SAND2007-6939C
**Proceedings of the 15th International Symposium on the Packaging and Transportation of Radioactive Materials**
**PATRAM 2007**
**October 21-26, 2007, Miami, Florida, USA**

# NRC SPENT FUEL TRANSPORTATION RISK ASSESSMENT METHODOLOGY

**Doug Ammerman**
Sandia National Laboratories*

**Carlos Lopez**
Sandia National Laboratories

**John R. Cook**
U.S. Nuclear Regulatory Commission

## ABSTRACT
In the approximately 30 years since the Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes (NUREG-0170) [1], there have been many changes in the shipment of spent fuel and the ability to analyze normal conditions and accident events that have a profound impact in calculated risks. Sandia National Laboratories is in the process of conducting a new spent fuel transportation risk assessment for the U.S. NRC. Unlike previous generic spent fuel risk assessments, this assessment will include certified cask designs and will also evaluate the effects of using welded or bolted inner canisters on spent fuel transport impacts. The intended result of the study is an updated assessment of the risks associated with the transport of spent fuel. This new risk assessment, which will be published as an NRC document, will be submitted to a peer review group, and will be published for public comment.

This paper will discuss the general analysis plan to be used in performing the risk assessment. Included will be the development of new event probability distributions for both truck and rail transport, the choice of real cask designs to be used in the study, the types of analyses to be performed, the analytical methodologies that will be used, and the schedule for completion of the study. A criticism of recent risk assessments has been that accident probability numbers are too old and do not represent the current transportation environment. The updated event probability distributions will address this concern. Parameters that influenced the cask selection included: type of wall construction, shipment mode, and whether or not a canister was used. Both structural and thermal response will be determined with detailed 3-D finite element calculations for a number of hypothetical accident conditions. The finite element analyses of the casks will be used to determine the presence or absence of any release pathways and analyses of the spent fuel contents will determine the source term available for release. Release information will be coupled with environmental and route factors to calculate risks using the computer code RADTRAN [2].

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* Sandia is a multi-program laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy under contract DE-AC04-AL85000.

## ACCIDENT RATES AND EVENT TREES

The accident rates will be updated prior to the risk analysis. The rail accident rates will be compiled from 5 years of rail accident data from the Federal Railroad Administration (FRA) and the past 30 years of Hazardous Materials Incident Reports (HMIR). Contacting the Hazardous Materials/Radioactive Materials Program Specialists from the FRA and the major rail companies will provide specifics for hazardous material accidents. Included will be updated accident event trees, such as the one shown in Figure 1.

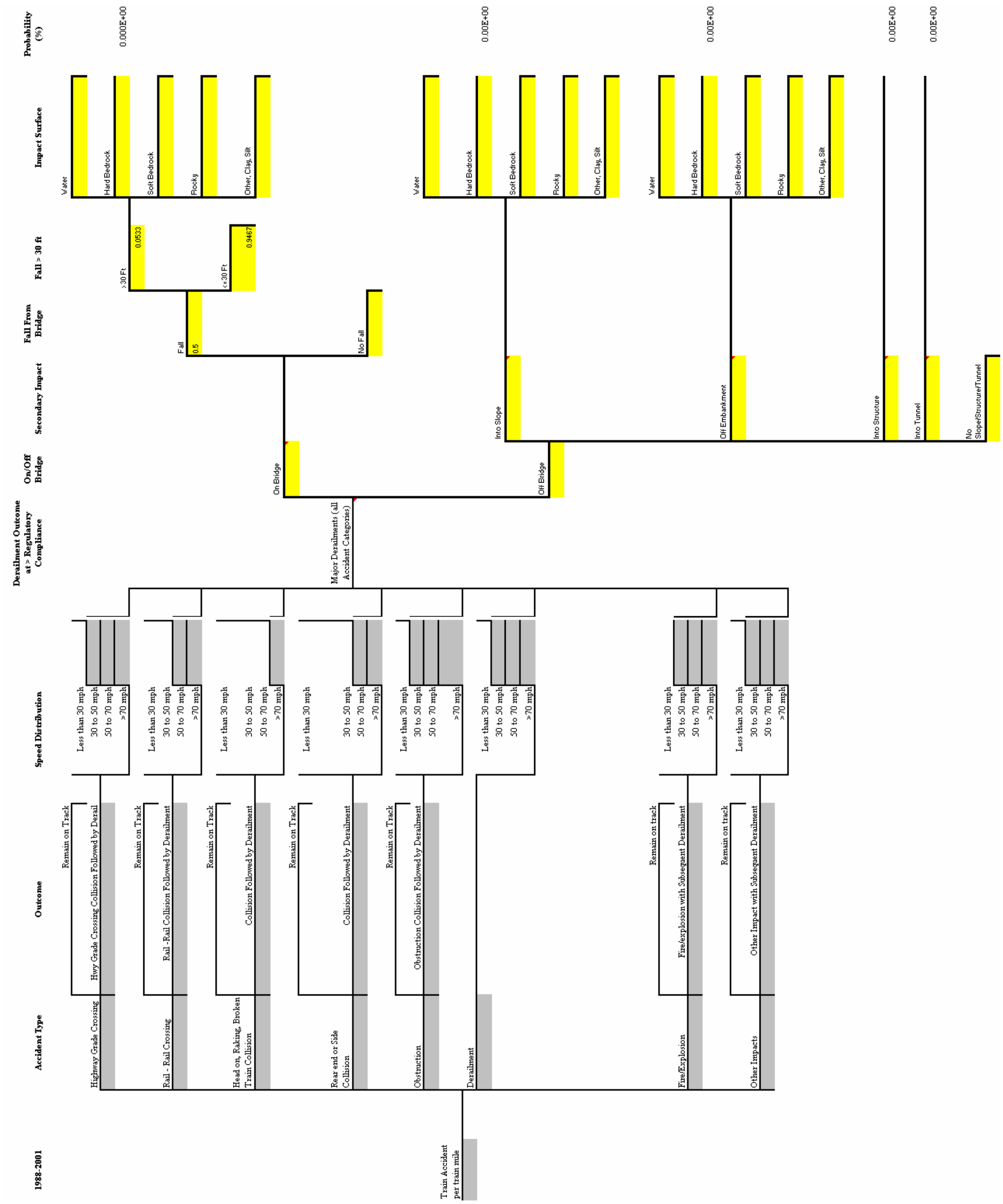


**Figure 1 - Structure of the rail accident event tree**

The truck accident rates will be compiled using state-by-state data from the past 5 years for large truck accidents and looking at the past 30 years of the HMIR. A comparison of large truck accidents, hazardous material accidents, and radioactive material accidents will also be investigated to see if adjustments within the truck accident rates should be made. A comparison of the truck accident rates will also be made to the WIPP TRUPACT-II, SGT, and SNF accident/incident rates if adequate data are available.

Records of RAM transportation accidents show that most such accidents do not involve the cargo. However, any accident or incident, however minor, will result in the vehicle sitting immobile at the accident scene for some period of time. SNF casks emit external radiation, and first responders, workers, inspectors, and members of the public could be exposed for the time that the vehicle is immobilized. Therefore, the times involved in accidents in which the vehicle is immobile will be investigated. Such accidents will be used in modeling accidents in which the spent nuclear fuel shipments do not result in a release but are stopped for a significant period of time.

## SELECTION OF CASKS

Past generic risk assessments for the transportation of spent fuel have used generic casks with features similar to real casks, but generally they are without all of the conservatisms that are part of real cask designs. In this effort, we intend to perform the generic risk assessment using actual cask designs with all of the features that contribute to their robustness. Because it is too costly and time consuming to examine all casks, a subset of casks to be used must be chosen. This paper will list the various spent fuel casks currently certified by the NRC, give options for the method of choosing the casks to be used, give some of the important features of the various cask designs, and finally make a recommendation about which casks to use.

Table 1 lists the casks that are currently (May 22, 2006) certified by the NRC for the transportation of irradiated commercial light water power reactor fuel assemblies [3]. The first 5 casks in the table are legacy casks that are not likely to be used in any significant quantities in a modern shipping campaign, so they were eliminated from further evaluations. Of the remaining casks there are two truck casks and seven rail casks. Of the two truck casks, the NAC-LWT has a steel-lead-steel wall construction and the GA-4 has a steel-DU-steel wall construction. Of the seven rail casks the TN-68 and HI-STAR 100 have multiple steel layers for the wall construction and the rest have steel-lead-steel wall construction. These four types of casks are the four types that were used in NUREG/CR-6672 [4] (although the all-steel rail cask in 6672 was assumed to be of monolithic wall construction). One of the primary purposes of this study is to determine the effect the addition of a fuel canister within the cask has on the risks associated with transportation. Therefore, it is imperative to evaluate at least one rail cask with a canister and one rail cask without a canister.

A key point in the cask selection decision was the number of casks to be analyzed and the types of casks that should be used. Table 2 lists the options that were considered.

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

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

**Table 2 - Cask selection options**

| Option | Features |
|---|---|
| One Truck, Two Rail | Includes both truck and rail, includes both canister and direct loaded rail casks, includes steel-lead-steel and all-steel casks |
| One Truck, Two and a Half Rail | Same as above but also includes the same rail cask both with and without a canister |
| Three Rail | No truck casks, both steel-lead-steel and all-steel casks, direct loaded and canistered fuel |
| Three and a Half Rail | Similar to above but includes both direct loaded and canistered for both steel-lead-steel and all-steel designs |
| Addition of a TAD Canister | Existing canistered casks with their certified canister replaced with a TAD canister |
| Addition of a Bolted Canister | Existing canistered casks with their certified canister replaced with a bolted canister |

The One Truck, Two Rail option was chosen for this study. The benefits of evaluating the response of a truck cask outweighed those of more fully characterizing the response of the rail casks. The casks chosen for this option are the GA-4 truck cask (because of its expected higher level of use in the Yucca Mountain transportation campaign), the HI-STAR 100 rail cask for the canister option, and the NAC-STC rail cask for the directly loaded option. With this choice of casks, there is one all-steel wall construction, one with DU shielding, and one with lead shielding. With this choice of casks it is easy to change to the One Truck, Two and a Half Rail option by performing an additional set of analyses on the NAC-STC with a canister. In addition, the effect of a bolted canister will also be considered.

## STRUCTURAL ANALYSES

The structural analysis will be performed using the explicit finite element code PRESTO [5], developed by Sandia National Laboratories (SNL). All casks will be modeled for closure end, closure corner, and flat side impacts at 13.4, 26.8, 40.2, and 53.6 m/s [30, 60, 90 and 120 mph] onto a flat rigid target. These velocities were chosen to enable direct comparisons with the results from NUREG/CR-6672 and because the historical accident record includes impacts at above

40.2 m/s [90 mph]. Impacts into/onto other targets (yielding surfaces) will use the same methodology that was used in NUREG/CR-6672 to determine equivalent impact velocities.

In performing the structural analyses, an attempt will be made to strengthen the structural analyses in NUREG/CR-6672. While the bolted joint in NUREG/CR-6672 contained an extremely short bolt with only one element through the section, the current bolt model will be more detailed. In addition, proper lid flange geometry and a nominal lid gap will be included. While the lid seal will not be explicitly modeled, the gap between the mating surfaces will be monitored to determine the extent of seal leakage. The inclusion of bolt preload into the model will also be investigated.

The cask impact limiter will be explicitly modeled using the orthotropic crush material model in PRESTO. The structural members will be modeled using shell or possibly hex elements, depending on the thickness of these components. If the design and geometry of the bolted connection permits, the attachment of the impact limiters may be modeled using the spot-weld feature in the code. This feature allows two surfaces to be attached using specified force displacement functions. This will require a separate finite element analyses to determine these functions for each bolted connection. Overall, this will be a more efficient method for modeling bolted connections in less critical areas (away from the closure flange area).

For the analyses of the cask with the canister, the contents (basket and fuel) will be modeled as a homogeneous mass using the orthotropic-crush material model. For the direct loaded casks, the fuel bundles will also be modeled as a homogeneous mass using the orthotropic-crush material model. However, some detail of the basket structure will be included in an effort to gain a better understanding of the diminution of force through the basket.

## THERMAL ANALYSES

The thermal analyses will be performed using the commercially-available finite element code, MSC PATRAN/Thermal [6], and the Sandia developed fire code, CAFE [7]. All casks will be modeled assuming that neither the body nor the impact limiters are damaged. This addresses the realistically conservative policy by recognizing that a severe mechanical impact immediately followed by a long-term co-located fire is highly unlikely. The initial condition for all fire analyses will be determined from steady-state analysis of the normal conditions of transport conditions defined in 10CFR71.71 [8]. The results of these analyses will also be compared with results presented in the SARs. The thermal analyses will use 3-D representations of the casks to calculate temperature distributions throughout the cask, including the seal region and the spent fuel region. Approximately 21 transient thermal analyses will be performed. These will include an 800°C [1472°F], a 1000°C [1832°F] , and a CAFE fire for each cask. 800°C and 1000°C are the two fire environment temperatures that were used in NUREG/CR-6672 and, therefore, will be used again for this work. Regarding fire duration, the 800°C and the 1000°C fires will be 30 minutes and a long (~11 hrs) fully engulfing fire will be simulated using the CAFE fire code with realistically calculated fire temperature distributions. The CAFE calculations will be performed assuming no wind conditions with the cask lying on the ground (only impact limiters touching ground). Cool-down analyses after different fire durations will be performed to capture internal peak temperatures, which, for thermally-massive objects, typically occur after the fire.

The analyses that will be performed for each cask are summarized below:

- one 30-minute, 800°C P/Thermal fire

- one 30-minute, 1000°C P/Thermal fire
- one long (~11 hrs) CAFE fire
- four P/Thermal cool-down analyses starting at different fire durations (1, 2, 3, and 11 hours)

The material properties presented in the SAR will be used in all models. The thermal characteristics of the fuel region (including decay heat) will be homogenized for all casks and benchmarked against SAR data. The models will have enough refinement to capture the temperature history at locations of interest, such as the seal and the fuel regions.

Some of the fire scenarios that are being considered for this project will be modeled using a computational fluid dynamics (CFD) code that is coupled with a finite element (FE) analysis code. The preferred coupled analysis codes are CAFE-3D and MSC PATRAN/Thermal. CAFE is a fast-running three-dimensional CFD and radiation heat transfer computer code that includes the dominant physics present in fires and is therefore capable of simulating fires realistically. This code, developed largely at Sandia National Laboratories, has been successfully coupled to commercially available finite element (FE) analysis computer codes. MSC PATRAN/Thermal (P/Thermal) is a commercially-available FE thermal code. CAFE calculates the fire field and provides time- and space-varying boundary condition information to the P/Thermal FE code, which calculates the three-dimensional heat transfer response of the object exposed to the fire modeled by CAFE. These two codes interact throughout the fire simulation, making the coupled CFD-FE analysis tool known as CAFE-P/Thermal.

## SOURCE TERM ANALYSES

The source term analysis will use the number of failed rods, rod impact energy, and containment hole sizes calculated by the structural analysis and the time of burst rupture (region specific) calculated by the thermal analysis for input. MELCOR [9] models of the HI-STAR 100/NAC-STC (from a compartment code point of view, these two casks are very similar except for the fact that the canister in the HI-STAR will be neglected for the STC) and of the GA-4 will be made to calculate blow-down times and internal deposition. The general method used in prior analyses conducted by Sandia for the U.S. NRC will be used to calculate the release fractions. In this analysis, an implicit assumption is that all fuel rods are initially intact.

## ROUTINE SHIPMENT RISK ANALYSIS

Using the selected rail and truck casks, routine exposure to the crew, workers, inspectors, and the public (the affected populations) can be determined using RADTRAN 6.0. The SAR for each of the casks will be used to determine the average gamma/neutron exposure rate. These exposure rates will be used to determine the exposure to the affected populations.

Unit risk factors will not be used in the routine analysis. RADTRAN 6.0 can analyze multiple routes and categorize each route state-by-state and by rural, suburban, and urban population zone. The routes will be selected using WebTRAGIS [10], a web-based transportation routing analysis and graphic information system computer code which replaced INTERLINE and HIGHWAY, and consideration of the routes selected from state groups. A comparison of some of the NUREG/CR-6672 routes will also be conducted.

A few cases will be selected to compare doses to the Reasonably Maximally Exposed Individual (RMEI) between RADTRAN 6.0 and RISKIND 2.0. The differences will be discussed with respect to the differences in modeling.

Probabilistic Risk Assessment (PRA) will also be conducted for each of the affected populations. Using the Incident-Free Importance Analysis Summary, the input parameters influencing each output will be distributed if necessary. A detailed discussion will be done for each of the parameters distributed.

## ACCIDENT RISK ANALYSIS

Upon completion of the structural and thermal analysis and the source term estimation, specific scenarios will be selected for analysis. Each of these scenarios will be assigned an accident probability, release fraction, aerosolized fraction, and respirable fraction. These scenarios are only for those accidents which involve release. Vehicle speeds and other accident parameters that could result in these scenarios will be discussed.

If one or more accident scenarios could result in a loss of lead shielding (LOS) for the rail cask (either due to lead slump determined in the impact analyses, or lead melt determined in the thermal analyses), the RADTRAN 6.0 LOS model will be used to determine the dose and dose-risk to the affected population groups. The LOS results will be compared with those done in NUREG/CR-6672.

The RADTRAN 6.0 stop model will be used to estimate the dose and dose risk to the populations of concern for accidents which involve the spent fuel package sitting somewhere for a period of time and result in no release of the package contents.

PRA will also be conducted for each of the affected populations. The input parameters influencing each output will be distributed if necessary. A detailed discussion will be included for each of the distributed parameters. Four different types of accidents will be analyzed using the PRA approach.

1. Accidents that involve a release of the package contents.
2. Accidents that involve a LOS for a rail cask.
3. Accidents that do not involve a LOS or release of the package contents (RADTRAN 6.0 stop model).
4. Accidents that lead to a criticality event (it is expected that the probability of this type of accident is vanishingly small; explicit demonstration of that fact will be included).

The PRA results will be compared with those done in NUREG/CR-6672.

A consequence analysis will be conducted for each accident scenario that results in a release. The analysis will be conducted for entire routes as well as selected locations within each of the population zones. The selected locations will not be named but will provide a general understanding of potential consequences involved within that particular type of population zone.

## SUMMARY

The results of this study will provide the U.S. NRC with a document that can be released to the public as a draft Environmental Impact Statement (EIS), updating the EIS for spent fuel transportation that has been used since the late 1970s. Unlike previous generic risk assessments, this one will use real casks, with all of their inherent factors of safety. The study will be performed in such a way as to allow direct comparison with the results from NUREG/CR-6672, the most recent generic risk assessment for spent fuel transportation carried out for the U.S. NRC.

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