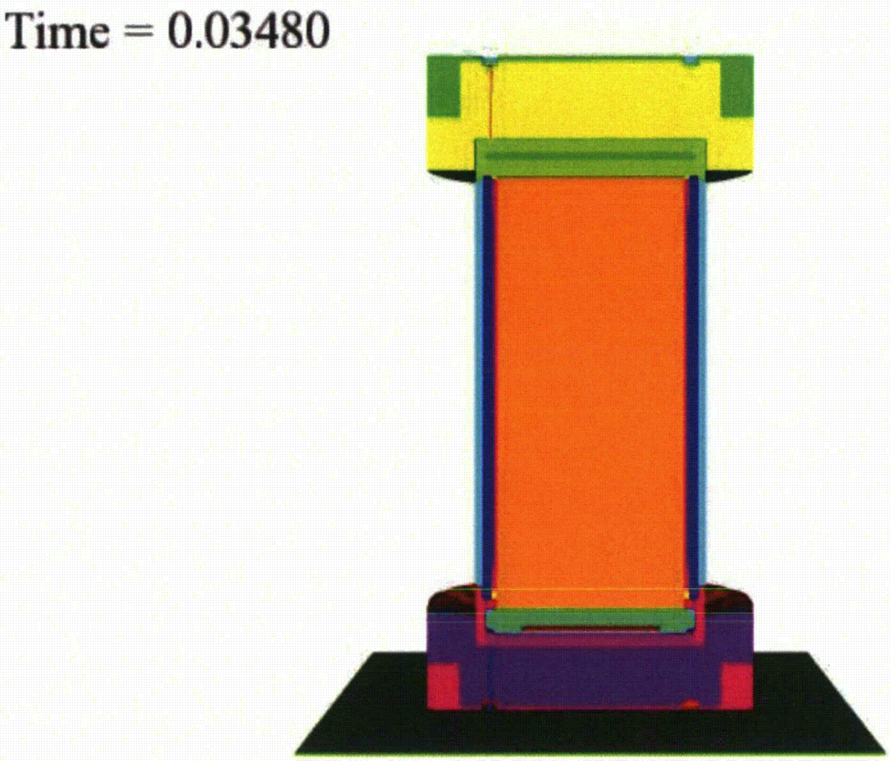


Figure III-29 Rail-Lead cask end impact at 48 kph (30 mph)

Time = 0.03480

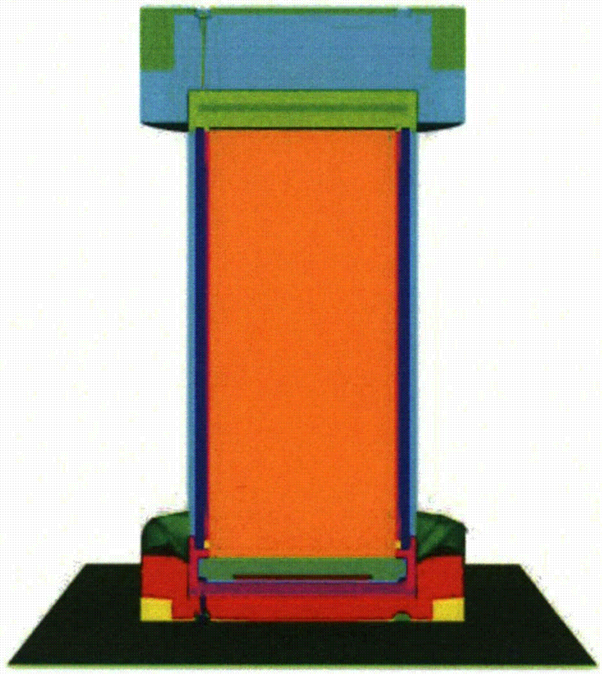


Figure III-30 Rail-Lead cask end impact at 97 kph (60 mph)

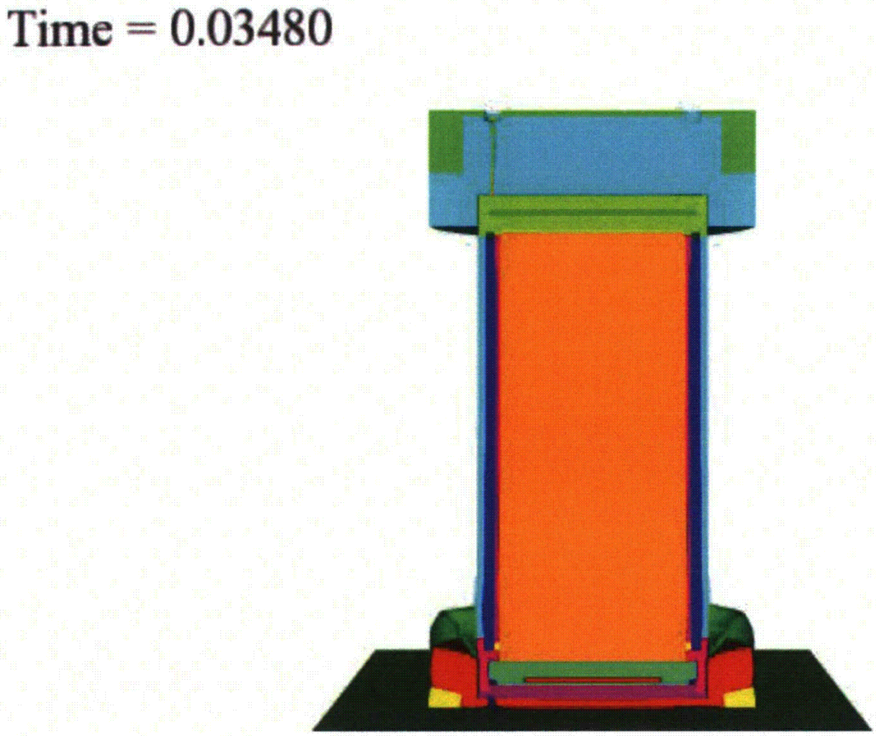


Figure III-31 Rail-Lead cask end impact at 145 kph (90 mph)

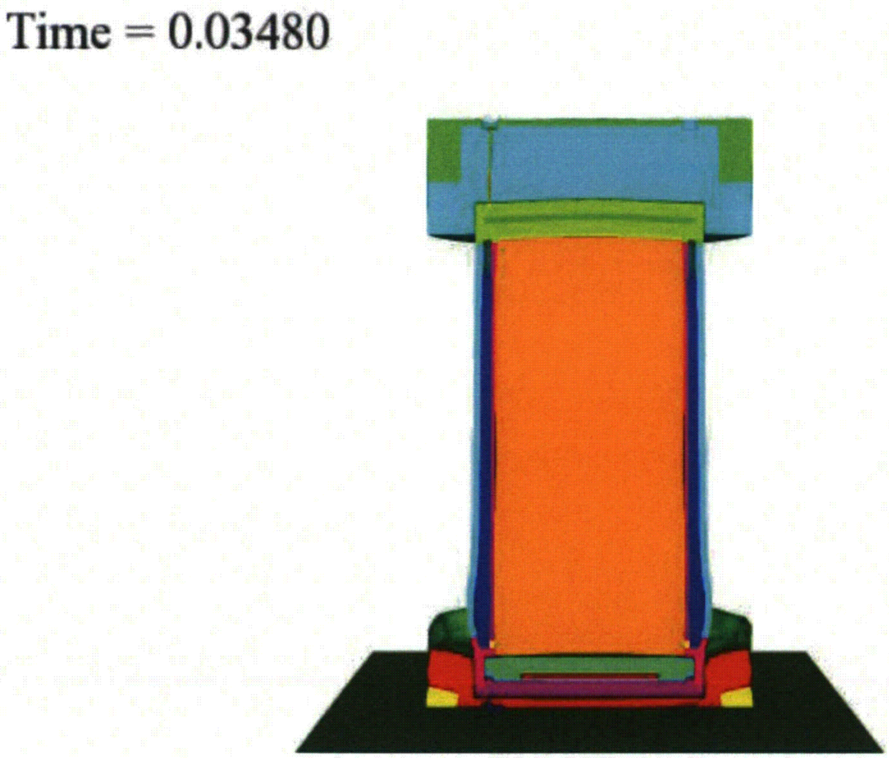


Figure III-32 Rail-Lead cask end impact at 193 kph (120 mph)

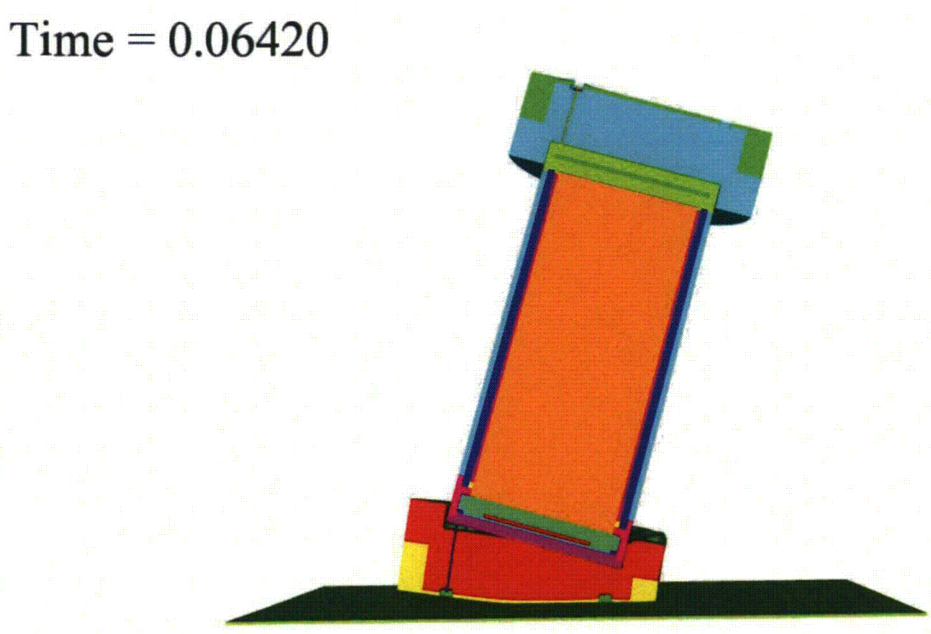


Figure III-33 Rail-Lead cask corner impact at 48 kph (30 mph)

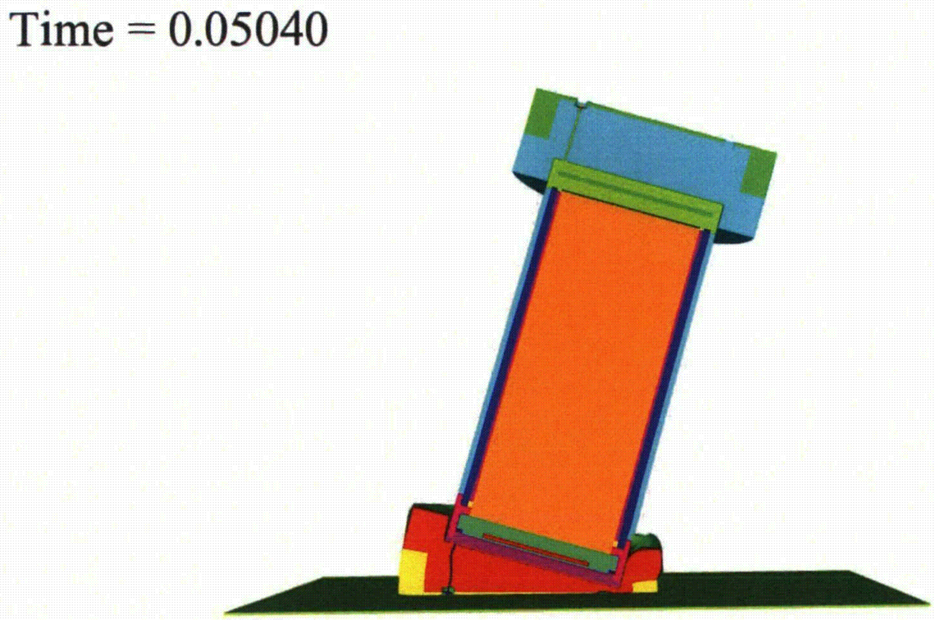


Figure III-34 Rail-Lead cask corner impact at 97 kph (60 mph)

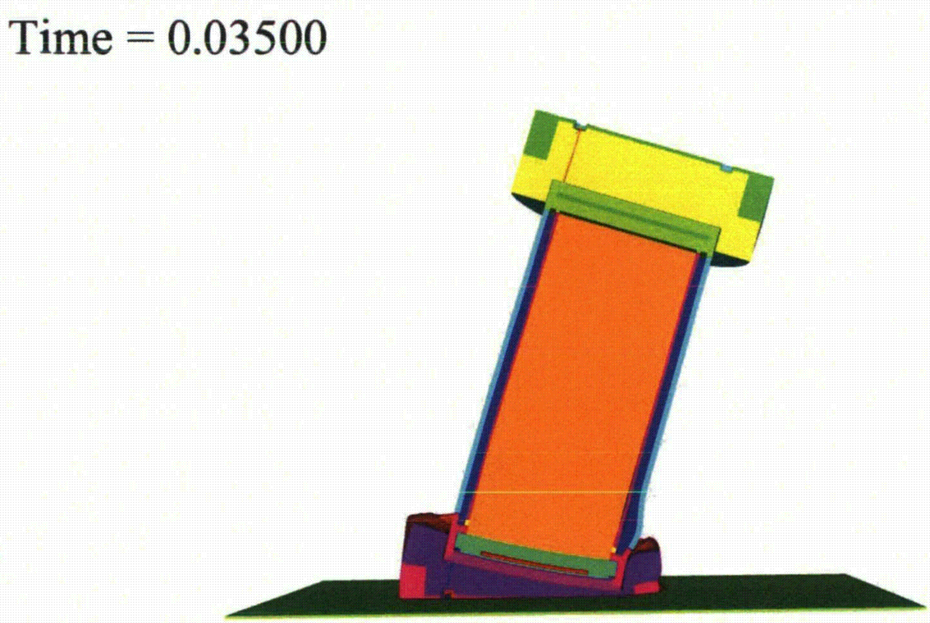


Figure III-35 Rail-Lead cask corner impact at 145 kph (90 mph)

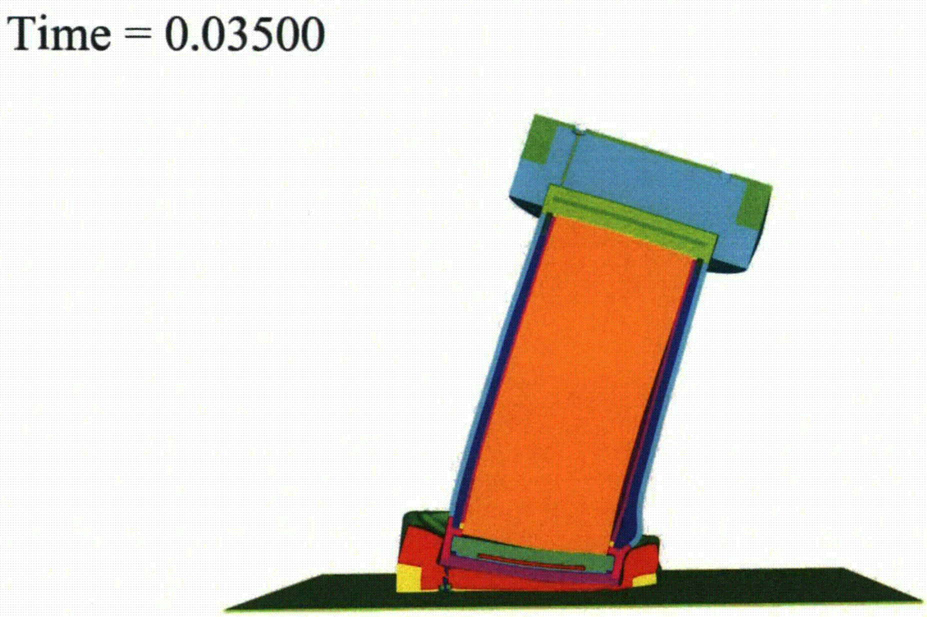


Figure III-36 Rail-Lead cask corner impact at 193 kph (120 mph)

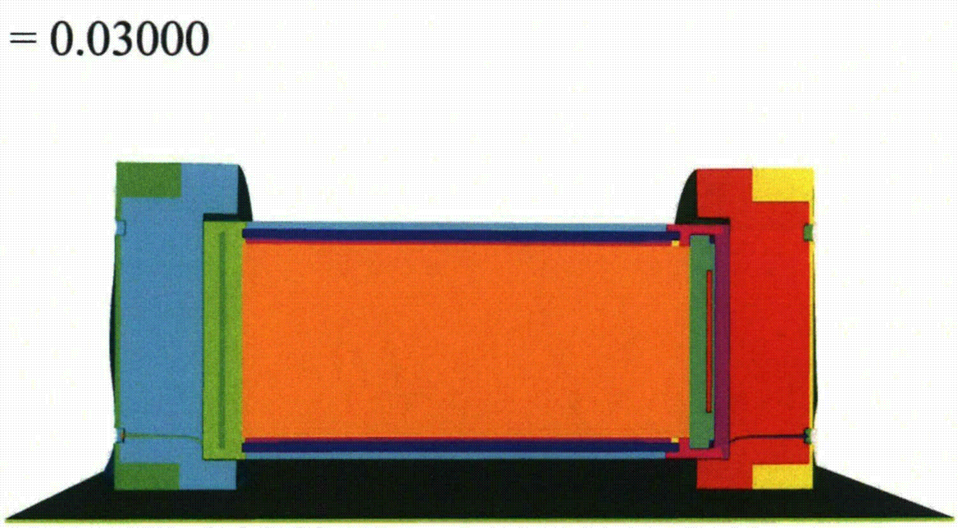


Figure III-37 Rail-Lead cask side impact at 48 kph (30 mph)

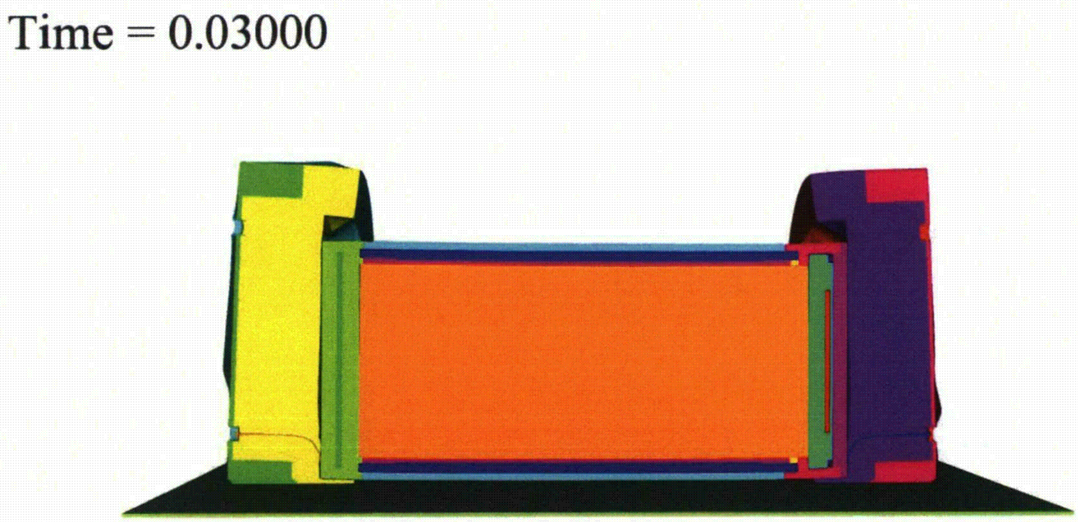


Figure III-38 Rail-Lead cask side impact at 97 kph (60 mph)

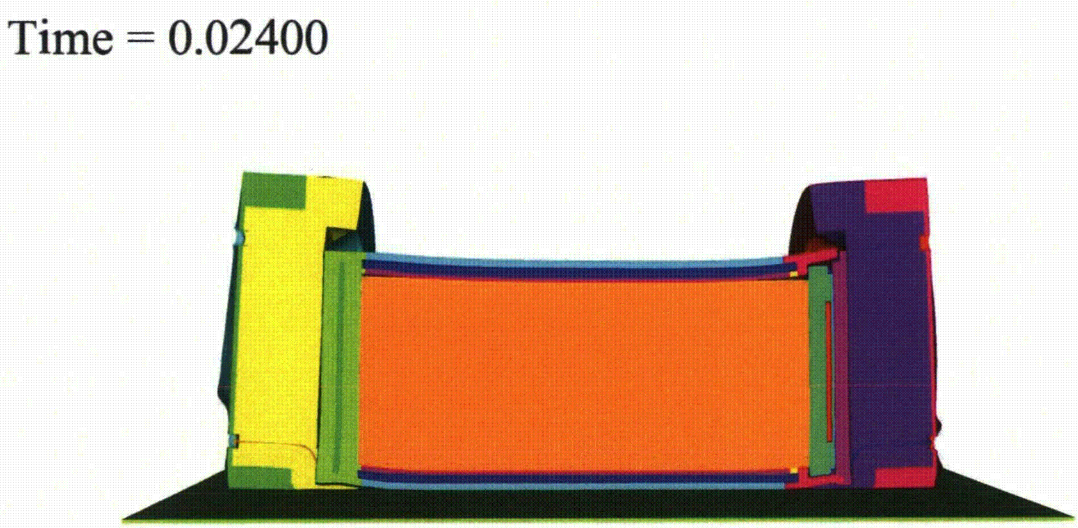


Figure III-39 Rail-Lead cask side impact at 145 kph (90 mph)

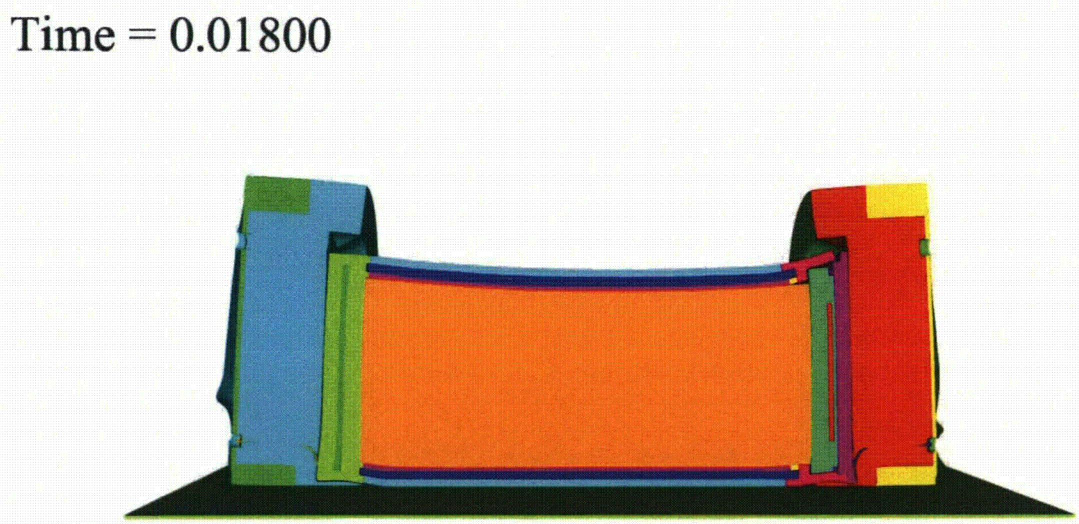


Figure III-40 Rail-Lead cask side impact at 193 kph (120 mph)

<u>Analysis Summary</u>

For the 48-kph (30-mph) impact analyses (the impact velocity from the regulatory hypothetical impact accident), the impact limiter absorbed almost all of the kinetic energy of the cask. No damage to the cask body occurred. The response of the Rail-Lead cask is more complicated than that of the Rail-Steel cask. **Error! Reference source not found.** gives the peak acceleration for each impact case. As expected, the accelerations for the side impacts are the highest and those for the corner impacts are the lowest. For the end orientation, as the impact velocity increases, there is initially additional damage to the impact limiter because it is absorbing more kinetic energy (this shows the margin of safety in the impact limiter design). At 97 kph (60 mph), there is no significant damage to the cask body or canister. At an impact speed of 145 kph (90 mph), 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, significant slumping of the lead gamma shielding material occurs, resulting in a loss of shielding near the end of the cask away from the impact point (Chapter 5 and Appendix V discuss this further). As the impact velocity is increased to 193 kph (120 mph), 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 multipurpose canister), some loss of radioactive contents could occur if the cask has metallic seals but not if the cask has elastomeric seals. A more detailed discussion of leakage is provided later in this section.

For the corner impacts at 97 and 145 kph (60 and 90 mph), there is some damage to the cask body, in addition to deformation of the impact limiter, which 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 it is enough to cause a leak if the cask is sealed with metallic o-rings. For a corner impact at 193 kph (120 mph), there is more significant deformation to the cask, more lead slump, and a larger gap between the lid and the cask body. Figure III-36 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 centimeters (12 inches).

Table III-2 Peak Acceleration from Each Analysis of the Rail-Lead Cask

Orientation	Speed, kph (mph)	Peak Accel. (g)
End	48 (30)	58.5
	97 (60)	111.6
	145 (90)	357.6
	193 (120)	555.5
Corner	48 (30)	36.8
	97 (60)	132.2
	145 (90)	256.7
	193 (120)	375.7
Side	48 (30)	76.1
	97 (60)	178.1
	145 (90)	411.3
	193 (120)	601.1

In the side impact, as the impact velocity increases from 48 kph (30 mph) to 97 kph (60 mph), the impact limiter ceases to absorb additional energy and permanent deformation of the cask and closure bolts occurs. The resulting gap between the lids and the cask body is sufficient to allow leakage if there is a metallic seal but not if there is an elastomeric seal. When the impact speed is increased to 145 kph (90 mph), 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 III-39 shows the deformed shape of the cask following this impact. The response of the cask to the 193-kph (120-mph) impact is similar to that from the 145-kph (90-mph) impact, except that the gaps between the lids and the cask are larger.

III.3.6 Determination of Lid Gaps

Possible gaps between the lids and the cask were extracted from the final drop results. The longitudinal orientation of the cask was along the y-direction, so the difference in y-direction displacement between the lid and the cask gave a measure of the gap. Researchers paired a node on the cask with the nearest node on the lid for this gap calculation. The nodes did not align exactly in the x-z plane. Researchers then calculated two gap values for the end drop orientation since the deformations were axisymmetric. For side down and CG over-corner orientations, researchers calculated gap values at five equally spaced locations around the half-circumference of the cask, as shown in Figure III-41 to Figure III-43.

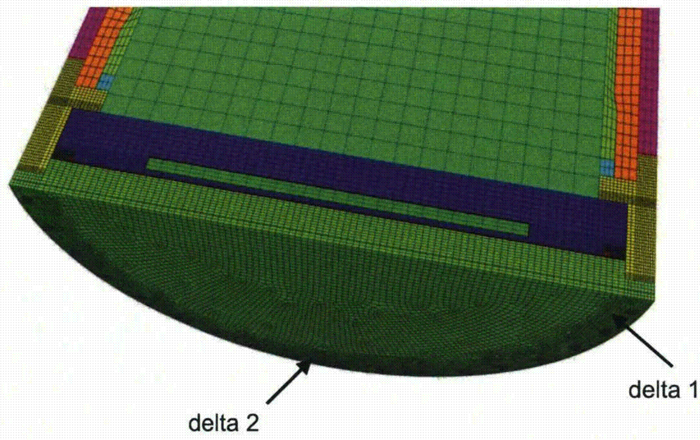


Figure III-41 Gap opening locations for end impact orientation

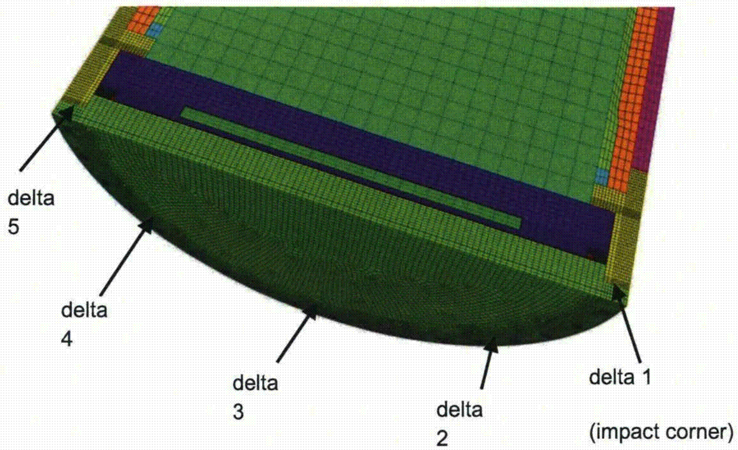


Figure III-42 Gap opening locations for corner impact orientation

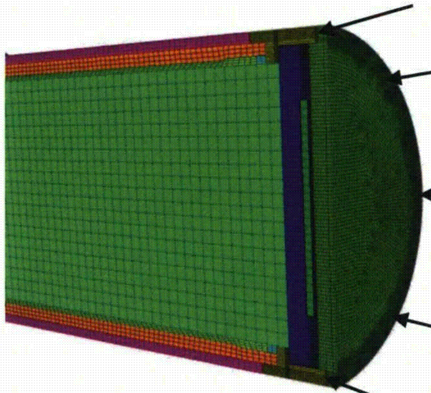


Figure III-43 Gap opening locations for side impact orientation

The next set of figures (Figure III-44 through Figure III-53) show plots of the gap sizes as a function of time for the inner and outer lid for each analysis case. All of the gaps calculated are somewhat conservative because the bolts did not include any preload. Preload decreased the gap size because the bolts do not start to elongate until the preload is overcome. As an example, if the 18-cm (7.1-inch) long inner lid bolts are preloaded to 50 percent of their yield strength (0.5 * 1,040 = 520 MPa (75.4 ksi)), the elastic elongation is 0.46 mm (0.018 inches). This indicates that the calculated gap for the inner lid is probably overestimated by this amount.

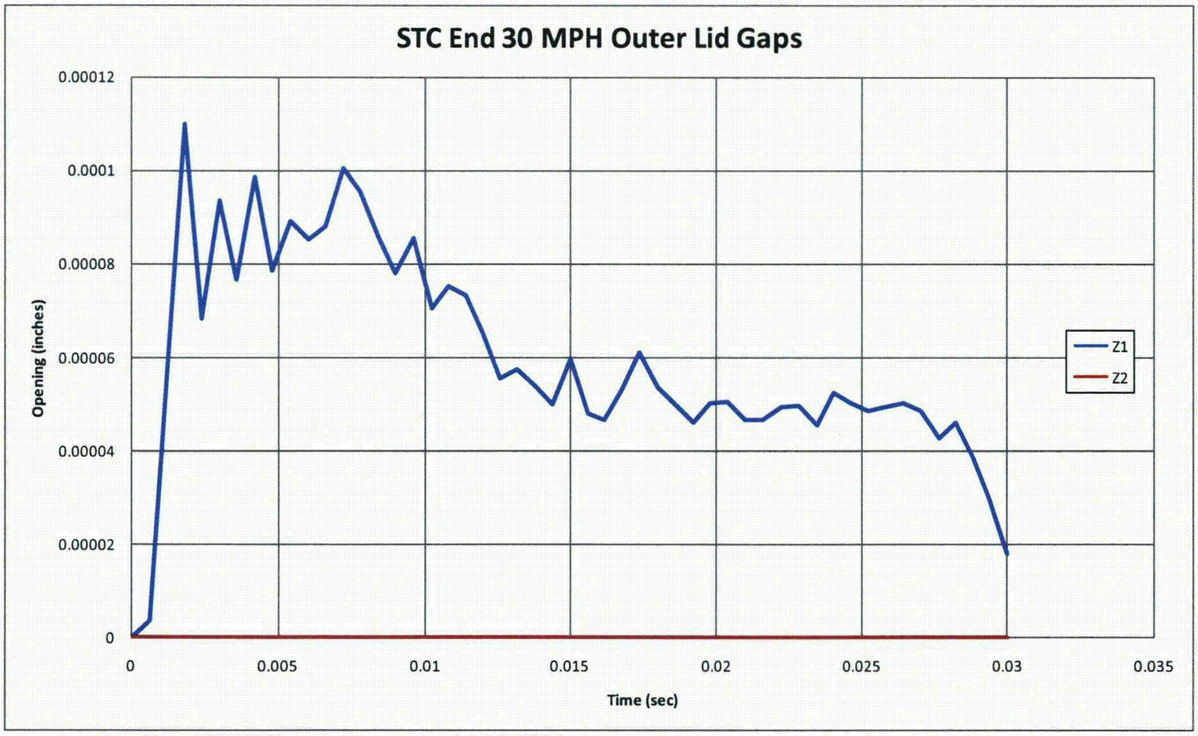
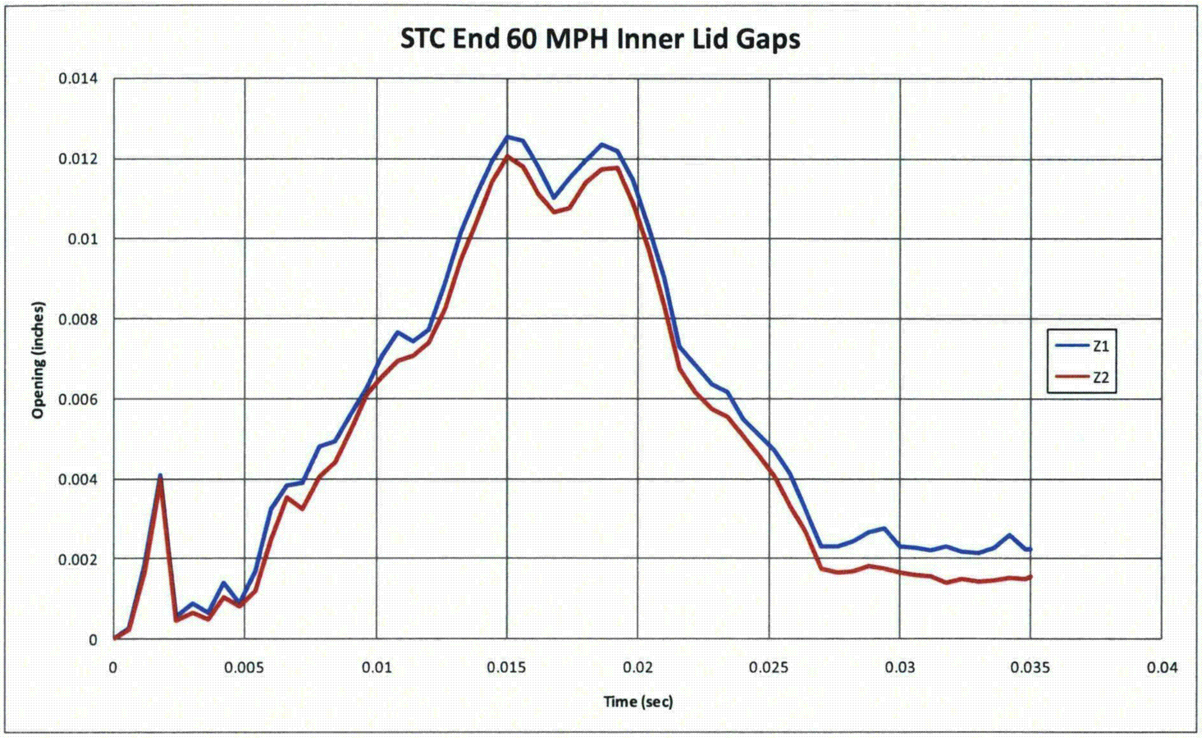


Figure III-44 Gaps in the inner and outer lids of the Rail-Lead cask from the end impact at 48 kph (30 mph)



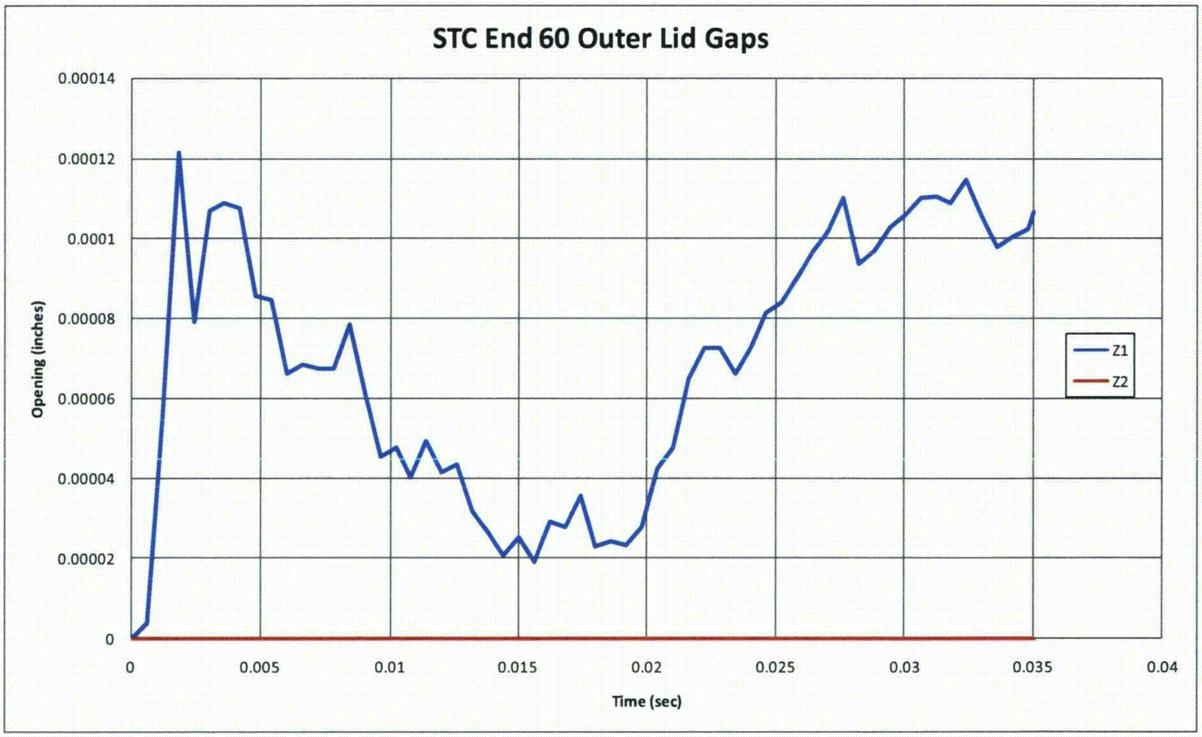
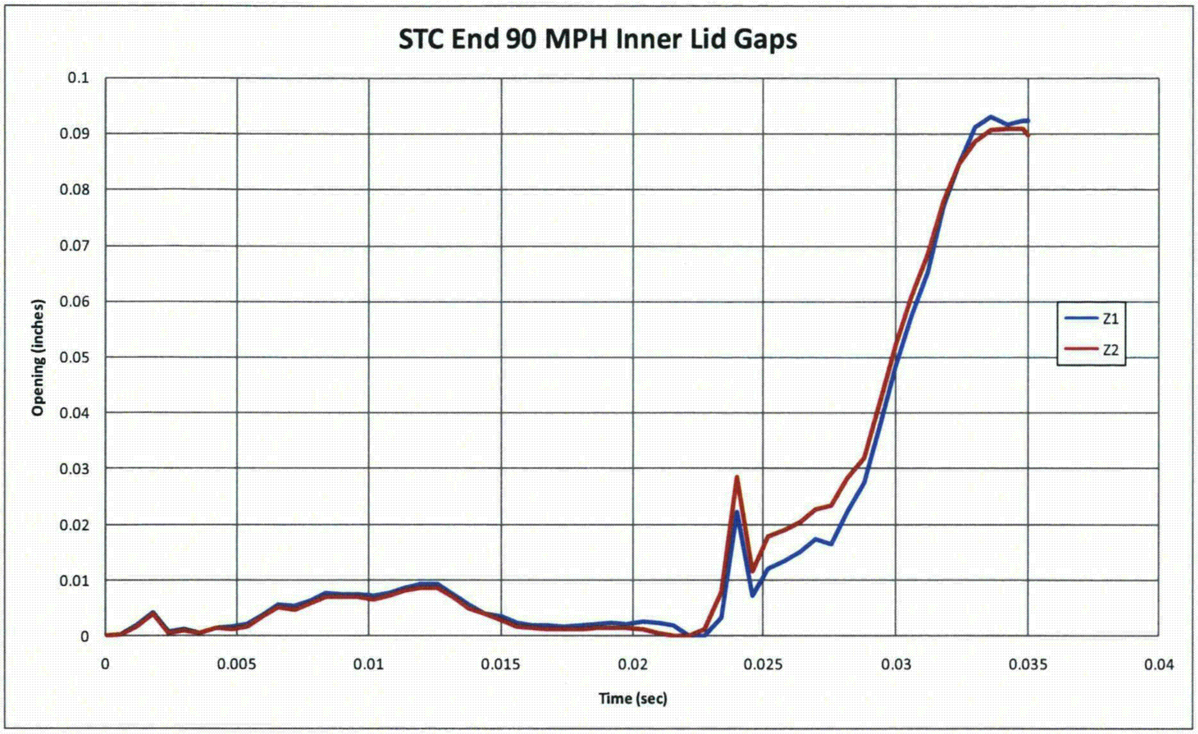


Figure III-45 Gaps in the inner and outer lids of the Rail-Lead cask from the end impact at 97 kph (60 mph)



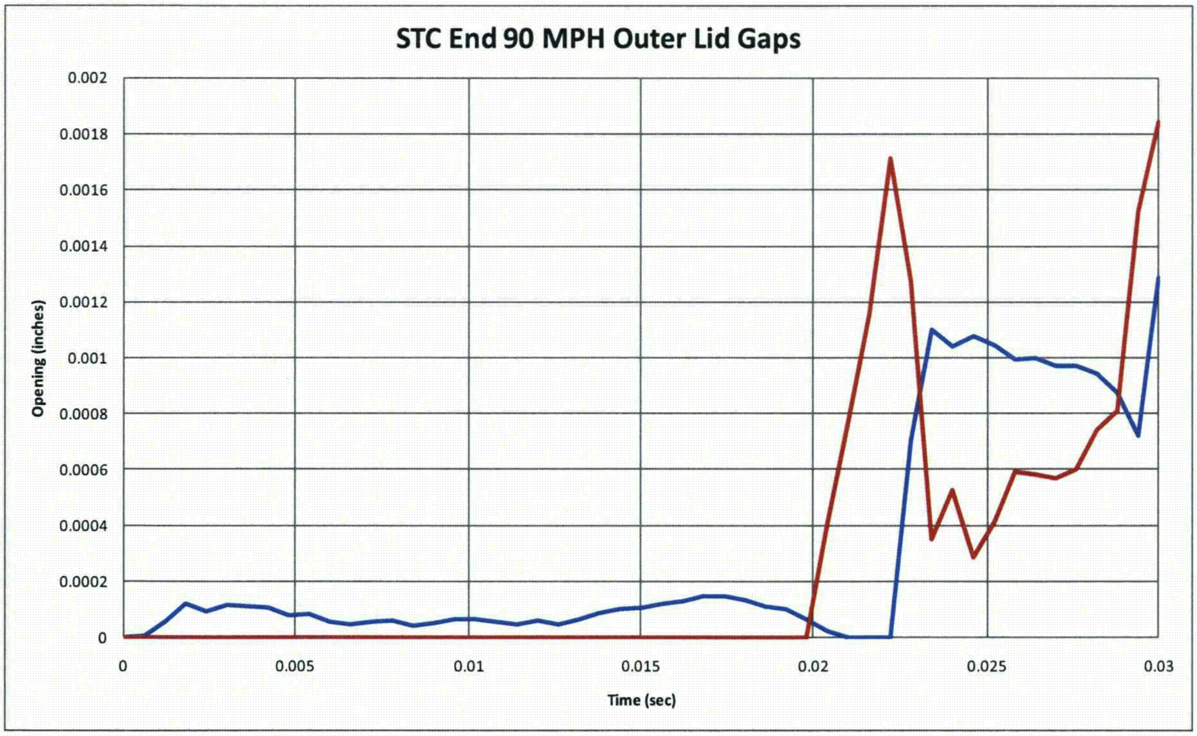


Figure III-46 Gaps in the inner and outer lids of the Rail-Lead cask from the end impact at 145 kph (90 mph)

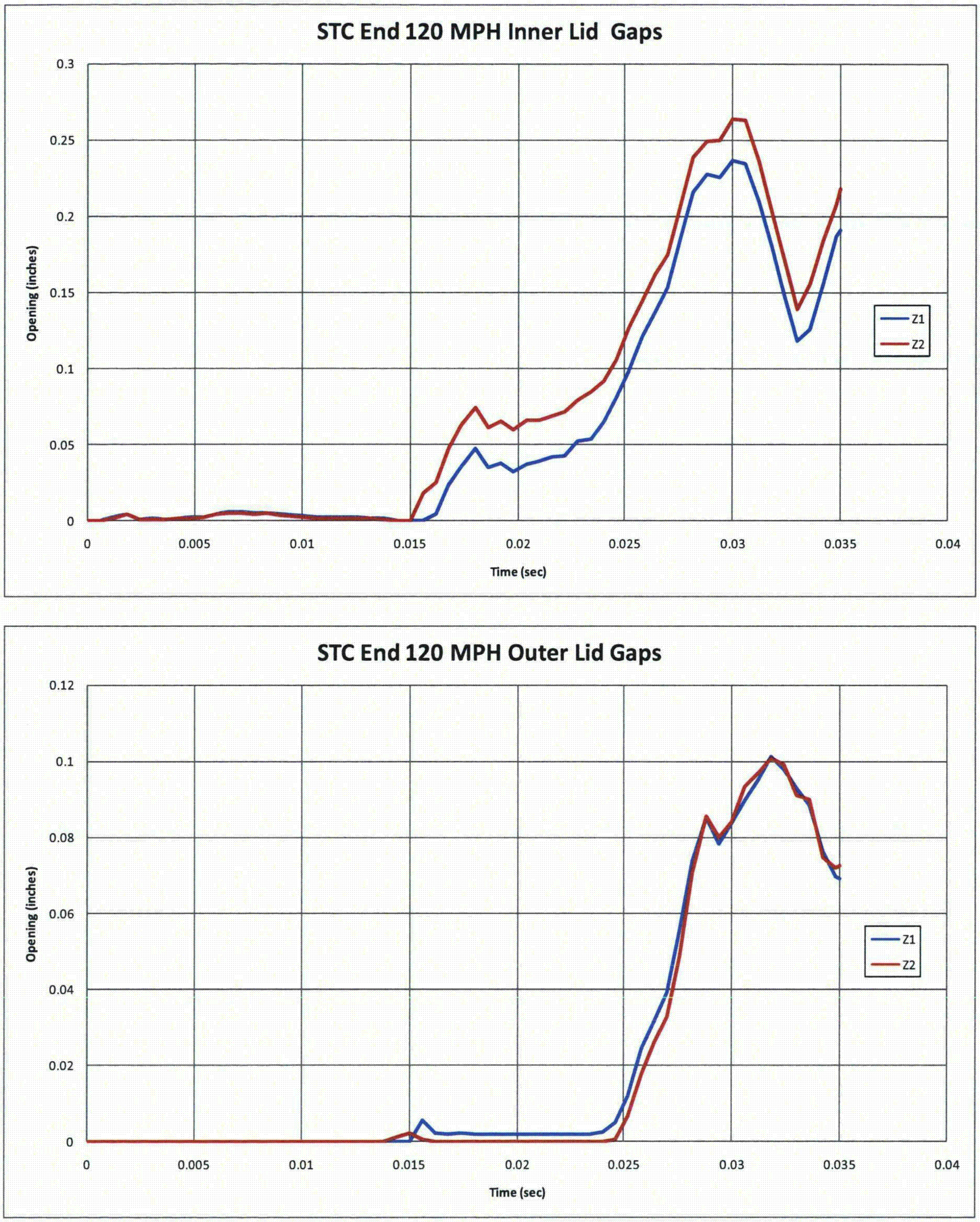
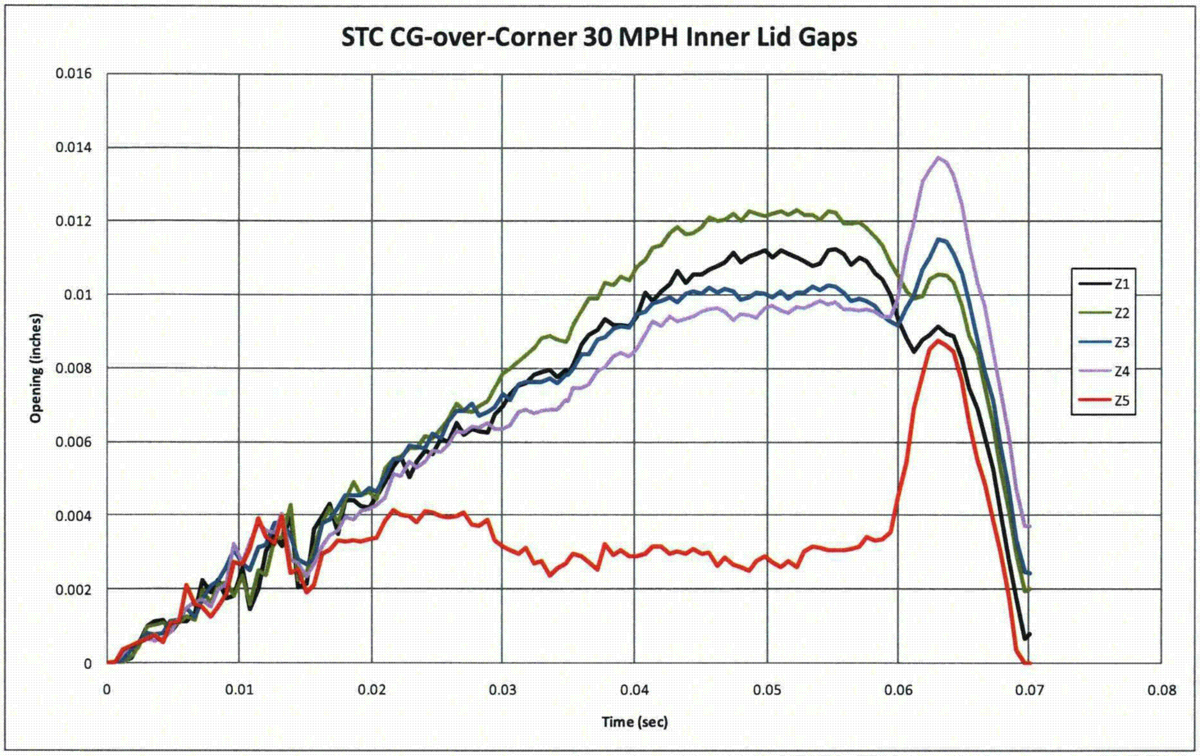


Figure III-47 Gaps in the inner and outer lids of the Rail-Lead cask from the end impact at 193 kph (120 mph)



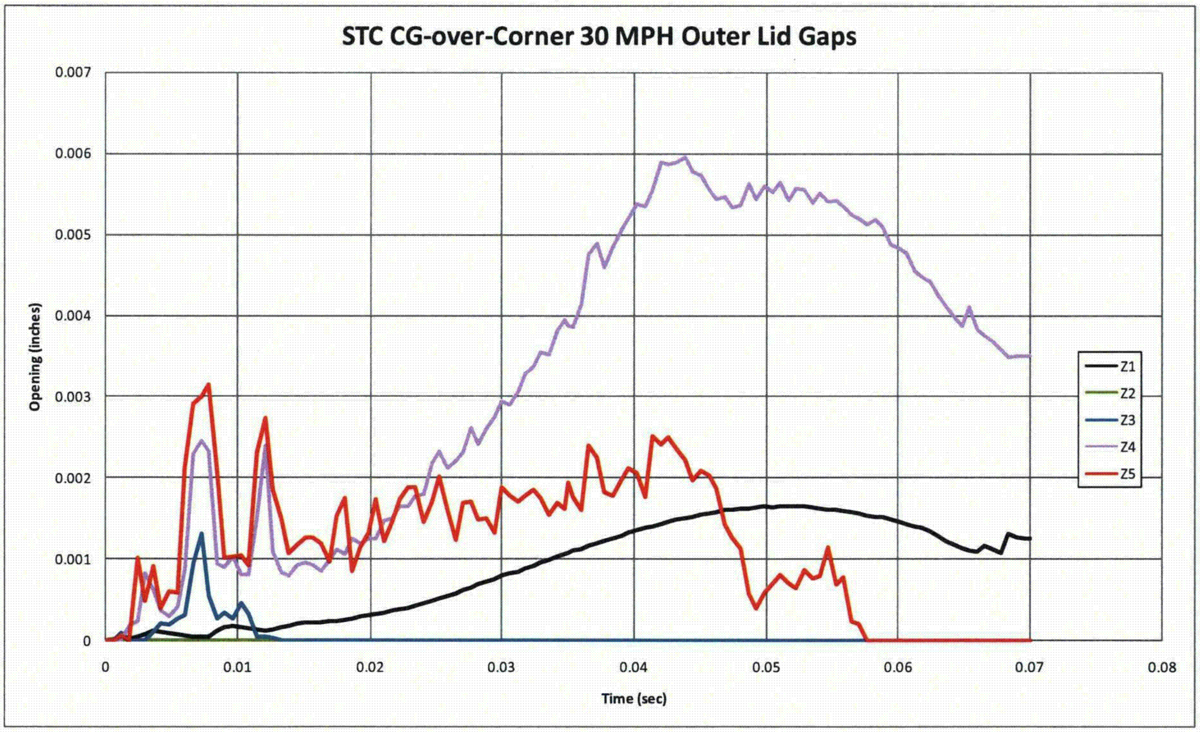
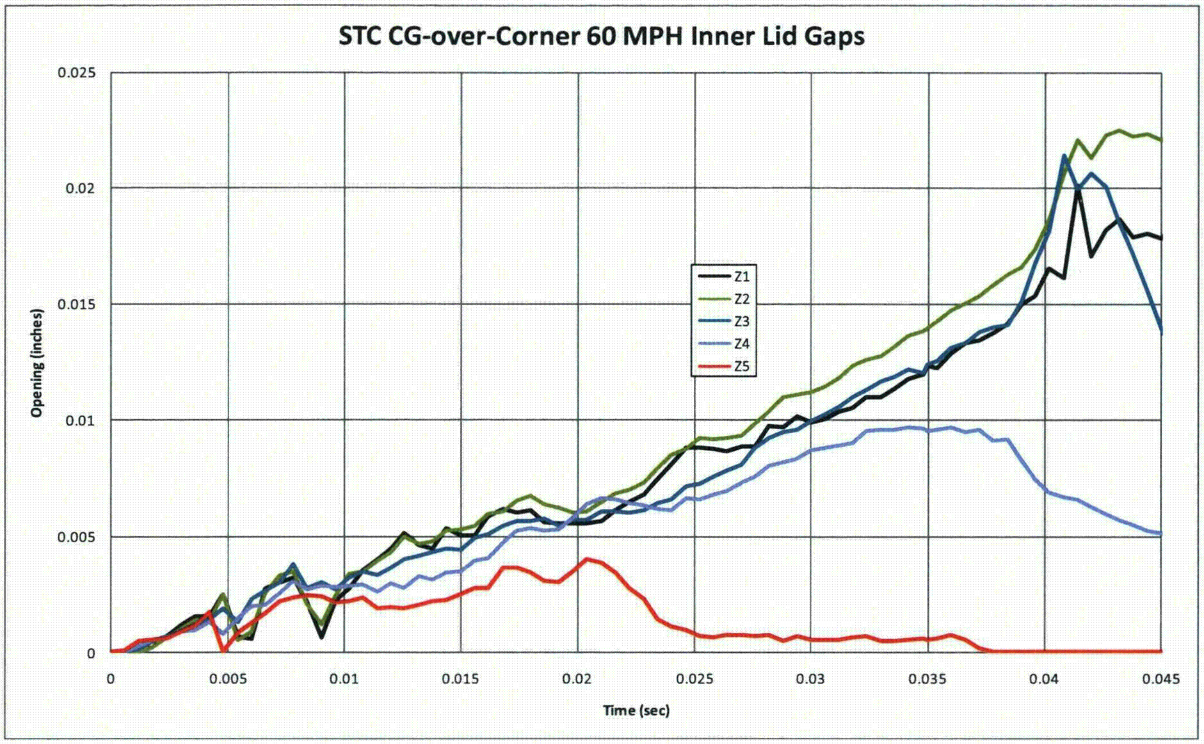


Figure III-48 Gaps in the inner and outer lids of the Rail-Lead cask from the corner impact at 48 kph (30 mph)



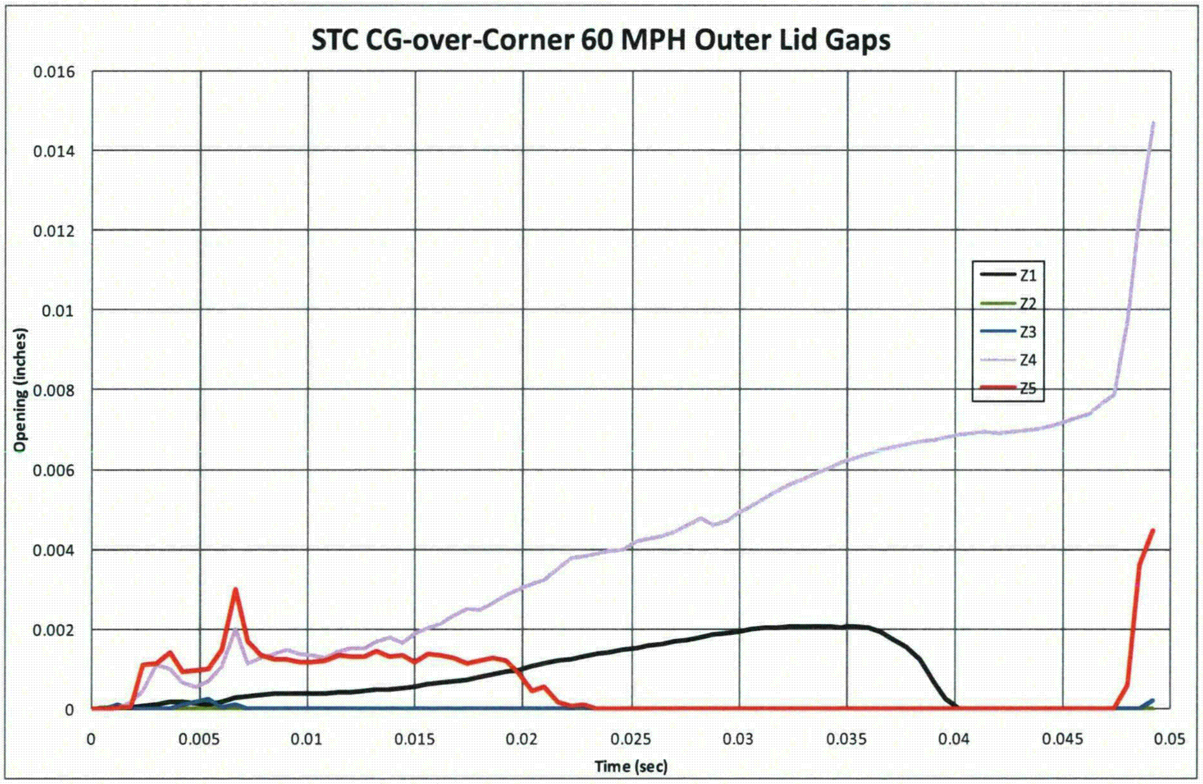
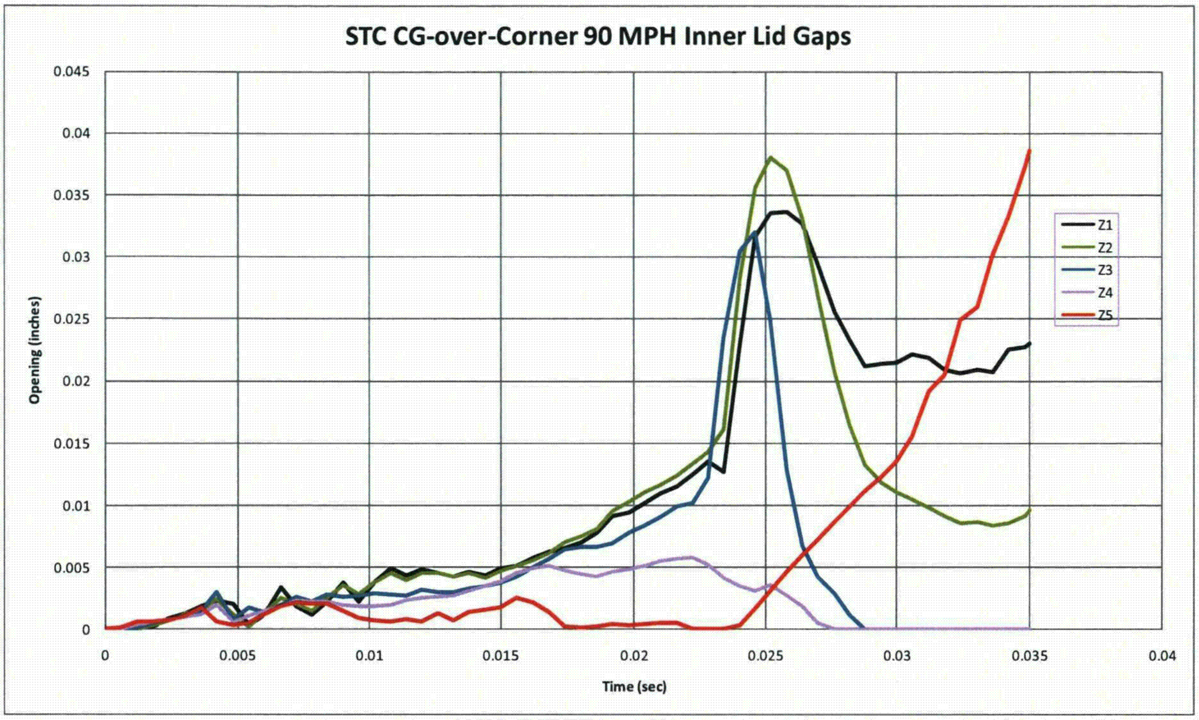


Figure III-49 Gaps in the inner and outer lids of the Rail-Lead cask from the corner impact at 97 kph (60 mph)



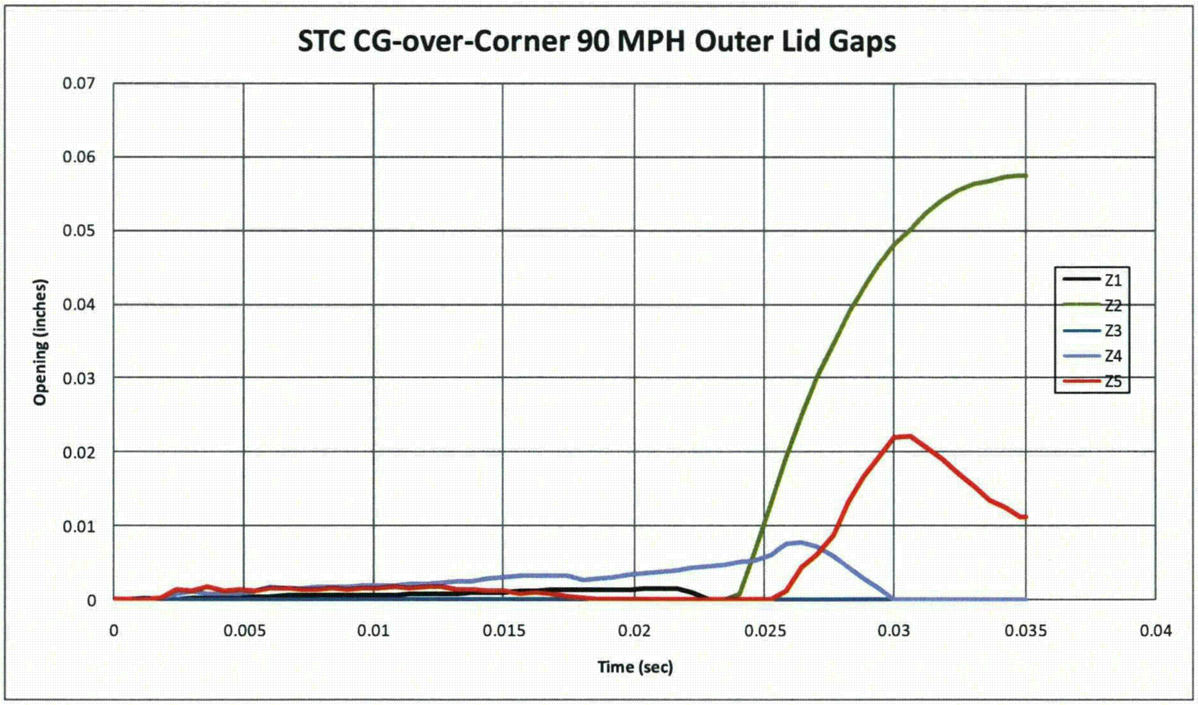
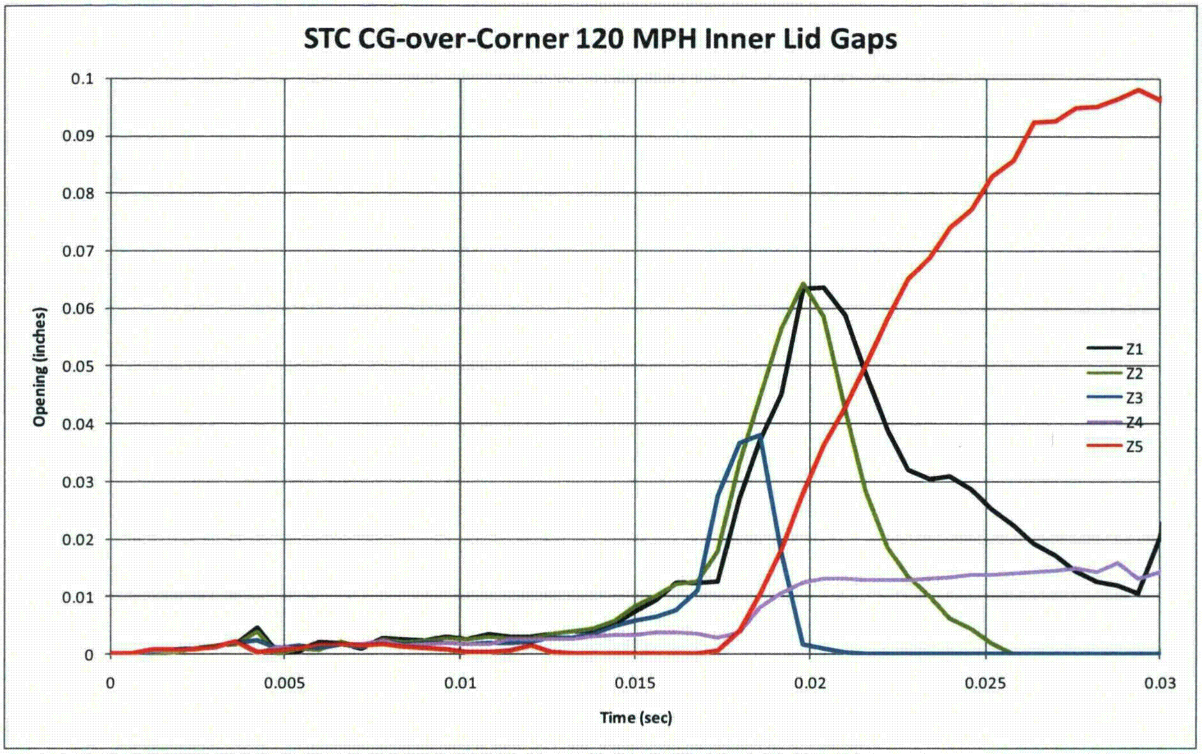


Figure III-50 Gaps in the inner and outer lids of the Rail-Lead cask from the corner impact at 145 kph (90 mph)



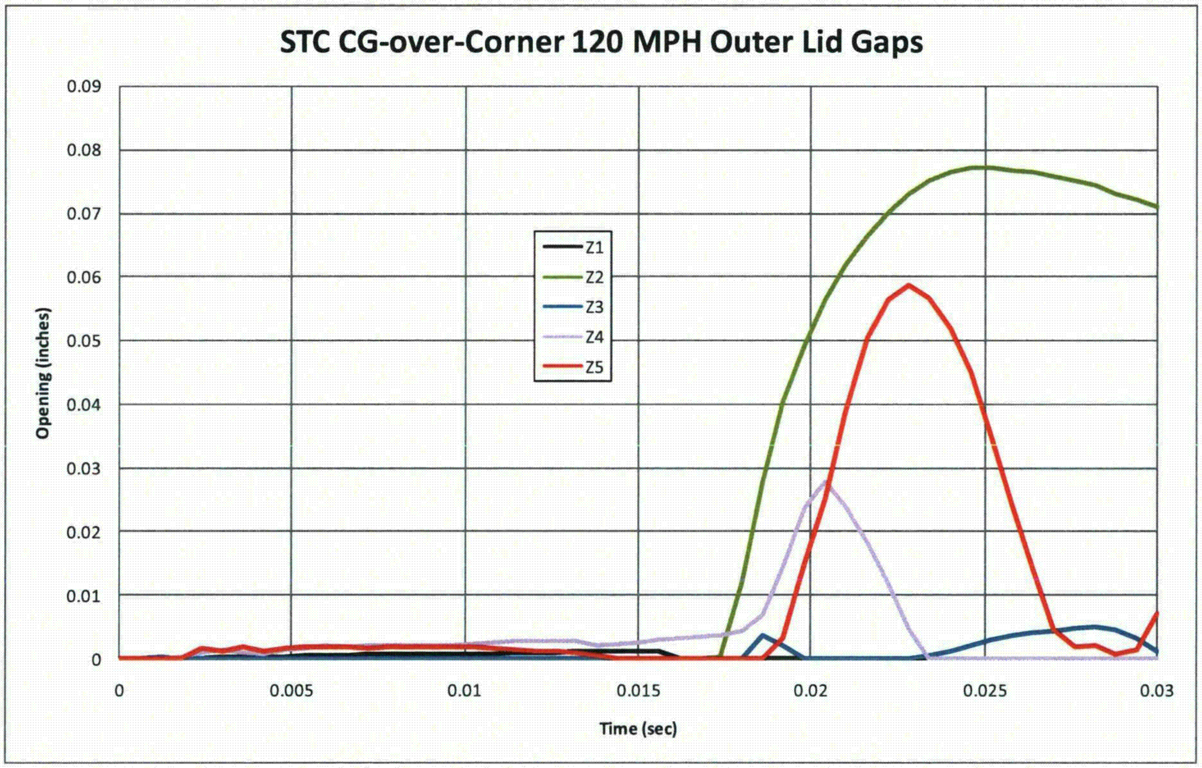
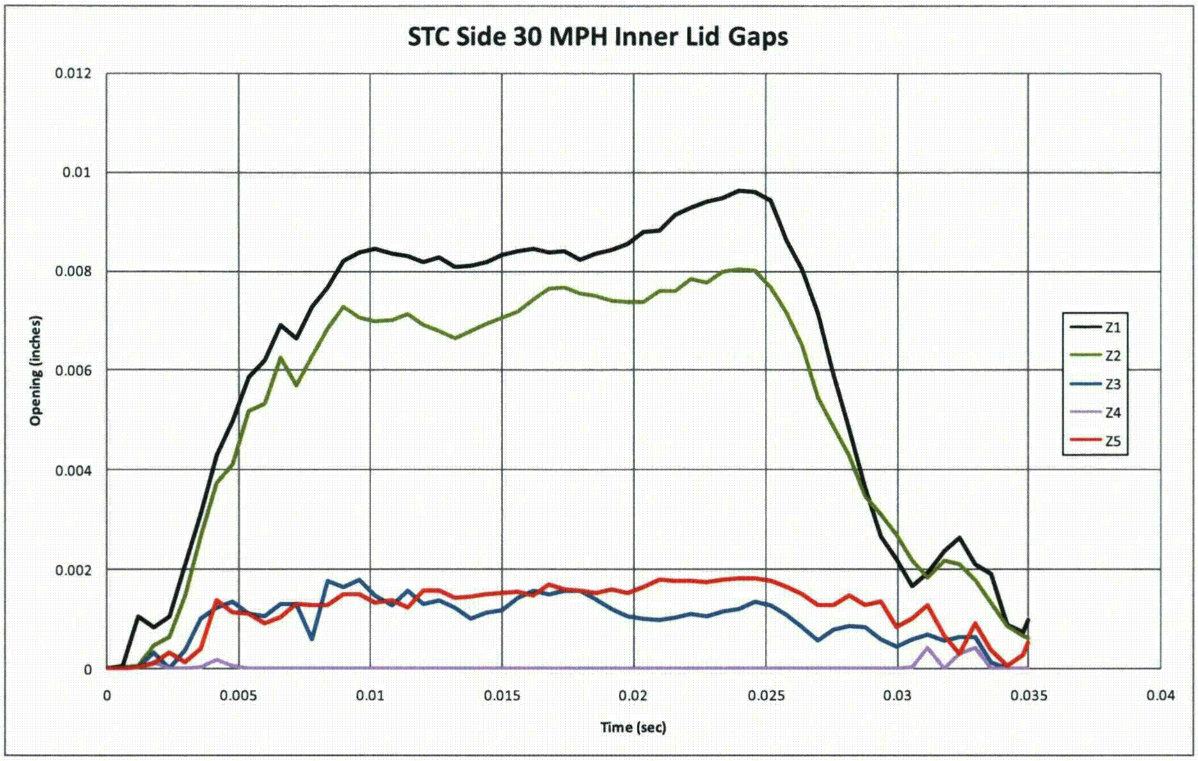


Figure III-51 Gaps in the inner and outer lids of the Rail-Lead cask from the corner impact at 193 kph (120 mph)



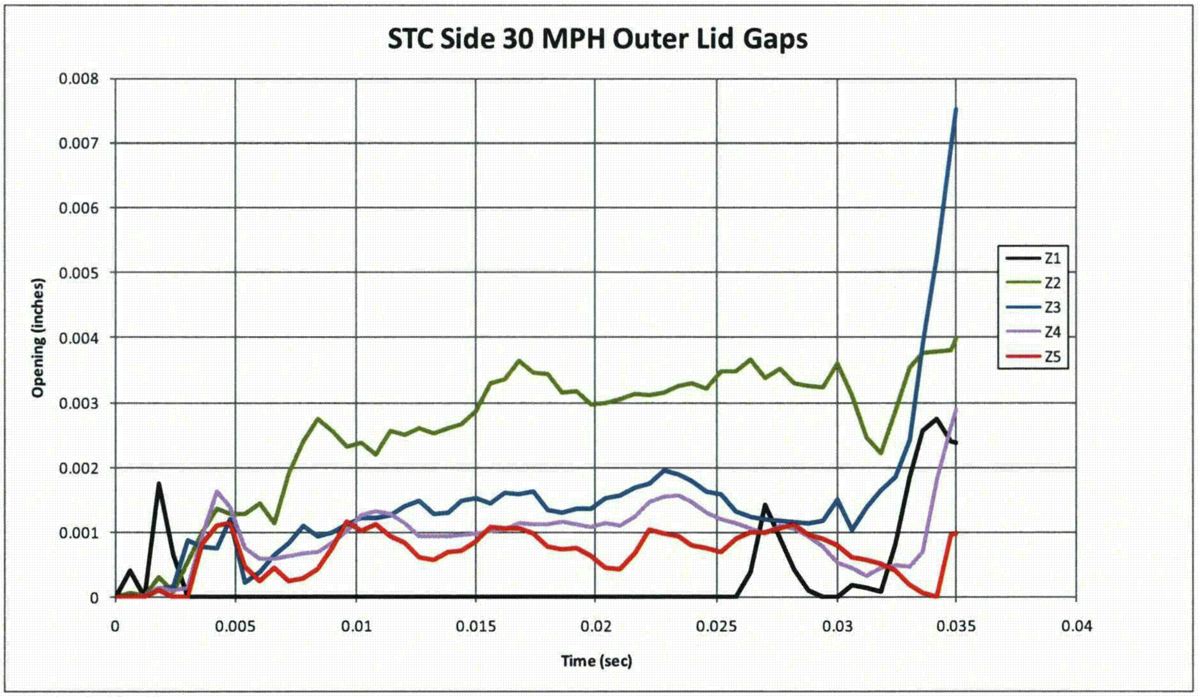
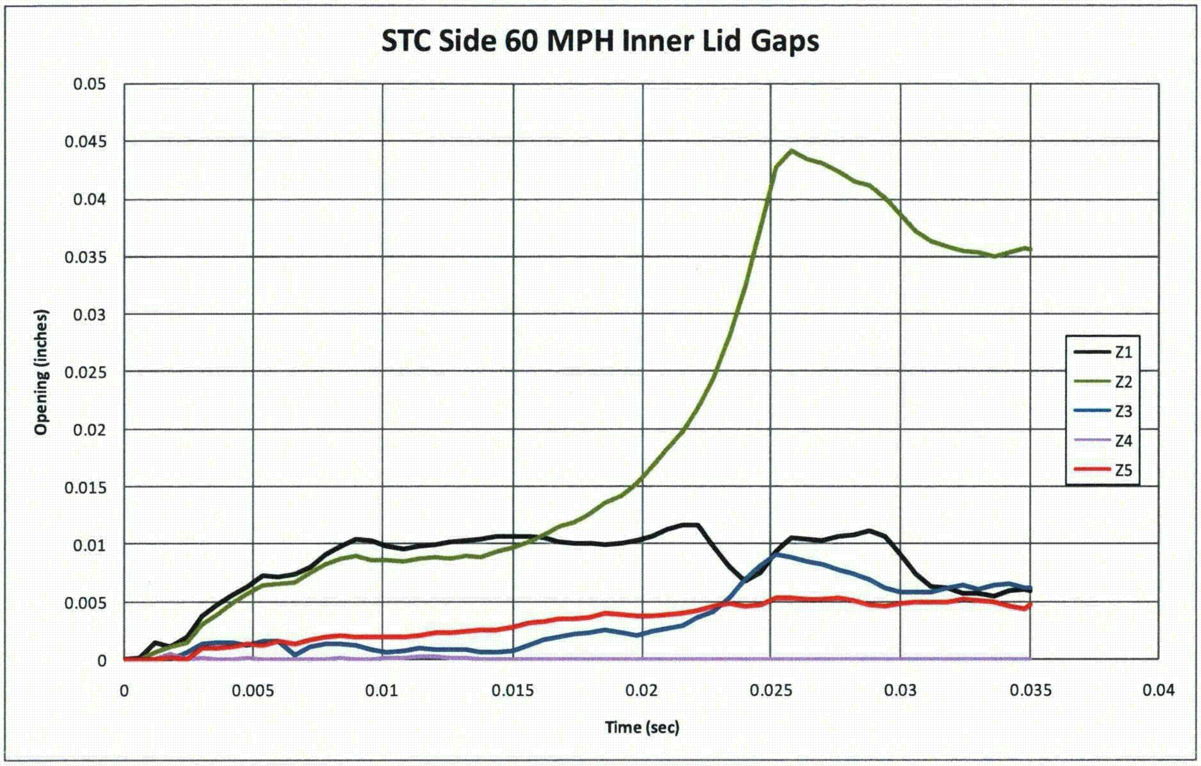


Figure III-52 Gaps in the inner and outer lids of the Rail-Lead cask from the side impact at 48 kph (30 mph)



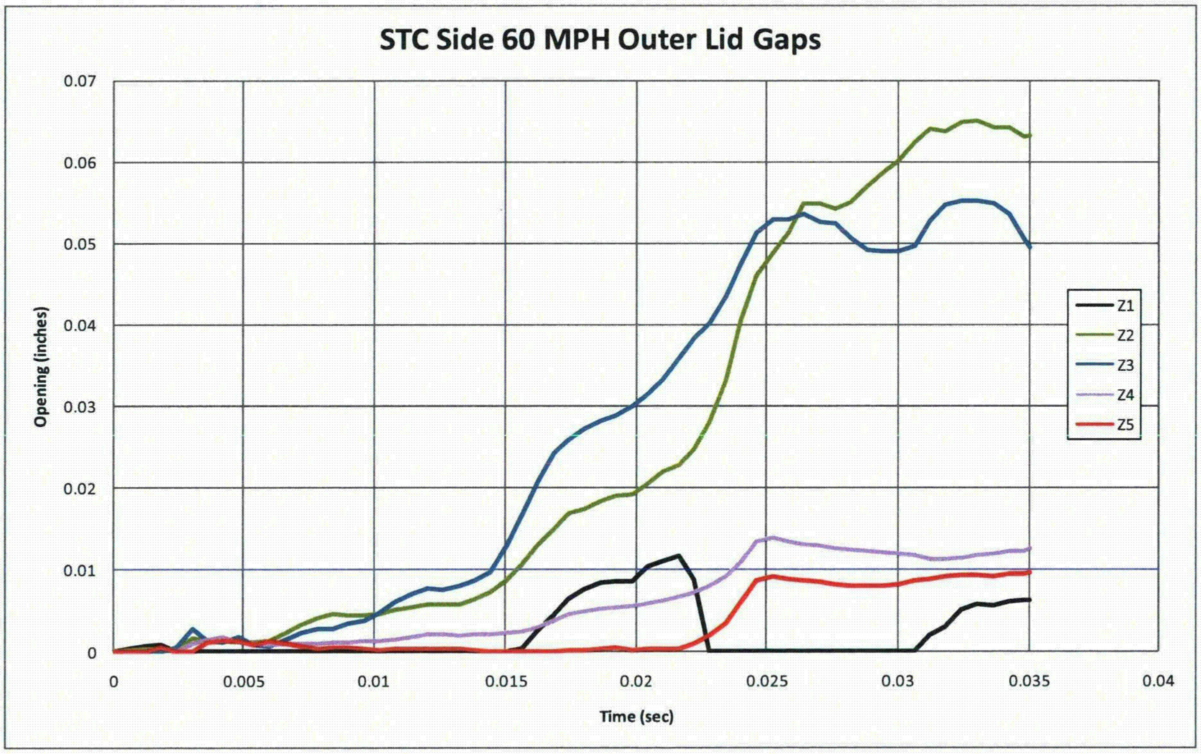


Figure III-53 Gaps in the inner and outer lids of the Rail-Lead cask from the side impact at 97 kph (60 mph)

To calculate any leak size based upon the gaps, researchers had to take the compliance of the o-rings into account. The Rail-Lead cask can be sealed with either elastomeric o-rings or metallic o-rings. Elastomeric o-rings of the types used in transportation casks can typically maintain a seal when the opening between the mating surfaces opens by 2.5 millimeters (0.10 inch). This number is used as the compliance for the cases with elastomeric o-rings. Unfortunately, no data are available for the specific o-rings used in this cask, and the actual compliance may be more or less than 2.5 millimeters. In any case, the hole sizes from the case with elastomeric o-rings are less than those for the cases with metallic o-rings. Metallic o-rings are much less tolerant to gaps, and a value of 0.25 millimeters (0.010 inch) is used as the compliance for the cases with metallic o-rings. For the end impact analyses, the gap size is uniform for the entire circumference of the seal, and the hole size is calculated by subtracting the compliance of the o-ring from the gap and multiplying by the circumference. If either the inner seal or the outer seal has a gap less than the compliance, then there is no leak area. For end impacts, the only case in which any leakage occurs is the 193-kph (120-mph) impact with metallic o-rings.

For the corner and side impacts, the amount of gap varies around the circumference of the seal, and a more complicated algorithm is needed to calculate the hole size. As in the end impact, the compliance of the seal is subtracted from the gap and a trapezoidal area between measurement locations is assumed. In the corner impact, none of the gaps are large enough to overcome the compliance of elastomeric o-rings. But some leakage would occur at impacts of 97 kph (60 mph), 145 kph (90 mph), and 193 kph (120 mph) for the case where the cask is sealed with metallic o-rings. The calculated hole sizes for these three cases are 65, 599, and 1,716 square millimeters (mm^2), (0.10, 0.928. and 2.66 in^2) respectively. In the side impact at 97 kph (60 mph), the gaps are not sufficient to cause a leakage with elastomeric seals. But with metallic seals, a hole size of 799 mm^2 (1.24 in^2) is calculated. In the 145-kph (90-mph) and 193-kph (120-mph) analyses, there are a number of failed bolts and very large openings between the lids and the cask body. In these cases, both the elastomeric and metallic seals fail and the resulting hole size is more than 10,000 mm^2(16 in^2). **Error! Reference source not found.** gives the final gap and hole sizes for each of the analyses.

Table III-3 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

III.4 Impacts onto Yielding Targets

III.4.1 Introduction

The finite element results discussed in the previous section apply only to impacts onto a rigid target. For this type of impact, the cask absorbs the entire kinetic energy of the impact. For finite element analyses, a rigid target is easily implemented by enforcing a no-displacement boundary condition at the target surface. In real life, the construction of a rigid target is impossible, but it is possible to construct a target that is sufficiently rigid that increasing its rigidity does not increase the amount of damage to the cask. This is because in real impacts there is a sharing of energy absorption between the cask and the target. If the target is much weaker than the cask, the target will absorb most of the energy. If the target is much stronger than the cask, the cask will absorb most of the energy. In this section, the partitioning of the drop energy between the four generic casks and several "real-world" targets will be developed to obtain impact speeds onto real surfaces that give the same damage as impacts onto rigid targets. Researchers considered impacts onto hard desert soil, concrete highways, and hard rock. They did not specifically test impacts onto water surfaces, but this scenario is also discussed. In addition, the probability of puncture of the cask caused by impact against a nonflat surface (or impact by a puncture probe) is developed.

III.4.2 Method

For each finite element calculation for impact onto a rigid target the total kinetic energy of the finite element model is output at 100 time-steps through the analysis. The total kinetic energy is one-half of the sum of the mass associated with each node times the velocity of that node squared. Figure III-54 shows kinetic energy time-histories for the steel-lead-steel truck cask for each orientation from the 197-kph (120-mph) impact analyses. From the time-history of kinetic energy, researchers derived a velocity time-history. They calculated the rigid-body velocity for each time-step by assuming that all of the kinetic energy of the model is caused by velocity in the direction of the impact. Equation III-1 shows this mathematically:

$$v_t = \sqrt{\frac{2KE_t}{\sum m_i}} \qquad \text{(III-1)}$$

Where v_t is the velocity at time t, KE_t is the kinetic energy at time t, m_i is the mass associated with node i, and the summation is over all of the nodes in the finite element model.

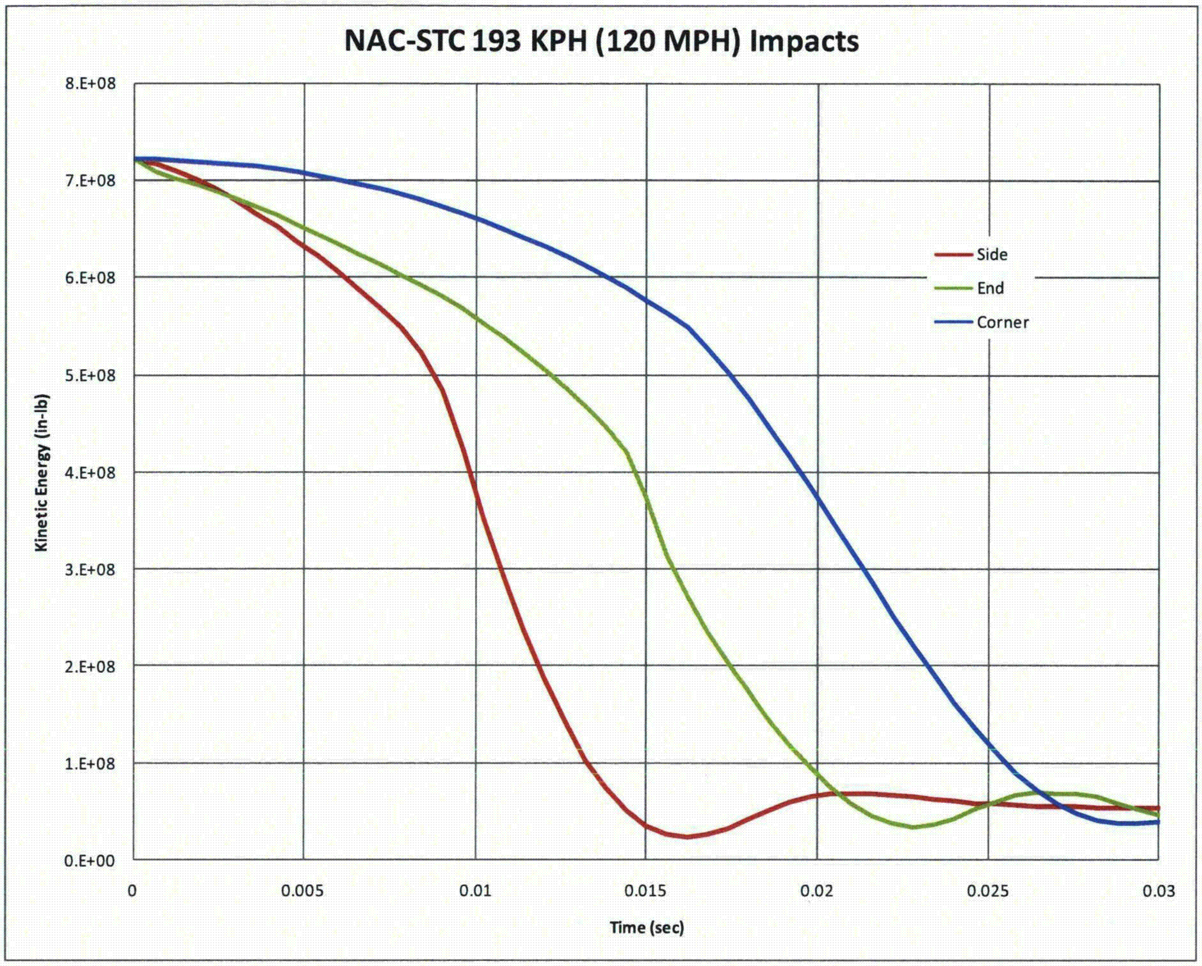


Figure III-54 Kinetic energy time-histories for the Rail-Lead cask from 193-kph (120-mph) impact analyses in the end, side, and corner orientations

For each analysis, the peak contact force is determined. **Error! Reference source not found.** lists these forces. For an impact onto a real target to be as damaging to the cask as the impact onto the rigid target, the target must be able to impart a force equal to this peak force to the cask. Because casks are complex structures and the rate of load application for impacts onto yielding targets is slower than for a rigid target, it is likely the actual damage to the cask for the yielding target impacts would be less than that calculated using this method. The wide variety of yielding target types makes it impossible to quantify the amount of conservatism that results.

The energy absorbed by the target in developing this force is added to the initial kinetic energy of the cask. This total absorbed energy is used to calculate an equivalent velocity by replacing KE_t in Equation III-1 with the total energy.

Table III-4 Peak Contact Force for the Rail-Lead Cask Impacts onto an Unyielding Target (Bold Numbers Are Cases In Which 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**

III.4.3 Soil Targets

The force that hard desert soil imparts onto a cask following an impact was derived from results of impact tests performed by Gonzales (1987), Waddoups (1975), and Bonzon and Schamaun (1976). The tests by Gonzales and Waddoups used casks that were comparable to Rail-Lead casks, but much smaller. The tests by Bonzon and Schamaun were with casks that were less stiff than the Rail-Lead cask. This large amount of test data was used to develop an empirical soil target force-deflection equation that is a function of impactor area. Figure III-55 shows the force-deflection curves for impact of the Rail-Lead cask onto a soil target. Corner impacts were assumed to have the same contact area on the soil target as the end impacts, so only two curves are shown. Similar curves were developed for each of the other casks. Comparison of Figure III-55 with the forces in **Error! Reference source not found.** shows that many of the impacts will result in very large soil penetrations. This is consistent with the results seen in the tests performed by Waddoups, where casks were dropped 610 meters (2,000 feet) from a helicopter. Penetration depths for these impacts were up to 2.4 meters (8 feet), and the equivalent rigid target impact velocity was less than 48 kph (30 mph). Integration of the force-deflection curve up to the peak contact force determines the amount of energy absorbed by the target.

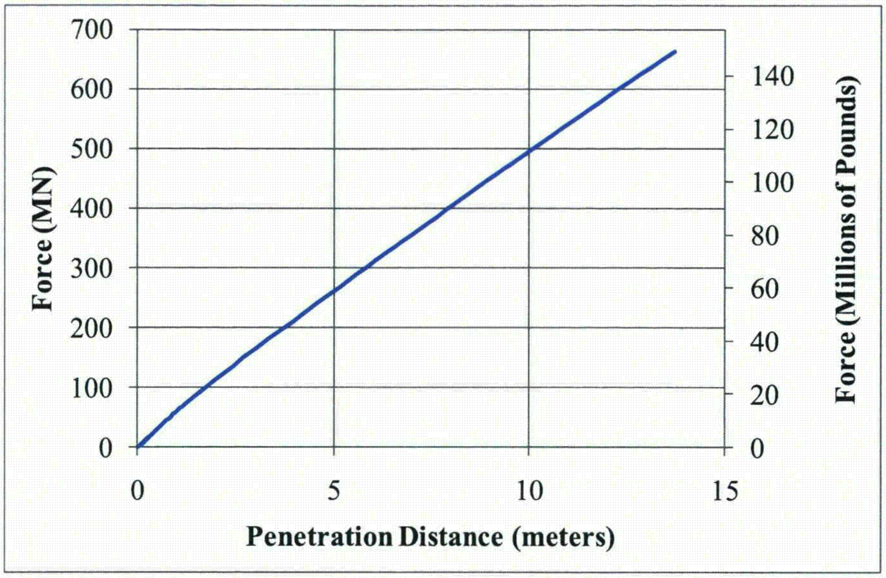


Figure III-55 Force generated by the Rail-Lead cask penetrating hard desert soil

III.4.4 Concrete Targets

The force imparted to a cask by impact onto a concrete target is derived from test results by Gonzales (1987). In his series of tests, a cask-like test unit impacted two types of concrete targets, one 12 inches thick and one 18 inches thick, at velocities from 48 to 97 kph (30 to 60 mph). All of the impacts were in an end-on orientation. Based on the results of these tests and engineering mechanics, researchers derived an empirical relationship between the force and energy absorbed. For impacts onto concrete slab targets, there are two mechanisms that produce large forces onto the cask. The first is the generation of a shear plug in the concrete. The force required to produce this shear plug is linearly related to the impact velocity, the diameter of the impacting body, and the thickness of the concrete. Equation III-2 gives the empirical equation for the force required to produce the shear plug:

(III-2) $$F_s = C_s v_e d_i t_c$$

Where F_s is the force required to produce the shear plug, C_s is an empirical constant (16.84), v_e is the equivalent impact velocity, d_i is the diameter of the impacting object, and t_c is the thickness of the concrete slab.

The energy absorbed in producing this shear plug is linearly related to the cask diameter, the square of the impact velocity, and the fourth root of the slab thickness. Equation III-3 gives the empirical equation for the energy required to produce the shear plug:

(III-3) $$E_s = C_e d_i v_e^2 t_c^{0.25}$$

Where E_s is the energy required to produce the shear plug, and C_e is an empirical constant (0.00676).

After the shear plug is formed, further resistance to penetration is achieved by the behavior of the subgrade and soil beneath the concrete. This material is being penetrated by the cask and the shear plug. Generally, the shear plug forms with 45-degree slopes on the side. Therefore, the diameter of the soil being penetrated is equal to the cask diameter plus twice the slab thickness. The behavior of the subgrade and soil is assumed to be the same as the hard desert soil used for the soil target impacts. Figure III-56 compares the empirical relationship with one of the tests performed by Gonzales.

For corner and side impacts, an equivalent diameter is calculated to fit with the empirical equations. For each case, the diameter is calculated by assuming the shear plug forms when the concrete target has been penetrated 5 cm (2 inches). The area of the equivalent diameter is equal to the area of the concrete in contact with the cask when the penetration depth is 5 cm (2 inches). To calculate the equivalent velocity for concrete targets, the force required to generate the shear plug must be compared to the peak contact force for the impact onto the rigid target. The velocity required to produce this force can be calculated from Equation III-2. The kinetic energy associated with this velocity is absorbed by a combination of producing the shear plug, penetration of the subgrade and soil beneath the concrete, and deformation of the cask. The energy absorbed in producing the shear plug is calculated by Equation III-3, the energy absorbed by the cask is equal to the kinetic energy of the rigid target impact, and the energy absorbed by the subgrade and soil is calculated in a manner similar to that for the soil impact discussed above. If the amount of energy to be absorbed by the soil is sufficiently high, the force in the soil will be higher than the force required to produce the shear plug. In this case, an iterative approach is necessary to derive an equivalent velocity so that the maximum force generated in penetrating the subgrade and soil beneath the concrete is equal to the peak contact force for the rigid target impact.

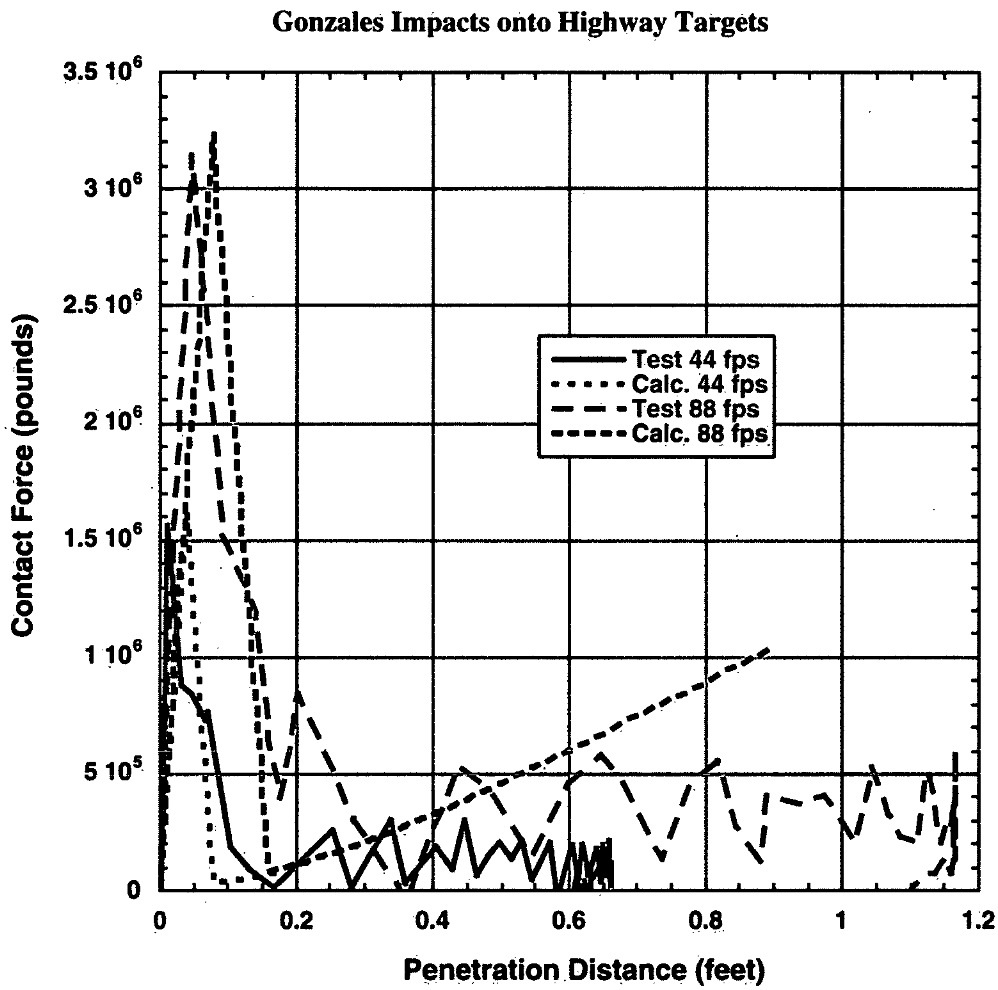


Figure III-56 Comparison of test force-deflection curves with those derived from the empirical equations

The only test data available on the orientation of impacts onto concrete targets is for end impacts. In this orientation, the contact area between the cask and the concrete does not increase with increasing penetration distance. To use the empirical relationships developed for end impacts with other impact orientations, an equivalent diameter must be determined. For both the side and corner impacts, the equivalent diameter was calculated to have an area equal to the area of the cask 5 mm (2 inches) above the contact point. For side impact orientations, this area is a rectangle. For corner impact orientations, this area is a truncated parabola. The shape of the contact area recognizes that there will be some deformation of the impact limiter before it generates sufficient force to fail the concrete and that the failure mode of the concrete is not a simple plug formation as it is for the end impact case.

III.4.5 Hard Rock Targets

For impacts onto hard rock targets, the target is assumed to be a semi-infinite half plane. The force and energy absorbed by the target is determined by the volumetric behavior of the rock.

For hard rock surfaces, this behavior is sufficiently stiff that the target absorbs very little energy. For this reason, these impacts are treated as rigid target impacts.

III.4.6 Results for Real Target Calculations

Error! Reference source not found. gives the results for impacts onto soil and concrete targets.

Table III-5 Equivalent Velocities for Rail-Lead Cask Impacts onto Various Targets (in 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

III.4.7 Impacts onto Water

Equivalent velocities for impacts onto water targets for velocities greater than the regulatory impact are assumed to be above the range of possible impact velocities (240 kph = 150 mph). The incompressible nature of water makes perfectly flat impacts quite severe. As the impact velocity increases, smaller deviations from the perfectly flat orientation are sufficient to cause the lack of shear strength in water to dominate the response. Because perfectly flat impacts are very improbable, this approach is justified.

III.5 Response of Spent Fuel Assemblies

III.5.1 Introduction

The response of spent power reactor fuel assemblies to impact accidents is not well understood. While this area has been investigated in the past (Sanders et al., 1992), those models tended to be relatively crude and imprecise. In addition, utility companies have renewed their interest in shipping higher burnup spent fuel. Therefore, it is essential to determine a more accurate response of spent fuel assembly to impact loads that may be affected by transportation or handling accidents or malevolent acts. Sandia has performed a series of computational analyses to predict the structural response of a spent nuclear fuel assembly that is subjected to a hypothetical regulatory impact accident, as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) 71.73, "Hypothetical Accident Conditions." This study performs a structural analysis of a typical pressurized-water reactor (PWR) fuel assembly using the Abaqus/Explicit finite element analysis code. The configuration of the pellet and cladding interface and the material properties of the pellet have been varied in the model to account for possible variations in actual spent fuel assemblies.

III.5.2 Description and Method

Figure III-57 shows a typical PWR fuel assembly, which consists of a series of fuel pins, or rods, grouped together in a square array. The fuels rods are held in place by a series of equally spaced grids. Within the array of fuel tubes are a series of guide tubes in which control rods are placed for controlling the fission reaction during operation. The guide tubes are attached to endplates, nozzles, or end fittings, which provide rigidity for handling.

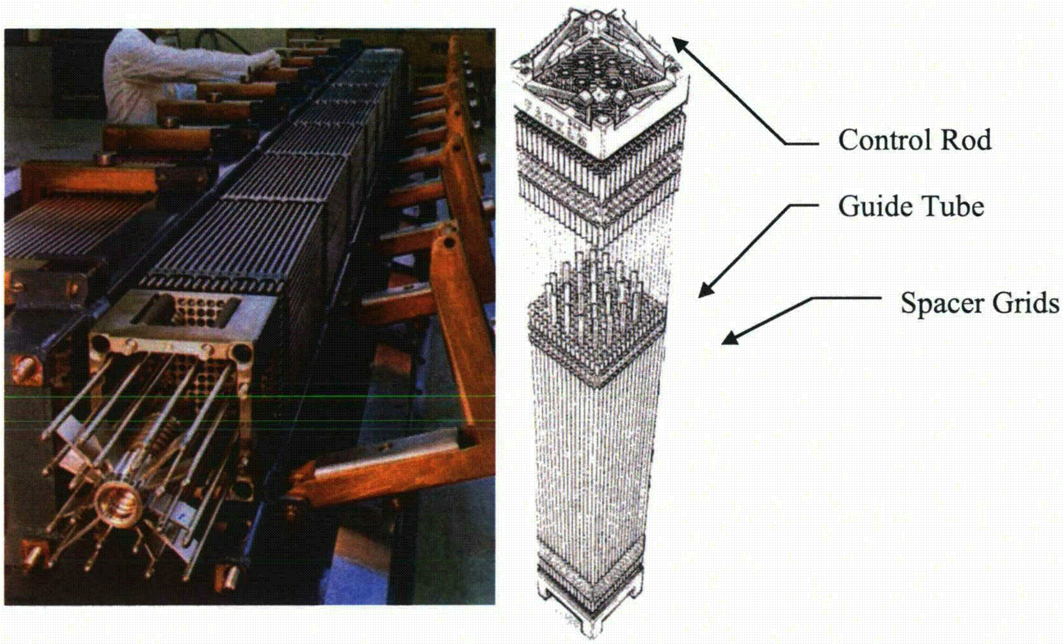


Figure III-57 PWR fuel assembly

Figure III-58 is a schematic representation of an individual fuel rod. This rod is constructed by stacking a series of uranium dioxide (UO_2) pellets inside a zirconium tube, placing a spring on the top of the pellet stack, and welding on end caps. A plenum is added at the top of the assembly to provide a sufficient volume to collect released fission gases.

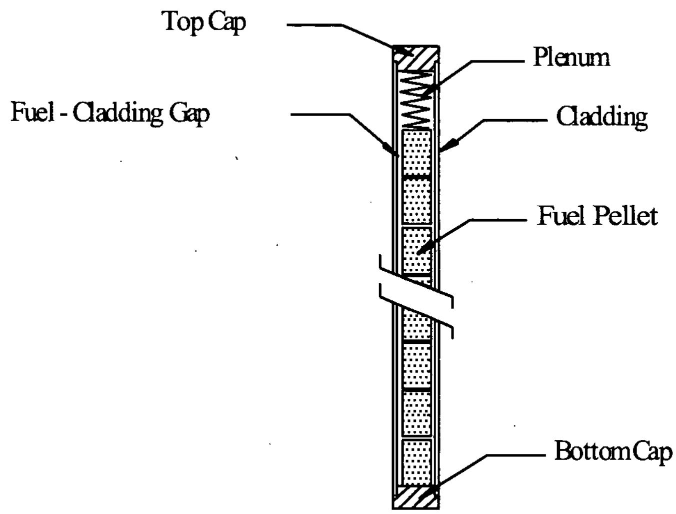


Figure III-58 Fuel rod schematic drawing

The working environment of a reactor is extremely harsh. The fuel rods are subjected to neutron radiation, large thermal gradients, large stress caused by external water pressure, and large local stress from contact between the pellet and the cladding. Upon the first power cycle, the uranium pellet cracks into pie-shaped pieces caused by the large radial temperature gradients across the pellet. Over a short period of time (months), the pellets shrink as fine porosity in the fuel is removed by radiation densifications. The cladding slowly creeps down onto the pellet because of its high operating temperature and the external pressure of the coolant. The pellet also begins to expand because of fission product swelling. Over a period of 1 to 2 years, the initial gap between the fuel rod and the pellet is eliminated. However, the contact between the cladding and the fuel pellet is not necessarily circular and uniform. This leads to local increases in the cladding stress. In addition, zirconium is one of the few elements that react with both oxygen and hydrogen. This can lead to a reaction between the zirconium dioxide (ZrO_2) layers on the inner cladding surface and the fuel pellet to form a bonding interface of $(U,Zr)O_2$ between the fuel pellet and the cladding, which in essence bonds the pellet to the cladding wall. In addition, hydride precipitants can also form in the Zircaloy cladding wall.

Upon removal from the reactor, the state of the spent fuel assembly at any future time depends on the spent fuel's environmental history, as well as on its condition upon removal from the reactor. The internal gas pressure in a fuel rod having been removed from the reactor now provides tensile hoop and axial stresses on the cladding. This stress, along with changes in cladding temperature, may allow hydrogen to precipitate out and possibly reform along the circumferential directions (direction of highest stress). Plastic creep in the cladding may cause a gap to develop between the cladding and the fuel pellet and void spaces to develop in the cracked pellets. The current material conditions and stress state of any particular rod at the time of an accident is complex and unknown. Therefore, the current material properties and geometric configuration will be varied over a small range to attempt to account for the actual unknown material and geometric variations.

Error! Reference source not found. lists the nominal dimensions of a 17×17 PWR fuel assembly. Because of the large number of rods and the large ratio between the fuel assembly length and the fuel rod diameter, modeling a complete assembly using the finite element method is challenging. To build the entire model using continuum and structural shell elements with a high enough resolution in each fuel rod would produce a model with so many degrees of freedom as to be computationally intractable. Therefore, the current analysis is broken down into three steps. In the first step, the entire assembly is modeled using structural beam and shell elements. In the second step, the loads from the highest loaded rod in the full assembly model are transferred to a single rod model constructed of continuum and structural shell elements. This model provides the detailed stress field necessary to determine the integrity of the fuel rod. Because of the severe nature of the reactor environments, there are significant material and geometric changes in the fuel rods. Very little, if any, test data is available for the Zircaloy-4 material under high irradiation conditions; therefore, as a third step, a series of parametric analyses were conducted with the continuum model to determine the sensitivity of the model to changes in the rod geometry and the pellet and cladding material properties.

Table III-6 Properties of Fuel Assembly

Assembly Type 17 × 17	
Cladding Material	Zircaloy-4
Assembly Cross-section, mm (in)	214.1-216.9 (8.43-8.54)
Number of Fuel Rods per Assembly	264
Fuel Rod OD, mm (in)	9.50 to 9.63 (0.374 to 0.379)
Minimum Cladding Thickness, mm (in)	0.58 (0.023)
Pellet Diameter, mm (in)	8.191 to 8.209 (0.3225 to 0.3232)
Maximum Active Fuel Length, m (in)	3.66 (144)

III.5.3 Finite Element Models

As described above, this analysis developed two major models. The first of these, the beam fuel assembly model, consists of beam and shell elements. This structural model determines the overall response of the fuel assembly. Using data from this model, researchers have developed a detailed continuum model of a single rod to determine a more detailed response of the most highly loaded rod. Several parametric analyses have been conducted, with the latter model to determine the effect of variations of rod material properties and geometry. In addition to these models, several smaller models have been developed to aid in the overall analysis. Initial models were developed to test the capabilities of the finite element codes. Researchers also developed small models when problems arose in the analyses. The following sections discuss all of these models, along with the final rod analysis.

Fuel Assembly Finite Element Model

Using the latest version of the Abaqus/Explicit finite element code, researchers constructed and analyzed a complete fuel assembly model (shown in Figure III-59), which incorporates three-dimensional beam elements for the fuel pins and control rods, shell elements for the spacer grid assemblies, and the support plates representing the basket walls. The endplates are modeled as solid plates using hexahedron elements so that the support rod beam elements can be attached. The model contains 265 fuel pins and 24 tie rods. There are a total of 129,440 elements, with 41,616 beam elements. The length of each fuel rod and support rod has 144 beam elements. Figure III-60 presents the location of the guide tubes in the cross-section of the fuel assembly.

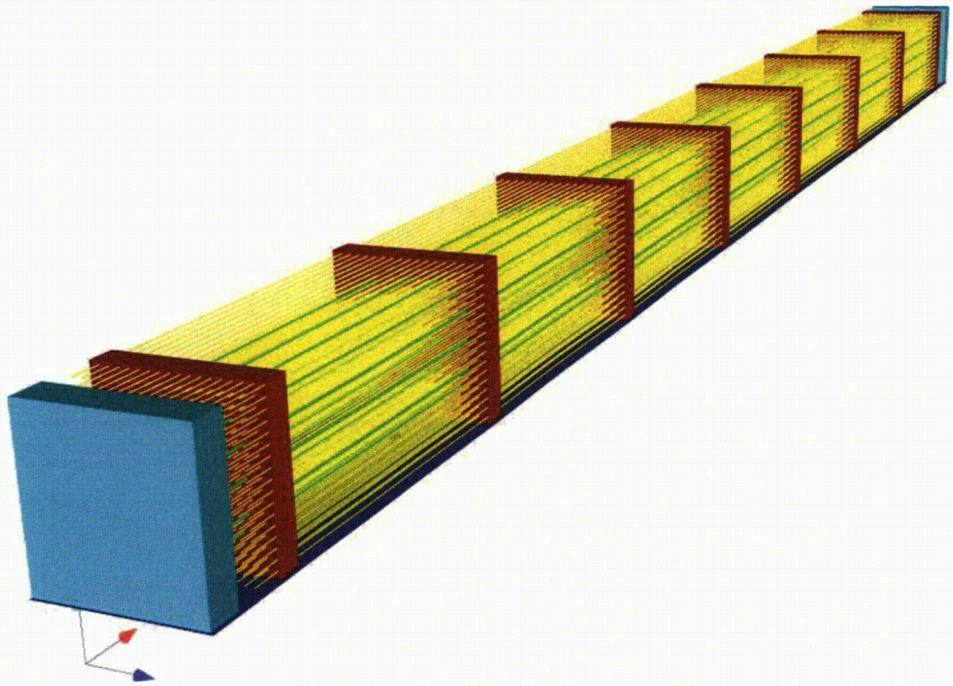


Figure III-59 Beam fuel assembly finite element model

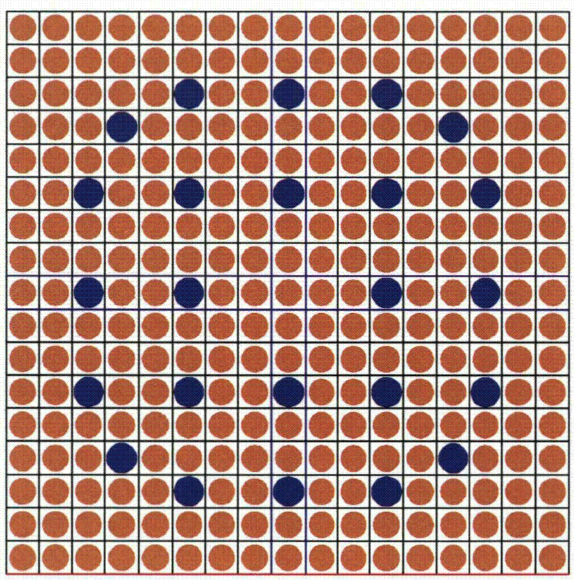


Figure III-60 Cross-section of 17×17 fuel assembly with guide tubes (in blue)

The fuel assembly model was loaded using acceleration curves developed from experimental data of a side impact drop test. Scientists used side loading because the fuel assemblies are much weaker in this loading direction and a previous study (Sanders et al., 1992) showed that the casks were more likely to fail from side loading than from other loading conditions. The full-scale data for the analysis was calculated from the ¼-scale test data. Figure III-61 presents a plot of the full-scale data. Researchers generated an additional curve from the full-scale data to yield a maximum acceleration of 100 g, while maintaining the same total impulse. The fuel rods are given an initial velocity of 13.4 meters per second (528 inches per second, 30 mph), which corresponds to a 9-meter (30-foot) drop test. The acceleration is applied to the lower plate, which represents the side of the fuel basket.

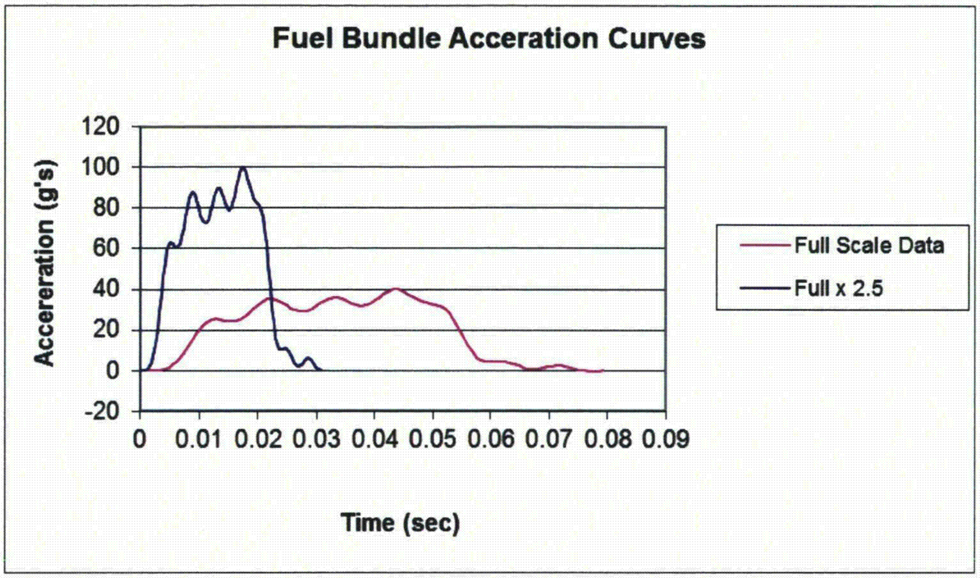


Figure III-61 Acceleration curves applied to fuel assembly beam model

The fuel rod material is modeled as unirradiated Zircaloy-4, using a power law hardening constitutive model fit to test data from the literature (Pierron et al., 2003). **Error! Reference source not found.** shows the calculated material parameters. These material properties are used for the fuel pins, the tie rods, and the support grid. This analysis models the fuel pins and tie rods as solid beams with a circular cross-section.

Table III-7 Zircaloy-4 Material Parameters

Elastic Modulus	89,600 MPa (13.0X10^3 ksi)
Yield Stress	448 MPa (65 ksi)
Luder Strain	0.00
Hardening Constant	714 MPa (103.5 ksi)
Hardening Exponent	0.845

Fuel Assembly Model Results

For the lower acceleration curve given in Figure III-61, which represents a rail cask, there is no plastic deformation in the fuel rods or the spacer grids. The entire model remains elastic. For the analysis with the higher acceleration curve, there is no plastic deformation in the fuel rods and some plastic deformation in the spacer grids. Figure III-62 shows the most highly strained spacer grid. The lower three sections of the spacer grid buckle, and a maximum plastic strain of 28 percent is calculated.

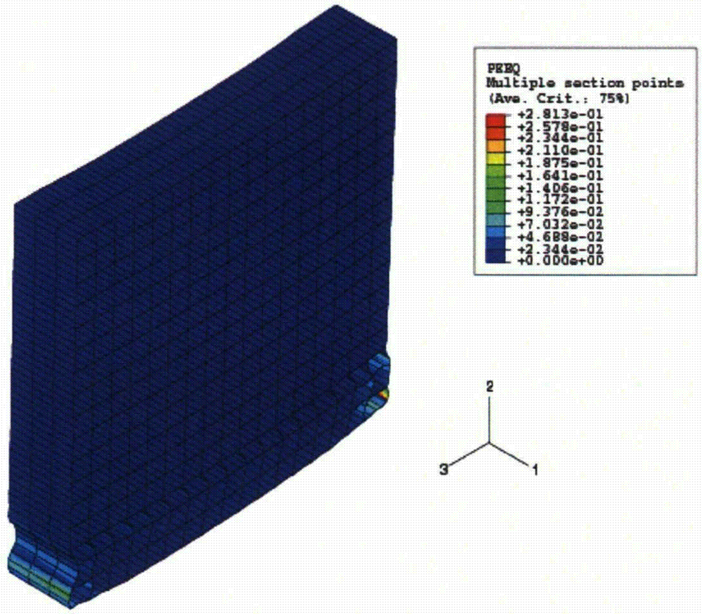


Figure III-62 Spacer grid 100-g analysis plastic strain

The contact forces from the beam fuel assembly model will be used as input to a single rod continuum model. Since these forces occur over very short durations during the analysis, it was necessary to obtain data points at each time-step in the fuel assembly model. Therefore, contact forces at a total of 20,349 time-steps were obtained from the fuel assembly analysis.

Beam Element Versus Solid Element Contact

In processing the contact forces from the beam fuel assembly model, researchers observed that the forces calculated during beam-to-beam contact were very large and acted over very short durations. These forces were much larger than those calculated in the model for the beam-to-shell contact. To investigate this difference in the magnitude and duration of the contact forces, researchers developed two additional models. The first, shown in Figure III-63, is a model of two impacting rods modeled with hexahedron elements. The second, shown in Figure III-64, is a model of two impacting rods modeled using beam elements. Since the beam elements in the beam fuel assembly model remain elastic, researchers evaluated these models for impact using elastic material properties.

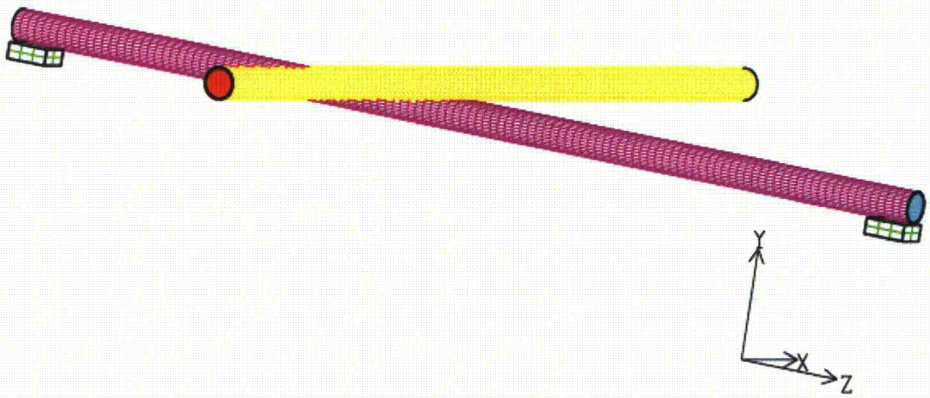


Figure III-63 Hexahedron test model for solid rod-to-rod contact in Abaqus/Explicit

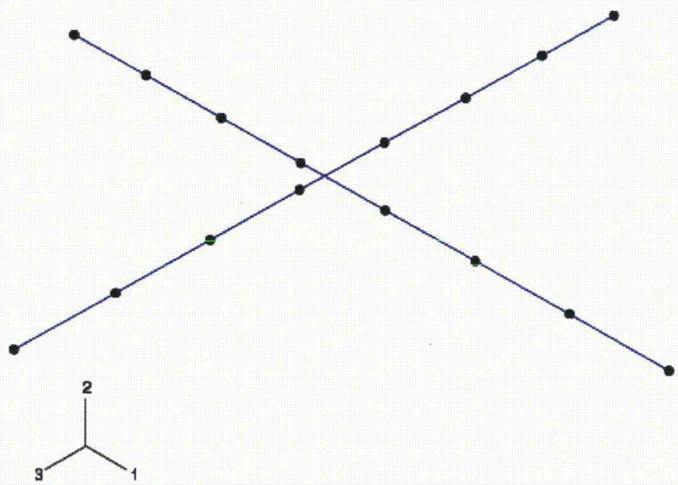


Figure III-64 Test model for beam-to-beam contact in Abaqus/Explicit

Figure III-65 and Figure III-66 show the results from the two finite element rod models. For the same mass, impact velocity, and cross-sectional geometry, the two models generate two different sets of contact forces. As shown in Figure III-65, the beam element impact forces are much larger and shorter in duration than those generated from the hex rod model. The magnitudes of the forces differ by about a factor of 7. Researchers made an additional check comparing the hexahedron Abaqus/Explicit model to a similar model run in the Sandia code PRONTO 3D. Both codes generated similar contact and reaction forces. Continued evaluation of the two models generated the curves shown in Figure III-66. For the velocity range of interest, there is a good linear fit for each curve. Therefore, in transferring the loads between the beam fuel assembly model and the continuum beam model, the magnitude of the forces were scaled in accordance with the curves in Figure III-66. The length of each beam element impulse was increased to keep the integral of the curve the same. That is, the total impulse was maintained to conserve the change in momentum. The ratio of contact forces from this simple crossed-rod problem was then applied to the more complex fuel assembly analyses using the model of Figure III-59.

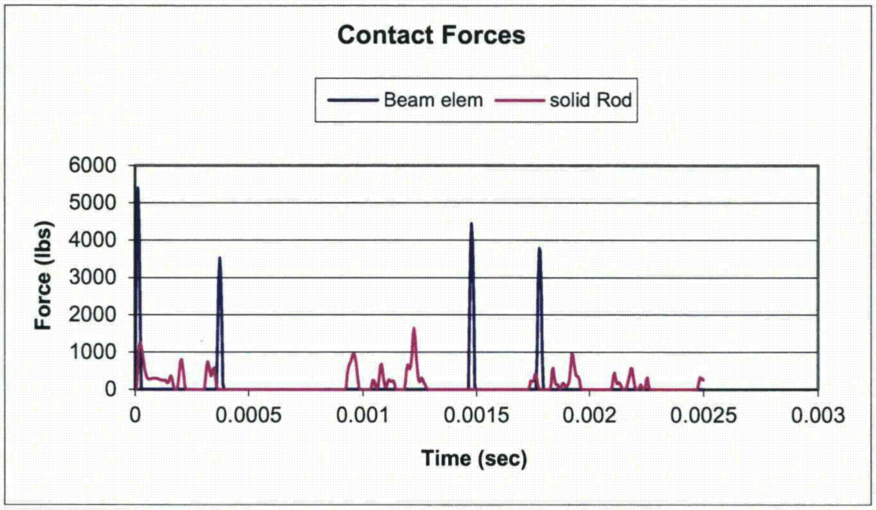


Figure III-65 Comparison of contact forces between solid rod and beam element rod

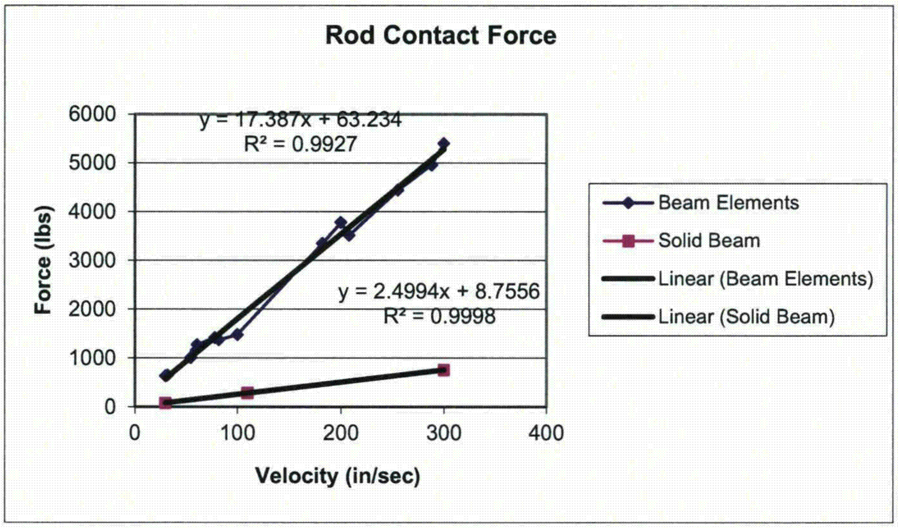


Figure III-66 Comparison of contact forces as a function of impact velocity

Continuum Rod Model

A continuum model was constructed using shell and hexahedral elements. The mesh is shown in Figure III-67, with a blowup of the end region showing the mesh density. The magenta-colored regions represent the locations of the spacer grids. A plane of symmetry occurs along the longitudinal axis of the beam. The symmetric model contains 162,000 elements, with 139,000 hexahedron elements used to model the UO_2 core and 23,000 shell elements used to model the Zircaloy-4 cladding. The hexahedron core has 16 elements across the diameter, and the semicircular arc of the cladding has 16 shell elements.

Researchers applied the contact forces obtained from the beam fuel assembly model for the 100-g loading to a set of shell nodes running along the top and bottom of the symmetry plane.

There are 1,446 nodes along each surface. Positive contact forces are applied to the bottom set of nodes and negative forces are applied to the upper nodes. As noted in the previous section, the forces from the beam fuel assembly model that result from beam-to-beam contact are scaled according to the curves in Figure III-66, and the duration of the load is then increased to conserve the change in momentum. In the region of the spacer grid, where there is beam-to-shell contact, the loads are not scaled. The new load curves are then interpolated from the element nodes in the beam fuel assembly model to a larger number of element nodes in the continuum model. Researchers give the rod model the same initial velocity as the beam fuel assembly model, 13.4 meters per second (528 inches per second).

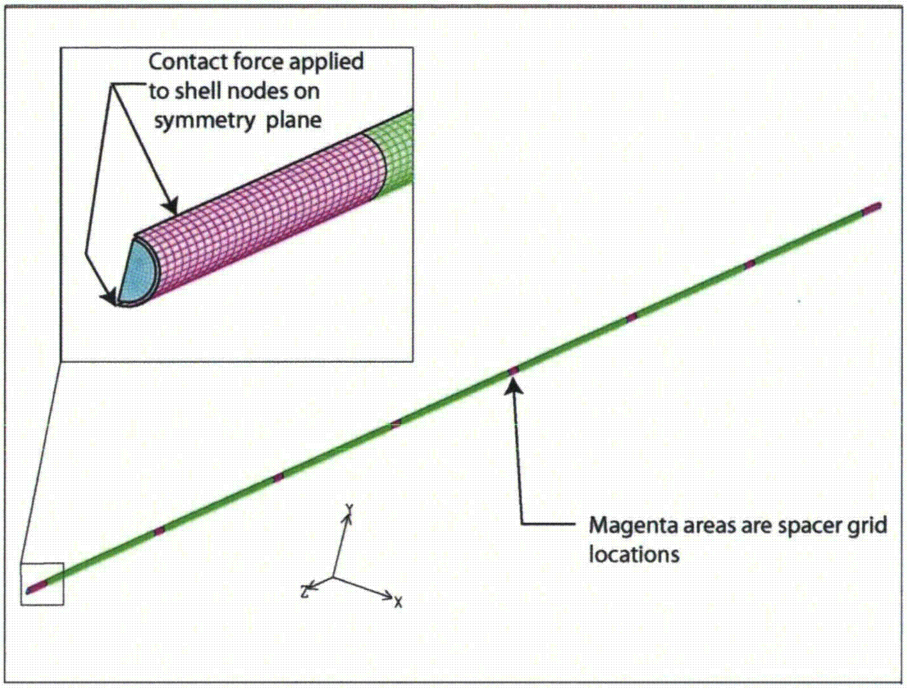


Figure III-67 Continuum rod finite element model

The rod materials are also modeled using a power-law hardening model. The parameters are presented in **Error! Reference source not found.**. The model was run for two different load cases, as shown in **Error! Reference source not found.**. In the first case, the outside diameter of the UO_2 core and the inside diameter of the cladding are the same, the Zircaloy-4 material is modeled as unirradiated fuel and the UO_2 is also assumed to be pristine. In the second load case, the cladding material is assumed unirradiated, while the modulus of the UO_2 is decreased by an order of magnitude to simulate a softer, crumbled material that has been irradiated. The results from both of these analyses are presented in the following section.

Table III-8 Standard Material Properties

	Zircaloy	Uranium Oxide
Elastic Modulus	89.6 X10^3 MPa (13.0X10^3 ksi)	193 X10^3 MPa (28.0X10^3 ksi)
Yield Stress	448 MPa (65 ksi)	149 MPa (21.6 ksi)
Luder Strain	0.00	0.00
Hardening Constant	714 MPa (103.5 ksi)	714 MPa (103.5 ksi)
Hardening Exponent	0.845	0.845

Table III-9 Load Case Parameter Changes

Load Case parameters			
Case	**Cladding Yield Strength, MPa (psi)**	**UO_2 Modulus, MPa (psi)**	**Cladding Gap (inches)**
Case 1	448 (65,250)	193x10^3 (28x10^6)	None
Case 2	448 (65,250)	193x10^2 (28x10^5)	None

Continuum Rod Results

Analysis Case 1

The first analysis case models unirradiated Zircaloy-4 material with no gap between the UO_2 rod and the cladding. Figure III-68 presents the resulting kinetic energy plot for this analysis. Almost all of the kinetic energy is lost from the rod, which indicates that the load impulse applied in the continuum model matches the impulse generated in the beam fuel assembly model. There is a large decrease in the kinetic energy at approximate 5.2 milliseconds. This corresponds to the large loads applied to the rod caused by beam-to-beam contact forces at locations between the spacer grids. Figure III-69 illustrates these impacts, which show the maximum EQPS in the rod cladding as a function of time for three intergrid locations. A maximum plastic strain of 1.5 percent is observed between spacer grid locations G and H. Figure III-70 presents a detailed contour plot of this region.

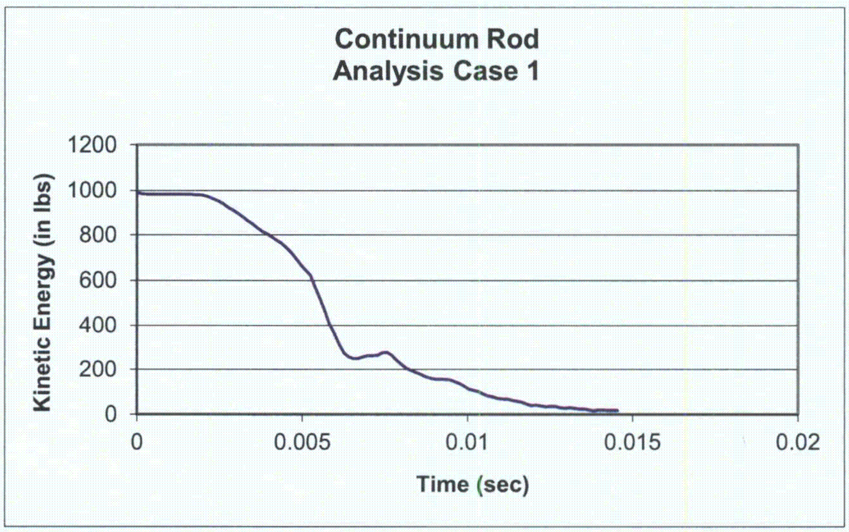


Figure III-68 Kinetic energy for Analysis Case 1

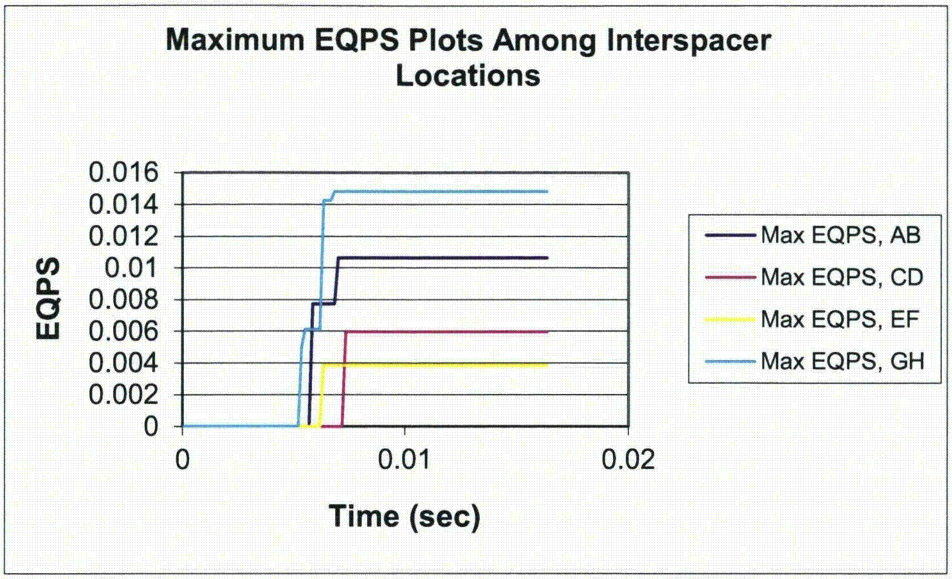


Figure III-69 Maximum equivalent plastic strain versus time for four interspacer grid locations. The spacer grids are specified by the letters in the legend (cf. Figure III-72).

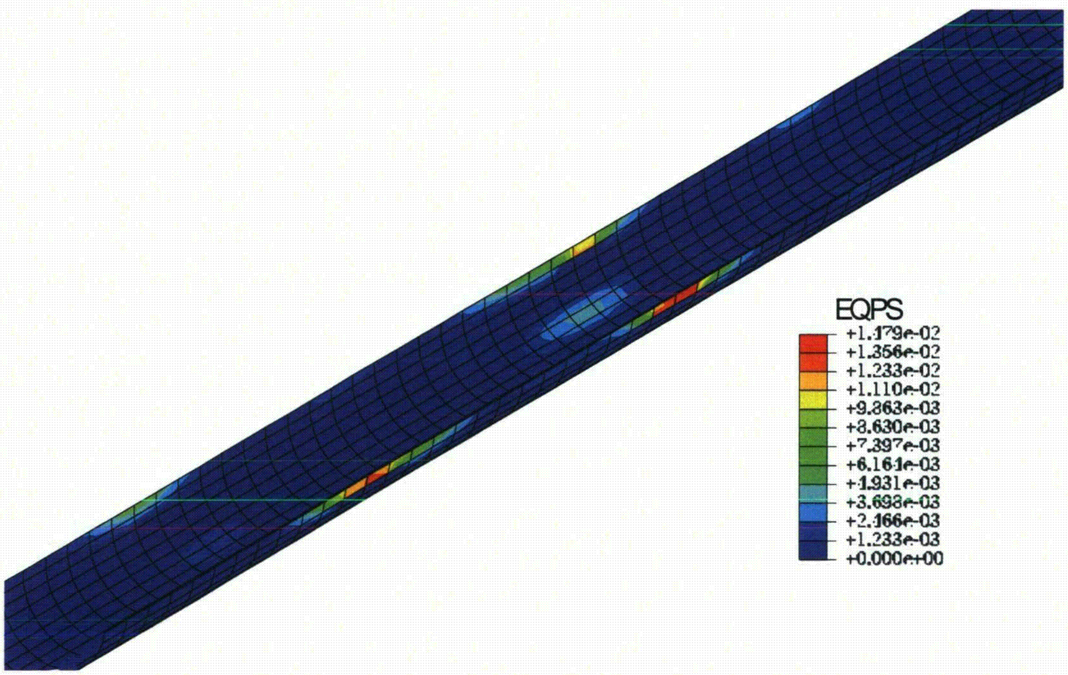


Figure III-70 Maximum equivalent plastic strain field in cladding for Analysis Case 1

Figure III-71 presents the plastic strain in the rods at several spacer grid locations. These strains are approximately an order of magnitude smaller than intergrid strains. This indicates that the spacer grid contact is much softer than beam-to-beam contact.

Figure III-72 shows the distribution of plastic strains along the length of the rod. The peak equivalent plastic strains are at the interrod locations between spacer grids G and H and between grids D and E. Strain at most of spacer grid locations along the rod remains elastic. The maximum plastic strain in the rod at a spacer grid is 0.06 percent at spacer grid C.

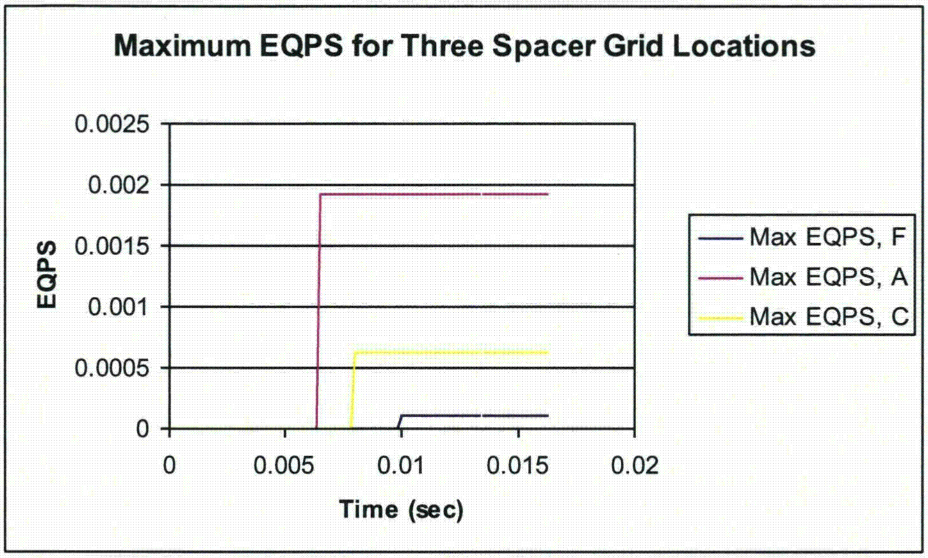


Figure III-71 Maximum equivalent plastic strain versus time for three spacer grid locations. The spacer grids are specified by the letters in the legend (cf. Figure III-72).

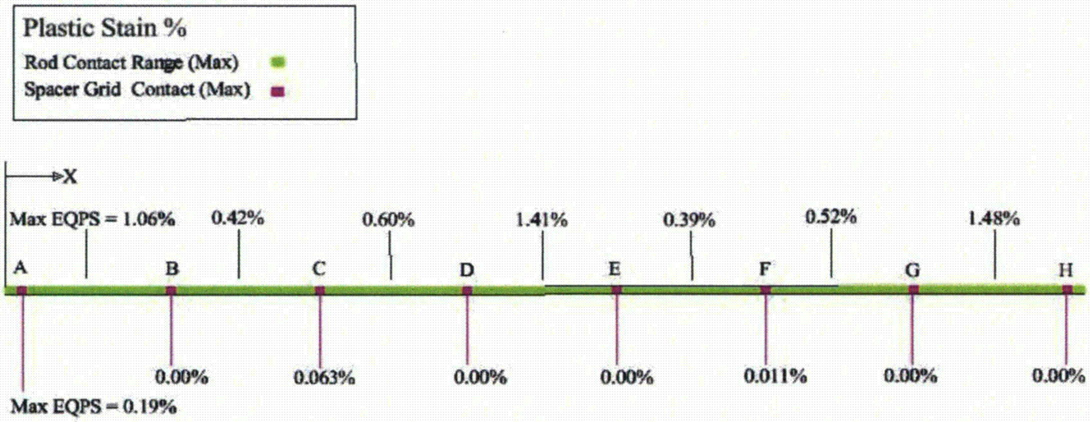


Figure III-72 Schematic showing maximum equivalent plastic strain for spacer grid and interspacer grid locations

A close examination of the strain distribution in Figure III-72 shows that they are not symmetric about the center of the beam, although the initial beam fuel assembly finite element model and its loading were symmetric. This artifice is a result of the beam contact algorithm in Abaqus/Explicit. As shown in Figure III-65, the impulses calculated for beam-to-beam contact are only a few microseconds long—or roughly equal to three analysis time increments. Since the resolution of the impulse and the analysis time-step are of the same order of magnitude, any accumulative numerical error on the position of the beam element nodes may result in a change in the time of contact and therefore the magnitude of the contact force and the subsequent position and velocity of the nodes. This results in a slight asymmetry in the calculated beam forces in the beam fuel assembly model. These forces are subsequently applied to the continuum model, and the result is the asymmetry of the strain fields shown in Figure III-72.

Analysis Case 2

For the second analysis case, the Zircaloy material properties remain the same, but the modulus of the UO_2 is decreased by an order of magnitude to provide a probable overestimation of the softness in the postreactor UO_2. The large cracks that develop in the fuel pellets during its in-core lifetime engender this softness. The largest plastic strains for this configuration are about one-third higher than those in the previous case of an unirradiated (pristine) UO_2 core. The maximum EQPS is reached between spacer grids A and B and has a value of 1.98 percent. A contour plot of this region is presented in Figure III-73, which shows an axial region about 2 inches long, with strain between 1 percent and 2 percent. Figure III-74 shows the maximum EQPS at four interspacer locations as a function of time; Figure III-75 shows the maximum EQPS for four spacer grid locations. These curves are similar in shape to those in Analysis Case 1, in which large strains occur at 5.2 milliseconds. For this configuration, plastic strains appear in the rod at all but one of the spacer grid locations, and the maximum value of plastic strain for a spacer grid location is 0.67 percent at spacer grid C. Figure III-76 depicts a distribution of plastic strain over the entire rod.

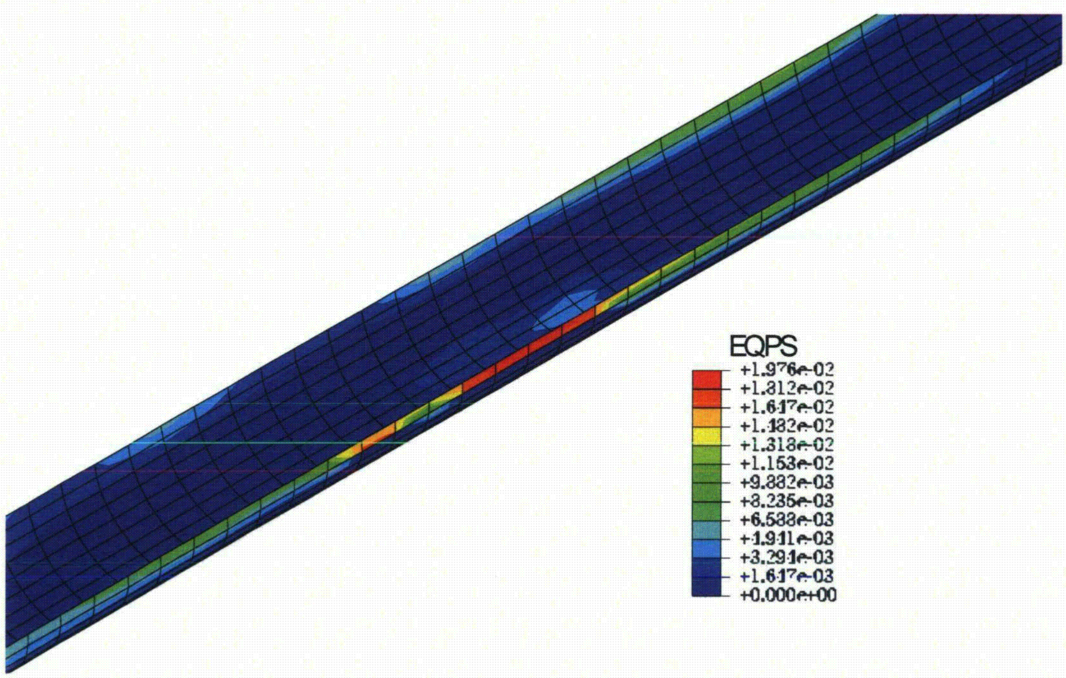


Figure III-73 Maximum equivalent plastic strain field in cladding for Analysis Case 2

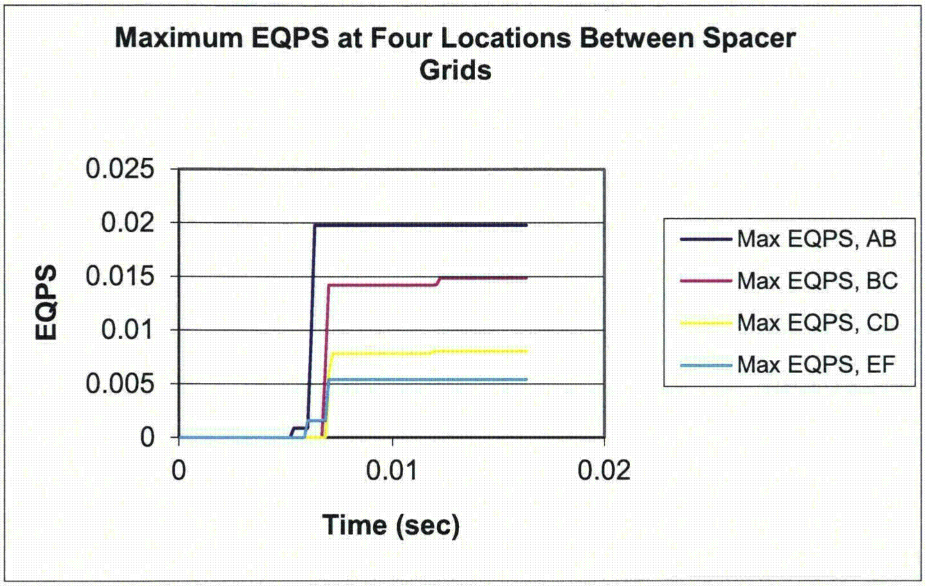


Figure III-74 Maximum equivalent plastic strain versus time for four interspacer grid locations. The spacer grids are specified by the letters in the legend (cf. Figure III-76).

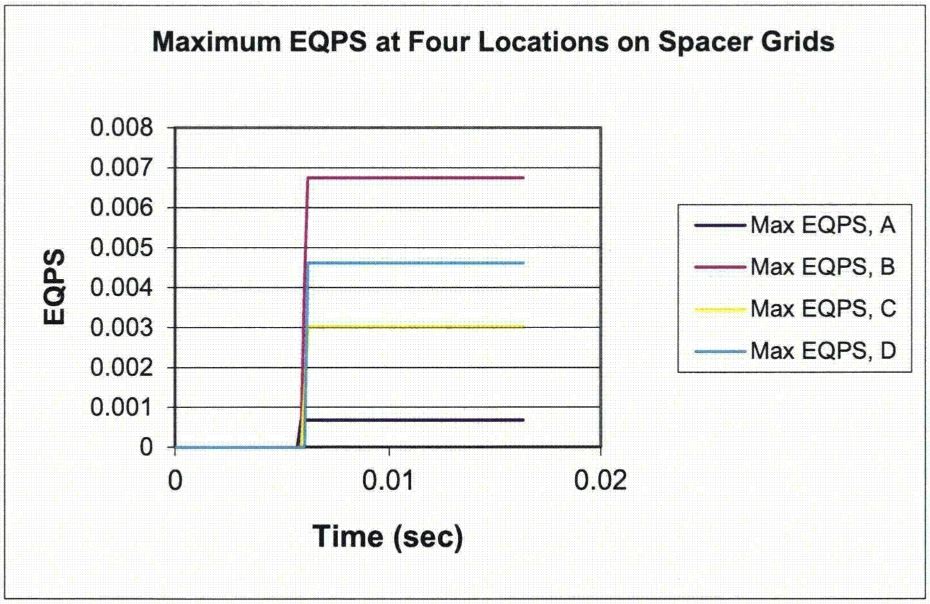


Figure III-75 Maximum equivalent plastic strain versus time for four spacer grid locations. The spacer grids are specified by the letters in the legend (cf. Figure III-76).

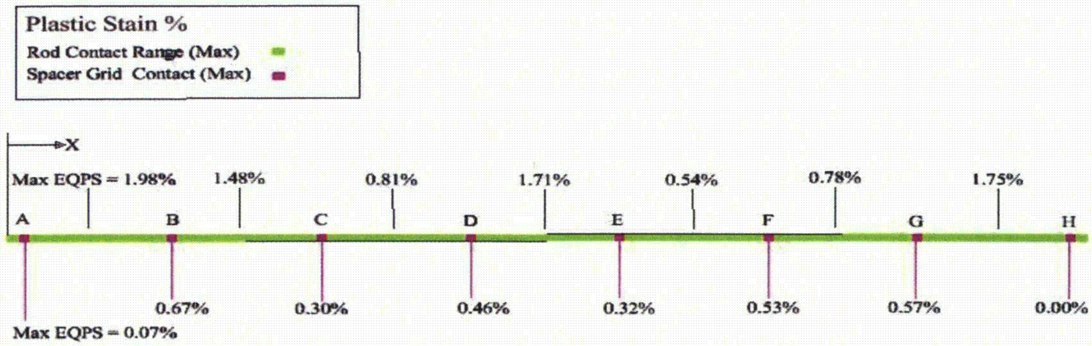


Figure III-76 Schematic showing maximum equivalent plastic strain for spacer grid and interspacer grid locations, Analysis Case 2

III.5.4 Discussion and Conclusions

In this study, the researchers conducted explicit dynamic finite element analyses of a PWR fuel assembly using two separate finite element models. The first model consisted of structural beam and shell elements and was used to determine the overall response of the complete fuel assembly to a regulatory side impact. Researchers applied loading data from this analysis to a continuum model of a single fuel pin to determine the localized stress and strain fields. They observed that during impact the largest loads on the rods were generated from beam-to-beam contact.

Because of the lack of experimental data and the variability in properties of stored spent fuel rods, researchers conducted a series of analyses with variations in the stiffness of the UO_2 core material. **Error! Reference source not found.** summarizes the parameters used in each analysis and the maximum plastic strain calculated in the cladding wall. From **Error! Reference source not found.**, it can be concluded that an order of magnitude change in the stiffness of the pellet material results in a 30-percent increase in the maximum plastic strain in the rod. The Case 2 maximum plastic strain is about half of the plastic strain to failure for the cladding of the fuel considered in this study. Thus, an acceleration pulse of about 200 g would be required to cause cladding failure. From **Error! Reference source not found.** and **Error! Reference source not found.** it can be seen that only the impacts at 145 and 193 kph (90 and 120 mph) onto rigid targets generate accelerations greater than 200 g.

Table III-10 Analysis Case Summary

Case	Cladding Yield Strength (psi)	UO_2 Modulus (psi)	Cladding Gap (inches)	Max EQPS (%)
Case 1	65,250	28×10^6	None	1.5
Case 2	65,250	28×10^5	None	1.96

The materials in this study were modeled as isotropic and homogenous using an elastic plastic power-law hardening model. It is not clear that this approximation accurately models the response of the UO_2 pellets. It is more likely that the initial response would not be a steep linear response as modeled, but would be nonlinear, with a soft initial reaction that would increase in stiffness as the pellet is squeezed. Researchers concluded that any attempt to estimate the nonlinear response of the pellet at this point would be pure conjecture.