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Examination of Risk Analysis Methods for MOX Land Transport in Japan

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Abstract

This report presents background information and methodology for a risk assessment of mixed oxide (MOX) reactor fuel transport in the nation of Japan to support their nuclear energy program. This work includes an extensive literature review, a review of other MOX activities worldwide, a survey of the statutory requirements for transporting nuclear materials, a discussion of risk assessment methodology, and calculation results for specific examples. Typical risk evaluations are given to provide guidance for later risk analyses specific to MOX fuel transport in Japan. This report also includes specific information that will be required for routes, cask types, accident-rate statistics, and population densities along specified routes, along with other detailed information needed for risk analysis studies pertinent to MOX transport in Japan. This information will be used in future specific risk studies.

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Executive Summary

ES.1 Introduction

This report presents background information and methodology, which can be used for a risk assessment of mixed oxide (MOX) reactor fuel transport in the nation of Japan to support their nuclear energy program. This work includes a literature review, a review of other MOX activities worldwide, a survey of the statutory requirements for transporting nuclear materials, a discussion of risk assessment methodology, and calculation results for specific examples. Typical risk evaluations are given to provide guidance for later risk analyses specific to MOX fuel transport in Japan. This report also includes specific information that will be required for routes, cask types, accident-rate statistics, and population densities along specified routes, along with other detailed information needed for risk analysis studies pertinent to MOX transport in Japan. This information will be used in future specific risk studies.

Extensive use has been made in this report of the recent Nuclear Regulatory Commission (NRC) Report, NUREG/CR-6672, "Reexamination of Spent Fuel Risk Estimates," that was conducted at Sandia National Laboratories (SNL).[ES-1] This prior study, called the Reexamination Study throughout this report, was used as a current example that demonstrates state-of-the-art risk analysis methodology.

ES.2 MOX Fuel Usage in Japan

Japan has used nuclear power ever since commercial operation began in the Ibaraki Prefecture in 1966. By 2001, 51 reactors were operating throughout Japan with a total output of over 44 GW, approximately one-third of its total electric power output.[ES-2] Nuclear power is considered important to future energy needs in Japan because of scarce natural resources and alleviation of environmental problems such as global warming and acid rain.[ES-2] Nuclear energy is thus expected to play a major role in the future in Japan as a power source for projected and increased electrical usage.

Japan Nuclear Fuel Limited (JNFL) was jointly established with the member companies of the Federation of Electric Power Companies Japan (FEC) as the major partners.[ES-2] JNFL is now working toward beginning the operation of a uranium (U) enrichment facility, a low-level radioactive waste storage facility, and a reprocessing facility in the Rokkasho-mura in the northern prefecture of Aomori.

Plutonium is obtained after reprocessing spent fuel under strict control during the nuclear fuel also cycle based on the principle that no surplus plutonium is produced. Japan plans to use MOX fuel pellets of uranium mixed with plutonium oxide in light-water reactors (LWRs).[ES-2] Although Japan was planning to gradually introduce this fuel for use in 16 of 18 of these reactors by 2010, this timetable is likely to be extended. A large MOX fabrication plant was commissioned in 2000.

Although MOX programs in Japan have been under research and development for decades, until now, Japan has contracted the reprocessing of spent nuclear fuel to the United Kingdom and France. Japan considers the recycling of spent fuel essential to establishing nuclear power as a domestic energy source that will be used to meet present and future energy needs.

Japan has also developed the Monju prototype fast breeder reactor, which generates more plutonium than is spent. Construction of the Monju FBR began in January 1983.[ES-3] Monju first achieved criticality on April 5, 1994. An accident caused by leaking sodium liquid occurred on December 8, 1995. The Nuclear Safety Commission is now examining the safety criteria for restarting Monju.

Plutonium, either separately or as MOX, has to be transported throughout the nuclear fuel cycle. Plutonium oxide is transported to Japan by sea for handling within the MOX plant and the fuel fabrication plant to supply MOX as fresh fuel to the nuclear power plants. Transportation will be required for spent fuel to be returned to the reprocessing plant, for return as reprocessed plutonium to the MOX fuel plant, and to a waste storage site before finally going to a disposal site for nuclear waste.

Nuclear fuel, contained in fuel rods, is typically made of ceramic pellets of uranium encased within a cylindrical cladding. In the case of MOX fuel, plutonium is combined with uranium to form a MOX. As fresh MOX is converted to spent fuel in the reactor, the plutonium content is reduced to roughly 1% by weight and becomes embedded in a highly radioactive matrix that deters its value for malevolent actions.[ES-4] The fuel rods are approximately 15 ft long and clad with a material such as zirconium or stainless steel as is used in Monju. This cladding provides a sealed environment for the fuel pellets. Fuel rods are bundled, depending on the reactor type, in either square or hexagonal arrays containing 50 to 300 rods. This bundle is contained in a supporting structure called a basket.

Additional information applicable to MOX transport in Japan discusses testing of the NFT-14P cask for transport of:

- high burn-up spent fuel,[ES-5]
- the TK-69 transport/storage cask,[ES-6]
- experiences with spent fuel transport in Japan,[ES-7]
- an overall assessment of nuclear fuel material transportation in Japan,[ES-8] and
- the performance of MOX transport casks against external water pressure.[ES-9]

A recent worldwide assessment of the transport of irradiated nuclear fuel and high-level waste has also been reported, including data for Japanese shipments.[ES-10]

ES.3 Risk Studies

The casks used to transport spent fuel must, in the United States, be certified by the U. S. NRC as being in compliance with 10 CFR 71.[ES-11] These U. S. regulations are almost identical in their requirements to internationally accepted IAEA standards[ES-12] that have been in effect for years.

Recently updated, these international regulations are intended to ensure protection of the public during both normal transport and in accident conditions.

In 1987, a study was completed to define and evaluate the responses of spent fuel casks exposed to severe highway and railway accident conditions.[ES-13] This report, generally referred to as the Modal Study, concluded that the radiological risks from spent fuel under the severe highway and railway accident conditions that were derived in the study are less than the risks estimated in the NUREG-0170, the U. S. NRC's generic Environmental Impact Statement (EIS) for transport of radioactive material (RAM), which was published in 1977.

In 2000, SNL reexamined the risks associated with the transport of spent nuclear fuel by truck and rail and compared them to the results published in NUREG-0170 and the Modal Study.[ES-1] The results of this study, together with the previous studies, demonstrate that the risks associated with the shipment of spent fuel by truck or rail are very small.

Basically two types of risks are associated with MOX fuel in addition to incident-free transport. The first is the release of plutonium and fission products into the environment as a result of accidents during transport and disposal, and the second is associated with terrorism. International safeguards are well established for transport modes and are followed according to national and international requirements.

Despite the fact that there never has been a significant release of radiological material from a transportation accident during the four decades during which nuclear transport has taken place, the safety of nuclear material containers in accident conditions remains an issue of public concern. Even though nuclear shipment containers are manufactured, tested, and used under very stringent regulations, the risks of nuclear shipments must be carefully evaluated to address and alleviate perceptions of risks.

Type B casks used for transport of spent nuclear fuel are typically manufactured in three weight classes: legal truck weight, overweight truck, and rail. They use three gamma-shielding materials: steel, lead, and depleted uranium (DU). For most applications, casks will be lead- and DU-shielded truck casks and steel- and lead-shielded rail casks.[ES-1] A publication titled *Shipping and Storage Cask Data for Commercial Spent Nuclear Fuel*, by JAI Corporation, provides specific information on cask construction and dimensions.[ES-14]

The Reexamination Study analyzed four generic casks for their reevaluation of risk analysis for spent fuel transportation.[ES-1] These casks are:

- Steel-lead-steel truck-cask configuration,
- Steel-DU-steel truck-cask configuration,
- Steel-lead-steel rail-cask configuration, and
- Monolithic steel rail-cask configuration.

ES.4 Japan's Safety Regulations

An article on safe transport in Japan outlines Japan's safety regulation system for the transport of radioactive materials[ES-15] The article includes procedures for application and approval for transport as well as emergency preparedness responsibilities.

Japan's safety standards for transport of radioactive materials are now regulated by the Ministry of Education, Culture, Sports, Science and Technology (MEXT); the Ministry of Economy, Trade and Industry (METI); and the Ministry of Land, Infrastructure and Transport (MLIT).

The methodology used to establish a risk determination for nuclear transport of radioactive materials is outlined below in Sections ES.5 through ES.10 for a representative set of accidents and casks. This methodology includes structural and thermal analysis, the response of cask seals, source term evaluation, the RADTRAN risk code, RADTRAN risk calculations, and consequence analysis.

ES.5 Structural and Thermal Analysis

PRONTO 3D[ES-16] is a three-dimensional, transient solid-dynamics code used to model the large deformation produced by impacts in serious accidents. It is especially useful for modeling the behavior of cask closures such as cask lids and bolt interfaces. PRONTO 3D was validated by comparing its predictions for a wide range of problems to test results, the predictions of other codes, and to simple-geometry theoretical solutions. At SNL, the Structural Evaluation Test Unit (SETU) Program compared experimental and PRONTO 3D analytical results for cask impacts of up to 60 mph.[ES-17]

Rod failure by burst rupture and time to fail in fire accidents were calculated in the Reexamination Study[ES-1] by using the PATRAN/Pthermal[ES-18] code that is available commercially.[ES-19] The code can be used to for one- , two- , or three-dimensional simulations to determine the heating rates of structures by conduction, convection, and thermal radiation. PATRAN/Pthermal, formerly called Q/TRAN, was validated by comparing its results to analytical solutions and to the predictions of other widely used codes.[ES-20, ES-21]

ES.6 Response of Elastomer and Metallic Seals

The regulatory Data Base Accident (DBA) defined by 10 CFR 71[ES-11] and 49 CFR 173[ES-22] is characterized as bound by a maximum impact load response of 0.2% maximum strain on the inner shell and a maximum thermal load of 260°C lead shield mid-thickness temperature.

For truck or rail casks with elastomer seals, failure is not assumed for impact loads and temperatures less than these DBA conditions.

Because radioactive materials packages are designed with large margins of safety, these packages would be capable of withstanding accident conditions more severe than the DBA. Recent tests and analyses at SNL using packages with elastomer seals have shown that this level of strain is reasonable for the DBA, and that the cask containment boundary does not fail for accidents even for inner shell strains of up to 20%.[ES-23] A conclusion is that cask containment boundaries will not fail for packages using elastomer seals for inner shell strains of less than 20%[ES-23] and for temperatures of 260°C or more. Metallic seals have a negligible rate of failure below 565°C.

Bolts used for seal closures must be carefully chosen. Inconel bolts are rated as high as 620°C and are to be used in place of high-strength carbon steel bolts rated to temperature of only 370°C for most uses.[ES-23]

A recent innovation at SNL provides a rebrazable metallic seal that eliminated the low temperature limit of elastomer seals and the impact limit for typical metallic seals.[ES-24, ES-25, ES-26] Because this seal forms a brazed joint with the cask wall material, it provides additional usage in matrix response regimes above those for either elastomer or metallic seals. Tests have shown brazing and rebrazing operations can be carried out up to 20 repetitions.

ES.7 Source Terms

An event tree is constructed and populated with branch point fractions, and it contains (by definition) a representative set of accident pathways. Accident statistics allow source term probabilities to be estimated for the accident pathways. The damage to a cask depends on the response of the cask to these accident conditions. Source terms are then calculated based on the damage to the cask and its contents.

Source terms are used to calculate the release of radioactive material, in the event of a serious accident, from the internal containment, such as fuel rods, to the cask interiors, and then from the cask interior, through a leak, to the environment. The Reexamination Study estimated the risks associated with accidents and the probabilities of these accidents for three broad classes of transportation accidents:[ES-1]

- Collisions without fires,
- Collisions that lead to fires, and
- Fires without collisions.

By definition, risk is the product of consequence magnitude and the probability of event occurrence. The consequence magnitude can be calculated using RADTRAN for radioactive material transportation accidents.[ES-27, ES-28] RADTRAN-produced values of consequence magnitude are calculated based on the accident source term, meteorological conditions for the accident event, population that could be exposed, and emergency response actions that result from the hypothetical accident event being studied.

ES.8 RADTRAN Risk Code

The first analytical step in ground transportation analysis is to assess the incident-free and accident-free risk factors on a per-shipment basis. Risk is the product of the probability of exposure and the magnitude of that exposure. Accident risk factors are then calculated for radiological and nonradiological traffic accidents. Incident free risk factors are calculated for crew and public exposure to radiation from the shipping containers as well as the nonradioactive transportation vehicle exhaust exposure.

RADTRAN 5 is a state-of-the-art risk code used to calculate both incident-free and accident risks for populations directly affected by the transport of radioactive material.[ES-29, ES-30] RADTRAN was developed by SNL to calculate population risks from radioactive material transportation by truck, rail, air, ship, or barge. The Transportation Incident Center Line Dose (TICLD) code is operated with RADTRAN to calculate doses to maximally exposed individuals. A previous version, RADTRAN 4,[ES-31] was used extensively until recently.

A 1978 study at SNL provided the basic concepts to initially bound or quantify the accident environments that a large nuclear materials transportation package would be subjected to in a transportation accident.[ES-32] The abnormal environments that were studied are as follows:

- Impact
- Crush
- Fire
- Puncture
- Immersion

Although many studies have been conducted since this basic early work was completed, the methodology used in this study is sound and provides insight into risk and probability evaluation methodology.

RADTRAN 5 is the most recent version of the RADTRAN code now available. It is being revised at the present time, and a Version 6 will soon be available. RADTRAN is the basis for the International Atomic Energy Agency (IAEA) INTERTRAN Code, and RADTRAN was the code used for the recent NRC NUREG/CR-6672 study to reexamine the earlier NUREG-0170 environmental impact study.

RADTRAN calculates consequences and risks for a specific radioactive material in a specific packaging transported along a specific route. The code examines both incident-free transport and the accident conditions that are postulated.

For an accident postulated to be so severe that a release of radioactive material would result, RADTRAN estimates the following.[ES-1]

- The probability of the assumed accidental release,
- Doses that could be received by humans downwind of the postulated accident location, and

- The radiological risks that would be incurred by the release (the product of the consequence and the probability of the release).

RADTRAN also calculates the radiation doses associated with accidents in which a packaging would lose shielding without releasing radioactive contents.

ES.9 RADTRAN Risk Calculations

By use of appropriate accident and input parameters, the Reexamination Study performed seven sets of RADTRAN calculations to evaluate the risk of truck and rail shipments of spent fuel and to compare these results to the earlier NUREG-0170 examination.[ES-1] Outputs from these RADTRAN calculations were used to compare the severity of risks and the parametric effects of the input parameters. Similar analyses can be conducted for specific transportation campaigns for MOX spent fuel in Japan, using the input parameters appropriate for the transport conditions and the casks that would be used.

ES.10 Consequence Analysis

Much of the recent development in nuclear material risk assessments has been in improving estimates in normal and severe accidents resulting in the release of radioactive materials. However, little new work has been performed on consequence analysis, even though this area is of importance in radioactive material transport. Because consequences are location-specific, considerable attention must be given to the details of consequences in the absence of specific information.[ES-33]

Because an enormous amount of news coverage would result from any event involving radiation, public concern is a major risk consideration in nuclear transportation. Emotional stress and associated health effects from an accident, real or not, would be expected. Devaluation of land, products, and agricultural commodities, either directly or indirectly associated with contamination is possible. It would be difficult to quantify such costs other than by using general bounding analyses.[ES-33]

ES.11 Conclusion

The steps involved in a transportation risk assessment have been summarized, and the Reexamination Study has been used to illustrate these steps. This Reexamination Study provided a recent and comprehensive set of calculations that are described in this report for their value in illustrating current risk methodology in evaluating population risk as a result of transporting radioactive materials. The Reexamination Study was concerned with truck and rail transport of spent fuel; however, the methods used are applicable to any radioactive material. MOX spent fuel transport will provide slightly different risk values for population dose rates than were calculated for uranium pressurized water reactor (PWR) and boiling water reactor (BWR) spent fuel by the Reexamination Study, but the methods used will be identical for both types of fuel.

The major conclusion of the Reexamination Study was that the NUREG-0170 estimates of spent-fuel transportation incident-free doses are somewhat conservative, and the NUREG-0170 accident population dose risk estimates are very conservative. This conclusion clearly demonstrates that the existing regulations governing spent-fuel transportation are adequate to protect public health and safety.[ES-1]

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Nomenclature

AAR	Association of American Railroads
ABWR	advanced boiling water reactor
AECL	Atomic Energy of Canada Ltd.
ANL	Argonne National Laboratory
ANSI	American Standards Institute
ASTM	American Society for Testing and Materials
ATR	advanced thermal reactor
BMFT	German Federal Ministry of Research and Technology
BNFL	British Nuclear Fuels Limited
BWR	boiling water reactor
CANDU	Canadian Deuterium-Uranium
CCDF	complementary cumulative distribution functions
CEA	Commission of the European Communities
CFR	Code of Federal Regulations
C/H	Crystal River Nuclear Plant, Florida to Hanford, Washington
CNRS	French National Research Council
COGEMA	a large power entity
CRL	Chalk River Laboratories
DBA	Design Basis Accident
DNFSB	Defense Nuclear Facility Safety Board
DOE	U. S. Department of Energy
DOT	U. S. Department of Transportation
DU	depleted uranium
EA	Environmental Assessment
EDF	Electricité de France
EIS	Environmental Impact Statement
FBR	fast breeder reactor
FEPC	Federation of Electric Power Companies Japan
FFF	fuel fabrication facility
GCR	gas cooled reactor
GmbH	Gesellschaft mit beschränkter Hafnung, a closed corporation under German law
HLW	high level waste
HUD	Housing and Urban Development
IAEA	International Atomic Energy Agency
INEEL	Idaho National Engineering and Environmental Laboratory
INIST	Institut de l'Information Scientifique et Technique

INSPEC	The Database for Physics, Electronics, and Computing
JAERI	Japan Atomic Energy Research Institute
JICST	Japanese Science and Technology
JNC	Japan Nuclear Cycle Development Institute
JNFL	Japan Nuclear Fuel Limited
K/SR	Kewaunee Nuclear Plant, Wisconsin to Savannah River Site, S. Carolina
LANL	Los Alamos National Laboratory
LHS	Latin hypercube sampling
LLW	low-level waste
LOS	loss of shielding
LWR	light-water reactor
MELOX	a fabrication plant in France
MEXT	Japan Ministry of Education, Culture, Sports and Technology
MFFP	Mixed Uranium Plutonium Oxide-Fresh Fuel Package
MITI	Japan Ministry of International Trade and Industry
MLIT	Japan Ministry of Land, Infrastructure and Transport
MOX	mixed oxide fuel
MRS	monitored retrievable storage
M/SR	Maine Yankee Nuclear Plant to Savannah River Site, S. Carolina
MS/V	Maine Yankee Nuclear Plant to Skull Valley, Utah
NEPA	U. S. National Environmental Policy Act
NASA	National Aeronautics and Space Administration
Np	neptunium
NRC	U. S. Nuclear Regulatory Commission
NRU	National Research Universal
NTIS	National Technical Information Service
ORNL	Oak Ridge National Laboratories
PacTec	Packaging Technology, Inc.
PATRAM	Packaging and Transportation of Radioactive Materials, an international conference
Pu	plutonium
PNC	Power Reactor and Nuclear Fuel Development Corp.
PR	permanent repository
PRA	probability risk assessment/probabilistic risk assessment
PWR	pressurized water reactor

R	rail
RAM	radioactive materials
RAMTRANS	*Journal of Radioactive Materials and Transport*
rem	roentgen equivalent man
ROD	Record of Decision
SDUS T	steel-DU-steel (truck)
SETU	Structural Evaluation Test Unit
SLS R	steel-lead-steel (rail)
SLS T	steel-lead-steel (truck)
SNL	Sandia National Laboratories
SST	Safe Secure Trailer
T	truck
TICLD	Transportation Incident Center Line Dose (code)
TRU	transuranic
TSD	DOE Transportation Safeguards Division
U	uranium

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1. Introduction

1.1 Scope of Work

This work supports plutonium transport studies performed by Sandia National Laboratories (SNL) for the Japan Nuclear Cycle Development Institute (JNC), under DE-F104-88AL52572, Appendix A-15, Statement of Work, and Section 1.2, Land Transport Emergency Response Technology. The scope of this task is to examine risk assessment techniques applicable to mixed oxide fuel (MOX) land transport in Japan, to examine the applicability of probability risk assessment (PRA) analysis techniques for MOX transportation, and to prepare recommendations for a future MOX fuel transport risk assessment program.

This report fulfills these tasks by (1) providing the methodology for examination of MOX land transport applicable to Japan for use in a future risk assessment program, (2) describing the RADTRAN risk code and source term determination methods for PRA analysis techniques, and (3) providing the criteria and latest methods for a MOX fuel transport risk assessment program.

This report reviews:

- Statutory regulations and requirements of risk analysis,
- Typical cask configurations for rail and truck transport,
- An overview of the methodology for calculating transportation risks,
- Container accident response and release fractions,
- Seal technology,
- Structural and thermal response in accident conditions,
- Source-term examples, and
- Examples of RADTRAN risk calculations.

This information provides an overview of the methods that will be required for detailed risk analyses for the Japanese MOX fuel program.

1.2 Overview

This report provides background information, which can be used for a risk assessment of MOX land transport in the nation of Japan to support nuclear energy programs there. This work includes a literature review, a review of other MOX activities worldwide, a survey of the statutory requirements for transporting nuclear materials, a discussion of risk assessment methodology, and calculations for specific examples. Typical examples are given to provide guidance for a later risk evaluation for MOX fuel transport in Japan. These examples consider transport by truck and rail for four generic Type B spent-fuel casks that were previously evaluated by SNL for the Reexamination Study (NUREG/CR-6672),[1-1] which developed new risk results and compared them to the results published in NUREG-0170[1-2] and the Modal Study.[1-3] This report also includes specific information that will be required for routes, cask types, accident-rate statistics, and population densities along specified routes, along with other detailed information needed for risk analysis studies pertinent to MOX transport in Japan. This information will be used in future specific risk studies.

To provide a complete understanding of MOX risk analysis, a comprehensive literature search was performed. Many of these references are cited in this report and will be useful in future route-specific risk studies.

Specific references were presented at PATRAM 2001 to provide the result of the Reexamination Study to international researchers in the area of radioactive materials transport.[1-4, 1-5, 1-6, 1-7, 1-8, 1-9, 1-10, 1-11]

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2. MOX Transportation

2.1 Role of MOX and Nuclear Power in Japan

Japan has used nuclear power ever since commercial operation began in the Ibaraki Prefecture in 1966. By 2001, 51 power reactors were operating throughout Japan with a total output of over 44 GW, approximately one-third of Japan's total electric power output.[2-1] Nuclear power is considered important to the future energy needs in Japan because of scarce natural resources and alleviation of environmental problems such as global warming and acid rain. Nuclear energy is thus expected to play a major role in the future in Japan as a power source for projected and increased electrical usage.

Table 2.1 lists 51 operating nuclear power plants in Japan; four under construction, four being planned, one closed, and two research reactors.

Japan Nuclear Fuel Limited (JNFL) was jointly established with the member companies of the Federation of Electric Power Companies Japan (FEPC) as the major partners[2-1] JNFL is in the beginning operational phases of a uranium (U) enrichment facility and a low-level radioactive waste storage facility in the northern prefecture of Aomori. JNFL is also working toward beginning operations of a reprocessing facility at the same site.

Plutonium is obtained after reprocessing spent fuel under strict controls during the nuclear fuel cycle based on the principle that no surplus plutonium is produced. JNFL plans to use MOX fuel pellets of uranium oxide mixed with plutonium oxide in light-water reactors (LWRs).[2-1] Although Japan was planning to gradually introduce this fuel for use in 16 to 18 of these reactors by 2010 or later, this timetable is likely to be extended. A large MOX fabrication plant was commissioned in 2000.

Although MOX programs in Japan have been under research and development for decades, until now, Japan has contracted the reprocessing of spent nuclear fuel to companies in the United Kingdom and France. Japan considers the recycling of spent fuel essential to establishing nuclear power as a domestic energy source that will be used to meet present and future energy needs.

Japan has also developed the Monju prototype fast breeder reactor (FBR), which generates more plutonium than is consumed. Construction of the Monju FBR began in January 1983.[2-2] Monju first achieved criticality on April 5, 1994. An accident caused by leaking sodium liquid occurred on December 8, 1995. The Nuclear Safety Commission is now examining the safety criteria for restarting Monju.

Plutonium, either separately or as MOX, has to be transported throughout the nuclear fuel cycle. Plutonium oxide is transported to Japan by sea for handling within the MOX fuel fabrication plant to supply MOX as fresh fuel to the nuclear power plants. Transportation will be required for spent fuel to be returned to the reprocessing plant, for return as reprocessed plutonium to the MOX fuel plant, and to a waste storage site before finally going to a disposal site for nuclear waste.

Table 2.1. Nuclear Power Plants in Japan

Site	Name of Plant	Unit Number	Company	Installed Capacity (MW)	Type of Reactor
1	Tomari	1	Hokkaido	579	PWR
		2		579	PWR
2	Onagawa	1	Tohoku	524	BWR
		2		825	BWR
3	Fukushima	1	Tokyo	460	BWR
	Daichi	2		784	BWR
		3		784	BWR
		4		784	BWR
		5		784	BWR
		6		1100	BWR
4	Fukushima	1	Tokyo	1100	BWR
	Daini	2		1100	BWR
		3		1100	BWR
		4		1100	BWR
5	Kashiwazaki	1	Tokyo	1100	BWR
	Kariwa	2		1100	BWR
		3		1100	BWR
		4		1100	BWR
		5		1100	BWR
		6		1356	ABWR
		7		1356	ABWR
6	Hamaoka	1	Chubu	540	BWR
		2		840	BWR
		3		1100	BWR
		4		1137	BWR
7	Shika	1	Hokuriku	540	BWR
8	Mihama	1	Kansai	340	PWR
		2		500	PWR
		3		826	PWR
9	Takahama	1	Kansai	826	PWR
		2		826	PWR
		3		870	PWR
		4		870	PWR
10	Ohi	1	Kansai	1175	PWR
		2		1175	PWR
		3		1180	PWR
		4		1180	PWR
11	Shimane	1	Chugoku	460	BWR
		2		820	BWR
12	Ikata	1	Shikoku	566	PWR
		2		566	PWR
		3		890	PWR
13	Genkai	1	Kyushu	559	PWR
		2		566	PWR
		3		890	PWR

(continued on next page)

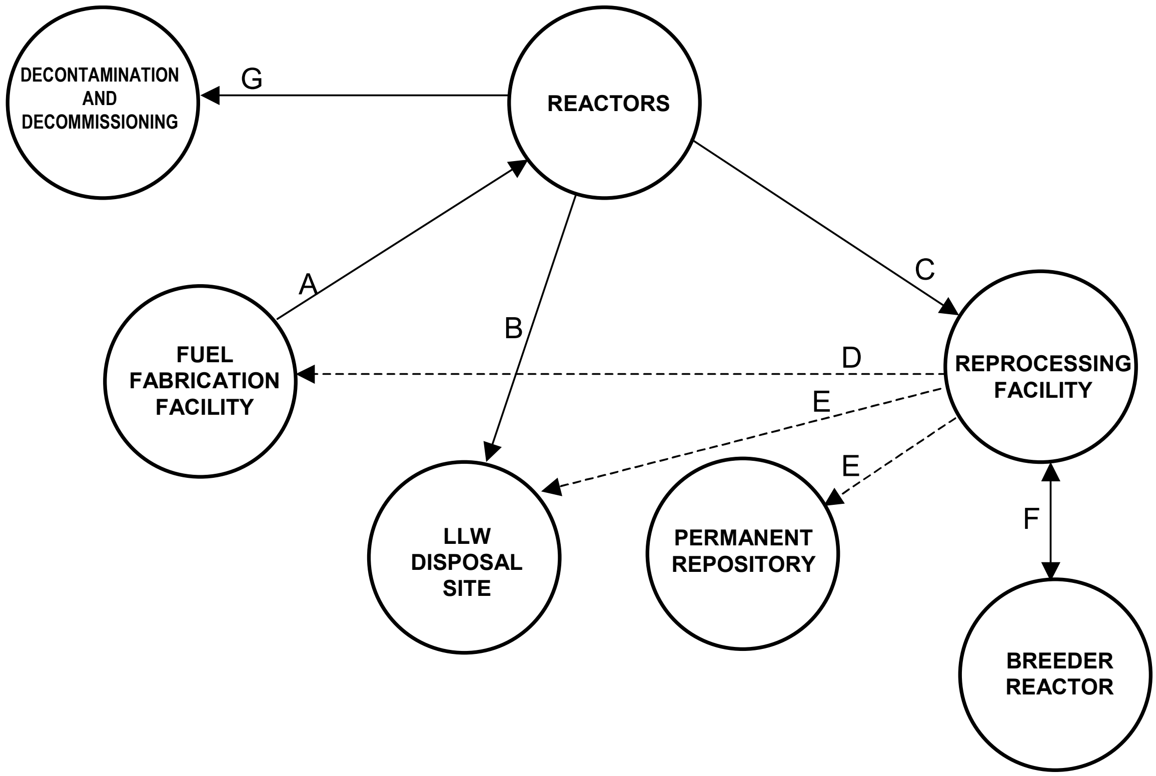
Table 2.1. Nuclear Power Plants in Japan (continued)

Site	Name of Plant	Unit Number	Company	Installed Capacity (MW)	Type of Reactor
14	Sendai	1	Kyushu	890	PWR
		2		890	PWR
15	Tokai Daini		Japan Atomic Power	1100	PWR
16	Tsuruga	1	Japan Atomic Power	357	BWR
		2		1160	BWR
	Totals	**51 Units**		**44,917 MW**	
			Under Construction		
2	Onagawa	3	Tohoku	825	BWR
17	Higashi-Dori	1	Tohoku	1100	BWR
6	Hamaoka	5	Chubu	1380	ABWR
7	Shika	2	Hokuriku	1358	ABWR
	Totals	**4 Units**		**4,663 MW**	
			Planned		
18	Maki	1	Tohoku	825	BWR
19	Ohma	1	EPDC	1383	ABWR
11	Shimane	3	Chugoku	1373	ABWR
1	Tomari	3	Hokkaido	912	PWR
	Totals	**4 Units**		**4,493 MW**	
			Closed		
	Tokai		Japan Atomic Power	166	GCR
			Others		
	Fugen		JNC	165	ATR
	Monju		JNC	280	FBR

Abbreviations:
PWR: pressurized water reactor
BWR: boiling water reactor
ABWR: advanced boiling water reactor
GCR: gas cooled reactor
ATR: advanced thermal reactor
FBR: fast breeder reactor

2.2 Types of Radioactive Material Shipped

To provide a thorough analysis of MOX transportation in Japan, detailed knowledge will be required for shipments between the facility pairs depicted in Figure 2.1 (modified) to support the power generation functions of MOX reactors.[2-3]



A. Fresh fuel from fuel fabrication facility (FFF) to the reactors
B. Low-level waste (LLW) from reactors to LLW disposal facility
C. Spent fuel shipments from reactors to off-site facility or reprocessing facility
D. Reprocessing facility shipments to FFF
E. Reprocessing facility shipments to LLW or PR, using different cask types
F. Reprocessing facility shipments to/from breeder reactor
G. Reactor and other shipments to decontamination and decommissioning areas

Figure 2.1. Schematic of Nuclear Fuel Cycle.

In addition to normal nuclear fuel cycle operations, transportation of materials involving breeder reactor functions, such as fresh fuel, low- and high-level waste (HLW), fresh fuel transport, and decontamination and decommissioning shipments must be evaluated in order to provide complete risk assessments for particular applications.[2-3, 2-4]

2.3 Past Fuel Shipments

Spent fuel from the Fugen nuclear power station will be transported to a reprocessing plant.[2-5] Transport of fresh MOX fuel assemblies for the prototype fast breeder reactor Monju initial core began in July 1992 and operated intermittently until March 1994, with 205 fresh MOX fuel assemblies transported in nine shipment campaigns.[2-6, 2-7] Minoru Kubo, in a 1994 article, describes the regulations, results, and trends of marine, land, and air transport of plutonium that was recovered from reprocessing spent fuel. This article includes a discussion of the transport from the French COGEMA La Hague installation to Japan by the "Akatsuki Maru."[2-8]

As described by S. Kikuchi, et al., in an article from the PATRAM '83 proceedings, a long-term cooperative program on transportation technology between PNC, later JNC, and the U. S. Department of Energy (DOE) has been in place to conduct risk assessments, develop structural models, conduct impact and fire tests, and evaluate leakage for special nuclear materials (including plutonium) in simulated overland transportation systems under normal and accident conditions.[2-9]

2.4 MOX Research in Japan

Extensive research in Japan has been conducted regarding the use of MOX as a nuclear power reactor fuel. The 1992 status of nuclear fuel transport in Japan was described by S. Aoki, S. Fukada, I. Tsuji, and H. Kuno in an article in the PATRAM '92 proceedings.[2-10] A more current status report was presented by S. Hamada, S. Fukada, and I. Nakazaki in the PATRAM '01 proceedings.[2-11]

Specific recent research from Japan on MOX related transportation issues was reported in the PATRAM '01 proceedings. Included are reports on:

- cask testing and safety,[2-12, 2-13, 2-14]
- spent fuel transport experience,[2-15, 2-16, 2-17]
- cask design and development,[2-18, 2-19, 2-20, 2-21]
- cask seals,[2-22]
- cask failure probability,[2-23]
- risk analyses,[2-24, 2-25, 2-26, 2-27]
- plutonium air transport packaging,[2-28, 2-29] and
- burnup credit.[2-30, 2-31]

A criticality analysis using the SCALE Code[2-32] was performed for MOX fuel criticality experiments at the Japan Atomic Energy Research Institute (JAERI) for infinite fuel rod arrays, varying the parameters of plutonium enrichment and lattice pitch.[2-33]

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3. Scope of Investigation

This report examines the methodology, with examples, to aid in preparing the environmental impact information reports required for Japan's land transport of MOX in support of nuclear energy programs in Japan. This report also reviews the statutory regulations and requirements of risk analysis; it describes typical cask configurations for rail and truck transport; it provides an overview of the methodology for calculating transportation risks, container accident response and release fractions, seal technology, structural and thermal response in accident conditions, source-term examples, and examples of RADTRAN risk calculations. This information provides an overview of the methods that will be required for detailed risk analyses for the Japanese MOX fuel program.

3.1 Aspects of Risk

The term "risk" reflects both the probability of a hazard and the consequences of exposure to this hazard; technically "risk" is evaluated as the product of a consequence and its probability of occurrence.[3-1] In probability theory, mean risk ($\hat{R}$) is the sum of the risks of a set of representative events whose probabilities of occurrence sum to one, and thus includes the "nothing happens" representative event. As will be explained in Section 14.4, mean risk is the area under a complementary cumulative distribution function (CCDF) curve.

Thus

$$\hat{R} = \sum_{i} P_i C_i \quad \text{provided} \quad \sum_{i} P_i = 1$$

where

$\hat{R}$ = mean risk
P_i = probability of occurrence for event i
C_i = consequence of event i

However, the risk of any "something happens" event is usually not a mean risk. Thus, for any i, $P_i C_i \neq$ mean risk.

Further refinement may be employed in risk assessment; for example, the risks of low-probability, high-consequence events may be numerically equal to those of high-probability, low-consequence events, but with very different consequences.

The definition of risk is inherently controversial; no one definition is suitable for all situations. An article on probabilistic risk assessment (PRA) for nuclear power plants provides a technique for integrating different aspects of design and operation to assess risks and to develop an information base for both specific and general issues.[3-2]

Land transport of nuclear material involves a risk arising from accident conditions during transport that could lead to the release of radioactive materials to the environment. Another risk results directly from exposures from radiation that occur when transport takes place without the occurrence of serious accidents (incident-free transport).

3.2 Shipping Container Accident Risk

Despite the fact that there never has been a significant release of radiological material from a transportation accident during the four decades that nuclear transport has been studied, the safety of nuclear material containers in accident conditions remains an issue of public concern. Even though nuclear shipment containers are manufactured, tested, and used under very stringent regulations, the risks of nuclear shipments must be carefully evaluated to address and alleviate perceptions of risks.

In 1977, the U. S. Nuclear Regulatory Commission (NRC) published NUREG-0170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes."[3-3] This report included an assessment of the likelihood and magnitude of radiological consequences associated with transport accidents during the shipment of radioactive materials (RAM). This assessment indicated that the radiological risk associated with all RAM shipments, including spent fuel, was small. This conclusion provided the technical basis for the NRC decision that the existing Code of Federal Regulations, 10 CFR 71, were adequate and not in need of immediate change (46 FR 21629, April 13, 1981). However, the NRC Commission also stated that "regulatory policy concerning transportation of radiological materials be subject to close and continuing review."

Nuclear fuel, contained in fuel rods, is typically made of ceramic pellets of uranium encased within a cylindrical cladding. In the case of MOX fuel, plutonium is combined with uranium to form a MOX. As spent fuel is produced in the reactor, the plutonium content is reduced to roughly 1% by weight[3-4] and becomes embedded in a highly radioactive matrix of the MOX fuel that deters its value for malevolent actions. The fuel rods are approximately 15 ft long and clad with a material such as zirconium or stainless steel as is used in Monju. This cladding provides a sealed environment for the fuel pellets. Fuel rods are bundled, depending on the reactor type, in either square or hexagonal arrays containing 50 to 300 rods. This bundle is contained in a supporting structure called a basket.

The casks used to transport spent fuel must, in the United States, be certified by the NRC as being in compliance with 10 CFR 71.[3-5] These regulations are almost identical in their requirements to internationally accepted standards that have been in effect for years. Recently updated, these international regulations are intended to protect the public during both normal transport and also during accidents.[3-6]

In 1987, a study was completed to define and evaluate the responses of spent fuel casks exposed to severe highway and railway accident conditions.[3-7] This report, generally referred to as the Modal Study, concluded that the radiological risks from spent fuel under the severe highway and railway accident conditions that were examined in the study are less than the risks estimated in the NUREG-0170 document published in 1977.

In 2000, SNL reexamined the risks associated with the transport of spent nuclear fuel by truck and rail and compared the new risk estimates to those published in NUREG-0170 and the Modal Study.[3-8] The results of this study, together with those of the previous studies, demonstrate that the risks associated with the shipment of spent fuel by truck or rail are very small.

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4. Other MOX Programs

4.1 MOX Description

The reactor fuel known as mixed oxide or MOX fuel is not a material directly useful in weapons.[4-1] By itself, fresh MOX is not explosive and, in pellet form, is not highly radioactive. The plutonium oxide concentration (approximately 3%) is diluted in the uranium oxide fuel matrix. After irradiation in a reactor and its subsequent removal as spent fuel, the fuel is highly radioactive because of the production of fission products.

Pu-239 is formed in reactors from U-238 by the following chain process:

$$\text{U-238} \xrightarrow{\text{neutron capture}} \text{U-239} \xrightarrow{\text{electron emission}} \text{Np-239} \xrightarrow{\text{electron emission}} \text{Pu-239}$$

Neutron capture by Pu-239 produces higher Pu isotopes:

$$\text{Pu-239} \xrightarrow{\text{neutron capture}} \text{Pu-240} \xrightarrow{\text{neutron capture}} \text{Pu-241} \xrightarrow{\text{neutron capture}} \text{higher Pu isotopes}$$

The longer U-238 is irradiated, the greater the percentage of heavier Pu isotopes.

Pu-239 is the preferred Pu isotope for nuclear weapons, thus the large proportion of Pu-240 and Pu-242 in reactor-grade plutonium makes it less than optimal for weapons.[4-2]

Plutonium is classified into grades, depending on the Pu-240 content. Weapon-grade plutonium contains less than 7% Pu-240, fuel-grade between 7 and 19%, and reactor-grade, greater than 19%.[4-1]

Plutonium is a silver-colored metal with a melting point of 641°C. In its bulk metal form, it slowly oxidizes at room temperature. Particles rapidly oxidize, and finely divided Pu metal spontaneously ignites at about 150°C. PuO_2 is the oxide of Pu used in MOX fuel.

Plutonium outside the body presents little risk because it emits primarily alpha particles. If plutonium is ingested in food or water, only one part in 1000 is absorbed by the gastrointestinal tract and migrates from there mainly to bones and the liver. The main hazard is inhalation of particles, which are then retained in the lungs and subsequently migrate into the blood or lymph system and finally into the liver and bones, where the alpha particles may eventually cause cancer.[4-1]

Without proper precautions, inhalation exposures can occur when MOX fuel is being fabricated because this is when the plutonium is in a powder form. To reduce the risk of contamination by the powder, the fuel is made in sealed glove boxes. In MOX fuel, the plutonium is embedded in the UO_2 matrix in a very dense ceramic pellet from which release under normal conditions is physically impossible.

Fabrication of MOX fuel is a well-established process. Approximately 32 European reactors are licensed to use MOX fuel and six in France are using MOX as 30% of their fuel.

Basically two types of risks are associated with MOX fuel. The first is the release of plutonium and fission products into the environment as a result of accidents during transport and disposal, and the second is associated with terrorism. International safeguards are well established for transport modes and are followed according to national and international requirements.

4.2 MOX Spent Fuel

In converting MOX to spent fuel, about two-thirds of the most fissile plutonium is consumed. Spent MOX fuel is similar to spent fuel from a reactor burning natural uranium oxide fuel. Both contain plutonium, other actinides, and fission products. In normal spent fuel, the plutonium concentration is about 0.4%; in MOX spent fuel it is about 1.4%.[4-1] Both natural uranium oxide and MOX spent fuels contain high levels of radioactivity. This provides a natural deterrent to diversion by anyone with malevolent or malicious intent because the radioactivity is too high to allow safe handling, and the concentration of plutonium is too low for use in nuclear weapons. Also, the plutonium isotope mix is far from ideal for weapons.

4.3 Mixed Oxide Fuel Program in France

The nuclear industries in Japan, France, and other European countries have a long-term global strategy of fuel utilization involving recycling and reuse of spent fuel.[4-3, 4-4, 4-5, 4-6, 4-7] Reprocessing started in France in 1958 at Marcoule.[4-5] In the closed fuel cycle, reprocessing of spent fuel extracts the plutonium and uranium to allow treatment, conditioning, storage, transportation, and final disposition. An assessment of the economics of the French reprocessing program was discussed in a paper by B. Lenail in 1986.[4-5] In 1994, the COGEMA installation in France was responsible for an average of 350 road, rail, and sea shipments of spent fuel to its La Hague reprocessing facility. Plutonium and MOX fuel have been safely transported in this operation for many years. The first MOX fuel elements were loaded in an experimental reactor in Belgium in the 1960s. By 1994, more than 300 metric tons of MOX fuel had been irradiated in Europe. By 2000, at least 16 Electricité de France (EDF) reactors were loaded with MOX fuel, and 15 to 20 other power plants will be operated with MOX fuel in Europe and Japan. A worldwide MOX fuel fabrication of 400 to 500 metric tons per year is expected in the near future, which will enable the recycling of up to 40 metric tons of plutonium annually.[4-4]

Today, MOX fuel can be loaded in up to 30% of the core of light-water reactors. In the future, the potential exists to use MOX in 100% of the cores in advanced reactors currently being developed.[4-5]

J. Malherbe of the Commission of the European Communities (CEA) published a comprehensive study of the management of radioactive waste as a result of reprocessing and disposal in 1991.[4-8] This report concluded that the generation of spent MOX fuel does not introduce a significant health threat into the nuclear fuel cycle.

C. Mattera et al. recently published a paper in PATRAM '01 that discussed the consequences of the use of average plutonium content for criticality evaluation of PWR MOX-fuel transport and storage packages.[4-9]

4.3.1 Spent Fuel Transportation in France

Transportation of nuclear materials has been progressively more important in linking the various elements of the nuclear fuel cycle. COGEMA continues to manage every year, directly or through its subsidiary, COGEMA Logistics and its affiliates, an average of 350 shipments of spent fuel to its La Hague reprocessing facility. Spent fuel is transported within continental Europe by rail in 25-m-long cars weighing 160 tons. For short distances between the power plants, rail terminals, and the La Hague plant, spent fuel is transported over public roads on special tractor-trailers. In France, the trailers weigh 130 metric tons and have 64 wheels on eight to ten axles.[4-5] Fuel is shipped between European countries in ships weighing 2000 metric tons. For transport to Japan, specially designed ships of 3000 metric tons are used. Equipment on these ships includes redundant engines, double hulls, radiation shielding, and advanced redundant communication systems.[4-5]

Approximately 150 shipments of plutonium were made from La Hague to MOX fuel fabrication plants in Europe between 1984 and 1994. Two shipments of plutonium from La Hague to Japan have been made, one in 1984 and another in 1992.

MOX transportation by COGEMA supports the La Hague reprocessing facility and three MOX fabrication plants: Cardache and MELOX, in southern France, and Belgonucléaire-Dessel in Belgium.[4-7] By 1997, the La Hague plant had reprocessed more than 12,000 tons of spent fuel, and 29 European reactors were loaded with MOX fuel in routine operations.[4-7]

COGEMA Logistics, a wholly owned subsidiary of COGEMA, has developed a wide range of casks for spent fuel as well as a specialized cask, the TN-28, for vitrified radioactive residues. COGEMA Logistics (the former Transnucléaire) has developed a total of more than 150 different types of casks for the nuclear fuel cycle.[4-7, 4-8]

H. Neau of COGEMA published a recent paper in PATRAM '01 describing the importance of transparency and dialogue in global acceptance of nuclear material transport.[4-10]

4.4 MOX Program in China

In the 1980s, China began to develop nuclear power reactors of the pressurized water type and formulated a closed fuel-cycle strategy.[4-11]

Spent fuel must be transported from the nuclear power plants mainly situated in southern and eastern coastal areas, while the reprocessing facility is in the northwest, some 3000 to 4000 km away. Casks were developed and sea, rail, and road transport plans were formulated. Radioactive waste management at interim storage facilities and repositories has been addressed.[4-11]

4.5 Mixed Oxide Fuel Program in England

British Nuclear Fuels Limited (BNFL) has a record of safely and efficiently transporting plutonium, including MOX fuel, for more than 30 years, both nationally and internationally.[4-12] They attribute their successful safety record to the design and development of robust casks and packagings and to the rigorous checks carried out at each stage of operations. Physical security is maintained by strict adherence to international conventions.

To further the public's education about the safety of their operations, BNFL encourages the public to visit their nuclear sites and marine operations. They have an "open door" policy at both Sellafield and at the BNFL Marine Terminal at Barrow.

Air transport of MOX fuel assemblies for the Dounreay prototype fast reactor began in 1978, and since then, several tons have been transported by air. MOX fuel rods, powder, and pellets have been transported between the United Kingdom and the European continent. Examples include air transport of MOX fuel assemblies to the Beznau reactor in Switzerland and sea transport of MOX assemblies to the Unterweser reactor in Germany.[4-12, 4-13]

4.6 MOX Program in Canada

An option being considered by the United States and the Russian Federation for disposition of excess plutonium from dismantled weapons is conversion to MOX fuel for use in Canadian Deuterium-Uranium (CANDU) power reactors. The U. S. DOE is participating in a demonstration project called Parallax (for parallel experiment) involving production of laboratory quantities of MOX fuel at the Los Alamos National Laboratory (LANL) and at the Bochvar Institute in the Russian Federation. The objective of the Parallax Project is to simultaneously irradiate test quantities of MOX fuel from the United States and the Russian Federation in a test reactor with operating conditions like those of the CANDU reactors. This irradiation is to take place in the pressurized loops of the National Research Universal (NRU) test reactor at the Atomic Energy of Canada Ltd. (AECL) Chalk River Laboratories (CRL).[4-14] The U. S. DOE plans to fabricate and transport up to 59.2 lb (26.8 kg) of MOX fuel as part of the Parallax Project.[4-15]

The MOX fuel consists of sintered oxide pellets encased in Zircaloy-4 cladding that forms sealed fuel elements. The MOX fuel produced at LANL is loaded into an AECL Model 4H shipping package certified as a Type B(U)F package. In compliance with the United States National Environmental Policy Act (NEPA), which requires all federal agencies to consider the environmental consequences of proposed actions before decisions are made, LANL developed a draft environmental assessment that DOE issued for public review and comment in August 1997.

After finalization of the environmental assessment and determination of no significant impact, DOE will apply to the Nuclear Regulatory Commission for a license to export the MOX to Canada.

Oak Ridge National Laboratory (ORNL) provides the overall management of the Parallax Project to DOE. Specific activities include transportation planning, truck route selection, carrier selection, and shipment tracking.[4-16]

4.7 United States MOX Program

The end of the Cold War produced a legacy of weapons-usable fissile materials in both the United States and the Russian Federation. In 1995, the United States announced that approximately 224 tons (203 metric tons) of weapons-usable fissile materials had been declared surplus to U. S. defense needs. Of this amount, 38 metric tons are weapons-usable plutonium.[4-17]

The DOE has analyzed strategies for disposition and storage of this surplus plutonium.[4-17] The environmental impact statement (EIS) that contains this analysis examined various methods for implementing the disposition of the surplus plutonium. On January 14, 1997, the DOE issued its Record of Decision on the storage and disposition of surplus nuclear weapons materials. DOE made a commitment to develop a program for (a) surplus plutonium disposition by immobilization by vitrification and (b) fabricating MOX reactor fuel for use in existing nuclear power plants.

The recommended approach would allow vitrification of surplus plutonium in glass or ceramic forms as well as the use of MOX fuel in existing reactors.[4-18] The reactors used for burning and irradiation of MOX fuel could be the existing domestic commercial LWRs and reengineered heavy-water-moderated reactors such as the CANDU reactors. The decision to undertake either or both of these disposition methods depends on technology, costs, environmental reviews, nonproliferation concerns, and negotiations with Canada and Russia.[4-18]

The glass vitrification facility will be located at either the Hanford Site in Washington or the Savannah River Site in South Carolina. A MOX fuel fabrication facility, and a pit disassembly and conversion facility, would be located at either Hanford, Savannah River, the Pantex Plant in Texas, or the Idaho National Engineering and Environmental Laboratory (INEEL).[4-18]

DOE will support a test and demonstration program for CANDU MOX fuel, with activities at Los Alamos National Laboratory and at the Chalk River Laboratories in Ontario, Canada. DOE will fabricate a limited amount of MOX fuel to facilitate testing and demonstration of the use of MOX fuel in CANDU reactors in a test program known as the Parallax Project.[4-15]

For fresh MOX fuel transportation for the demonstration program, the initial plan was for the rod samples to be transported in a Type A shipping package, such as the Model 4H Enriched Fuel Bundle Shipping Package. This package, designed and manufactured in Canada, is a 55-gal (208-L) metal drum with a sealable lid. The MOX fuel will be shipped by commercial truck carrier in accordance with U. S. Department of Transportation and Canadian regulations.

The environmental assessment for the Parallax Project[4-18] considered transportation modes other than truck, performed route analyses, evaluated environmental issues, considered environmental justice, estimated population radiation doses, calculated radiological-incident-free doses to the public and truck crew during single shipments, calculated radiological dose-risks for accidents, evaluated accident source terms, and calculated accident doses and radiation induced latent cancer fatalities for all routes.

Oak Ridge National Laboratories provided a conceptual design of a revised transport package that could be used as an alternative to maximize the number of fresh MOX fuel assemblies that can be carried in a Safe Secure Trailer (SST) between the MOX fuel fabrication facility and the reactors. Other reports by Oak Ridge National Laboratories discuss the programmatic and technical requirements and the shipment plan for the fresh fuel package.[4-19, 4-20, 4-21]

An economics study was performed in 1983 to survey the impact of transuranic waste storage on the conceptual design of a stand-alone monitored retrievable storage facility.[4-22] The study concluded that current technology is available to store transuranic waste safely and economically from light water reactor (LWR) fuel reprocessing and MOX fuel prefabrication operations.

In November 1999, DOE published a Surplus Disposition Final EIS that analyzed the impacts of three facilities for the disposition of surplus plutonium, including MOX, and the impact of using MOX fuel in reactors.[4-23] In January 2000, DOE issued a Record of Decision (ROD) for the final EIS that identified a hybrid approach for disposing up to 50 metric tons of surplus U. S. weapons plutonium by immobilizing up to 17 metric tons and irradiating 33 metric tons as MOX fuel. Ultimately, both approaches would involve disposal in a geological repository pursuant to the Nuclear Waste Policy Act.

In March 1999, DOE signed a contract with a consortium of Duke, COGEMA, and Stone and Webster to design and obtain a commercial MOX fuel system consisting of a MOX fuel fabrication facility, reactor modifications necessary for the use of MOX fuel, and construction and management of these facilities.[4-23] The MOX fabrication facility will be U. S. government-owned and located at a DOE site at Savannah River, South Carolina. The facility will be shut down when the plutonium disposition is completed.

A MOX fresh fuel package, the Mixed Uranium Plutonium Oxide Fresh Fuel Package (MFFP) has been designed by Packaging Technology, Inc. (PacTec) as part of the Duke, COGEMA, and Stone and Webster consortium. This package will be used to support the DOE program for domestic MOX fuel fabrication and irradiation for the purpose of disposing of surplus weapons usable plutonium.[4-24]

The Materials Identification and Surveillance Program has been established at Los Alamos National Laboratory (LANL) in response to the Defense Nuclear Facility Safety Board (DNFSB) recommendations to establish parameters for safe storage of plutonium destined for DOE materials disposition in accordance with DOE standard (DOE-STD-3013-99).[4-25]

4.8 German MOX Transport Packages

Fuel assemblies manufactured for the SNR 300 fast breeder reactor are currently stored at different locations in Germany and abroad. These fuel assemblies are to be moved to a single storage site in Germany.[4-24] In addition to the SNR 300 assemblies, approximately 800 MOX fuel rods are also to be stored. Their single fuel assembly container, the ESBB, was developed for this packaging. This B (U) licensed package is capable of holding 91 SNR 300 fuel rods or 40 MOX fuel rods, with a plutonium content up to approximately 10 kg.[4-26]

A new ANF-18/MOX transport system has been designed for shipment of nonirradiated PWR MOX fuel assemblies from the fuel fabrication facility to the power plant.[4-27] This packaging is based on the recently developed ANF-18 shipping container for nonirradiated uranium fuel assemblies.[4-25]

4.9 Mixed Oxide Fuel Program in Russia

On June 4, 2000, the United States and the Russian Federation released a fact sheet[4-28] announcing that a key arms control and nonproliferation agreement had been reached "providing for the safe, transparent, and irreversible disposition of 68 metric tons of weapons-grade plutonium."

Under this agreement, each party must dispose of 34 metric tons of plutonium by irradiating it as fuel in reactors or by immobilizing it with high-level radioactive waste, rendering it suitable for geologic disposal. The United States intends to use 25.5 tons as fuel and to immobilize 8.5 tons; the Russian Federation intends to use 34 tons as fuel.[4-28]

The agreement requires each party to begin operation of industrial-scale facilities by 2007 to achieve a disposition rate of at least two metric tons of weapons-grade plutonium per year and to work with other countries to find ways to double that rate.[4-28]

The work in the Russian Federation closely parallels the reported work described above for the United States and Canadian programs.

The VVER-1000-type Balakovo Nuclear Power Plant was chosen to dispose of the plutonium designated from the Russian weapons program.[4-29] Criticality calculations were performed for fresh fuel assemblies and for spent fuel in storage and transportation configurations.[4-29, 4-30] The SCALE Code[4-31] was used for those criticality calculations.

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5. Packaging and Representative Shipment Configurations

5.1 Packaging Overview

Although several federal, state, and international organizations are involved in regulating radioactive material, primary responsibility in the United States resides with the Department of Transportation (DOT) and the Nuclear Regulatory Commission (NRC). All transportation activities must be in accordance with the applicable regulations of these agencies specified in 49 CFR Part 173 (DOT 1997) and 10 CFR Part 71 (NRC 1997). The international regulations are covered in International Atomic Energy Agency (IAEA) ST-1, formerly IAEA Safety Series 6.

Transportation packagings for small quantities of radioactive materials must be designed, constructed, and maintained to contain and shield their contents during normal transport conditions. For large quantities and for more highly radioactive material, such as plutonium, MOX, or spent fuel, packagings must contain and shield their contents in the event of a severe accident. The type of packaging used is determined by the total radioactive hazard posed by the material packaged. Four basic types of packaging are used:

- Excepted
- Industrial
- Type A
- Type B

Another packaging option, "Strong, Tight," is still available for some domestic shipments.

Regulations require that commercial quantities of plutonium be shipped in Type B packages. As a result, the following discussion will be limited to Type B packages.

NRC regulations, 10 CFR 71, require a Type B package when transporting more than an activity level of material (A_2 quantity), which in the case of plutonium, is a small amount. A Type B package must comply with a number of regulations that address structural integrity, criticality, shielding, containment, and heat dissipation in both normal conditions and accidents conditions. The package must be able to withstand the following events without loss of containment:

- A 9-m drop onto an unyielding surface
- A 1-m drop onto a prescribed unyielding punch
- A fully engulfing 800°C fire for 30 minutes
- Immersion in 15 m of water for 8 hours

In addition to these tests, the package must also satisfy containment specifications for leakage rate. In the United States, safe secure trailers are used to provide physical protection of specified and classified nuclear material cargos.

5.2 Cask Descriptions

Type B casks used for transport of spent nuclear fuel are typically manufactured in three weight classes: legal-weight truck, overweight truck, and rail. They use three gamma-shielding materials: steel, lead, and depleted uranium (DU). For most applications, truck casks will be lead- and DU-shielded, and rail casks will be steel- and lead-shielded.[5-1] A publication titled *Shipping and Storage Cask Data for Commercial Spent Nuclear Fuel*, by JAI Corporation, provides specific information on cask construction and dimensions.[5-2]

The Reexamination Study analyzed four generic casks to support its reevaluation of the risks of spent fuel transportation.[5-1] These generic casks are:

- Steel-lead-steel truck cask, Figure 5.1 (Reexamination Study, Table 4.1),
- Steel-DU-steel truck cask, Figure 5.2 (Reexamination Study, Table 4.2),
- Steel-lead-steel rail cask, Figure 5.3 (Reexamination Study, Table 4.3), and
- Monolithic steel rail cask, Figure 5.4 (Reexamination Study, Table 4.4).

The capacity of these casks was assumed to be 24 pressurized water reactor (PWR) or 52 boiling water reactor (BWR) assemblies for the rail casks and one PWR or two BWR assemblies for the truck casks. All of the generic casks were assumed in the Reexamination Study to have elastomeric o-ring seals inboard of the closure bolt locations. The closure on all the casks is recessed into the cask body with a face-seal orientation as the usual configuration.

A guideline for anticipated transportation, packaging, and facility handling operations likely at MOX fuel fabrication and commercial reactor facilities was prepared by ORNL.[5-3] This information was intended for use by prospective contractors to DOE for MOX fuel assembly irradiation in commercial LWRs.

An article that provides a general analysis of the transportation requirements for post-fission radioactive wastes produced from the commercial LWR fuel cycle discusses transport of the following types of radioactive waste:[5-4]

- Spent fuel
- Solidified HLW
- Fuel residues (cladding wastes)
- Plutonium
- Non-high-level transuranic (TRU) wastes

This article describes transportation for wastes generated in three fuel-cycle options: once-through fuel cycle, uranium recycle only, and recycle of uranium and plutonium.

Name	Weight (pounds)	Material	Closure Bolts (no/size)	Wall Thickness (inches)	Outside Diameter (inches)	Cavity Diameter (inches)	Length (inches)	Impact Limiter	Design Heat Rejection (kW)	Seal Material	C of C
NAC-LWT	52,000	stainless	12 1"	0.75,5.75,1.2	44.2	13.375	199.80	honeycomb	2.5	both	71-9225
NAC-1	49,000	stainless	6 1.25"	0.31,6.63,1.25	38	13.5	214	balsa	11.5	Elastomer	71-9183
NLI-1/2*	49,250	stainless	12 1"	0.5,2.125Pb 2.75DU,0.875	47.125	13.375	195.25	balsa	10.6	metal	71-9010
TN-8**	79,200	steel	16 1.25"	0.23,5.32,0.79	67.6	~30	217.2	balsa	35.5	Elastomer	71-9015
TN-9**	79,200	steel	16 1.25"	0.23,5.04,0.79	67.6	~21	226.6	balsa	24.5	Elastomer	71-9016
TN-FSV	47,000	stainless	12 1"	1.12,3.44,1.5	31.0	18.0	207	wood	0.36	Elastomer	71-9253
Modal Study	N.A.	stainless	N.A.	0.5,5.25,1.25	27.5	13.5	193	yes	0.8-5.4	N.A.	-
Generic	**50,000**	**stainless**	**12 1"**	**0.5,5.5,1.0**	**27.5**	**13.5**	**205**	**yes**	**2.5**	**Elastomer**	**-**

* This cask has a steel-lead-DU-steel wall configuration and was therefore not used to characterize the generic cask.
** These casks are overweight-truck casks and were therefore not used to characterize the generic cask.

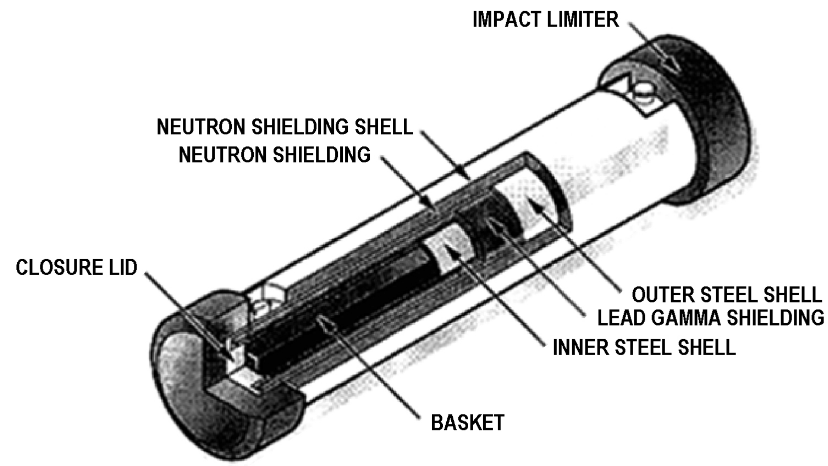

Figure 5.1. Steel-Lead-Steel Truck Cask Configuration.

Name	Weight (pounds)	Material	Closure Bolts (no/size)	Wall Thickness (inches)	Outside Diameter (inches)	Cavity Diameter (inches)	Length (inches)	Impact Limiter	Design Heat Rejection (kW)	Seal Material	C of C
FSV-1	47,600	stainless	24 1.25"	0.67,3.5,0.91	28.0	17.7	208	yes	4.1	Elastomer	71-6346
GA-4	53,610	stainless	12 1"	0.375, 2.64,1.5	39.75	18.16 sq.	187.75	honeycomb	2.47	Elastomer	71-9226
GA-9	54,000	stainless	12 1"	0.25,2.45,1.75	39.75	18.16 sq.	198.3	honeycomb	2.12	Elastomer	-
NLI-1/2*	49,250	stainless	12 1"	0.5,2.125Pb, 2.75DU,0.875	47.125	13.375	195.25	balsa	10.6	metal	71-9010
Generic	**50,000**	**stainless**	**12 1"**	**0.5,3.5,0.9**	**28**	**18**	**200**	**yes**	**2.5**	**Elastomer**	**-**

* This cask has a steel-lead-DU-steel wall configuration and was therefore not used to characterize the generic cask.

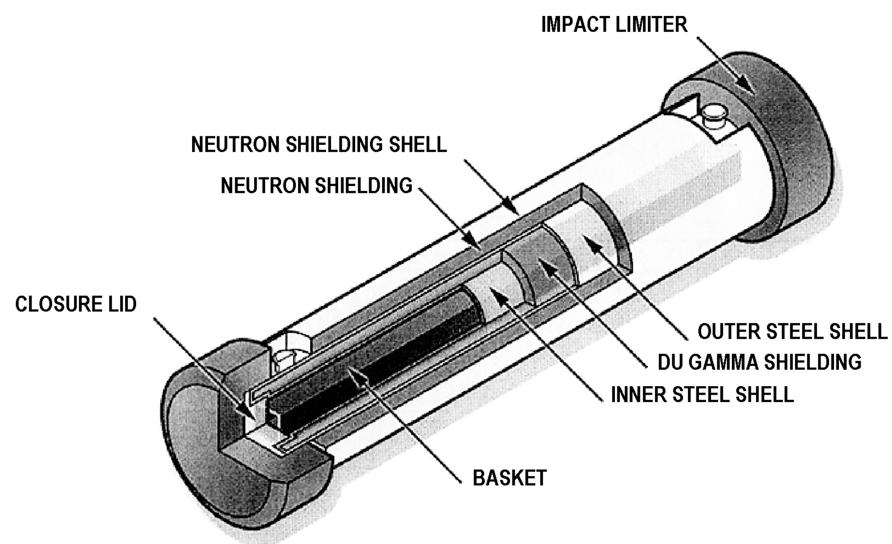

Figure 5.2. Steel-DU-Steel Truck Cask Configuration.

Name	Weight (pounds)	Material	Closure Bolts (no/size)	Wall Thickness (inches)	Outside Diameter (inches)	Cavity Diameter (inches)	Length (inches)	Impact Limiter	Design Heat Rejection (kW)	Seal Material	C of C
NAC-STC	250,000	stainless	42 1.5"	1.5,3.7,2.65	87.0	71.0	193	wood	22.3	metal	71-9235
TranStor	244,000	stainless	N.A.	N.A.	87.0	67.0	210.0	honeycomb	26	metal	-
125B	181,500	stainless	32 1.5"	1.0,3.88,2.0	65.5	51.25	207.5	foam	0.7	Elastomer	71-9200
Excellox -6	194,000	ferritic steel	N.A.	N.A.	83.23	32.8	200.5	yes	N.A.	N.A.	-
NLI-10/24	194,000	stainless	16	.75,6,2	96.0	45.0	204.5	balsa	70	both	71-9023
BR-100	202,000	stainless	32 2.5"	1.0,4.5,1.75	82	58.5	202	wood	15	Elastomer	-
Modal Study		stainless	N.A.	0.5,5.25,1.5	52	37.5	193	yes	3.4-24	N.A.	-
Generic	**225,000**	**stainless**	**24 1.75"**	**1.0,4.5,2.0**	**80**	**65**	**200**	**yes**	**24**	**Elastomer**	**-**

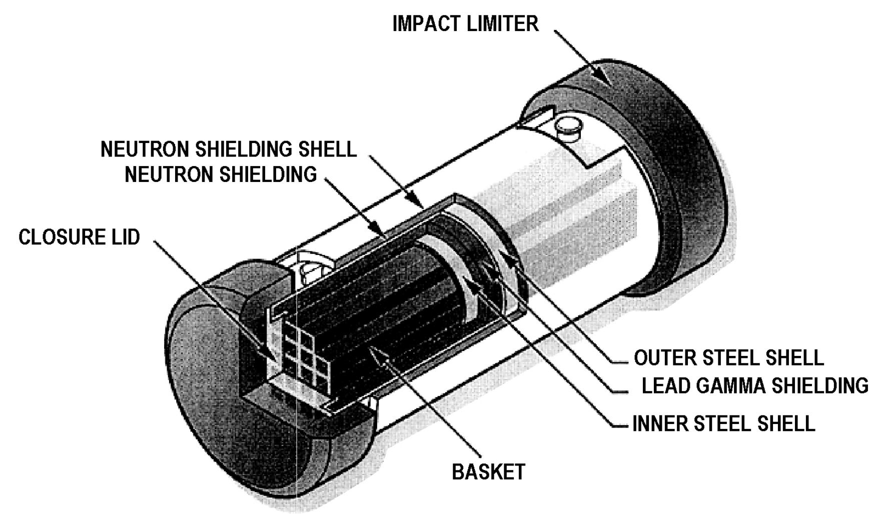

Figure 5.3. Steel-Lead-Steel Rail Cask Configuration.

Name	Weight (pounds)	Material	Closure Bolts (no/size)	Wall Thickness (inches)	Outside Diameter (inches)	Cavity Diameter (inches)	Length (inches)	Impact Limiter	Design Heat Rejection (kW)	Seal Material	C of C
TN-24**	224,000	SA-350	N.A.	9.5	92.4	57.25	186.8	none	24	metal	72-1005
REG	225,000	SA-350	48 1.625"	9.25	90.25	71.25	180	redwood	2.7	both	71-9206
BRP	215,000	SA350 LF3	48 1.625"	9.62	83.25	64	190.5	redwood	3.1	both	71-9202
Hi-Star 100	244,000	ferritic steel	N.A.	13.6	95.9	68.75	202.9	?	23.4	N.A.	71-9261*
C-E Dry Cap	224,000	steel	N.A.	12.7	90.0	64.6	196.9	none	N.A.	N.A.	-
TN-12	144,800	ferritic steel	40 1.65"	15.9	78.74	33.2	210	wood	120	Elastomer	-
Castor-V/21**	234,000	NCI	N.A.	15.0	93.9	60.1	192.4	none	28	metal	72-1000
Generic	**224,000**	**stainless steel**	**24 1.75"**	**10**	**85**	**65**	**190**	**yes**	**24**	**Elastomer**	**-**

* Certificate pending
** These casks are only licensed for storage in the United States but are used for transportation in other countries.

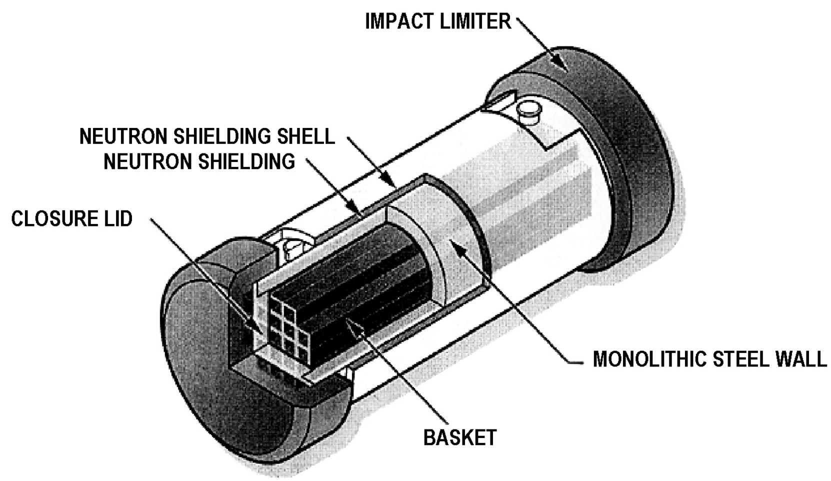

Figure 5.4. Monolithic Steel Rail Cask Configuration.

For U. S. transport of plutonium, the DOE Transportation Safeguards Division (TSD) sponsored an evaluation at Sandia National Laboratories of the risks associated with the transportation of special nuclear materials. Individual models were developed for nine container systems that might be used for plutonium transport from the year 2000 to the year 2004.[5-5] Container response characteristics are specified for accidents involving heating, impact, crush, puncture, and fire.

A new Type B nuclear fuel shipment container for the Monju reactor was developed and described in an article in *Genshiryoku Kogyo*.[5-6] This article lists design characteristics of the container, transportation methods, and potential issues.

At the end of 1994, COGEMA decided to analyze the feasibility of a multicontent MOX packaging, which was described in an article in *RAMTRANS*.[5-7] This article describes the background, testing, and performance assessment of an optimized design for a new packaging for transport of MOX fuel from COGEMA sites, including the associated interfacing equipment at COGEMA and partner sites. The resultant Type B packaging, named the FS 65, was developed in 20 months.

A 1986 article summarizes the COGEMA experience gained in transport of spent fuel to the La Hague reprocessing facility by road, rail, and sea.[5-8] Because this MOX transport experience covered 15 years of operation, COGEMA is willing to share its experience in order to aid other countries or utilities.

Another article about this experience, which appeared in *RAMTRANS* in 2000, describes the COGEMA and COGEMA Logistics (the former Transnucléaire) experience with MOX transportation and the FS 65 and TN 17 packagings.[5-9] The safety and security standards used with these packages are in accordance with the International Atomic Energy Agency (IAEA) document INFRIC 225, Revision 3, in addition to national regulations and guidelines.

A conference paper on transport of irradiated nuclear fuel from an industry standpoint by J. Charlton describes BNFL's experience during more than 30 years transporting MOX.[5-10] The regulatory framework, packaging regulations, physical protection standards, safety, and public acceptance are discussed.

5.3 References

5-1 J. L. Sprung, D. J. Ammerman, N. L. Breivik, R. J. Dukart, F. L. Kanipe, J. A. Koski, G. S. Mills, K. S. Neuhauser, H. D. Radloff, R. F. Weiner, and H. R. Yoshimura, *Reexamination of Spent Fuel Shipment Risk Estimates* (generally known as the Reexamination Study), NUREG/CR-6672, Vol. 1, SAND2000/0234. Sandia National Laboratories, Albuquerque, New Mexico, March 2000.

5-2 JAI Corporation, *Shipping and Storage Cask Data for Commercial Spent Nuclear Fuel*, JAI-421, Fairfax, Virginia, July 1996.

5-3 S. B. Ludwig, D. E. Welch, R. E. Best, and S. P. Schmid, *Transportation and Packaging Issues Involving the Disposition of Surplus Plutonium as MOX Fuel in Commercial LWRs*, ORNL/TM-13427. Oak Ridge National Laboratory, Oak Ridge, Tennessee, August 1997.

5-4 Battelle Pacific Northwest Laboratory, *Technology for Commercial Radioactive Waste Management*, DOE/ET-0028 (Vol. 4 of 5); Chapter 6, "Transportation of Radioactive Fuel Cycle Wastes." Richland, Washington, May 1979.

5-5 Marvin E. Larson and John M. Clauss, *Container Characteristics, Environments, and Responses in Support of DPTRA '99,'* SAND 2000-1045. Sandia National Laboratories, Albuquerque, New Mexico, May 2000.

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5-10 J. Charlton, "Transport of Irradiated Nuclear Fuel: An Industry Perspective," Proceedings International Conference on the Management of Irradiated Nuclear Fuel, IMechE, November 1992.

6. United States, IAEA, and Japanese Regulations Applicable to Type B Casks

6.1 United States Nuclear Transportation Regulations

The Nuclear Regulatory Commission (NRC) regulates the transportation of radioactive byproduct, source, and special nuclear materials within the United States. The Department of Transportation (DOT) regulates all radioactive materials in interstate commerce. International transport regulations are consistent with the standards of the International Atomic Energy Agency (IAEA), with the DOT serving as the "competent authority" within the United States.[6-1] NRC and the affected state agencies control United States shipments that are neither in interstate nor foreign commerce nor in air transport.

Chapter I of Title 10 of the Code of Federal Regulations contains the rules and regulations of the NRC. The parts most applicable to radioactive material transport are Parts 20, 70, 71, and 73, which present the following rules/regulations:

Reference	Rule/Regulation
10 CFR 20	Standards for Protection Against Radiation
10 CFR 70	Special Nuclear Material
10 CFR 71	Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions
10 CFR 73	Physical Protection of Plants and Materials

The DOT has regulatory responsibility for safety in transportation, and DOT regulations governing truck and rail transport of nuclear materials in the United States are covered in 49 CFR 171–179. DOT regulations governing packaging of radioactive materials are in Part 173 (49 CFR 173) and Part 178 (49 CFR 178). These regulations are consistent with the guidelines of Part 71 (10 CFR 71). The DOT regulations governing air and ship or barge transport are in Parts 175 and 176 (49 CFR 175 and 49 CFR 176).

NUREG-0170 is an environmental statement about the potential risks of nuclear material transportation. Although this statement originally set out to evaluate the safety of radioactive material transport by air, other modes of transport were examined, and the relationship between package designs and radiation exposures of humans under conditions of both normal transport and transport involving an accident was also examined.

This NUREG-0170 environmental statement was started in May 1975 and was completed before President Carter's April 7, 1977, message on nuclear power policy regarding deferral of commercial reprocessing and plutonium recycling.

The NUREG-0170 risk analysis is associated with nuclear materials transportation for 25 different radioactive materials including spent fuel, by plane, truck, train, ship, and barge.[6-1] The estimate of radiation doses and latent cancer fatalities that might be associated were made using Version I of the RADTRAN Code, which was written for that study.[6-2]

The validity of the risk assessment in NUREG-0170 was "seriously challenged" by staff members of the NRC working on that statement.[6-1] The challenge was that the assessment was excessively conservative and showed the risk to be greater than a more realistic assessment would show. A recent examination of the risks of nuclear transportation performed in 2000 by SNL and sponsored by the NRC showed that these challenges to the 1977 NUREG-0170 study were well founded.[6-3] In fact, the original assessment was found to be quite conservative in that it calculated greater risk than is realistically expected.

6.2 International Atomic Energy Agency Safety Standards

The IAEA first published Safety Series No. 6 in 1961 for application to the national and international transport of radioactive materials by all modes. Subsequent reviews were made over the years, the latest being the 1996 edition (Revised) or ST-1, Revised.[6-4] The IAEA previously published two companion documents to Safety Series No. 6, which are Safety Series No. 7 and Safety Series No. 37. No. 7 provides explanatory information on the intent and rationale of the regulations. No. 37 provides advisory information about the technical requirements of the regulations and about methods and technology that may be used to meet the regulations.

6.3 Transport Regulations in Japan

An article on safe transport in Japan outlines Japan's safety regulation system for the transport of radioactive materials.[6-5] The article includes procedures for application and approval for transport as well as emergency preparedness responsibilities.

Japan's safety standards for transport of radioactive materials are now regulated by the Ministry of Education, Culture, Sports, Science and Technology (MEXT); the Ministry of Economy, Trade and Industry (METI); and the Ministry of Land, Infrastructure and Transport (MLIT).

6.4 References

6-1 *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes*, NUREG-0170, Vol. 1. U. S. Nuclear Regulatory Commission, Washington, D. C., December 1977.

6-2 J. M. Taylor and S. L. Daniel, "RADTRAN: A Computer Code to Analyze Transportation of Radioactive Material," SAND76-0243. Sandia National Laboratories, Albuquerque, New Mexico, April 1977.

6-3 J. L. Sprung, D. J. Ammerman, N. L. Breivik, R. J. Dukart, F. L. Kanipe, J. A. Koski, G. S. Mills, K. S. Neuhauser, H. D. Radloff, R. F. Weiner, and H. R. Yoshimura, *Reexamination of Spent Fuel Shipment Risk Estimates* (generally known as the Reexamination Study), NUREG/CR-6672, Vol. 1, SAND2000/0234. Sandia National Laboratories, Albuquerque, New Mexico, March 2000.

6-4 IAEA Safety Standards Series, "Regulations for the Safe Transport of Radioactive Material," 1996 Edition (Revised), No. TS-R-1 (ST-1, Revised), International Atomic Energy Agency, Vienna, Austria, 1996.

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7. General Transportation Route Selection

7.1 United States Route Selection

According to DOE guidelines, nuclear material shipment in the United States must comply with requirements of both the U. S. Nuclear Regulatory Commission (NRC) and the U. S. Department of Transportation (DOT). While the NRC regulations specify the packaging and transport of spent fuel and nuclear waste, the DOT specifically regulates carriers, routing, handling and storage, and driver requirements.[7-1]

Highway routes for nuclear material are established in compliance with DOT regulations 49 CFR 171-179, as discussed in Chapter 6, and 49 CFR 397 for commercial transport.[7-2] For DOE's Transportation Safeguards Division (TSD) shipments, specific routes cannot be made public in advance because of national security.[7-1]

DOT routing regulations require that a "highway controlled quantity" of radioactive material be shipped over a preferred highway network. The DOT prefers routing on interstate highways, using bypasses and beltways around cities. A state or tribe may designate a preferred route in accord with DOT guidelines.[7-3]

The primary criterion for selecting a preferred route is travel time. Preferred routes are selected based on accident rates, transit time, population density, activities, road conditions, time of day, and day of week.

A computerized atlas, the HIGHWAY Code, is used for selecting highway routes in the United States.[7-4] This atlas includes all 240,000 miles of interstate highway and most of the main state, city, and local roads. INTERLINE is the corresponding railway code. TRAGIS is under development at ORNL to succeed both HIGHWAY and INTERLINE.

The HIGHWAY Code allows the users to select routes that conform to DOT regulations. HIGHWAY also includes population densities along routes. The distances and populations obtained from the HIGHWAY Code are then used for risk analyses and EISs.

Earlier work by SNL, in support of NRC risk evaluations, was performed to evaluate the risk associated with transportation of radioactive materials through densely populated urban areas.[7-5] This work evaluated the effects of urban routing for 20 large cities in the United States.

Similar route selection criteria using INTERLINE are used for rail transport of radioactive materials.

7.2 Security Aspects

In the United States, certain shipments are transported in Safe Secure Trailers (SSTs) for security reasons. Specific routes cannot be publicly identified in advance for these shipments of the DOE

TSD for reasons of national security. The state of Nevada has published issues that they have studied concerning nuclear waste terrorism and sabotage.[7-6]

7.3 German Security Operations and Security Vehicle

The authors in *Transportation of MOX Fuel* briefly describe the security operations in Germany for MOX fuel assemblies in transport.[7-7] These security operations include stringent technical and administrative security measures that are imposed during MOX transport. A security vehicle used by Nuclear Cargo and Service GmbH of Hanau, Germany, is described in a 1992 PATRAM article.[7-7]

7.4 References

7-1 U. S. Department of Energy, "Draft Environmental Impact Statement for the Production of Tritium in a Commercial Light Water Reactor," Appendix E: Evaluation of Human Health Effects for Overland Transportation, DOE/EIS - 0288D, Volume 1, Washington, D. C., August 1998.

7-2 DOT (U. S. Department of Transportation), "Shippers-General Requirements for Shipments and Packagings," 49 CFR 173. Washington, D.C., October 1997.

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7-4 P. E. Johnson, S. D. Joy, D. B. Clarke, J. M. Jacobi, *HIGHWAY 3.1, An Enhanced Transportation Routing Model: Program Description, Methodology, and Revised User's Manual*, ORNL/TM-12124. Oak Ridge National Laboratory, Oak Ridge, Tennessee, March 1993.

7-5 S. F. Herreid, N. C. Finley, and L. T. Ritchie, *A Method for Determining Radioactive Material Shipment Patterns in Urban Areas*, NUREG/CR-1117; SAND79-2071. Raytheon Service Company, Sandia National Laboratories, Albuquerque, New Mexico, May 1981.

7-6 R. J. Halstead, J. D. Ballard, F. Dilger, *Nuclear Waste Transportation Terrorism and Sabotage: Critical Issues*, Proceedings of PATRAM 2001, September 3-7, 2001, Chicago, Illinois.

7-7 R. Christ, H. Geiger, and J. Jahn, *Transportation of MOX Fuel*, TopFuel '97, Vol. 1, Proceedings, BNES Conference, Manchester, United Kingdom, June 9-11, 1997, ISBN: 07277 26196, pages 3.78-3.79, Nuclear Cargo and Service GmbH, Hanau, Germany, June 1997.

7-8 R. Christ, I. Fay, and J. Jahn, *Implementation and Operation of a Vehicle Location System*, PATRAM '92, The 10th International Symposium on the Packaging and Transportation of Radioactive Materials, Yokohama, Japan, September 13-18, 1992, INIS-JP-018, pages 1639-1640, Nuclear Cargo and Service GmbH, Hanau, Germany, 1993.

8. Methodology for Calculating Transportation Risks

8.1 Introduction

The initial step in determining a risk assessment is to select the route by using a code such as the HIGHWAY Code in the United States or similar codes in other countries. The selected route would consider environmental issues and require approval by or notification to the appropriate regulatory bodies. In the United States this approval or notification would be the obtained from the U. S. Nuclear Regulatory Commission (NRC), the U. S. Department of Transportation (DOT), and the affected states, municipalities, and tribes.

The first analytical step in ground transportation analysis is to assess the incident-free transport doses on a per-shipment basis. Incident-free doses are calculated for crew and public exposure to radiation from the shipping containers as well as the nonradioactive transportation vehicle exhaust exposure. Risk is the product of the probability of exposure and the magnitude of that exposure. Accident risk factors are next calculated for radiological and non-radiological traffic accidents. Figure 8.1 summarizes the transportation risk assessment methodology based on that used in a recent environmental impact statement.[8-1]

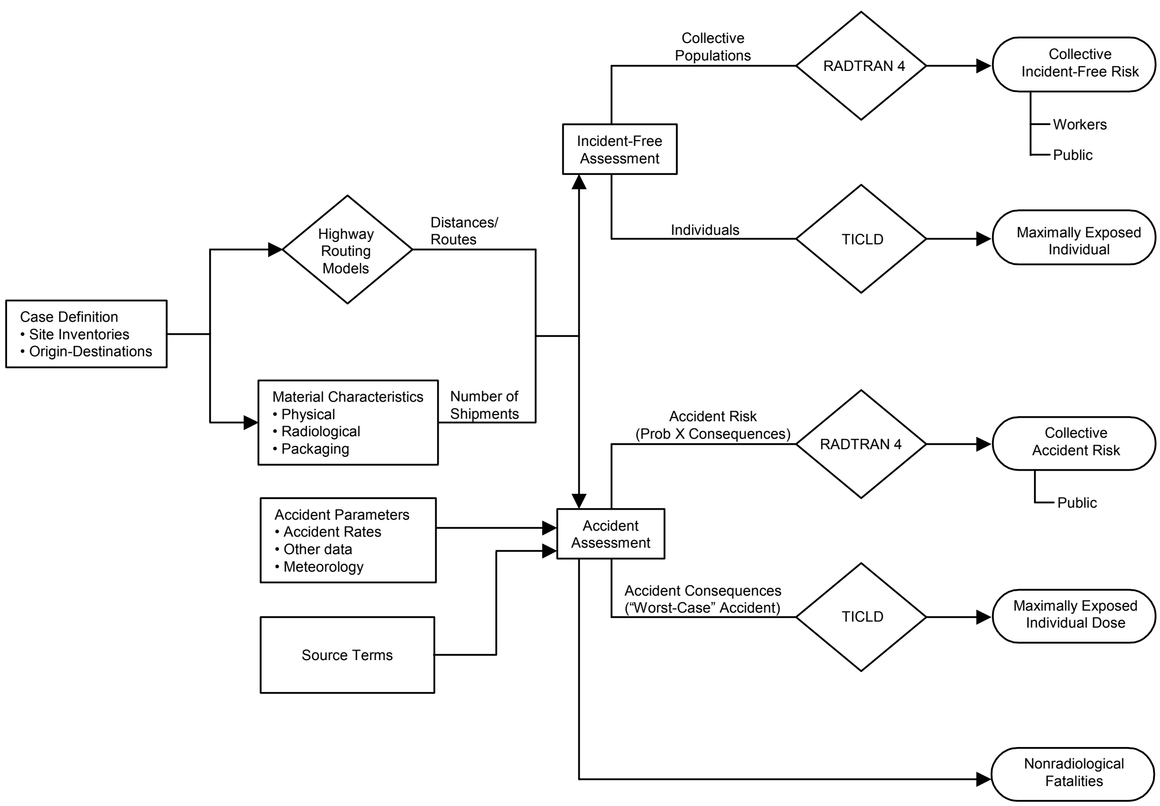

Figure 8.1. Overland Transportation Risk Assessment.

A major part of the work involved in risk analysis is the analysis required to develop a representative set of accidents and accident conditions. The thermal and structural response of the casks analyzed must then be determined for the predicted accident conditions.

Source terms are evaluated based on the cask inventory, the fraction of the inventory released by a given accident scenario (the release fraction), and the fraction of all accidents that lead to this release (the severity fraction).

Figure 8.2 describes in more detail the methodology involved in evaluating source terms and their role in risk assessment determinations.

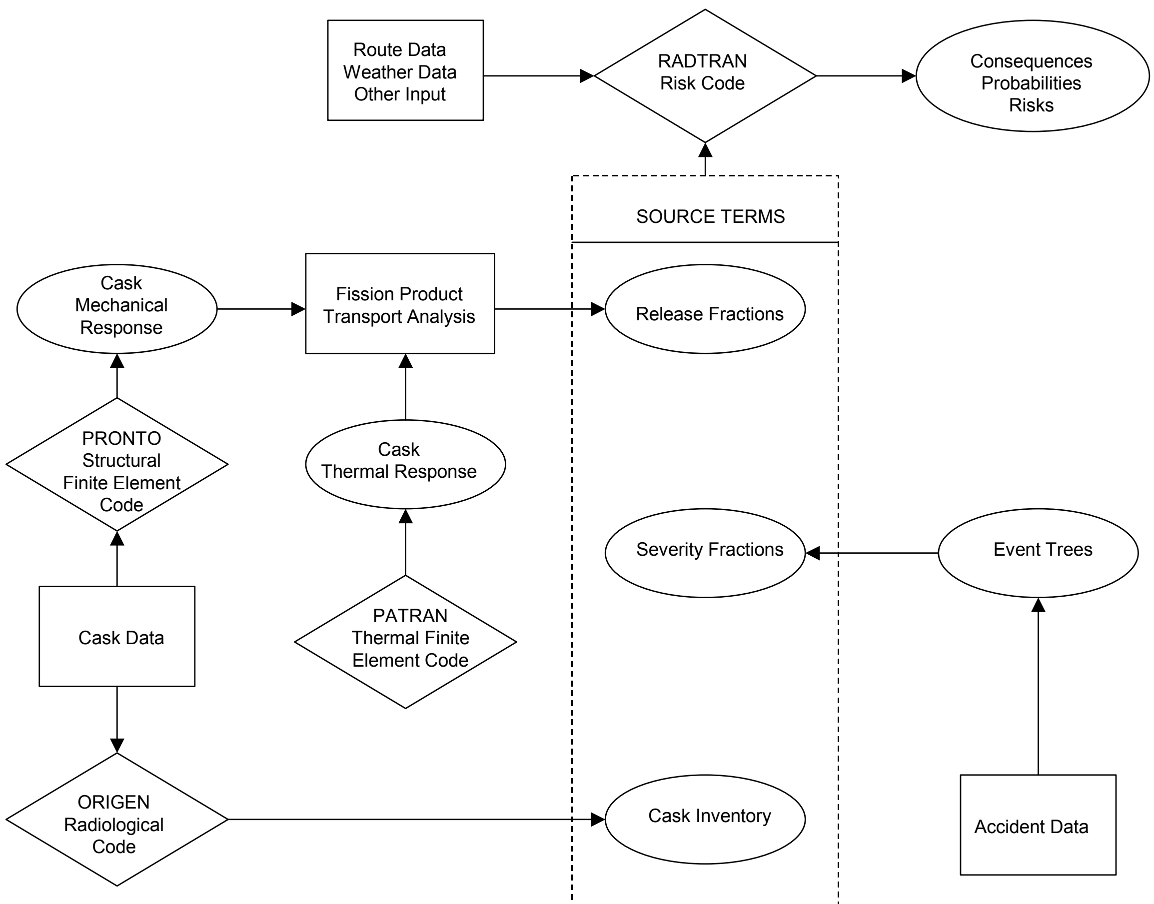

Figure 8.2. Source Term Methodology.

Radiological doses for a single shipment are expressed in units of roentgen equivalent man (rem). These doses are then multiplied by dose conversion factors and by the estimated numbers of shipments to estimate the number of latent cancer fatalities[8-2] that would result from incident-free transport or transport in which accidents might occur.

Multiplication of these estimated numbers of latent cancer fatalities by their probability of occurrence per shipment (numerically equal to 1 for incident-free transport, <<<1 for transport accidents) yields estimates of the risks associated with the transport of the radioactive material.

Risks are assessed for both incident-free transport and potential accidents. The incident-free risks are calculated both for populations of potentially exposed people and for potential maximally exposed individuals. The accident value has two components:

1. An accident probability risk assessment, which takes into account the likelihood of a range of possible accidents, including low-probability accidents that have high consequences and high-probability accidents that have lower consequences.

2. Accident consequence assessments that consider only the more severe transportation accidents postulated, those that might lead to the release of radioactive materials from the damaged transport package.

8.2 RADTRAN

The RADTRAN 5 computer code is a state-of-the-art code used to calculate both incident-free and accident doses and risks for collective populations exposed to radiation as a result of the transport of radioactive material.[8-3, 8-4] RADTRAN was developed by SNL to calculate population risks from radioactive material transportation by truck, rail, air, ship, or barge. The Transportation Incident Center Line Dose (TICLD) Code is operated with RADTRAN to calculate doses to maximally exposed individuals. A previous version, RADTRAN 4, was used extensively until recently.[8-5]

An earlier study at SNL provided the basic concepts to initially bound or quantify the accident environment that a large nuclear materials transportation package might be subjected to as a result of transportation accidents.[8-6] The following accident environments were studied:

- Impact
- Crush
- Fire
- Puncture
- Immersion

Although many studies have been conducted since this basic early work was completed, the methodology used in this study is sound and provides insight into risk and probability evaluation methodology.

RADTRAN 5 is the most recent version of the RADTRAN code now available. It is being revised at the present time, and a Version 6 will soon be available. RADTRAN is the basis for the International Atomic Energy Agency (IAEA) INTERTRAN code, and RADTRAN was the code used for the recent NRC NUREG/CR-6672 study[8-7] to reexamine the earlier NUREG-0170 risk environmental study.

The methodology employed in RADTRAN is widely accepted. Code changes are verified by a quality assurance plan consistent with the American National Standards Institute (ANSI) recommendations. Validations of RADTRAN 4 have been published[8-8, 8-9] and, because of the very slight difference in the two models (RADTRAN 4 and RADTRAN 5), the Version 4 validation is also applicable to Version 5.

RADTRAN calculates consequences and risks for a specific radioactive material in a specific packaging transported along a specific route. The code examines both incident-free transport and a postulated set of accident conditions.

For an accident postulated to be so severe that a release of radioactive material would result, RADTRAN estimates the following:[8-7]

- Doses that could be received by populations located downwind of the postulated accident site,
- The probability of the assumed accidental release during transport along the specified routes, and
- The radiological risks that would be incurred by the release (the product of the radiological consequence and the probability of the release).

RADTRAN also calculates radiation doses associated with accidents in which a packaging would lose shielding without releasing radioactive contents.

8.3 RADTRAN Input

The RADTRAN Code requires values for a large number of input parameters, and the RADTRAN User's Guide provides guidelines for assigning values to these parameters.[8-4]

The following parameters need to be specified by the user of RADTRAN, and because they vary greatly, depending on the individual shipments, representative values were estimated for the sake of general observations. Exact parameters would, of course, be substituted for analyses of individual shipments over specific routes. The main parameters affecting consequences and risk are as follows:

1. Route lengths,
2. Breakdown of route length into urban, suburban, and rural fractions,
3. Population densities and accident rates on each segment of the route,
4. Population of drivers and passengers in other vehicles along each segment of the route,
5. Number of stops, locations, and duration of stops along the route,
6. Weather predictions along the segment of the route for the time of the shipment,
7. Surface dose rate of the packaging,
8. Amount of each radionuclide contained in the packaging (the package inventory),
9. The fractions of those radionuclides that might be released to the atmosphere for each accident scenario considered,
10. The probability of the release, and
11. The time required to evacuate nearby populations in the event of an accident release.

8.3.1 Route Parameters

The Reexamination Study considered a number of hypothetical routes, interim storage sites, route fractions, population densities along the routes, accident rates, and various other input values in order to provide a meaningful representation of route parameters for the purpose of evaluating the risk of spent fuel transportation in the United States.[8-7] For that study, the HIGHWAY and INTERLINE Codes were used to define the characteristics of 492 routes for spent fuel transport by truck or rail. In addition, INTERLINE and HIGHWAY calculations had been performed for 249 different routes in a previous spent fuel transportation risk study.[8-10] Thus, a total of 741 different truck and 741 different rail route parameters were used by the Reexamination Study to evaluate the general risk of transportation of spent fuel. A specific study for the MOX shipments in Japan would use the appropriate route parameters for each specific shipment, mode, and route.

In the absence of specific real truck or rail routes, analytical representations of representative route parameters for hypothetical spent fuel or other radioactive material shipments can be generated by sampling existing distributions using structured Monte Carlo sampling (Latin Hypercube Sampling) methods.[8-11]

Accident rates for route segments are not calculated by HIGHWAY or INTERLINE. Accident distributions have to be developed in separate calculations for both truck and rail accidents.

8.3.2 Weather Parameters

Prevailing weather conditions, most importantly atmospheric turbulence and wind speed, are important factors that are required to evaluate radioactive plume dispersion following a transport accident release and any possible radiation exposures to populations. RADTRAN develops accident consequences for six sets of prevailing weather conditions that correspond to the six Pasquill-Gifford atmospheric stability classes[8-12] using national average frequencies of occurrence for each of the stability classes. RADTRAN assumes that all wind directions are equally probable and uses a uniform population density for each route segment determined by the population density distributions developed by the HIGHWAY or INTERLINE Codes.

8.3.3 Package Inventories and Surface Dose Rates

Package inventories are calculated using the ORIGEN Code.[8-13] The surface dose rate of a package can be calculated from the package inventory and specific packaging design.

8.3.4 Accident Source Terms

Source terms are calculated based on the quantities of radionuclides available for release that are calculated by ORIGEN, the number of rods that fail, and the fraction of rod inventory released on failure. This number is reduced by the amount of release deposited on the interior walls and surfaces of the packaging. The fraction of gas-borne radionuclides that are transported out of the cask and that are therefore available for atmospheric release is determined by the fraction of cask gases that escape from the cask after pressurization from blowdown of failed rods and heating of the cask gases as a result of exposure to fire accident conditions. The term “severity fraction” is used to describe the conditional probabilities (conditional on the occurrence of an accident) of accident source terms estimated from the probabilities of accident speeds, cask impact orientations, impact surface hardness, fire probability, and fire durations.[8-7]

8.3.5 Source Term Probabilities

The probability of occurrence of a specific accident source term during shipment along a specified route is the product of the accident occurrence chance during shipment and the fraction of all possible accidents that yield source terms similar to that source term. Event trees that depict a representative set of possible truck and train accidents are developed. Each possible accident scenario on the event tree is then multiplied by the chance that the accident falls into specified speed ranges and by the chance that the scenario involves a fire that heats the cask to specified temperature ranges. Modal Study data is used to provide the probabilities of speeds falling within given ranges and of fires heating the cask to a given temperature.[8-14]

This may be expressed as

$$P_{ST} = L_{route}\, P_{accident\ scenario}\, P_{speed\ range}\, P_{fire}\, P_{fire\ temperature\ range}$$

where

P_{ST}	=	probability of a specific accident source term
L_{route}	=	route length
$P_{accident\ scenario}$	=	probability of the specific accident
$P_{speed\ range}$	=	probability the accident speed falls within a given range
P_{fire}	=	probability of fire occurrence
$P_{fire\ temperature\ range}$	=	probability of the fire heating the cask to a given temperature range.

8.3.6 Source Term Magnitudes

The amount of radioactive material that might be released from a failed spent fuel cask as the result of a fire or collision accident is called the accident source term.[8-7] The source term is the product of four numbers:

1. The inventory of each radionuclide in the spent fuel cask,
2. The fraction of fuel rods that fail in the accident,
3. The fraction of the inventory of a single rod that is released to the cask interior, and
4. The fraction then released from the cask interior to the environment.

Precise cask inventories are calculated by the ORIGEN Code.[8-13] Release fractions from failed fuel rods to the cask for noble gases, compounds of cesium and ruthenium, and particulates, as well as CRUD have been studied and reviewed.[8-15, 8-16, 8-17, 8-18, 8-19]

The MELCOR Code[8-20, 8-21] is used to evaluate the transport of fission products through the interior of a spent fuel cask to the cask leak and out to the environment.

Thus MELCOR calculations reflect cask surface-to-volume ratios and the effect of leakage path cross sections and lengths on release from the cask to the environment. The Reexamination Study scaled MELCOR results using leak cross sectional areas to obtain results for a range of leak areas.

8.4 Response of Representative Casks to Accident Conditions

Because cask leak rates depend on possible cask leak-areas, evaluation of potential leak areas depends on the cask design and the specifics of the cask response to specific accident conditions. In the Reexamination Study, the response of the four generic casks described above was analyzed by finite element structural response calculations together with one-dimensional heat-transfer calculations.[8-7] Evaluation of cask puncture probabilities during collision was estimated by reviewing rail tank car accident data. For a specific shipping campaign, finite element structural calculations would be performed to predict and model impact damage (both failure of the cask closure seals and of spent fuel rods) for the individual cask being used. Heat transfer calculations would be performed to evaluate heating times and temperatures in engulfing fires required to cause cask seals to fail and also to cause rod failure by burst rupture.

8.4.1 Finite Element Impact Calculations

PRONTO 3D[8-22] is a three-dimensional, transient solid-dynamics code used to model the large deformations produced by impacts in serious accidents. It is especially useful for modeling the behavior of cask closures such as cask lids and bolt interfaces. PRONTO 3D results were validated by comparison to test results for a wide range of problems, to predictions of other codes, and to simple-geometry theoretical solutions. At SNL, the Structural Evaluation Test Unit (SETU) Program compared experimental and PRONTO 3D results for cask impacts of up to 60 mph.[8-23]

Cask seal leakage and the calculation of leak areas are modeled by examining radial and circumferential displacements of cask closures caused by impact. Results and calculations from the Reexamination Study suggest that rail casks may leak when impact speeds onto an unyielding surface are as low as 60 mph, and both truck and rail casks are likely to leak at accident speeds of 90 and 120 mph.[8-7]

For any impact speed and cask orientation, the damage resulting from impact onto an unyielding target would be greater than that caused by a corresponding impact onto a yielding target that would absorb a significant fraction of the impact energy. This difference is determined by evaluating the cask velocity time-history that is calculated from the kinetic energy-time history. Combining the calculated force-time history with the displacement-time history will then produce a force-deflection curve that can be used to evaluate the impact based on the unyielding target impact. The energy partitioning method that was used determined the increased energy that must be available for a yielding surface impact to cause the same damage as the unyielding surface impact. This increased energy is added to the energy of the unyielding surface impact. For more detail on energy partitioning methods, refer to Section 5.2 of the Reexamination Study.[8-7]

8.4.2 Rod Failure Fractions

The fractions of fuel rods that are failed by a cask impact in a particular orientation and at a particular impact velocity are estimated by predictions of finite element calculations of peak rigid-body accelerations. Rod-cladding strains were calculated for a typical boiling water reactor assembly and a pressurized-water reactor assembly for a 100-G side impact onto an unyielding surface.[8-24] The 100-G side impact strain map, obtained from finite element calculations, was scaled to higher accident conditions (impact speeds) using the calculated peak acceleration experienced by the cask as a scale factor. These scaled results were then compared to a 4% strain failure criterion for cladding failure in typical spent fuel rods. This 4% strain failure criterion was calculated in the Reexamination Study for a dependence of rod failure strain levels on burn-up and on the current and projected amounts of spent fuels of different burn-up levels in the U. S. inventory. Any rod with scaled strains that exceeded the 4% criterion was assumed to fail.

8.4.3 Thermal Calculations

Rod failure by burst rupture and time to fail in fire accidents were calculated in the Reexamination Study[8-7] by using the PATRAN/PThermal Code[8-25] that is available commercially.[8-26] The code can be used to for one- , two- , or three-dimensional analysis simulations to determine the heating rates of structures by conduction, convection, and thermal radiation. PATRAN/ PThermal, formerly called Q/TRAN, was validated by comparing its results to analytical solutions and to predictions of other widely used codes.[8-27, 8-28]

For the four generic casks, PATRAN/PThermal was used to determine the duration of a fully engulfing, optically dense fire that would heat the cask to the temperature at which spent fuel rods would fail by burst rupture.[8-7] The probability of fires of this duration was then used for accident severity fraction determination.

Data on temperatures that cause seals to leak are published in the literature[8-29] For Viton elastomer seals, the heating of a cask to 400°C is assumed to cause complete seal failure as a result of extensive thermal degradation. Also, bolt softening during cask heating from a long duration hot fire would be expected to eliminate seal compression.

8.5 RADTRAN Calculations

By use of the accident and input parameters discussed above, the Reexamination Study performed seven sets of RADTRAN calculations to evaluate the risk of truck and rail shipments of spent fuel and to compare these results to the earlier NUREG-0170 examination.[8-7] Outputs from these RADTRAN calculations were used to compare the severity of risks and the parametric effects of the input parameters. Similar analyses would be conducted for specific transportation campaigns for MOX spent fuel in Japan, using the input parameters appropriate for the transport conditions and the casks that would be used. RADTRAN calculation results will be discussed in Chapter 14.

8.6 Risk-Based Monitoring Program

In a recent paper presented at PATRAM '01, T. McSweeney suggested using a monitoring system based on key performance parameters for transportation risk assessments.[8-30] Measurements of these parameters would then be used as a monitoring instrument for actual transportation operations.

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9. Parameters and Assumptions

Certain assumptions can be used to formulate a descriptive risk basis for unspecified routes and casks, which can later be applied for an actual shipment. This was the approach adopted in the DOE Environmental Impact Statement (EIS) involved with tritium production.[9-1]

9.1 External Dose Rates

For general applications with unspecified selections and types of casks, the external dose rate is conservatively estimated by assuming regulatory limits, even though the actual external dose rate would be much smaller in most cases.

External radiation from a package must be below specified limits to minimize the exposure of handling personnel as well as the general public. This external dose limit, according to 49 CFR 173, must not exceed:

- 10 millirem per hour at any point 2 meters from the vertical planes projected by the outer lateral surfaces of the transport vehicle and
- 2 millirem per hour in any normally occupied position in the transport vehicle.

Additional restrictions apply to surface contamination levels.

9.2 Health Risk Conversion Factors

Health risk conversion factors from the radiological protection recommendations can be used to estimate expected cancer fatalities.[9-2] These values are 5×10^{-4} and 4×10^{-4} fatal cases of cancer per person-rem for members of the public and workers, respectively. Incidences of fatalities from cancer occur as latent fatalities in the exposed populations.

9.3 Accident Involvement Rates

To calculate nonspecific accident parameters, vehicle accident and fatality rates can be taken from the International Commission on Radiological Protection.[9-2] Accident rates are the number of accidents (or fatalities) in a given year per unit of travel in that same year. This rate is the accident-involvement count divided by total travel distance (vehicular activity), typically determined as an average for a multi-year period. In the DOE EIS for tritium production, the total number of expected accidents or fatalities are calculated by multiplying the total shipment distance for a specific case by the appropriate accident or fatality rate.[9-1]

For truck transportation, the accident rates given in data in the DOE EIS are based on an Argonne National Laboratory (ANL) review of interstate freight accidents for heavy trucks.[9-3] These heavy combination trucks combine a separable tractor unit with one to three freight trailers, which is the configuration typically used to transport radioactive materials. The accident rate values used in the Ref. 9-1 study were computed for shipments through particular states (United States) based on statistics compiled by the U. S. Department of Transportation Office of Motor Carriers for 1986 to 1988. A review of statistics from states on interstate truck freight accidents was used to provide accident involvement and fatality counts (to the public or attributable to the accident 30 days thereafter), estimated kilometers traveled in each state, and the average accident involvement, fatality, and injury rates for three years.[9-3]

Rail accidents are computed in a manner similar to truck accident rates, with the rail car being the unit of haulage.[9-3] The study cited in Ref. 9-1, used state-specific accident and fatality rates.[9-1] These rates are based on statistics compiled by the U. S. Federal Railroad Administration for 1985 through 1988. Rail accident rates include both main-line accidents and accidents in rail yards.

The statistics from the ANL study are for heavy interstate combination truck shipments independent of cargo.[9-3] Shippers and carriers of radioactive materials have an above-average awareness of the risk of transport, and they prepare both the cargoes and drivers accordingly.[9-3] This preparedness should reduce both equipment failure and the human error as contributors to accident rates. No credit was taken for the effects of this preparedness in the accident assessment of Ref. 9-1 or in the ANL accident rate study of Ref. 9-3.

9.4 Consequence Analysis

Much of the recent development in nuclear material risk assessments has been in improving estimates of the severe accident conditions that can lead to the release of radioactive materials. However, little new work has been performed on consequence analysis, even though this area is of importance in radioactive material transport. Because consequences are location-specific, considerable attention must be given to the details of consequences in the absence of specific information.[9-4]

For a source of radioactive material released to the environment, consequence-analysis methods should consider the following.[9-4]

- Dispersion and transport of radionuclides through air, water, and food chains;
- Pathway radioactive concentrations and decay modes;
- Populations at risk;
- Consequences of exposure, including the effects of contaminating land and food and the direct effects on humans;
- Economic impact from the release, including evacuation, relocation, cleanup, and mitigation costs; also economic impact resulting from releases in populated areas: disruption of daily activities, the impact of having to relocate, and reduced productivity;

- Security aspects and vulnerability to malevolent actions; and
- Emergency response and relocation costs.

Methods of release of radioactive materials to the environment and humans include the following principle pathways.[9-4]

- External radiation directly from casks or released material;
- Air dispersion by plumes yielding external radiation from cloud immersion and internal radiation from inhalation of radionuclides;
- Dispersion of radionuclides in surface and ground water contaminating irrigation and drinking water;
- Deposition of radionuclides from the air to the ground and structures yielding external radiation, and into plants and water yielding internal radiation from contaminated food sources and contaminated water; and
- Resuspension of deposited radionuclides yielding additional external exposures.

The Brookhaven study of risk assessment provides details for each of these principle pathways.[9-4]

The usual terminology for exposure pathways is as follows:

- Cloudshine – external radiation from the contaminated cloud,
- Inhalation – internal radiation due to inhalation of cloud materials,
- Groundshine – external radiation from radioactive materials deposited on surfaces such as ground or structures,
- Resuspension inhalation – internal radiation due to inhalation of materials resuspended from the ground and structures, and
- Ingestion – internal radiation from ingestion of contaminated food or water.

9.5 Cost Consequence

A study from 1980 addressed the costs associated with transportation accidents involving radioactive materials.[9-5]

Initial costs of accidents include the costs of emergency response and monitoring to assess the extent of radiation. These initial costs for emergency response would be related to the numbers of persons involved, and related to the severity of both the radiological and nonradiological effects of the accident.[9-4]

Subsequent or secondary costs would be related to the amount of radiation released and dispersion into the environment. Secondary costs include litigation, loss of economic value of contaminated land and property, loss of crops, loss of tourist trade, and loss of habitability. Associated economic loss resulting from public concern, apprehension, and litigation would also be expected.

A later study conducted at SNL provides an estimate of attributed costs from plutonium dispersal accidents.[9-6]

9.5.1 Disruption of National Power Production

National economic security is clearly a dominant factor for nuclear risk concerns. To date, nuclear risk analysis has predominately focused on public health issues arising from both accident and incident-free transportation dose effects. However, the economic security of a nation as a result of nuclear accidents causing any shutdown of electrical power production would be a severe financial impact for the nation as a whole.

9.5.2 Litigation

A major financial risk expected from accidental release of radiation as a result of a transportation accident is in potential litigation based on claims that cancers or adverse genetic effects were caused by exposure.[9-4] The short-term (acute) effects of an accident, without punitive damage awards, are relatively straight-forward assessments. This is clearly not the case for long-term effects resulting in birth defects and cancers.

9.5.3 Costs Resulting from Public Concern

Because an enormous amount of news coverage would result from any event involving radiation exposures, public concern is a major risk of nuclear transportation. Emotional stress and associated health effects from an accident, real or not, would be expected. Devaluation of land, products, and agricultural commodities, either directly or indirectly associated with contamination are possible. It would be difficult to quantify such costs other than by general bounding analyses.[9-4] The disruption of some economic activities in the contaminated region may result in increased economic activity in noncontaminated locations.

9.6 Security

The opportunity for malevolent actions from terrorist sources is a significant concern for the nuclear transportation community. Whether the terrorist intent is to acquire radioactive materials or to cause an accident that could lead to the release of radioactive materials and exposure of people to radiation, security during transport is important. The impact on nuclear activities would be great, even if a limited terrorist action were carried out. Public concern would be aroused, and the costs would be enormous: improved security would be costly as would possible severe limits on transporting radioactive materials.

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10. Container Accident Response Characteristics and Release Fractions

10.1 Response of Cask to Impact and Thermal Loads

As previously discussed, the Modal Study was the result of an initiative taken by the U. S. Nuclear Regulatory Commission (NRC) to refine the analysis of NUREG-0170[2] for spent nuclear fuel shipping casks.[10-1, 10-2] While the NUREG-0170 analysis used best engineering judgments and presumptions about cask accident response, the Modal Study used sophisticated structural and thermal engineering analysis and a probabilistic assessment of the conditions that could be expected in severe transportation accidents.[10-3] The Modal Study based its analyses on casks that comply with national codes and standards with design parameters that meet minimum test criteria of 10 CFR 71.[10-4] Therefore, the Modal Study is considered to provide realistic, however conservative, results for accidents.[10-3]

In the Modal Study, a matrix was developed to categorize damage to a cask according to the combined mechanical forces of impact and the thermal forces from fire for accidents of varying levels of severity. In this consideration, severity is independent of the specific accident sequence. This accident severity matrix is designed to account for all potential transportation accidents including accidents with low probability and high consequences and those with high probability and low consequences.

Each severity category in the matrix represents a set of accidents for a combination of mechanical and thermal forces. A conditional probability of occurrence (the probability that if an accident occurs and is of a particular severity) is assigned to each category.

This range of combined structural and thermal responses for a cask is represented by the response matrix shown in Figure 10.1 (Modal Study, Figure 4-5).[10-1] The ordinate of this response matrix represents the structural response in terms of maximum strain on the inner shell in percentage, the abscissa is the thermal response in terms of the mid-thickness temperature of the lead shielding layer in degrees Fahrenheit.

Thus, there are 20 response regions denoted by $R(S_i, T_j)$ (shaded in Figure 10.1): S_i is the structural response level and T_j is the thermal response level. The first region, R(1,1) is for cask response to combined mechanical and thermal loads within 0.2% strain and 500°F temperatures. Radioactive releases in R(1,1), if they occur, would be within regulatory limits. The region R(4,5) represents the most extreme combined loads possible.

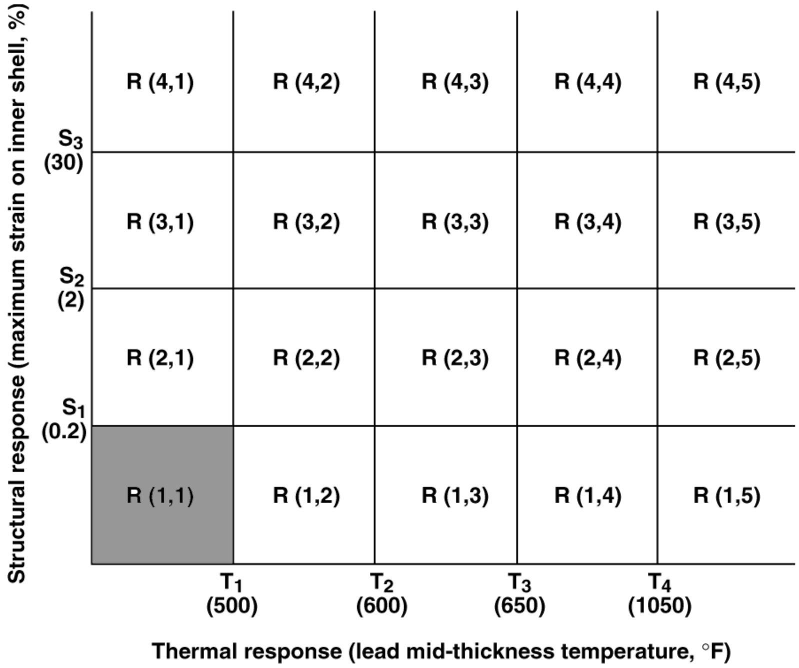
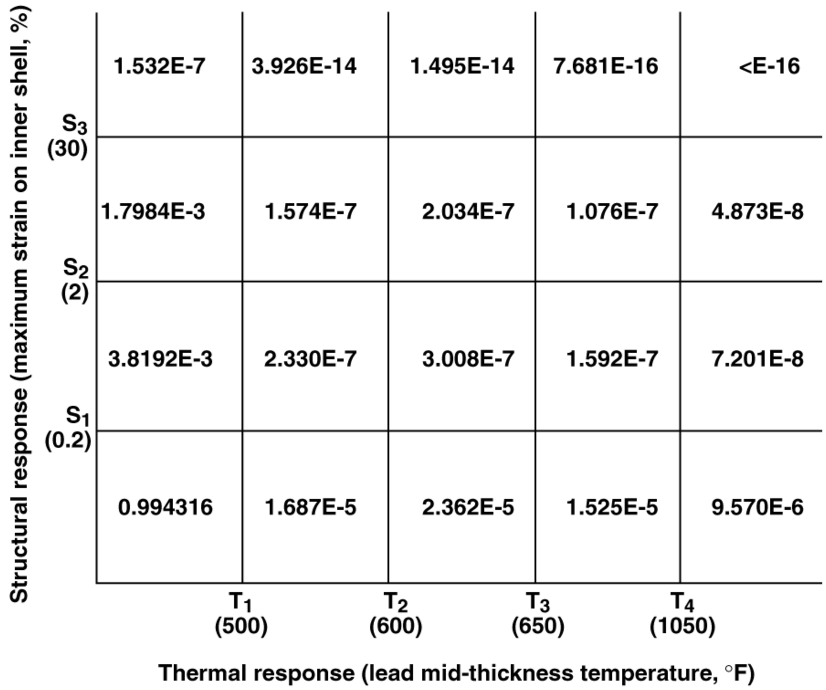


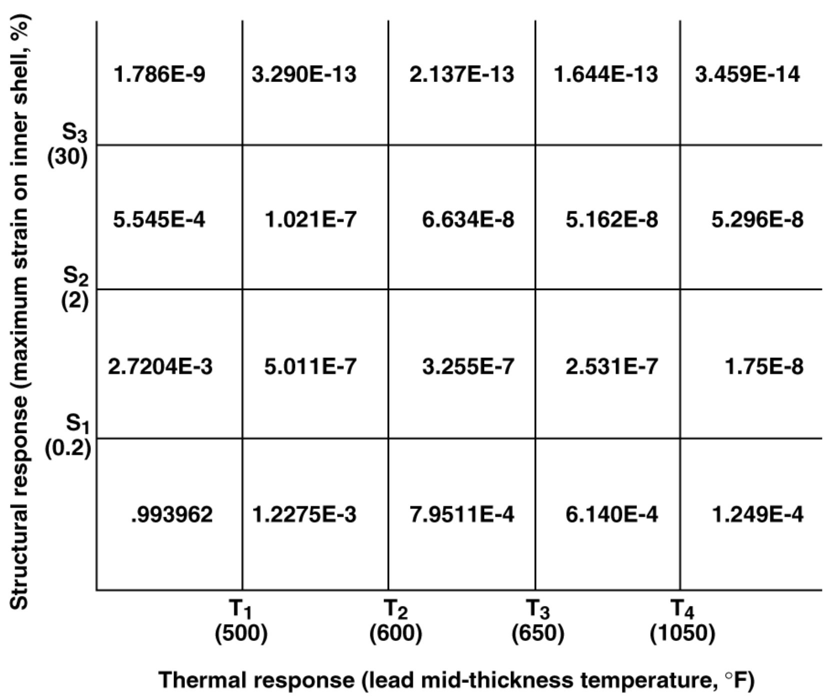
Figure 10.1. Matrix of Cask Response Regions for Combined Mechanical and Thermal Loads.

The probability of occurrence of each region decreases with increasing severity of the loads. After lengthy and detailed analyses, the Modal Study[10-1] expressed these probabilities as fractions of accidents that could result in response levels designated by the response matrix, assuming that an accident occurs. Figure 10.2 (Modal Study, Figure 7-10) shows this matrix for truck accidents.[10-1] The Transportation Accident Scenario Probabilities Code (TASP) was used to determine these probabilities.[10-1] Figure 10.3 (Modal Study, Figure 7-11) shows this matrix for rail accidents.[10-1]



Note: E + x = 10^x

Figure 10.2. Fraction of Truck Accidents that Could Result in Responses Within Each Response Region, Assuming an Accident Occurs.



Note: E + x = 10^x

Figure 10.3. Fraction of Rail Accidents that Could Result in Responses Within Each Response Region, Assuming an Accident Occurs.

To determine a predicted frequency of an accident of a given severity for shipment along a route of given length, the conditional probability in each category is multiplied by the baseline accident rate and the route length. This entire spectrum of accident scenarios is then considered for accident risk assessments.

10.2 Response of Elastomer and Metallic Seals

The regulatory Design Basis Accident (DBA) defined by 10 CFR 71[10-4] and 49 CFR 173[10-5] is characterized as bound by a maximum impact load response of S (0.2% maximum strain on the inner shell), and a maximum thermal load of T_1 (260°C lead shield mid-thickness temperature).

For truck or rail casks with elastomer seals, failure is not assumed for impact loads and temperatures less than these DBA conditions.[10-3]

Because radioactive materials packages are designed with large margins of safety, these packages would be capable of withstanding accident conditions substantially more severe than the DBA. Recent tests and analyses at SNL using packages with elastomer seals have shown that the cask containment boundary does not fail for accidents even for inner shell strains of up to 20%.[10-6] A conclusion is that cask containment boundaries will not fail for packages using elastomer seals for inner shell strains less than S_1 or 20%.[10-3]

Because a packaging that has metallic seals will lose its sealing function as a result of slight amounts of closure movement typical of extra-regulatory impacts, it has been previously assumed that any impact load above S_1 will result in cask containment boundary failure for casks using metallic seals.[10-3] However, the probability of failure for a metallic seal below T_4 (565°C) is negligible. The American Society for Testing and Materials (ASTM) Type 304 stainless-steel materials used in radioactive materials containers are used in industrial applications routinely at operating temperatures up to 565°C. ASTM Type 304 stainless steel is rated by the American Society of Mechanical Engineers Code, Section III at 122 Mpa (17.7 ksi) for a 10-hour exposure to temperatures of 565°C.[10-3]

Bolts used for seal closures must be carefully chosen. Inconel bolts are rated as high as 620°C and are to be used in place of high-strength carbon steel bolts rated to temperature of only 370°C for most uses.[10-3]

10.3 Rebrazable Seal Method

A recent innovation at SNL provides a rebrazable metallic seal that eliminates the low temperature limit of elastomer seals and the impact limit for typical metallic seals.[10-7, 10-8, 10-9] Because this seal forms a brazed joint with the cask wall material, it provides additional usage in matrix response regimes above those for either elastomer or metallic seals. Tests have shown brazing and rebrazing operations can be carried out up to 20 repetitions.

10.4 References

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10-3 DOT (U. S. Department of Transportation), "Shippers-General Requirements for Shipments and Packagings," 49 CFR 173. Washington, D.C., October 1997.

10-4 10 CFR 71, *Packaging and Transportation of Radioactive Material*, Code of Federal Regulations, January 1, 1987. 10 CFR Part 71, U. S. Nuclear Regulatory Commission, Washington, D. C., 1987.

10-5 49 CFR Part 173 *Shippers - General Requirements for Shipments and Packaging*, 11th Code of Federal Regulations, 49 CFR Part 173, October 1, 1997. U. S. Department of Transportation, Washington, D. C.

10-6 D. J. Ammerman and J. G. Bobbe, *Testing of the Structural Evaluation Test Unit*, Proceedings of PATRAM '95, Las Vegas, Nevada, Sandia National Laboratories, Albuquerque, New Mexico, December 1995.

10-7 J. D. Pierce, J. J. Stephens, C. A. Walker, F. M. Hosking, R. M. Curlee. *Development of a Re-brazeable Containment System for Special Nuclear Material Storage and Transport*, SAND95-0206C, PATRAM '95 Conference (International Conference on Packaging and Transportation of Radioactive Materials, December 3-8, 1995, in Las Vegas, Nevada). Sandia National Laboratories, Albuquerque, New Mexico, December 1995.

10-8 J. J. Stephens, F. M. Hosking, J. D. Pierce, C. V. Robino, J. G. Bobbee, and C. A. Walker, *Materials-Related Advances in Re-brazeable SNM Packages*, SAND98-0966A. Fall Transportation Management System (TMS) Meeting, Rosemont, Illinois, October 12-15, 1998. Sandia National Laboratories, Albuquerque, New Mexico, August 1998.

10-9 J. D. Pierce, J. J. Stephens, and C. A. Walker, U. S. Patent 5,950,906, "Reversible Brazing Process," Sandia Corp., Albuquerque, New Mexico, September 14, 1999.

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11. Structural Response Example

11.1 Introduction

As reported in the Reexamination Study, finite element calculations were conducted to determine the responses of the generic casks discussed above for impacts onto rigid targets.[11-1] The method used was the SNL developed non-linear transient dynamics finite element program, PRONTO-3D.[11-2, 11-3, 11-4] This code, especially developed for impact analyses, uses a time-marching explicit integration of the equations of motion to determine structural response throughout the impact event. Inputs to the code are the cask geometry, material properties, initial velocities, and boundary conditions. This code updates the position of each material node at every time step, thus allowing for material and geometric nonlinearities. PRONTO has been benchmarked for analysis of cask responses.[11-5, 11-6]

For each of the four generic casks analyzed in the Reexamination Study, calculations were made for impacts in which the cask impacted in end-on, center-of-gravity over-corner, and side-on orientations. These orientations are considered to bound intermediate impact orientations.

To shorten analyses times and simplify calculations, the analysis reported in the Reexamination Study conservatively assumed that the cask impact limiters were pre-crushed to their lock-up regime. The amount of energy absorbed by the impact limiter before lock-up is equivalent to the kinetic energy from the regulatory drop test. Then, by using the pre-crushed impact limiters, analyses were conducted with impact velocities of 30, 60, 90, and 120 mph for each generic cask and orientation. Thus, if the energy required to crush the impact limiters is added to the initial kinetic energy of the casks, the actual impact velocities correspond to 42, 67, 95, and 124 mph.

11.2 Finite Element Results

In the Reexamination Study, finite element calculations were used to investigate areas that could cause a loss of cask containment.[11-1] The main factors are maximum tensile plastic strains in the containment boundary, maximum tensile plastic strains in the closure bolts, and deformation in the region of seals. For the sandwich-wall casks, the containment boundary is the inner shell, although the outer shell could remain intact and prevent containment loss.

None of the finite element calculations in the Reexamination study indicated strains above 70% for the inner wall of the sandwich-wall casks studied. True strain at failure for 3045 stainless steel is greater than 120%, so this would indicate that no tearing of the inner-wall would take place. Table 11.1 (Reexamination Study, Table 5.3) includes the finite-element calculations from the Reexamination Study for the sandwich-wall cask configurations.[11-1]

Table 11.1. Maximum Plastic Strain in the Inner Shell of the Sandwich Wall Casks

Cask	Corner Impact Speed	Strain (%)	End Impact Speed	Strain (%)	Side Impact Speed	Strain (%)
Steel-Lead-Steel Truck	30 mph	12	30 mph	3.9	30 mph	NA
	60 mph	29	60 mph	12	60 mph	16
	90 mph	33	90 mph	18	90 mph	24
	120 mph	47	120 mph	27	120 mph	27
Steel-DU-Steel Truck	30 mph	11	30 mph	1.8	30 mph	6
	60 mph	27	60 mph	4.8	60 mph	13
	90 mph	43	90 mph	8.3	90 mph	21
	120 mph	55	120 mph	13	120 mph	30
Steel-Lead-Steel-Rail	30 mph	21	30 mph	1.9	30 mph	5.9
	60 mph	34	60 mph	5.5	60 mph	11
	90 mph	58	90 mph	13	90 mph	15
	120 mph	70	120 mph	28	120 mph	NA

For the monolithic rail cask, the maximum strain on the interior surface of the cask is less than 60% for all impact configurations.[11-1] The maximum strain was calculated at the lid-cask interface, where the plasticity is caused by compression. This eliminates the possibility of material failure.

The finite element calculations of the Reexamination Study also evaluated the strain in the cask closure bolts.[11-1] The amount of deformation between the cask body and the lid at the seal location was calculated, and leak areas were determined for the rail cask types under differing orientations. Table 11.2 (Reexamination Study, Table 5.5) shows these results.

In the Reexamination Study the casks considered for cask closure bolt analysis had no gap between the lid and the lid well, and this geometry meant that any closure deformation forced the lid into the well, thus reducing bolt strain. However, because the cask impact limiter was assumed to be fully crushed to lockup by a 30-mph impact onto an unyielding surface in these analyses, the considerable design margin of the impact limiters was neglected. These two assumptions tended to largely cancel each other. For example, it was predicted that rail casks might leak if they hit an unyielding surface in a center-of-gravity over-corner orientation at 60 mph. However, more detailed finite element calculations, performed for a follow-on study, with the impact limiter in place and with a gap between the cask lid and well, have shown that failure does not occur below 80-mph impact velocities.

If we compare the Reexamination Study's calculation results for unyielding targets for the four generic casks with the calculation results from the Modal Study (Figures 10.1, 10.2, and 10.3 in this report or 4-5, 7-10, and 7-11 in the Modal Study),[11-7] we observe that the strain results that were obtained are all at levels above the R(1,j) regulatory impact threshold. In fact, these calculated strains occur below the R(3,j) strain level for only three of the generic casks and only for the end-impact configuration.

The Reevaluation Study also evaluates loss of shielding from lead-slump caused by impact of the steel-lead-steel casks.[11-1] As expected, calculations reveal that shielding material loss occurs mostly in the end-on impact orientation, with a lesser amount in the center-of-gravity over-corner orientation.

If lead slump causes loss of shielding and the cask does not tip over following impact, then the unshielded view factor faces upwards. This effect was neglected in the Reexamination Study to provide a conservative estimate of risk.

Closure failures related to deformations between the cask lid and cask body as well as bolt failure were also considered.[11-1] In order to be somewhat conservative, bolt failure was considered likely at strain levels greater than 50%. The results of this analysis are shown in Table 11.2 (adapted from Reexamination Study, Table 5.5). Note that other analyses indicate that bolt strains above 25% could result in failure.

For the monolithic rail cask, the maximum strain on the cask inner surface is shown in Table 11.3 (Reexamination Study, Table 5.4). Note that for the maximum strain for the 120 mph side impact case, most of the plasticity is caused by compression, with little possibility of failure.

The calculated rail cask closure leak paths results from the Reexamination Study are given in Table 11.4 (Reexamination Study, Table 5.7).

Table 11.2. Maximum True Strain in Closure Bolts from the Reexamination Study

Cask	Speed mph	Corner Impact Strain	End Impact Strain	Side Impact Strain
		Strain %	Strain	Strain
Steel-Lead-Steel	30	3	1	NA
Truck	60	6	3	2
	90	9	5	5
	120	11	7	10
Steel-DU-Steel	30	5	0	1
Truck	60	9	3	4
	90	19	7	10
	120	22	9	18
Steel-Lead-Steel	30	19	6	14
Rail	60	37	3	106
	90	60	9	151
	120	102	16	NA
Monolithic	30	14	4	15
Rail	60	40	14	32
	90	67	35	104
	120	80	58	170

Table 11.3. Maximum Plastic Strains on the Inside of the Monolithic Rail Cask

Corner Impact Speed	Strain (%)	End Impact Speed	Strain (%)	Side Impact Speed	Strain (%)
30 mph	<10	30 mph	<2	30 mph	<10
60 mph	<20	60 mph	<5	60 mph	<30
90 mph	<30	90 mph	<10	90 mph	<50
120 mph	<50	120 mph	<17	120 mph	<60

Table 11.4. Calculated Rail Cask Closure Leak Path Areas

Cask	Velocity (mph)	Orientation	Opening Displacement (inches)	Opening Width (inches)	Leak Path Area (in^2)
Steel-Lead-Steel	90	Corner	0.243	12.76	0.54
Rail	120	Corner	0.512	19.14	3.2
Monolithic Rail	60	Corner	0.103	6.38	0.00028
	90	Corner	0.216	12.76	0.40
	120	Corner	0.439	19.14	2.5
	120	Side	0.123	6.38	0.014

11.3 Impacts onto Real Targets

The finite element results presented above from the Reexamination Study were calculated for cask impacts occurring on a rigid or unyielding target. In real impacts onto likely targets, the target and cask would both absorb some of the impact energy. The Reexamination Study investigated several likely real targets for impact of the four generic casks considered. These were hard desert soil, concrete highways, hard rock, and water.[11-1]

The analysis of the Reexamination Study conservatively assumed that the impact limiter had been fully crushed before impact.

For hard soil targets, the results of impact tests found in the three studies by A. Gonzales, I. G. Waddoups, and L. L. Bonzon were used to obtain the force-deflection curve.[11-8, 11-9, 11-10]

For concrete targets, the test results and empirical results found in a study by A. Gonzales were used to develop force-deflection curves.[11-8, 11-1]

The researchers treated impacts onto hard rock as equivalent to impacts onto a rigid target.[11-1]

To evaluate the effect of real targets, the Reexamination Study calculated velocities for real target impacts that were equivalent to the rigid target velocities previously calculated.[11-1] The equivalent velocities that were calculated for the real targets were substantially greater than those for the rigid targets. In fact, in many cases, equivalent velocities were greater than the 150-mph top of the accident speed distributions of the Modal Study.

11.4 Puncture Analysis

The Reexamination Study includes data from the Association of American Railroads (AAR) tank car accident database regarding punctures of tank car shells.[11-1] The data show that tank cars with a shell thickness of at least one inch rarely experience punctures. The steel-lead-steel generic rail cask has a two-inch outer shell thickness and a one-inch inner containment shell. Thus, punctures of this cask are unlikely. Truck casks have thinner outer walls than rail casks, but their composite wall strength is significantly greater than that of the strongest tank cars. The probability that these casks will fail by puncture is also very low.[11-1]

11.5 Failure of Rods

The Reexamination Study presents an analysis of strains that could cause rod failure for differing burnup level fuels.[11-1] As a result of many considerations, the conclusion of the Reexamination Study is that high burnup (55 to 60 GWD/MTU) spent fuel is assumed to fail at 1% strain, intermediate burnup (40 to 45 GWD/MTU) spent fuel at 4% strain, and low burnup (0 to 25 GWDT/MTU) spent fuel at 8% strain. A mass weighted summation of these cladding strains by burnup range provides an average failure strain level of 3.6%. A sensitivity analysis concluded that mean accident doses and dose risks are not particularly sensitive to the average value chosen for the strain criteria for rod failure during collisions.[11-1]

Tables 11.5 and 11.6 provide the fraction of rods failed by end, corner, and side impacts onto an unyielding surface for varying impact speeds for each of the generic casks. Table 11.5 (Reexamination Study, Table 7.18a) presents the rod fractions failed for PWR fuel assemblies, while Table 11.6 (Reexamination Study, Table 7.18b) shows BWR fuel assembly data.

Tables 11.5 and 11.6 show the percentage of rods with strains greater than 4%.

Table 11.5. PWR Rod Failure Fractions (percent) for Four Generic Casks

Cask	Impact Orientation	Impact Speed (mph)			
		30	60	90	120
Steel-Lead-Steel Truck	end	27	60	100	100
	corner	7	73	100	100
	side	0	0	13	27
Steel-DU-Steel Truck	end	27	33	60	87
	corner	13	100	100	100
	side	7	27	60	87
Steel-Lead-Steel Rail	end	13	60	100	100
	corner	0	13	33	100
	side	0	0	13	87

Table 11.5. PWR Rod Failure Fractions (percent) for Four Generic Casks (Continued)

Cask	Impact Orientation	Impact Speed (mph)			
		30	60	90	120
Monolithic Steel Rail	end	13	100	100	100
	corner	0	33	100	100
	side	0	13	33	73
All	end	20.0	63.3	90.0	96.8
	corner	5.0	54.8	83.3	100.0
	side	1.8	10.0	29.8	68.5
All	All	5.1	45.3	71.8	92.8

Table 11.6. BWR Rod Failure Fractions (percent) for Four Generic Casks

Cask	Impact Orientation	Impact Speed (mph)			
		30	60	90	120
Steel-Lead-Steel Truck	end	0	0	14	29
	corner	0	0	57	100
	side	0	0	0	0
Steel-DU-Steel Truck	end	0	0	0	0
	corner	0	29	100	100
	side	0	0	0	0
Steel-Lead-Steel Rail	end	0	0	14	43
	corner	0	0	0	43
	side	0	0	0	0
Monolithic Steel Rail	end	0	29	57	71
	corner	0	0	29	57
	side	0	0	0	0
All	end	0	7.3	21.3	35.8
	corner	0	7.3	46.5	75.0
	side	0	0.0	0.0	0.0
All	All	0	5.6	34.8	56.2

11.6 Bridge Section Crush Accident

Japan Nuclear Cycle Development Institute (JNC) is leading Japan's programs in fast breeder reactor research and development. As part of their comprehensive safety assessments, JNC has regularly ensured safe MOX fuel transport by performing regulatory tests and supporting analyses on their various Type B packagings. Motivated by the severe earthquake in Kobe, Japan, in 1995, specific detailed studies have been performed to estimate Type B packaging performance in such severe accident conditions. The two studies cited in References 11-11 and 11-12 provide results of an analytical study for a postulated accident involving a 1500-metric-ton concrete bridge section falling 10.4 meters onto the Monju-F fresh-fuel-package-laden truck traveling over an asphalt highway. This package was designed by JNC and was certified in 1990 to meet IAEA regulations for Type B nuclear material transport containers. The analytical detailed study involved a series of plane strain large-deformation finite element analysis using the ABAQUS/ Explicit finite element code.[11-13] These analyses results showed that although deformations in the primary containment vessel and hexagonal fuel assemblies were substantial, the primary containment vessel and fuel cladding maintained their integrity throughout the bridge crush earthquake accident. This result was verified by comparing similar analytical calculations with experimental measurements for the regulatory 9-m side drop tests onto an unyielding target. Conclusions of these studies showed that the safety margin against failure for the Monju-F package is large. The bridge-crush accident is a low-probability severe loading condition, yet the packagings containment boundaries would likely remain intact.[11-11, 11-12]

11.7 Conservatism in Calculating Structural Response

The structural calculations in the Reexamination Study contain a number of conservative assumptions; that is, they describe an accident more severe than one would realistically expect. Some of these assumptions follow.

- Treating all corner impacts as center-of-gravity-over-corner impacts. For corner impacts other than this type, some of the kinetic energy of the cask will be converted into rotational kinetic energy. This rotational energy will be absorbed by a secondary impact on the opposite end of the cask.

- Assuming all end and corner impacts occur on the closure ends of the cask. Because deformations on the opposite end of the cask are much smaller than deformations on the impact end, no releases resulting from closure deformations would result, even at 120-mph impacts, were the impact to occur on the nonclosure end of the cask.

- Assuming velocity vectors for all accidents to be perpendicular to the impact surface. In reality, a distribution of velocity vectors would be expected with only the component perpendicular to the impact surface causing cask damage.

- Treating impact limiters as completely locked from a 30-mph impact. This neglects the design margin that cask designers incorporate into impact limiter designs.

- In the finite-element calculations, zero-thickness shell elements are used to represent the sandwich walls for lead and DU-shielded casks. This results in an over-prediction of lead slump for loss of shielding analyses.

- Omitting neutron shielding and associated linear structures. This assumption ignores the energy that will be absorbed by these components, which could be important for higher velocity impacts.

- Seal leak path areas were calculated at the location of one of the two O-rings typically used. In reality, both O-rings provide containment.

- Use of minimal material properties for closure bolts. The specified bolt material (SA-540, Grade B23, Class 5) can have yield strengths more than 50% higher than the values used for the calculations.

- Soil properties based on desert soils. Desert soils are hard compared to tillable soils.

- Highway surface impacts assumed to be concrete. Asphalt highway surfaces are more yielding than concrete.

- Hard rock assumed to be a rigid surface. In reality, hard rock impacts would absorb energy by cracking and spalling.

- Risk of puncture assumed to be 0.1% for truck casks and 1% for rail casks in rail-coupling accidents and 0.1% for other impacts. The puncture structural review indicates the probability for puncturing a cask with a 1-in. wall thickness to be extremely remote.

- Estimating strains in spent fuel rods for severe impacts by scaling strains in spent fuel rods calculated for a 100-G impact without accounting for energy consumed deforming the basket and assembly structures.

11.8 References

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12. Thermal Analysis Example of Generic Cask in a Long Duration Fire

12.1 Introduction

In the Reexamination Study, thermal analyses were performed on the four generic casks for both an 800°C regulatory fire and a 1000°C fully engulfing, optically dense fire.[12-1] PATRAN/Pthermal was used to model the casks as one-dimensional axisymmetric cylinders, including a neutron shield.

12.2 Thermal Response: 1000°C Long Duration Fire

Figure 12.1 (Reexamination Study, Figure 6.6), shows the calculation results for interior surface temperature histories for the four generic casks.

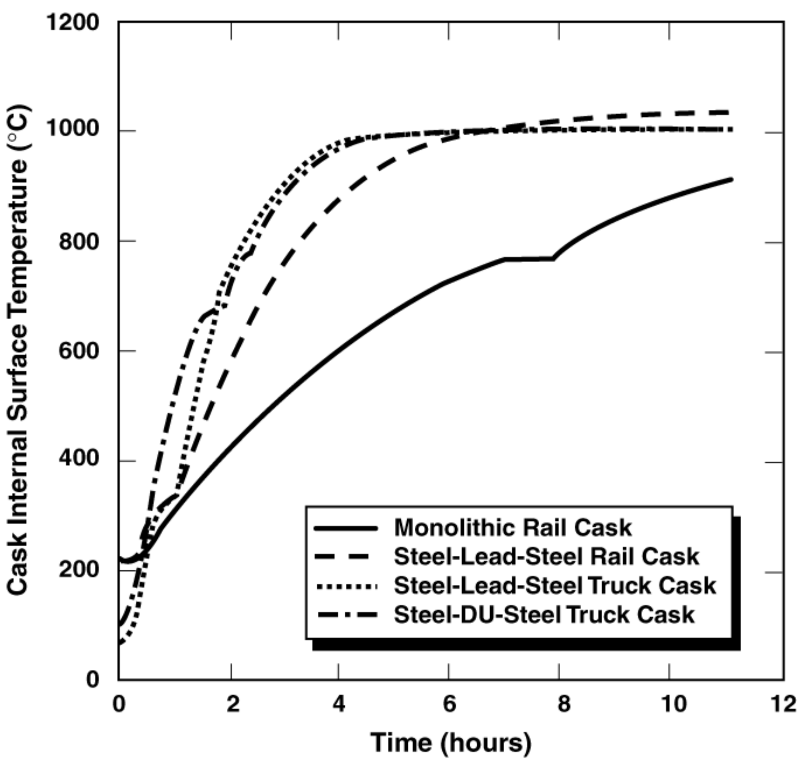

Figure 12.1. Internal Surface Temperature Histories of the Generic Casks in a 1000°C Long Duration Fire (Figure 6.6, Ref. 12-1, page 6-6).

Times to reach three characteristic temperatures were evaluated: 350°C for thermal degradation of elastomer seals, 750°C for burst rupture of spent fuel rods, and 1000°C where the cask equilibrates with the fire.

Table 12.1 contains the results from Table 6.5 of the Reexamination Study.

Table 12.1. Time in Hours Required for the Generic Cask Internal Surface to Get to the Three Characteristic Temperatures in a Long Duration Engulfing, Optically Dense, 1000°C Fire

Temperature (°C)	Truck Casks		Rail Casks	
	Steel-Lead-Steel	Steel-DU-Steel	Steel-Lead-Steel	Monolithic Steel
350	1.04	0.59	1.06	1.37
750	2.09	1.96	2.91	6.57
1000	5.55	5.32	6.43	>11

The internal heat load in the casks was calculated by the ORIGEN Code.[12-2] This heat load is generated by the decay of radionuclides in the spent fuel.

12.3 Thermal Response to a Long Duration 800°C Fire

In a similar analysis to the 1000°C fire discussed above, the 800°C regulatory fire condition was also calculated in the Reexamination Study for a long duration fire. Table 12.2, taken from Table 6.7 of the Reexamination Study, shows the time that is required for the generic cask internal surfaces to reach the characteristic temperatures of 350°C and 750°C as discussed above.[12-1]

Table 12.2. Time in Hours Required for the Generic Cask Internal Surface to get to the Two Characteristic Temperatures in a Long Duration Engulfing, Optically Dense, 800°C Fire

Temperature (°C)	Truck Casks		Rail Casks	
	Steel-Lead-Steel	Steel-DU-Steel	Steel-Lead-Steel	Monolithic Steel
350	1.77	1.06	1.69	2.37
750	4.88	5.07	6.32	>11

12.4 Conservatism in the Thermal Calculations

Three-year high burnup spent fuel was assumed for the thermal analysis of the Reexamination Study. Because ten-year fuel will typically be transported, large conservatism is noted: Three-year pressurized water reactor (PWR) fuel assemblies, those considered, produce approximately 2.8 kW; however, ten-year average burnup fuel produces less than 600 W of decay heat. This resulted in the calculation of shorter times for seal degradation and rod burst than would result if the actual decay heat load had been used.

The phase change of the neutron shield material at the cask outside region was not included. A water neutron shield, or a solid hydrogen-based shield, would delay internal heating lengthening the heating times presented in the Reexamination Study.[12-1]

In summary, the conservative assumptions in the thermal analysis are:

- The generic casks considered were not optimized for thermal response.
- The casks were assumed to be fully engulfed in the fire.
- The fire was assumed to be large enough to be optically dense.
- Heat fluxes for a fully engulfing optically dense 1000°C fire were assumed.
- The water in the neutron shield was assumed to be immediately replaced by air.

12.5 References

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13. Source Term and Source Term Probabilities

13.1 Introduction to Event Trees

An event tree is a diagram that depicts possible accident scenarios as paths through event space. By definition, the set of scenarios on a properly constructed event tree constitutes a representative set. Values of the branch points on an event tree are estimated from past accident data or from route data.

Transportation accident event trees were developed for truck and rail accidents during the performance of the Modal Study.[13-1] Slightly modified versions of the Modal Study event trees were used in the Reexamination Study.[13-2]

These event trees are shown in Figure 13.1 (Reexamination Study, Figure 7.1) for truck accidents and Figure 13.2 (Reexamination Study, Figure 7.2) for rail accidents. Each path on the event tree constitutes a unique sequence of events, such that the product of all the branch point probabilities on a path provides the probability of that particular accident scenario. This probability is conditional because it depends on the occurrence of an accident. Thus, if an accident occurs, the event tree predicts the probability of the accident taking place in the described manner. The sum of the branch point probabilities at each level of branching on the tree must add to exactly one.

The Modal Study analyzed the response of generic casks to the mechanical and thermal conditions that characterize each scenario on their two event trees.[13-1] Paths that are considered capable of causing a Type B spent fuel cask to fail by impact are indicated with an asterisk after the path number, or path accident index.

The Reexamination Study re-evaluated the probability values for route wayside surfaces in Figures 13.1 and 13.2 using recent data. These modified event trees are shown in Figure 13.3 (Reexamination Study, Figure 7.3) for truck accidents and Figure 13.4 (Reexamination Study, Figure 7.4) for rail accidents. This reevaluation incorporated better estimates of route wayside surface frequencies of occurrence into the Modal Study event trees.

Data used for this revision were obtained from references cited in the Reexamination Study.[13-3, 13-4, 13-5, 13-6, 13-7]

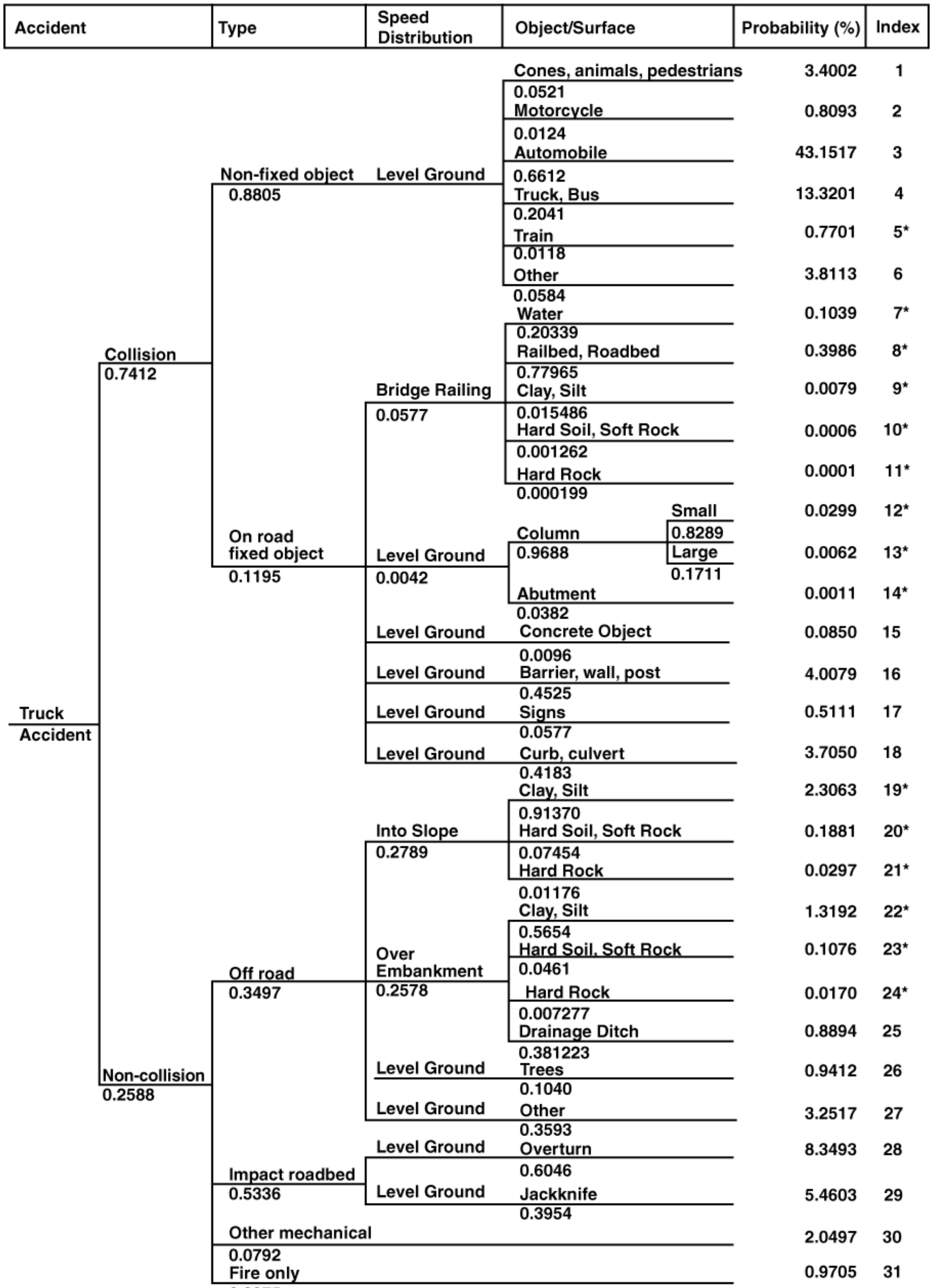

Figure 13.1. Modal Study Truck Accident Event Tree.

*Paths that are considered capable of causing a Type B cask to fail are marked with an asterisk on the index number.

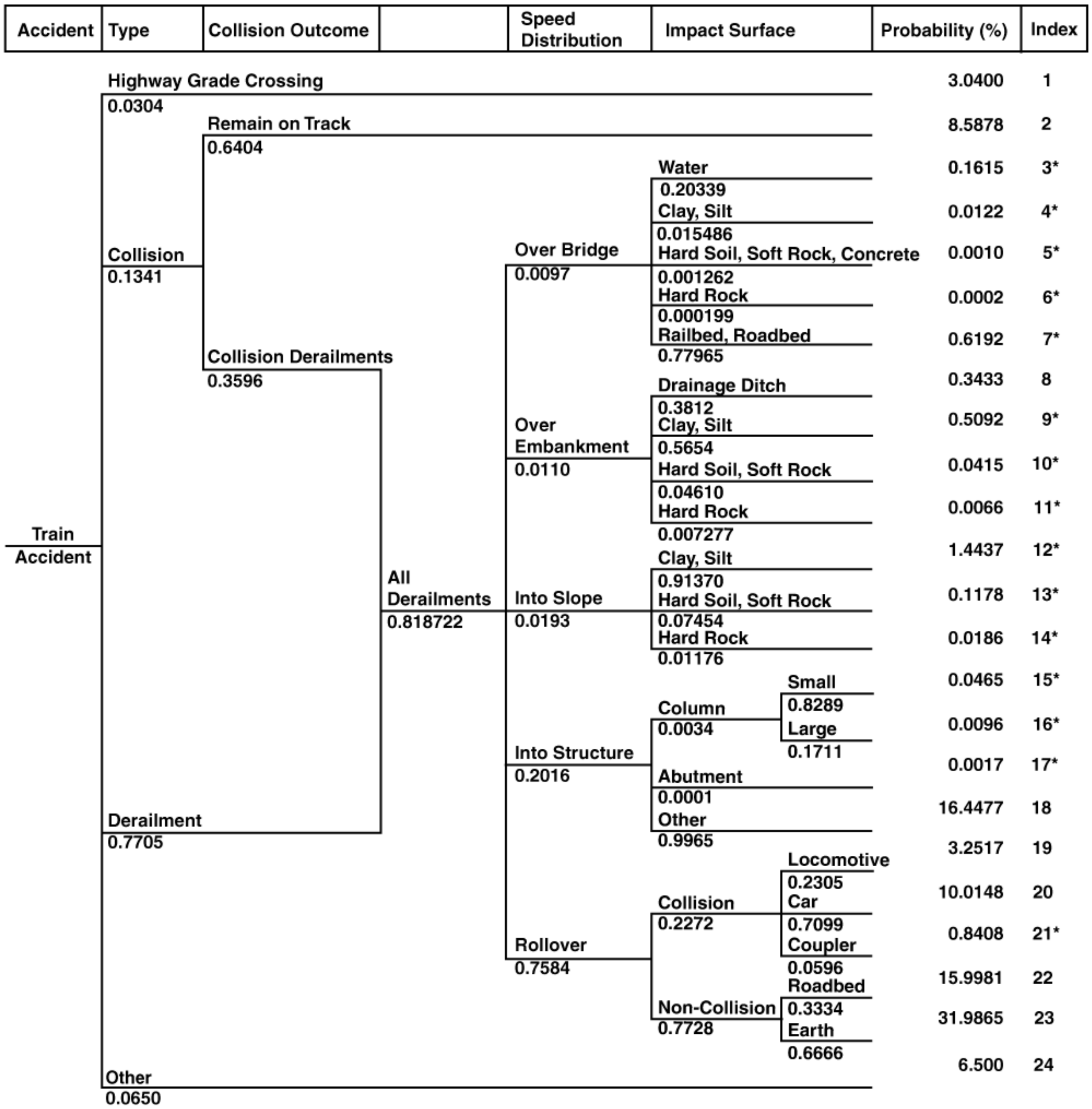

Figure 13.2. Modal Study Train Accident Event Tree.

*Paths that are considered capable of causing a Type B cask to fail are marked with an asterisk on the index number.

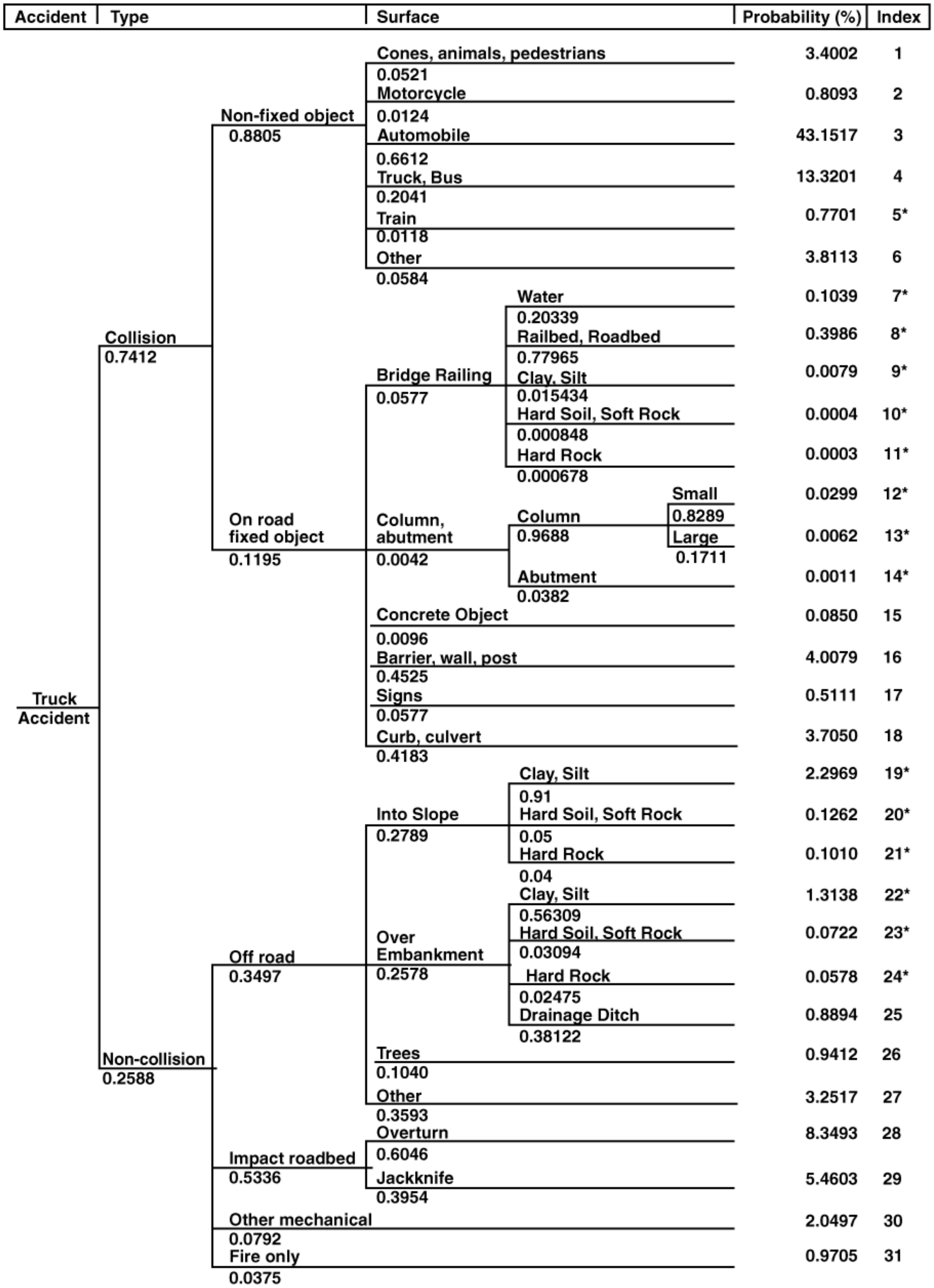

Figure 13.3. Modified Modal Study Truck Accident Event Tree.

*Paths that are considered capable of causing a Type B cask to fail are marked with an asterisk on the index number.

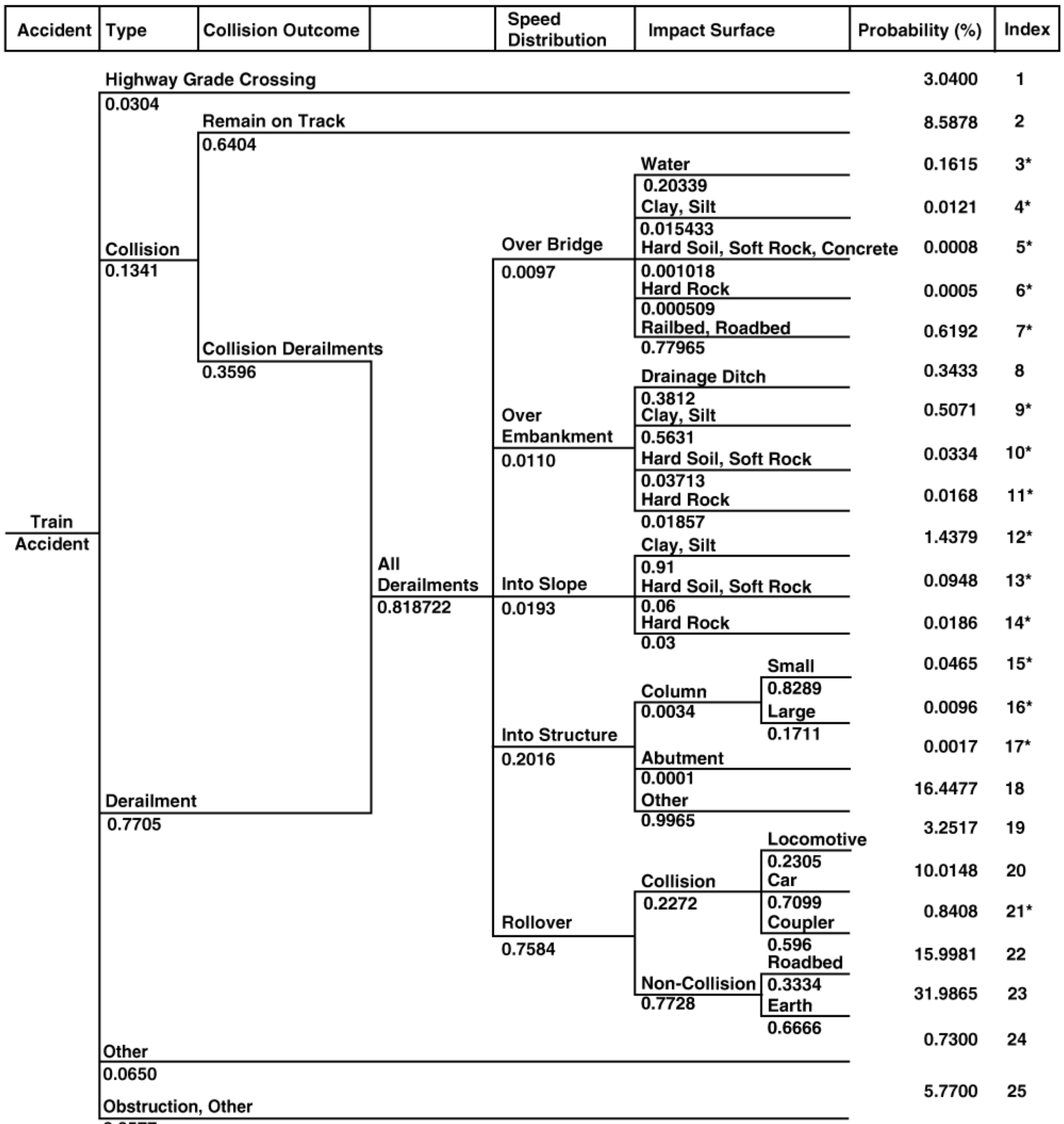

Figure 13.4. Modified Modal Study Train Accident Event Tree.

*Paths that are considered capable of causing a Type B cask to fail are marked with an asterisk on the index number.

13.2 Source Terms

By definition, risk is the product of consequence magnitude and the probability of event occurrence. The consequence magnitude can be calculated using RADTRAN for radioactive material transportation accidents.[13-8, 13-9] RADTRAN-produced values of consequence magnitude are calculated based on the accident source term, meteorological conditions for the accident event, population that could be exposed, and emergency response actions that result from the hypothetical accident scenario being studied.

Source terms specify the amounts of radioactive materials that might be released by a particular accident scenario. The Reexamination Study estimated the source terms for accident scenarios and the probabilities of these accident scenarios for three broad classes of transportation accidents:[13-2]

A. Collisions without fires
B. Collisions that lead to fires
C. Fires without collisions.

The Reexamination Study estimated source terms as the product of the cask inventory and two release fractions: a fuel rod to cask release fraction and a cask to environment release fraction.[13-2] For example, the following equation was used in the Reexamination Study as (ST_{jk})[13-2] to estimate the source term for accident scenario (j) and cask (k). The expression is summed for each radionuclide (i) in the radioactive material contained in the cask.

$$\left(ST_{jk}\right) = \sum_i \left(ST_{ijk}\right) = \sum_i I_{ik}\ f_{release,\ ijk} =$$
$$= f_{rod,\ jk} \sum_i I_{ik}\ f_{RCijk} f_{CEijk}$$

Where

ST_{ijk}	=	the amount of radionuclide i released from cask k during accident scenario j
I_{ik}	=	the number of curies in nuclide i in the inventory of cask k
$f_{release,ijk}$	=	the fraction of the inventory of radionuclide i in cask k that is released to the environment for the accident scenario j
$f_{rod,jk}$	=	the fraction of rods in cask k that fail during accident scenario j
f_{RCijk}	=	the fraction of radionuclide i that is released by accident scenario j to the interior of the cask k from each failed rod
f_{CEijk}	=	the fraction of the amount of each radionuclide released to the cask interior that is transported to the environment through the cask leak.

The inventories of single fuel assemblies were evaluated in the Reexamination Study by using the ORIGEN Code for generic pressurized water reactor (PWR) and boiling water reactor (BWR) assemblies.[13-10, 13-11] Cask inventories were then calculated by multiplying the single assembly inventories (less negligible radionuclides that do not contribute significantly to radiation doses) by the number of assemblies transported in each of the casks.

Fuel burnup controls inventory size in spent fuel. The Reexamination Study used a DOE report, *Spent Nuclear Fuel Discharges from U. S. Reactors*, to identify BWR and PWR burnups for U. S. reactors.[13-12]

An ORIGEN inventory contains approximately 800 radionuclides, a number that can be reduced to those contributing significantly to health hazards by sorting inventory amounts divided by A_2 values.[13-13, 13-14] The RADSEL Code can be used to perform this reduction.[13-15]

Radioactive gases, such as tritium and Kr-85, were evaluated in the Reexamination Study.[13-2] Kr-85 is the most important member of the noncondensible gas chemical element group. Tritium was not included for the BWR and PWR inventories because of its small amount calculated as present.

Radioactive deposits called CRUD[13-16] are formed from the corrosion products deposited on fuel assembly surfaces from the reactor cooling water system. Activation of these deposits by neutron bombardment produces radionuclides, most importantly Co-60. During normal transport and accidents involving impact and/or fire, these materials could be released to the atmosphere if the cask suffered a containment loss.

The final generic PWR and BWR assembly inventories used in the Reexamination Study are presented in Table 13.1 (Reexamination Study, Table 7.9).[13-2] Of course, for MOX fuel assemblies, the individual nuclide mix would include larger amounts of Pu isotopes.

To simplify the development of accident source terms, fission products are assigned to chemical element classes. In the Reexamination Study, five chemical element classes were designated to describe spent fuel.[13-2] These are listed below:

Representative Element	Description
Xe	Noble (noncondensible) gases
Cs	Condensible gases
Ru	Single element group
Co	Fission products found in CRUD
Part	All other fission products

Table 13.1. Generic High Burnup, Three-Year Cooled, Fuel Assembly Inventories for RADTRAN Calculations (Ci/assembly)

Generic BWR Assembly		Generic PWR Assembly	
Nuclide	**Amount**	**Nuclide**	**Amount**
Co-60	6.40e+01	Co-60	5.78e+01
Kr-85	1.74e+03	Kr-85	1.74e+03
Sr-90	1.59e+04	Sr-90	5.36e+04
Y-90	1.59e+04	Y-90	5.36e+04
Ru-106	1.42e+04	Ru-106	4.43e+04
Cs-134	2.15e+04	Cs-134	6.99e+04
Cs-137	2.59e+04	Cs-137	7.90e+04
Ce-144	1.03e+04	Ce-144	3.87e+04
Pm-147	8.49e+03	Pm-147	2.58e+04
Pu-238	1.67e+03	Eu-154	8.42e+03
Pu-239	7.44e+01	Pu-238	4.81e+03
Pu-240	1.36e+02	Pu-239	2.14e+02
Pu-241	2.91e+04	Pu-240	4.28e+02
Am-241	2.05e+02	Pu-241	6.52e+04
Am-242M	8.09e+00	Am-241	4.36e+02
Am-243	1.22e+01	Am-242M	1.33e+01
Cm-242	1.82e+02	Am-243	2.51e+01
Cm-243	1.42e+01	Cm-242	3.76e+02
Cm-244	2.95e+03	Cm-243	2.88e+01
		Cm-244	5.62e+03

13.3 Collision-Only Scenarios: Release Fractions

The analysis performed for the Reexamination Study showed that collisions that do not cause fires must be unusually severe to cause seal leakage. This reference assumed the following based on finite element calculations:

1. Seal leakage occurs for Type B packages for equivalent unyielding surface collisions at or above 60 mph for rail casks and 120 mph for truck casks.
2. The leakage area produced is about 1mm^2.
3. At least some of the rods in a spent fuel cask will fail.

The total release fraction ($f_{release}$), for release of fission products from failed rods in a spent fuel cask to the environment is given in the Reexamination Study by

$$f_{release} = f_{rod,impact} f_{RC} \left(1 - f_{deposition}\right)\left(1 - \frac{p_{atm}}{p_{Imp}}\right)$$

where

$f_{rod,impact}$ = the fraction of rods in the cask that are failed by impact,
f_{RC} = the fraction of materials in a spent fuel rod that is released to the cask interior from rod failure,
$f_{deposition}$ = the fraction of materials that rapidly deposit onto the cask interior surfaces on release from the failed spent fuel rod,
p_{atm} = the atmospheric pressure,
p_{Imp} = the cask internal pressure after fuel rod depressurization.

13.4 Collisions that Lead to Fires: Release Fractions

The Reexamination Study considered accident scenarios where the collision initiates fires of various durations. Heating of the cask to three temperatures of interest was examined. T_s is the temperature at which elastomer seals fail; T_b is the temperature at which rods fail by burst rupture; and T_f is the average temperature of a hydrocarbon pool fire. Fire duration determines which of these temperatures are reached by the cask. The release fraction derived for these cases is given by the following equation (Reexamination Study, page 7-23):

$$f_{rel} = f_{imp} f_{RCimp}\left(1 - f_{dep}\right)\left\{\left[1 - \frac{p_{atm}}{p_{imp}}\frac{T_a}{T_s}\right] + \left[\frac{p_{atm}}{p_{imp}}\frac{T_a}{T_s}\right]\left[1 - \frac{T_s}{T_b}\right] + \left[\frac{p_{atm}}{p_{imp}}\frac{T_a}{T_b}\right]\left[1 - \frac{p_{atm}}{p_b}\frac{T_b}{T_f}\right]\right\}$$
$$+ f_{bur} f_{RCf}\left(1 - f_{dep}\right)\left\{\left[1 - \frac{p_{atm}}{p_b}\frac{T_b}{T_b}\right]\right\}$$

where

f_{rel} = release of fission products from failed rods to the environment during Category 5 accidents
f_{bur} = $1 – f_{imp}$, because rods not failing by impact are assumed to fail by thermal bursting when the burst temperature T_b is reached.
T_a = cask internal temperature during normal transport under ambient conditions
T_f = temperature of the fire
T_s = temperature where the cask seal begins to leak
T_b = temperature where the remaining rods fail by burst rupture
T_f = temperature of fire
p_{Imp} = cask pressure after rod failure due to impact
p_{atm} = atmospheric pressure
f_{Imp} = fraction of the rods failed by impact
f_{bur} = fraction of rods failed by thermal burst rupture
f_{RCimp} = the release fraction for fission products to the cask interior from a rod failed by impact
f_{dep} = the fraction of the materials released from failed rods to the cask interior that deposits rapidly onto cask internal surfaces

f_{RCf} = the release fraction for fission products to the case interior from a rod failed by burst rupture due to a fire

p_b = cask pressure after rod failure due to burst rupture

13.5 References

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13-2 J. L. Sprung, D. J. Ammerman, N. L. Breivik, R. J. Dukart, F. L. Kanipe, J. A. Koski, G. S. Mills, K. S. Neuhauser, H. D. Radloff, R. F. Weiner, and H. R. Yoshimura, *Reexamination of Spent Fuel Shipment Risk Estimates* (generally known as the Reexamination Study), NUREG/CR-6672. Vol. 1, SAND2000/0234, Sandia National Laboratories, Albuquerque, New Mexico, March 2000.

13-3 *Ibid*., page 7-2. The Reexamination Study cites ARC/INFO Coverages and Arc view Project, K. C. Bayer Digital Map of the U. S., purchased from Geologic Data Systems, Inc., 1600 Emerson St., Denver, Colorado, 80218.

13-4 *Ibid*., page 7-2. The Reexamination Study cites State Soil Graphics (STATSGO) Data Base, available on the Internet at ftp.ftw.nrcs.usda.gov/pub/statsgo.

13-5 *Ibid*., page 7-12. The Reexamination Study cites Accidents of Motor Carriers of Property (for the years 1973 through 1983), Bureau of Motor Carrier Safety, Federal Highway Administration, U. S. Department of Transportation, Washington, D. C., 1975–1984.

13-6 *Ibid*., page 7-7. The Reexamination Study cites D. B. Clauss, *et al*., *A Statistical Description on the Types and Severities of Accidents Involving Tractor Semi-Trailers*, SAND93-2580, Sandia National Laboratories, Albuquerque, New Mexico, June 1994.

13-7 *Ibid*., page 7-9. The Reexamination Study cites Accident/Incident Bulletins Nos. 145 through 151, Office of Safety, Federal Railroad Administration, U. S. Department of Transportation, Washington, D. C., 1977–1983.

13-8 *Ibid*., page 7-13. The Reexamination Study cites K. S. Neuhauser and F. L. Kanipe, *RADTRAN 4, Volume II, Technical Manual*, SAND89-2370, Sandia National Laboratories, Albuquerque, New Mexico, May 1994.

13-9 *Ibid*., page 7-13. The Reexamination Study cites K. S. Neuhauser and F. L. Kanipe, *RADTRAN 4, Volume III, User Guide*, SAND89-2370, Sandia National Laboratories, Albuquerque, New Mexico, January 1992.

13-10 *Ibid*., page 7-14. The Reexamination Study cites A. G. Croff, *ORIGEN2 – A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, Oak Ridge, Tennessee, July 1980.

13-11 *Ibid*., page 7-14. The Reexamination Study cites *ORIGEN2 Isotope Generation and Depletion Code*, CCC-371, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1991.

13-12 *Ibid*., page 7-14. The Reexamination Study cites Spent Nuclear Fuel Discharges from U. S. Reactors, Energy Information Administration, U. S. Department of Energy, SR/CNEAF/96-01, 1994.

13-13 *Ibid*., page 7-15. The Reexamination Study cites *International Atomic Energy Agency, Safety Series No. 7, IAEA Safety Guides, Explanatory Material for the IAEA Regulation for the Safe Transport of Radioactive Material* (1985 Edition), 2nd Edition, International Atomic Energy Agency, Vienna, Austria, 1987.

13-14 *Ibid*., page 7-15. The Reexamination Study cites the Code of Federal Regulation, Volume 49, Part 173-435 (40 CFR 173-435) for this information.

13-15 *Ibid*., page 7-15. The Reexamination Study cites RADSEL, unpublished Sandia code, available on request, for this information.

13-16 *Ibid*., page 7-16. The Reexamination Study cites R. P. Sandoval, *et al*., *Estimate of CRUD Contribution to Shipping Requirements*, SAND88-1358, Sandia National Laboratories, Albuquerque, New Mexico, January 1991, for this information.

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14. RADTRAN Calculation Examples

14.1 Overview of Calculation Examples

The set of calculations in the Reexamination Study demonstrates how to analyze the potential risk of spent fuel transport using up-to-date assumptions about parameters. These calculations provide a comprehensive array of risk conditions and their consequences.[14-1] As such, this work provides both a template for risk analysis procedures and definitive risk calculations for spent fuel shipments in the United States. The methodology of this study could be utilized, with proper input of parameters, for MOX shipments in Japan. The discussion below was largely taken from the Reexamination Study to provide a concise description of the methodology used.

In the Reexamination Study, seven sets of RADTRAN calculations were performed for the four generic cask designs discussed above in Section 5.2, two shipment modes, two sets of routes, and three sets of accident source terms. The four cask designs are steel-lead-steel truck cask, steel-lead-steel rail cask, steel-DU-steel truck cask, and monolithic steel rail cask. The two shipment modes are truck and rail. The two sets of routes are a compilation of 200 representative routes determined by Latin Hypercube Sampling (LHS) and a set of four specific routes together with the NUREG-0170 shipment route. The three sets of accident source terms were the NUREG-0170 source term, the Modal Study source terms, and the new source terms developed for the Reexamination Study.

The seven sets of RADTRAN calculations are explained below and are shown in Table 14.1 (Reexamination Study, Table 8.1).

1. The first set of RADTRAN calculations determined the risk associated with shipping PWR and BWR spent fuel by truck (T) in steel-lead-steel (SLS T) and steel-DU-steel (SDUS T) casks.

2. The second set determined the risk for rail (R) in steel-lead-steel (SLS R) and monolithic steel (Mono R) casks.

3. The third set determined the risk for PWR spent fuel by truck in a steel-lead-steel cask over five illustrative (Illus) routes:

   - Crystal River Nuclear Plant in Florida to Hanford, Washington (C/H),
   - Maine Yankee Nuclear Plant in Maine to Skull Valley, Utah (M/SV),
   - Maine Yankee Nuclear Plant to the Savannah River Site in South Carolina (M/SR),
   - Kewaunee Nuclear Plant in Wisconsin to the Savannah River Site (K/SR), and
   - The representative truck route examined by NUREG-0170.

4. The fourth set of RADTRAN calculations determined the risk for rail shipments of PWR spent fuel for the routes of the third set, above, using a monolithic steel cask.

Table 14.1. Characteristics of Sets of RADTRAN Calculations[14-2]

Set	Calc.	Routes		Inventory[2]			Severity and Release Fractions														Exp. Paths	
				This Study			This Study								NUREG-0170				Modal Study			
		LHS	Illus	PWR	BWR	0170	SLST		SDUST		SLSR		Mono R		Mod 1		Mod 2				All	Inhal
							PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	T	R	T	R	T	R		
1	1	X		X			X														X	
	2	X			X			X													X	
	3	X		X					X												X	
	4	X			X					X											X	
2	5	X		X							X										X	
	6	X			X							X									X	
	7	X		X									X								X	
	8	X			X									X							X	
3	9		C/H	X			X														X	
	10		M/SV	X			X														X	
	11		M/SR	X			X														X	
	12		K/SR	X			X														X	
	13		0170	X			X														X	
4	14		C/H	X									X								X	
	15		M/SV	X									X								X	
	16		M/SR	X									X								X	
	17		K/SR	X									X								X	
	18		0170	X									X								X	
5	19	X		X											X						X	
	20	X		X													X				X	
	21	X				X											X				X	
	22	X				X											X					X
6	23	X				X									X							X
	24	X		X															X		X	
7	25	X				X										X						X
	26	X				X												X				X
	27	X		X																X	X	

5. The fifth set examines the influence on spent fuel truck accident risks of the inventory source term, and exposure pathway models used in NUREG-0170.

6. The sixth set calculates risks for spent fuel truck accidents using Modal Study and NUREG-0170 source terms.

7. The seventh set repeats the above set for corresponding rail shipments.

14.2 RADTRAN 5 Computational Scheme

The RADTRAN 5 Risk Code provides a calculated estimate of the various risks associated with the shipment of a single radioactive material along a single route.[14-2, 14-3] Inputs for the calculation are the material, package specifications, route information, weather, accident source term, and emergency response actions. Outputs from RADTRAN 5 include calculations of population dose for either a specified accident or incident-free dose. Because RADTRAN's computational methodology allows calculational repetitions over additional route segments, weather conditions, and accident source terms, the codes uses a Latin Hypercube Sampling (LHS) method[14-4] for examination of a large number of cases. LHS allows large sets of parameters to be selected by distribution sampling and provided to RADTRAN 5 as separate input files.

LHS is a Monte Carlo sampling method. It was used in the Reexamination Study to generate representative sets of values for RADTRAN 5 input parameters.[14-1]

14.3 Input Parameters

For the Reexamination Study, the RADTRAN 5 calculations studied spent fuel Type B casks. All routes considered had three segments: one urban, one suburban, and one rural segment.

- The radionuclides were chosen to represent three-year cooled, high burnup PWR and BWR inventories for 60 and 50 gigawatt-days per metric ton of uranium respectively.
- Routing was either the 200 representative routes, one illustrative route, or the NUREG-0170 route.
- Traffic density and speeds, average vehicle occupancy, accident rates, population densities, and lengths for each of the three aggregate route segments were also input parameters.
- The number of times the spent fuel truck or train stops, the duration and type of stop, and the number of people that might be exposed to radiation as a result of the stop were input parameters.
- The package dose rate at 1 m from the packaging surface was used.
- Weather conditions were included.

- Either the 19 sets of truck accident release fractions or the 21 sets of train accident release fractions developed in the Reexamination Study, the eight sets of NUREG-0170 release fractions, or the 20 sets from the Modal Study were used.

- The severity fraction associated with each release fraction, i.e., the fraction of all possible accidents estimated to cause each of the release fractions, was an input parameter.

- The evacuation time, or time after occurrence of an accident when the population that might have been exposed has been evacuated, was considered.

- Other RADTRAN 5 parameters that do not depend on the type of radioactive material shipped, the shipment route, the accident source term, prevailing weather, or emergency response actions were used.

For these input parameters, RADTRAN 5 was used in the Reexamination Study to calculate the following:

- Incident-free doses by population groups along the routes or involved within the proximity of the shipment. The sum of doses for each population group and for all population groups together was calculated to provide the total incident-free dose.

- Accident doses, or the dose that people might receive in the event of an accident that could release radioactive material to the environment.

14.4 Complementary Cumulative Distributions Functions

RADTRAN calculation results are typically displayed as complementary cumulative distribution functions (CCDFs). CCDFs are plots of the probability of occurrence of an accident population dose of a given size or larger. Thus, the probability associated with each consequence value on a CCDF is the sum of that probability value and the probabilities associated with all larger consequence values of the CCDF. The area under the CCDF is the mean population dose risk in person-rem for the set of accidents considered.

The Reexamination Study provided four compound CCDFs for each case: the mean or expected curve, and the 5th, 50th (median), and the 95th percentile results.

14.5 Truck Cask Results for the Generic Steel-Lead-Steel and Steel-DU-Steel for 200 Representative Routes

Figure 14.1 (Reexamination Study, Figure 8.2) shows the four compound CCDFs of the set of 200 CCDFs that were calculated by authors of the Reexamination Study using the PWR source terms developed for the generic steel-lead-steel truck cask and the representative LHS sample of 200.[14-1] This case was calculated for the steel-lead-steel truck cask transporting one PWR spent fuel assembly.

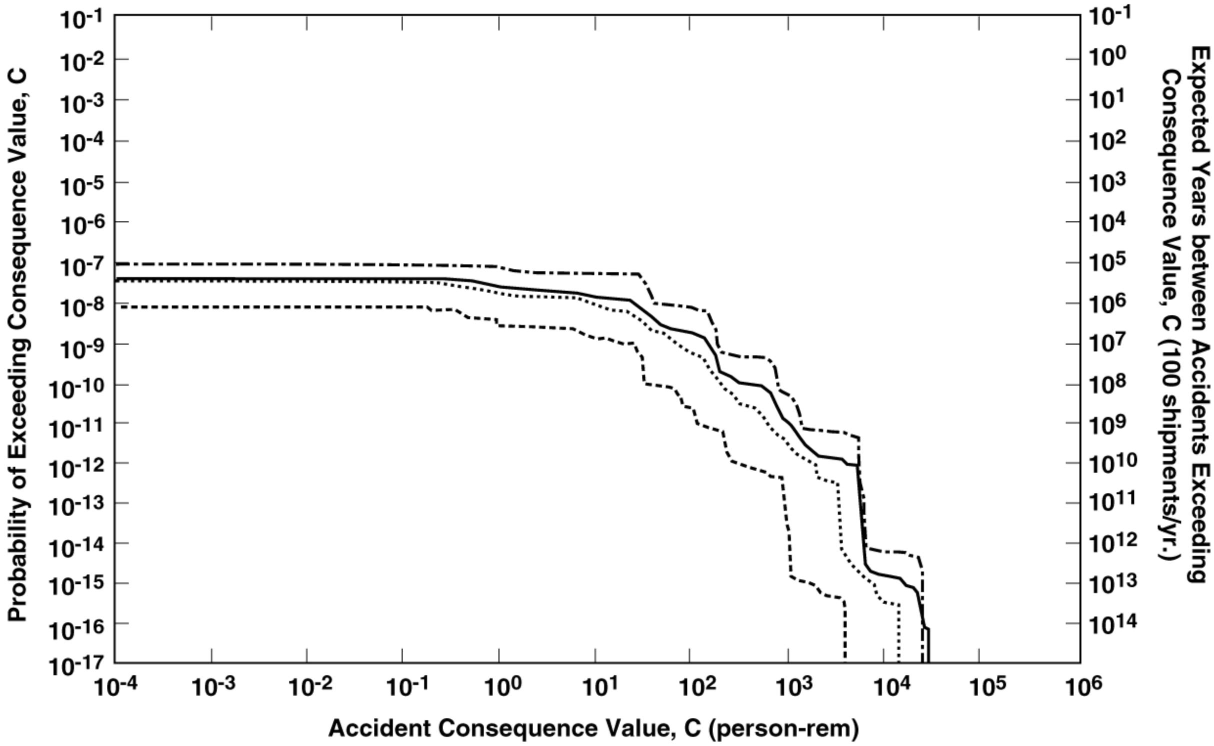

Figure 14.1. Truck Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Steel-Lead-Steel Truck Cask Over the 200 Representative Truck Routes. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 19 Representative Truck Accident Source Terms.

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

The CCDF ordinate is the probability of exceeding the corresponding consequence value, together with a second ordinate calculated to provide the expected years between accidents that exceed the consequence value, assuming an arbitrary but easily scaled value of 100 shipments per year; i.e., the years per accident is equal to the reciprocal of the product of the accidents per shipment and the shipments per year.

Figure 14.2 (Reexamination Study, Figure 8.3) is the calculated CCDF for the generic steel-lead-steel truck cask carrying BWR spent fuel.[14-1] Figure 14.3 (Reexamination Study, Figure 8.4) is the calculated CCDF for the generic steel-DU-steel truck cask with PWR fuel. Figure 14.4 (Reexamination Study, Figure 8.5) is the calculated CCDF for the generic steel-DU-steel truck cask carrying BWR spent fuel.

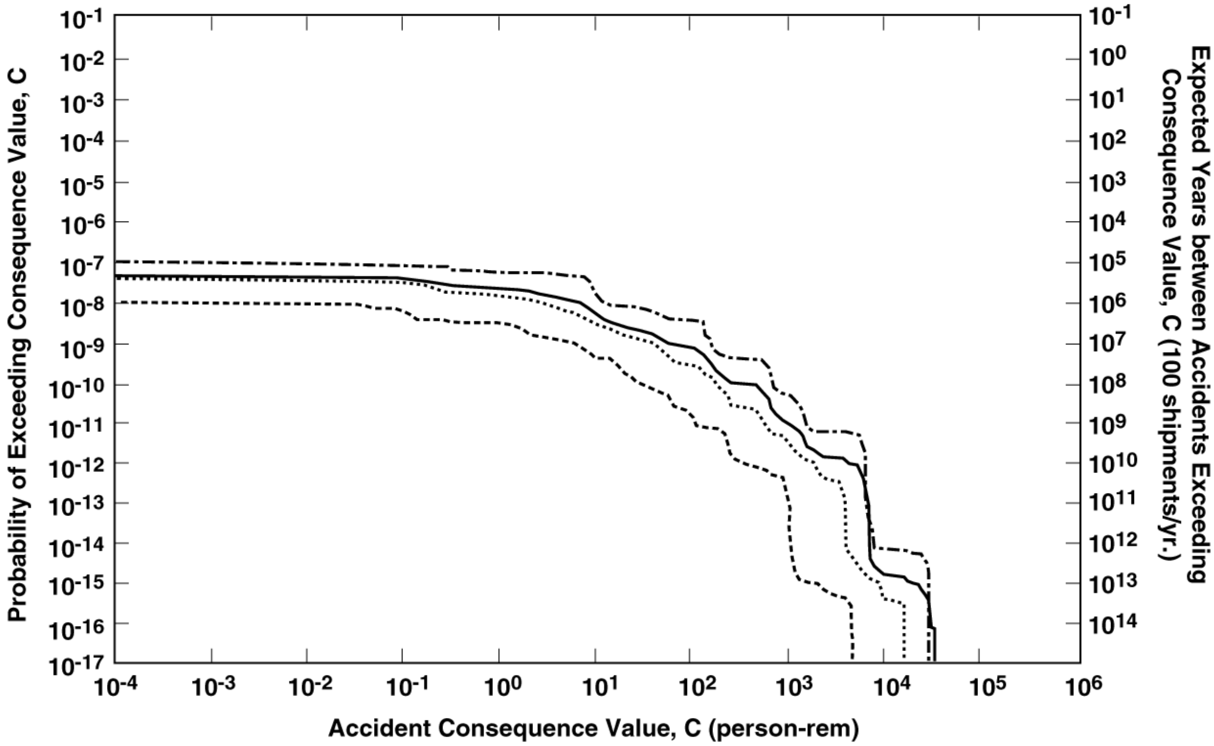

Figure 14.2. **Truck Accident Population Dose Risk CCDFs for Transport of BWR Spent Fuel in the Generic Steel-Lead-Steel Truck Cask Over the 200 Representative Truck Routes. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 19 Representative Truck Accident Source Terms.**

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

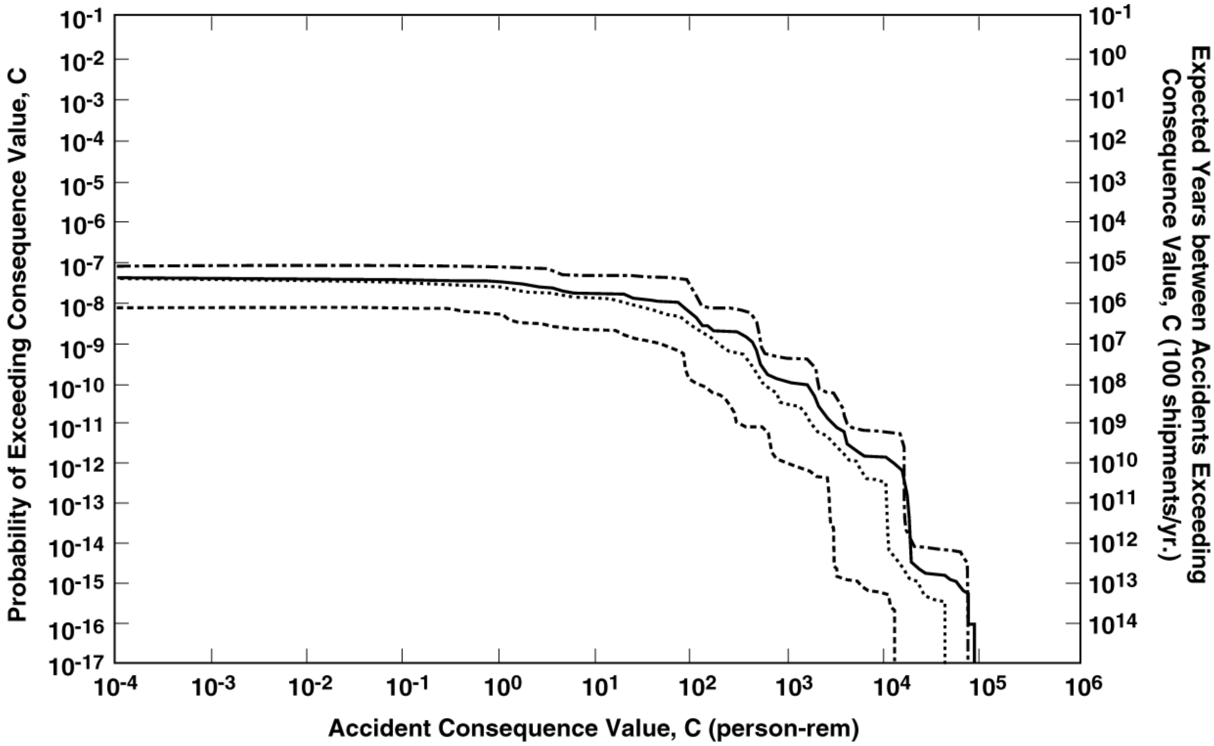

Figure 14.3. **Truck Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Steel-DU-Steel Truck Cask Over the 200 Representative Truck Routes. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 19 Representative Truck Accident Source Terms.**

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

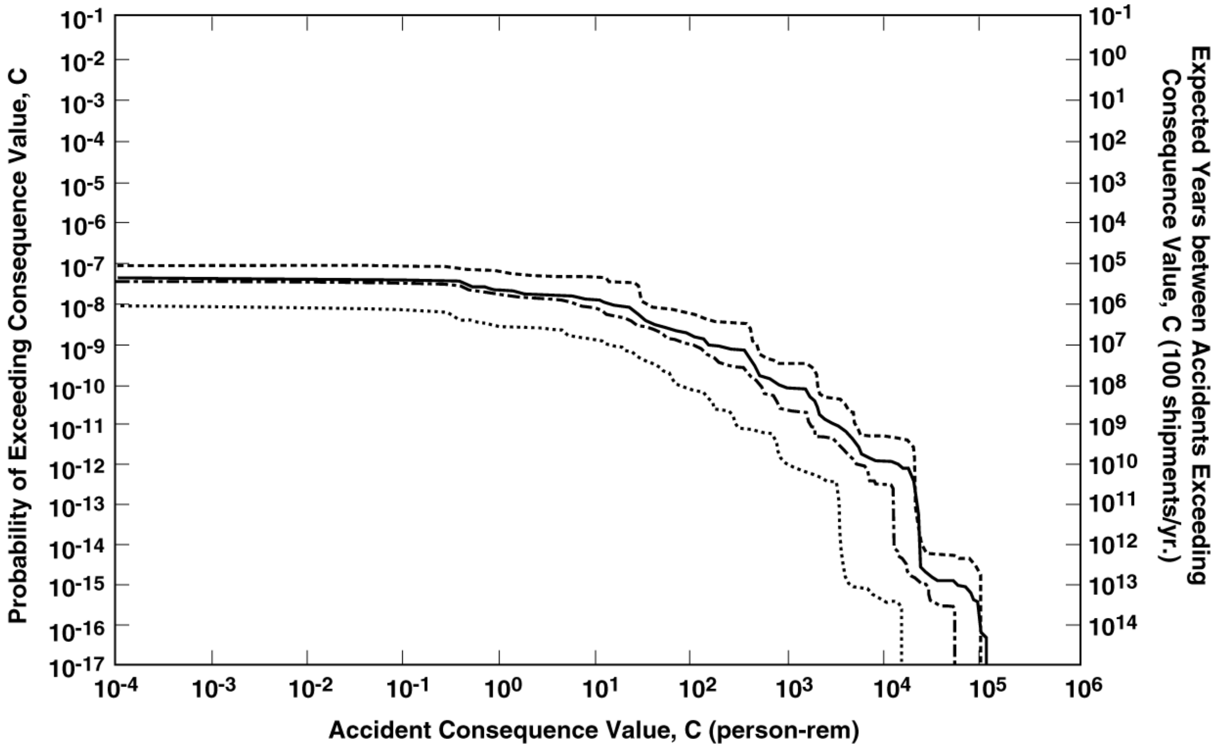

Figure 14.4. Truck Accident Population Dose Risk CCDFs for Transport of BWR Spent Fuel in the Generic Steel-DU-Steel Truck Cask Over the 200 Representative Truck Routes. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 19 Representative Truck Accident Source Terms.

Mean (———) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

The area under the mean or expected CCDF shown in Figures 14.1, 14.2, 14.3, and 14.4 give the expected values of truck accident population dose for the set of RADTRAN 5 calculations performed in the Reexamination Study for spent-fuel truck transport for each generic truck cask and type of spent fuel.[14-1]

Table 14.2 (Reexamination Study, Table 8.4) shows these expected truck accident population doses and compares them to the expected, or average, values of three incident-free population doses (stop, other, and total incident-free dose) that were also calculated in the Reexamination Study. Note that all incident-free doses have a probability of occurrence of one for an accident-free transport. Thus, the average of all the values of any specific incident-free population dose is also the expected, or mean, value of that incident-free dose risk.

The model used in RADTRAN 5 for calculating stop doses to people has two radial distance intervals centered on the stopped truck with ranges of 1 to 10 m and 10 to 800 m. The population density of the closer interval is assumed to be 30,000 people per square kilometer, and for the second interval, the population density is set equal to the value of suburban portions of the route. No shielding is assumed for the first interval, and a shielding factor of 0.2 is assumed for the second interval to account for buildings and intervening parked vehicles.

Table 14.2. Incident-Free and Accident Population Dose Risks for Truck Transport

Metric	Population Dose Risks (person-rem)					
	Incident-Free					Accident
	Stops[a]		Other[b]	Total		
	Sleep[c]	No Sleep[d,e]		Sleep[c]	No Sleep[d]	
PWR Spent Fuel; Steel-Lead-Steel Cask; 1 Assembly						
Mean =	0.427	0.0153	0.0288	0.456	0.0441	8.00E-07
Standard Deviation =	0.296	0.0106	0.0238	0.297	0.0261	8.53E-07
Maximum =	1.840	0.0657	0.1340	1.974	0.1997	4.38E-06
Minimum =	0.017	0.0006	0.0024	0.019	0.0030	4.06E-08
PWR Spent Fuel; Steel-DU-Steel Cask; 3 Assemblies						
Mean =	0.427	0.0153	0.0288	0.456	0.0441	2.29E-06
Standard Deviation =	0.296	0.0106	0.0238	0.297	0.0261	2.44E-06
Maximum =	1.840	0.0657	0.1340	1.974	0.1997	1.24E-05
Minimum =	0.017	0.0006	0.0024	0.019	0.0030	1.14E-07
BWR Spent Fuel; Steel-Lead-Steel Cask; 2 Assemblies						
Mean =	0.427	0.0153	0.0288	0.456	0.0441	3.30E-07
Standard Deviation =	0.296	0.0106	0.0238	0.297	0.0261	3.61E-07
Maximum =	1.840	0.0657	0.1340	1.974	0.1997	1.99E-06
Minimum =	0.017	0.0006	0.0024	0.019	0.0030	1.68E-08
BWR Spent Fuel; Steel-DU-Steel Cask; 7 Assemblies						
Mean =	0.427	0.0153	0.0288	0.456	0.0441	1.08E-06
Standard Deviation =	0.296	0.0106	0.0238	0.297	0.0261	1.20E-06
Maximum =	1.840	0.0657	0.1340	1.974	0.1997	6.51E-06
Minimum =	0.017	0.0006	0.0024	0.019	0.0030	5.22E-08

a. Exposures at rest, food, and refueling stops.
b. Sum of on-link, off-link, and crew doses.
c. Sleep means that the truck makes a rest stop of 8 hours once every 24 hours so the crew can sleep.
d. No Sleep means that the truck does not make any rest stops to allow the crew to sleep.
e. The No Sleep stop dose is obtained by dividing the Sleep stop dose by 28.

To illustrate the levels of stop doses, the Reexamination Study calculated stop doses for the Crystal River to Hanford route for the case of the truck stopping for the driver to sleep. Using the Reexamination Study's assumed stop time upper value estimate of 0.011 hours per kilometer, the total stop time for the 4818.5-km route is 53 hours. Thus, the aggregate stop dose for persons in the first interval over all stops was calculated to be 0.128 person-rem and 5.4×10^{-4} person-rem for the second interval.

To estimate stop doses for the case of the route traveled without stopping for sleep (for a two-person crew, alternating sleep periods en route in the truck), the Reexamination Study authors developed the following equation for the Crystal River to Hanford route, together with numerical values for that route:

$$\mathrm{Dose}_{\text{no sleep}} = \left(D_1 + D_2\right)\left(\frac{t_{\text{rest,nosleep}}}{t_{\text{rest,sleep}}}\right) + \left(D_1\left[\frac{\rho_{\text{suburban}}}{\rho_{\text{rest}}}\right]\right) + \left(D_2\left[\frac{1}{f_{\text{shielding}}}\right]\right)\left(\frac{t_{\text{inspections}}}{t_{\text{rest,sleep}}}\right)F_{\text{population}}$$

Where

D_1	=	the dose to persons exposed in the first radial interval = 0.128 person-rem
D_2	=	the dose to persons exposed in the second radial interval = 5.4×10^{-4} person-rem
$f_{\text{shielding}}$	=	the shielding factor assumed for persons in the second radial interval = 0.2
$t_{\text{rest,sleep}}$	=	the stop time at rest stops when sleep stops are made = 53 hrs
$t_{\text{rest,nosleep}}$	=	the stop time at rest stops when sleep stops are made = 1.9 hrs = (0.5 hrs) (4818.5 km/1280 km)
$t_{\text{inspections}}$	=	the time spent at inspection stops = 4.4 hrs = (70 min/60 min per hr) (4818.5 km/1280 km)
ρ_{rest}	=	the population density of the first radial interval = 3×10^4 persons/km^2
ρ_{urban}	=	the population density of urban portions of the Crystal River-to-Hanford route = 2190 persons/km^2
ρ_{suburban}	=	the population density of suburban portions of the Crystal River-to-Hanford route = 331 persons/km^2
ρ_{rural}	=	the population density of rural portions of the Crystal River-to-Hanford route = 7.5 persons/km^2
f_{urban}	=	the urban length fraction of the Crystal River-to-Hanford route = 0.01
f_{suburban}	=	the suburban length fraction of the Crystal River-to-Hanford route = 0.15
f_{rural}	=	the rural length fraction of the Crystal River-to-Hanford route = 0.84

where

$$F_{\text{population}} = f_{\text{urban}}\left(\frac{\rho_{\text{urban}}}{\rho_{\text{suburban}}}\right) + f_{\text{suburban}}\left(\frac{\rho_{\text{suburban}}}{\rho_{\text{suburban}}}\right) + f_{\text{rural}}\left(\frac{\rho_{\text{rural}}}{\rho_{\text{suburban}}}\right)$$

Substitution of values into these equations yielded the value of

$$\mathrm{Dose}_{\text{no sleep}} = 4.69 \times 10^{-3} \text{ person rem}$$

By comparison of this value to the previously calculated result for the case where the truck stopped for the driver to sleep, the Reexamination Study estimate is

$$\mathrm{Dose}_{\text{sleep}} / \mathrm{Dose}_{\text{no sleep}} = 27.4$$

Note that the Reexamination Study comparison of dose rates between accident conditions and incident-free transport resulted in the expected value of the total incident-free population dose risk exceeding the accident population dose risk by a least a factor of 2×10^4 if no stops are made for sleeping, and 1.4×10^6 if stops for sleeping are made. The conclusion was that for any truck shipment, incident-free dose risks greatly exceed accident dose risks.

Note that all the individual incident-free doses that were calculated are within regulatory limits and are small when compared with normal yearly background radiation doses.

Also note that the Reexamination Study calculation results for the expected accident population doses for PWR and BWR on a per fuel assembly basis are, respectively, 7.8E-7 and 1.6E-7 person-rem per assembly. Thus, truck transport of PWR fuel provides an expected accident population dose per assembly that is approximately five times greater than the dose would be if BWR fuel were being transported. This was expected because the failure of PWR spent fuel is expected to be about twice the value for BWR spent fuel, and the curie amounts of the radionuclides most responsible for population dose in three-year cooled, high burnup PWR fuel are about three times greater than corresponding values for BWR fuel.

14.6 Rail Cask Results: Generic Steel-Lead-Steel and Monolithic Steel Casks for 200 Representative Routes

In a calculation set analogous to the truck cask results, the Reexamination Study includes calculations of the risk of PWR and BWR spent fuel transported in rail casks.[14-1] Figure 14.5 (Reexamination Study, Figure 8.7) is the calculated CCDF for the generic steel-lead-steel rail cask carrying PWR fuel over the 200 representative routes. Figure 14.6 (Reexamination Study, Figure 8.8) is the CCDF for the same rail container carrying BWR fuel. Figure 14.7 (Reexamination Study, Figure 8.9) shows the CCDF for the monolithic steel rail cask containing PWR fuel, while Figure 14.8 (Reexamination Study, Figure 8.10) is the same cask carrying BWR fuel.

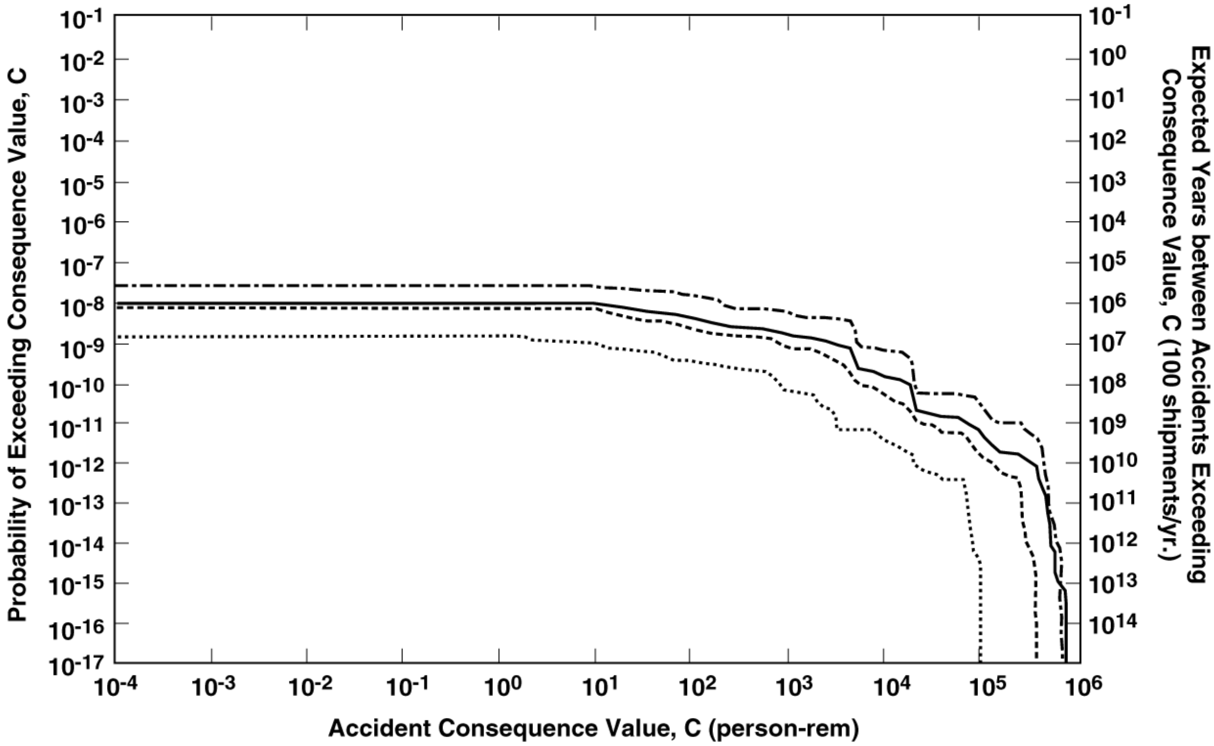

Figure 14.5. Rail Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Steel-Lead-Steel Rail Cask Over the 200 Representative Rail Routes. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 21 Representative Rail Accident Source Terms.

Mean (————) CCDF, and 95th (–-–-–-–-–), 50th (··········), and 5th (–––––––) quantiles

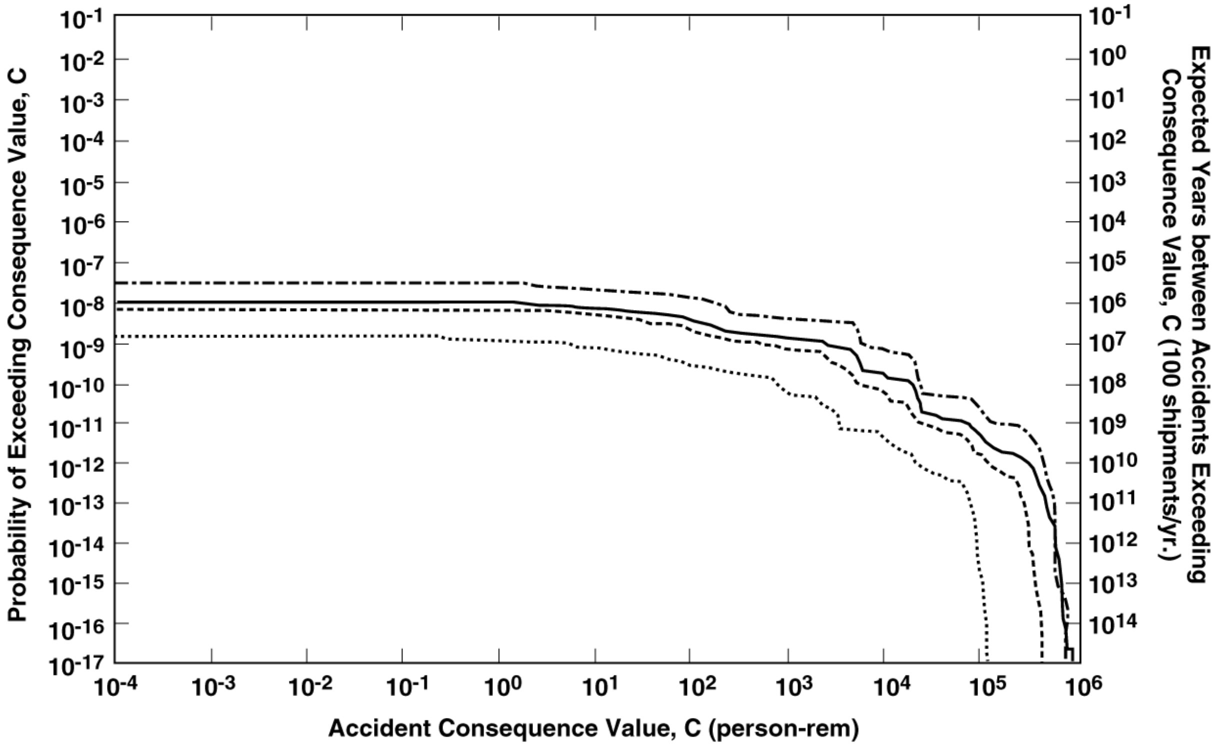

Figure 14.6. **Rail Accident Population Dose Risk CCDFs for Transport of BWR Spent Fuel in the Generic Steel-Lead-Steel Rail Cask Over the 200 Representative Rail Routes. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 21 Representative Rail Accident Source Terms.**

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

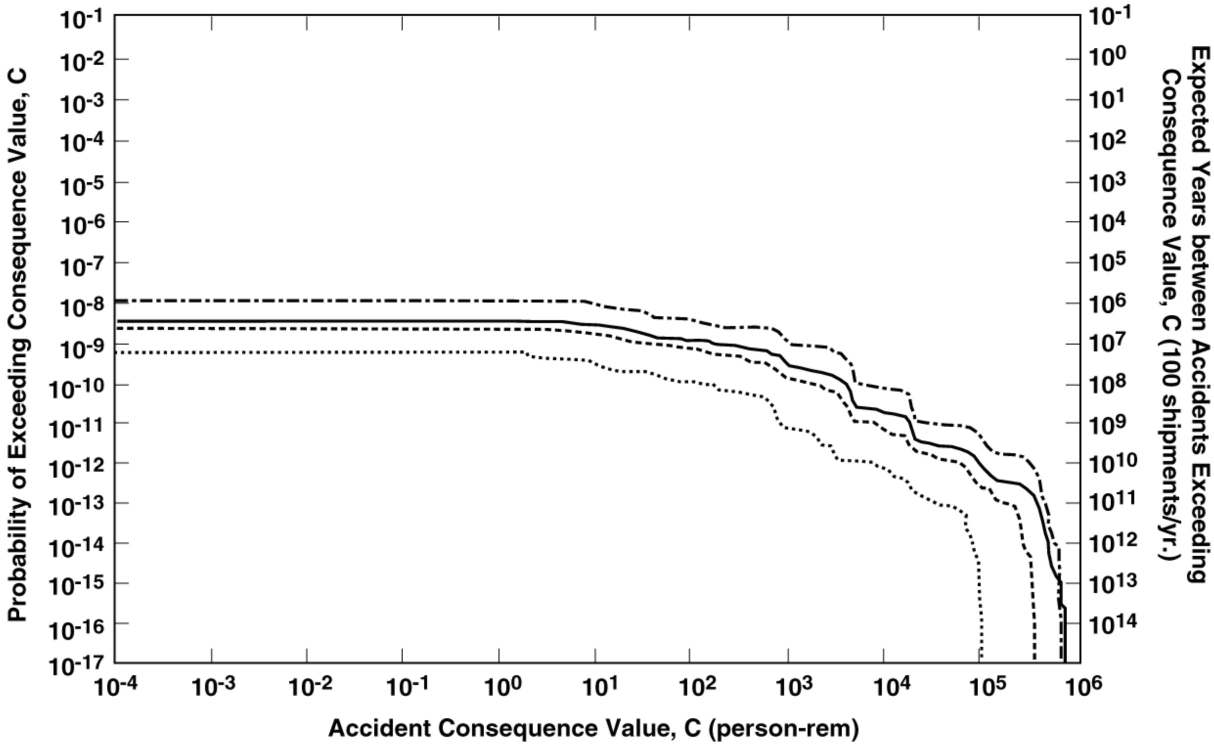

Figure 14.7. Rail Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Monolithic Steel Rail Cask Over the 200 Representative Rail Routes. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 21 Representative Rail Accident Source Terms.

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

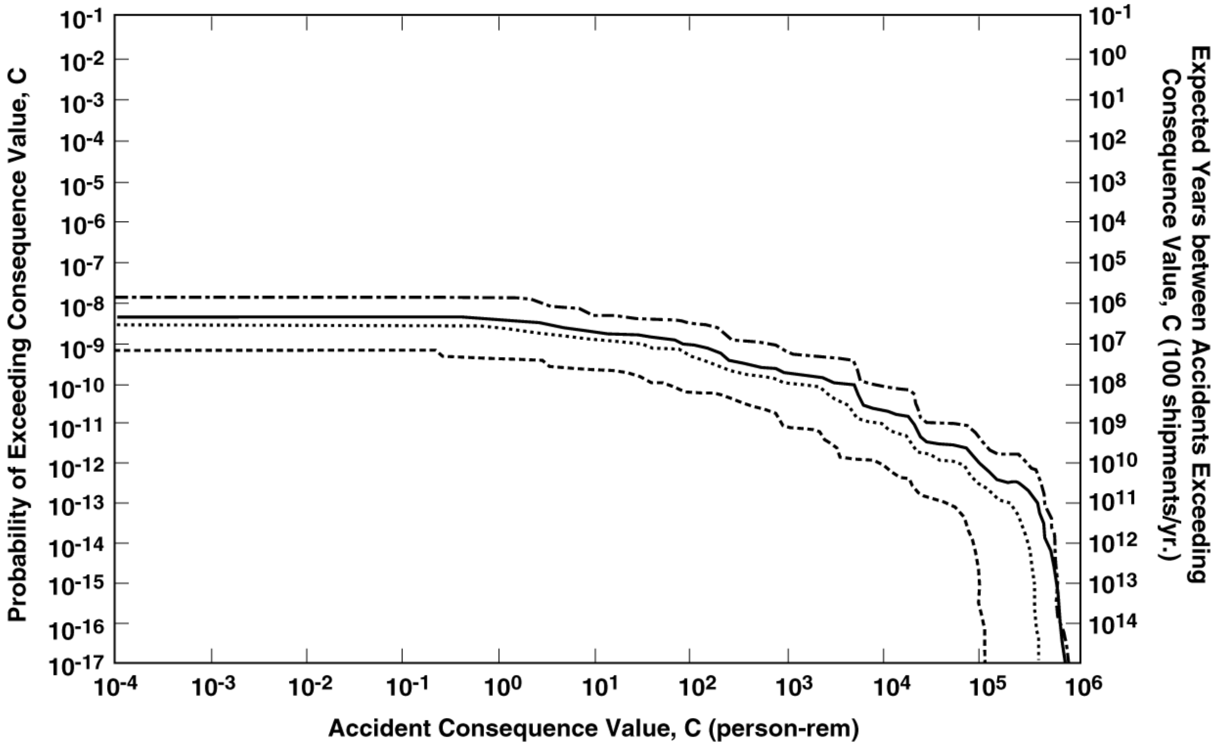

Figure 14.8. Rail Accident Population Dose Risk CCDFs for Transport of BWR Spent Fuel in the Generic Monolithic Steel Rail Cask Over the 200 Representative Rail Routes. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 21 Representative Rail Accident Source Terms.

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

Table 14.3 (Reexamination Study, Table 8.5) compares the expected incident-free population doses to the corresponding accident-related population doses. Similar to the truck transport cases, the values of total incident-free rail transport population dose rate greatly exceed the values for accidents by factors of approximately 10^3 to 10^4. Note that the rail incident-free stop doses are lower (by a factor of 3.6) than for the incident-free rail transport values. This is in contrast to the result obtained in the Reexamination Study for the case of truck transport, and is explained by the lower population densities and the greater shielding at rail yards than at truck stops.

Table 14.3. Incident-Free Population Dose Risks for Rail Transport

Metric	Population Dose Risks (person-rem)			
	Incident-Free			Accident
	Stops[a]	Other[b]	Total	
PWR Spent Fuel; Steel-Lead-Steel Cask; 24 Assembly				
Mean	4.37E-03	1.59E-02	2.03E-02	9.43E-06
Standard Deviation	2.58E-03	1.38E-02	1.40E-02	1.18E-05
Maximum	1.29E-02	8.26E-02	9.55E-02	6.32E-05
Minimum	1.73E-03	3.57E-04	2.08E-03	3.39E-08
PWR Spent Fuel; Monolithic Steel Cask; 24 Assemblies				
Mean	4.37E-03	1.59E-02	2.03E-02	1.99E-06
Standard Deviation	2.58E-03	1.38E-02	1.40E-02	2.47E-06
Maximum	1.29E-02	8.26E-02	9.55E-02	1.35E-05
Minimum	1.73E-03	3.57E-04	2.08E-03	8.08E-09
BWR Spent Fuel; Steel-Lead-Steel Cask; 52 Assemblies				
Mean	4.37E-03	1.59E-02	2.03E-02	9.23E-06
Standard Deviation	2.58E-03	1.38E-02	1.40E-02	1.18E-05
Maximum	1.29E-02	8.26E-02	9.55E-02	6.19E-05
Minimum	1.73E-03	3.57E-04	2.08E-03	2.79E-08
BWR Spent Fuel; Monolithic Cask; 52 Assemblies				
Mean	4.37E-03	1.59E-02	2.03E-02	1.46E-06
Standard Deviation	2.58E-03	1.38E-02	1.40E-02	1.86E-06
Maximum	1.29E-02	8.26E-02	9.55E-02	9.94E-06
Minimum	1.73E-03	3.57E-04	2.08E-03	4.87E-09

Table 14.3 also indicates that the accident population dose risk per assembly for PWR spent fuel exceeds the corresponding value for BWR by factors of between 2 to 3. The cause of this difference is the same as for truck casks, namely higher rod failure fractions for PWR fuel and curie amounts of radionuclides responsible for a population dose that is higher for PWR fuel than for corresponding three-year cooled, high burnup BWR fuel.

14.7 Comparison of Truck and Rail Spent Fuel Transport Mean Risks

Comparing Tables 14.2 and 14.3 shows that incident-free doses for truck-stop doses exceed rail-stop doses by a factor of 100 if the trucks make sleep stops, and by a factor of 35 if no sleep stops are made. Total truck incident-free doses exceed total train incident-free doses by a factor of 22.5 if truck sleep stops are made, and by a factor of 2 if stops are not made for sleep. Because the same routing was used in the Reexamination Study for rail and truck transport, the other truck and train incident-free population doses are quite similar. These dose rates are similar because the surface dose rates are limited by regulation. Thus, even though rail casks carry many more assemblies than truck casks, inner assemblies are shielded by outer assemblies, and the cask is designed to limit the external dose rate so that it does not exceed the regulatory limit.

The Reexamination Study authors also point out that because the truck casks contain fewer assemblies than rail casks, more truck shipments are required to equal the number of assemblies carried by a rail shipment. In fact, between eight and 24 truck shipments of PWR spent fuel would be required to transport an amount equal to one rail shipment. For BWR fuel, between 7.4 and 26 truck shipments would be required to transport an amount equivalent to that transported in one rail cask. Thus, according to the Reexamination Study, truck incident-free doses would exceed rail incident-free doses by factors between 180 and 585 for a given shipment campaign. This factor is really not a problem because all individual incident-free doses are within regulatory limits and are small compared to normal yearly background radiation doses.

Accident dose rates for rail accidents are expected to be larger than for single truck accidents because of the larger number of assemblies carried by rail. Comparison of Tables 14.2 and 14.3 shows truck accident dose risk similar to or about ten times greater than mean rail accident dose risks. However, as pointed out in the Reexamination Study, for any shipment campaign, truck transport will require eight to 26 times as many shipments as for rail. Accordingly, truck accident dose risks will exceed rail by eight to 26 times on a per campaign basis.

14.8 Illustrative Real Routes

The results presented above were calculated in the Reexamination Study by using 200 sets of RADTRAN 5 calculations containing 200 different hypothetical truck or rail routes, typical of but not exact to any continental United States routes. The authors investigated four real routes for illustration purposes together with the NUREG-0170 representative truck and rail routes and compared the results to the 200 representative truck or rail routes discussed previously. For these calculations, the transport of high burnup PWR fuel was examined in the generic steel-lead-steel truck cask and in the generic monolithic-steel-rail cask. Table 14.4 (Reexamination Study, Table 8.7) shows the parameter values.[14-1]

Table 14.4. NUREG-0170 and Illustrative Real Truck and Rail Routes

Origin	Destination	Length (km)	Fraction of Total Length			Population Density[a]			Stop Time[b]
			Rural	Subur ban	Urban	Rural	Subur ban	Urban	
Truck Routes									
Crystal River, FL	Hanford Site, WA	4818.5	0.84	0.15	0.01	7.5	331	2190	53.0
Maine Yankee, ME	Skull Valley, UT	4228.7	0.74	0.24	0.02	9.2	296	2286	46.5
Maine Yankee, ME	Savannah River Site, SC	1917.5	0.52	0.43	0.05	18.3	282	2565	21.0
Kewaunee, WI	Savannah River Site, SC	1765.0	0.63	0.32	0.05	16.3	358	2452	19.4
NUREG-0170		2530.0	0.90	0.05	0.05	6.0	719	3861	8.0
Route Parameter Distribution Mean Values		2550.0	0.76	0.23	0.01	10.1	336	2195	28.0
Rail Routes									
Crystal River, FL	Hanford Site, WA	5178.6	0.83	0.15	0.02	7.9	360	2063	231
Maine Yankee, ME	Skull Valley, UT	4488.7	0.75	0.22	0.03	8.9	337	2429	208
Maine Yankee, ME	Savannah River Site, SC	2252.7	0.52	0.38	0.10	14.3	325	2738	134
Kewaunee, WI	Savannah River Site, SC	1917.2	0.64	0.32	0.04	14.1	351	2268	122
NUREG-0170		1210.0	0.90	0.05	0.05	6.0	719	3861	24
Route Parameter Distribution Mean Values		2560.0	0.75	0.22	0.03	9.6	356	2280	144

a. People per square kilometer.
b. Sum of all stop durations (hours) for the entire shipment. For truck shipments, includes stop time for sleep stops.

14.8.1 Steel-Lead-Steel Truck Cask

Figures 14.9 through 14.13 (Reexamination Study, Figures 8.12 through 8.16) are the CCDFs for accident dose risks for the steel-lead-steel truck cask over the four routes chosen for illustration and the NUREG-0170 representative truck route. Table 14.5 (Reexamination Study, Table 8.8) presents the incident-free dose risks as calculated by use of RADTRAN 5 in the Reexamination Study.[14-1]

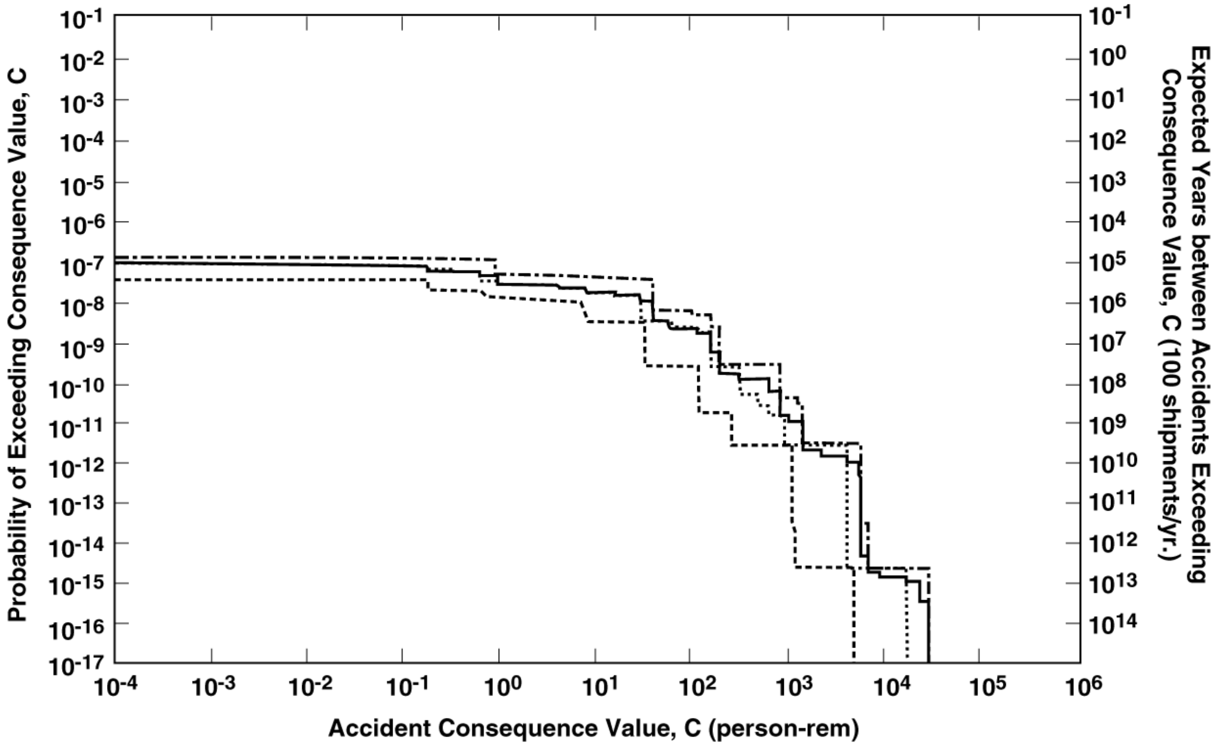

Figure 14.9. Truck Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Steel-Lead-Steel Truck Cask Over the Crystal River to Hanford Illustrative Truck Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 19 Representative Truck Accident Source Terms.

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

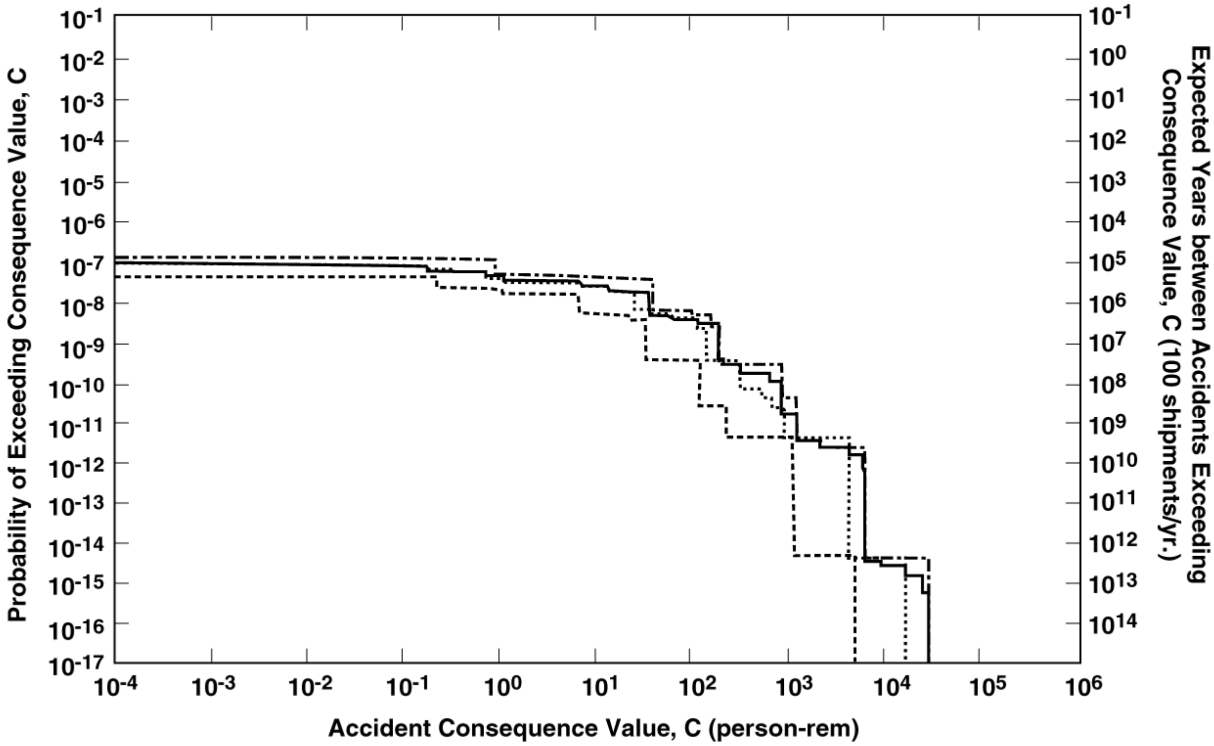

Figure 14.10. Truck Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Steel-Lead-Steel Truck Cask Over the Maine Yankee to Skull Valley Illustrative Truck Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 19 Representative Truck Accident Source Terms.

Mean (————) CCDF, and 95th (–·–·–·–·–), 50th (··········), and 5th (–––––––) quantiles

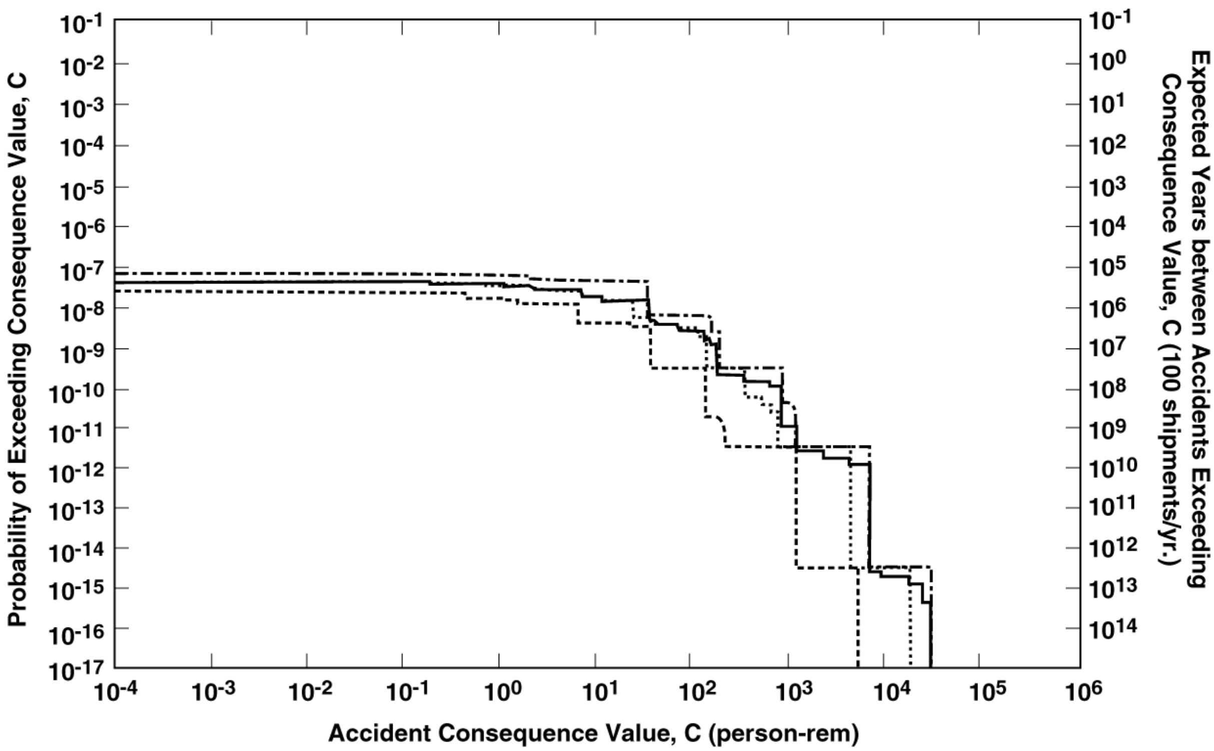

Figure 14.11. Truck Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Steel-Lead-Steel Truck Cask Over the Marine Yankee to Savannah River Site Illustrative Truck Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 19 Representative Truck Accident Source Terms.

Mean (———) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

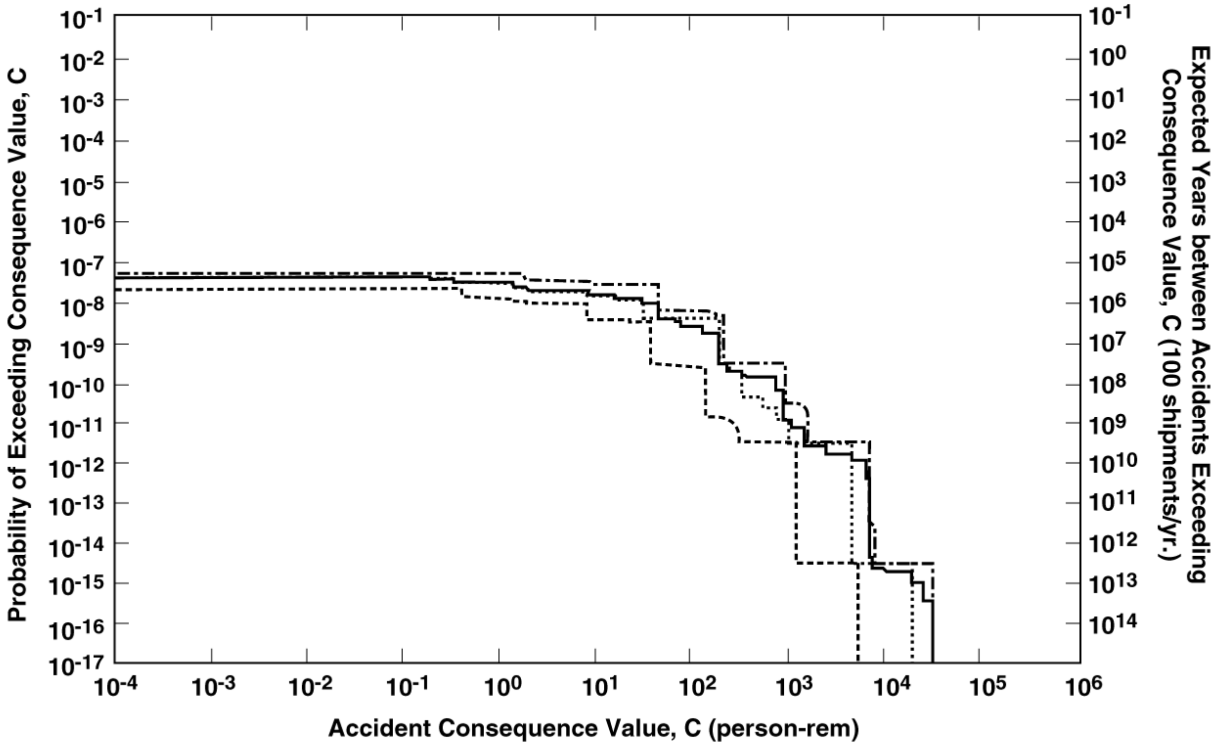

Figure 14.12. Truck Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Steel-Lead-Steel Truck Cask Over the Kewaunee to Savannah River Site Illustrative Truck Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 19 Representative Truck Accident Source Terms.

Mean (———) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

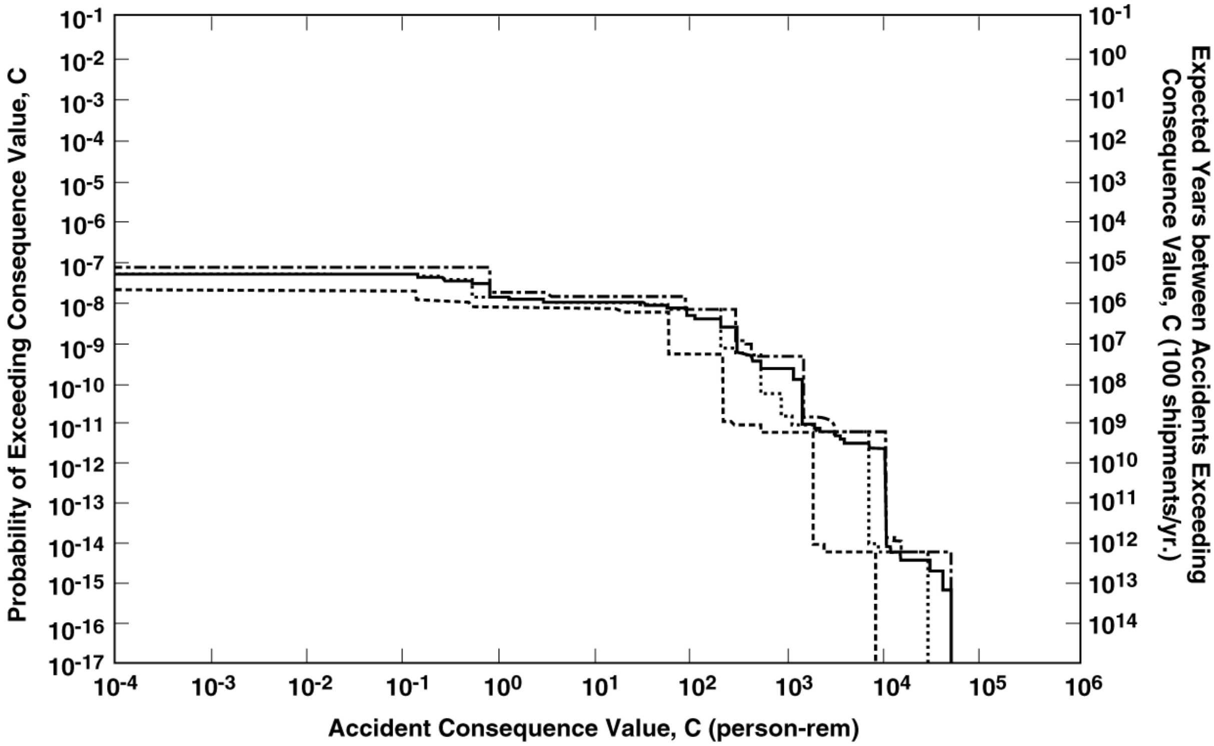

Figure 14.13. **Truck Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Steel-Lead-Steel Truck Cask Over the NUREG-0170 Representative Truck Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 19 Representative Truck Accident Source Terms.**

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

Table 14.5. Incident-Free Population Dose Risks for Truck Transport of PWR Spent Fuel in a Generic Steel-Lead-Steel Truck Cask over Illustrative Routes

Metric	Population Dose Risks (person-rem)					
	Incident-Free					Accident
	Stops[a]		Other[b]	Total		
	Sleep[c]	No Sleep[d,e]		Sleep[c]	No Sleep[d]	
Crystal River Nuclear Plant to Hanford Site						
Mean	1.470	0.0525	0.0581	1.530	0.111	9.53E-07
Standard Deviation	0.722	0.0258	0.0281	0.722	0.038	5.92E-07
Maine Yankee Nuclear Plant to Skull Valley						
Mean	1.300	0.0464	0.0524	1.350	0.099	1.29E-06
Standard Deviation	0.637	0.0228	0.0252	0.637	0.034	7.81E-07
Maine Yankee Nuclear Plant to Savannah River Site						
Mean	0.585	0.0209	0.0252	0.610	0.046	1.14E-06
Standard Deviation	0.288	0.0103	0.0122	0.288	0.016	6.73E-07
Kewaunee Nuclear Plant to Savannah River Site						
Mean	0.541	0.0193	0.0231	0.564	0.042	1.01E-07
Standard Deviation	0.257	0.0092	0.0112	0.257	0.011	5.93E-07
NUREG-0170 Truck Route						
Mean	0.779	0.0321	0.0304	0.810	0.063	1.28E-06
Standard Deviation	0.383	0.0137	0.0147	0.383	0.020	6.68E-07

a. Exposures at rest, food, and refueling stops.
b. Sum of on-link, off-link, and crew doses.
c. Sleep means that the truck makes a rest stop of 8 hours once every 24 hours so the crew can sleep.
d. No Sleep means that the truck doesn't make any rest stops to allow the crew to sleep.
e. The No Sleep stop dose is obtained by dividing the Sleep stop dose by 28.

Comparison of these truck transport results to the previous section, where 200 representative truck routes were constructed by LHS sampling, shows similar results for the accident risk CCDFs as well as for the incident-free doses.

14.8.2 Monolithic Steel Rail Cask

Figures 14.14 through 14.18 (Reexamination Study, Figures 8.18 through 8.22) are CCDFs for the accident population dose risks, and Table 14.6 (Reexamination Study, Table 8.9) illustrates the incident-free doses as calculated by RADTRAN 5 for the generic monolithic steel cask over the four illustrative rail routes and the NUREG-0170 rail route.

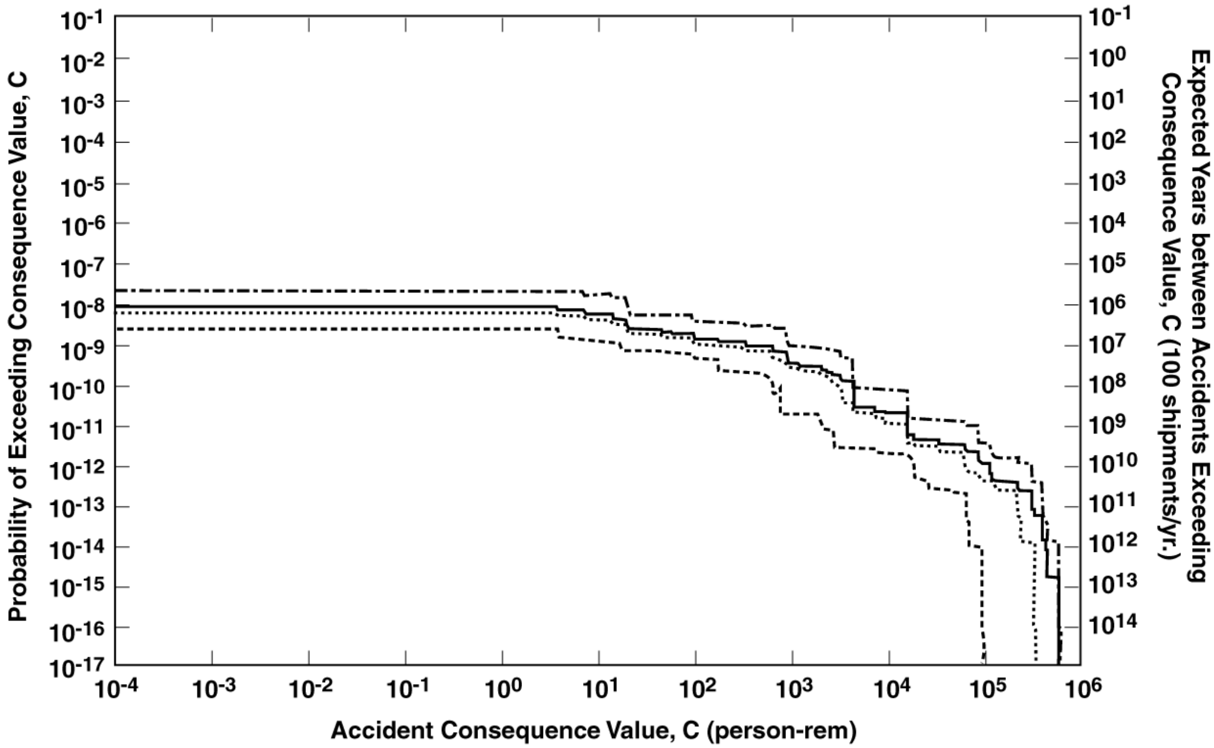

Figure 14.14. Rail Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Monolithic Steel Rail Cask Over the Crystal River to Hanford Illustrative Rail Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 21 Representative Rail Accident Source Terms.

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

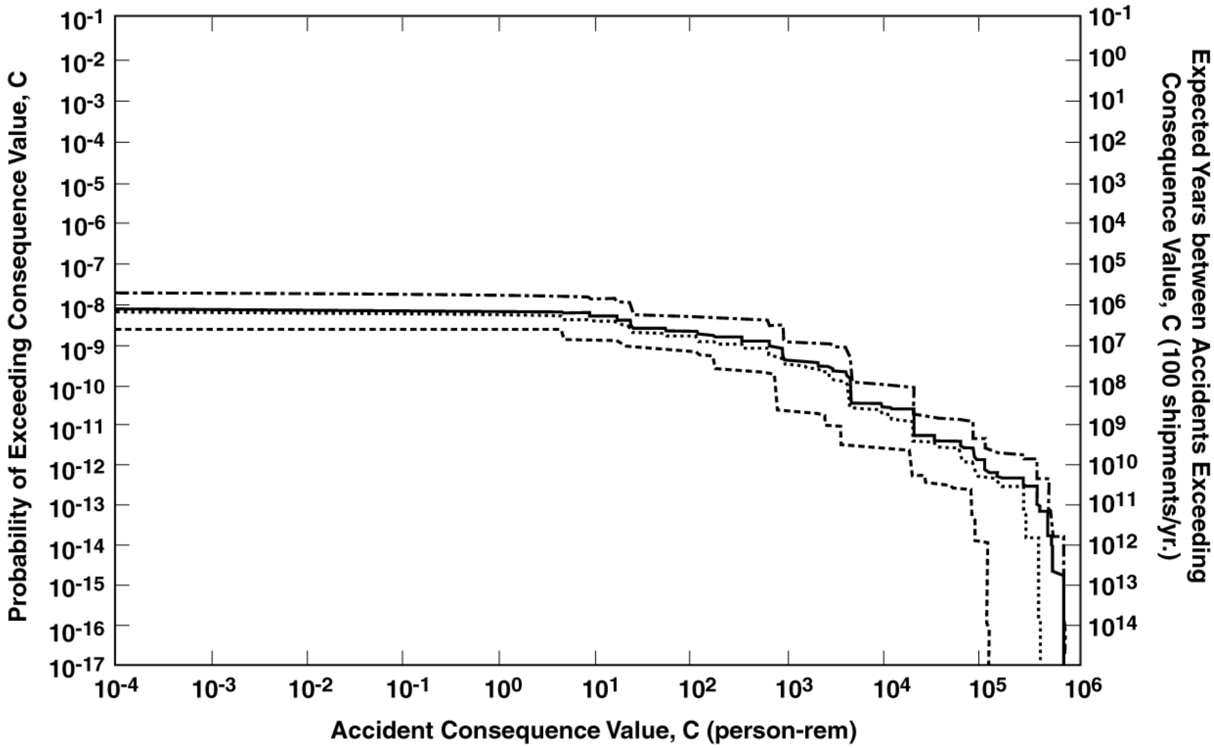

Figure 14.15. Rail Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Monolithic Steel Rail Cask Over the Maine Yankee to Skull Valley Illustrative Rail Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 21 Representative Rail Accident Source Terms.

Mean (————) CCDF, and 95th (–·–·–·–·–), 50th (··········), and 5th (–––––––) quantiles

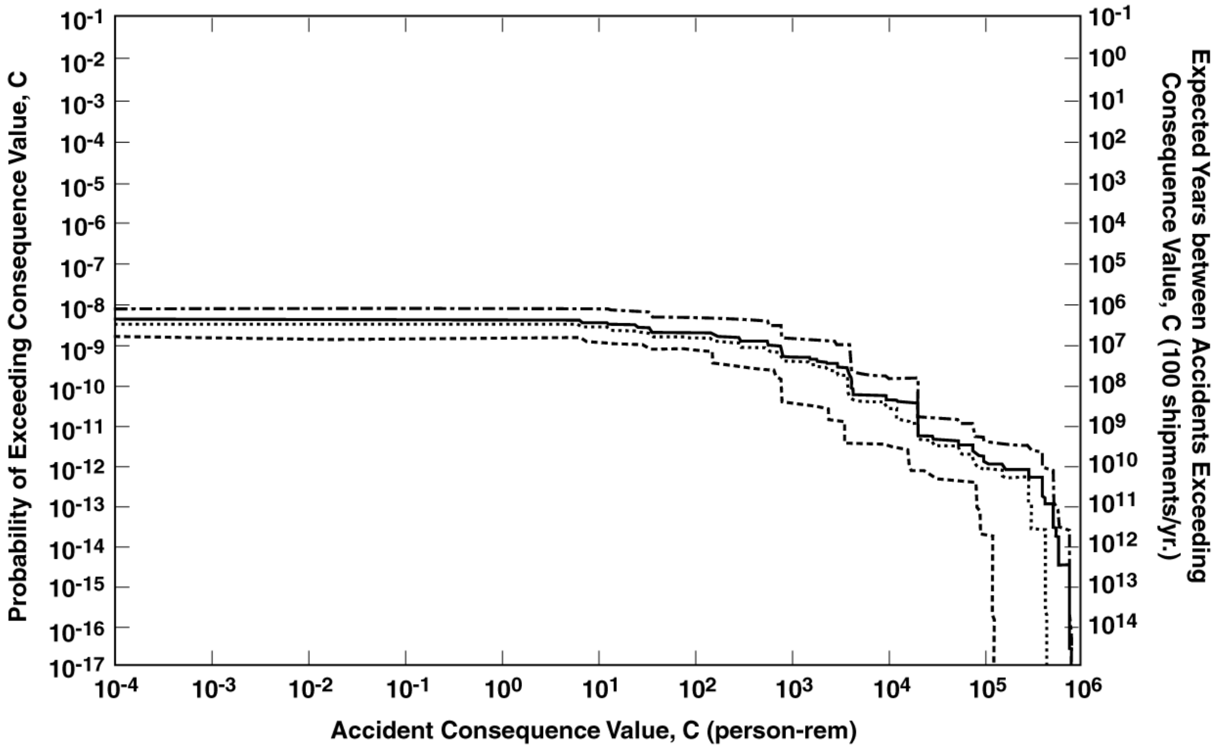

Figure 14.16. Rail Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Monolithic Steel Rail Cask Over the Maine Yankee to Savannah River Site Illustrative Rail Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 21 Representative Rail Accident Source Terms.

Mean (———) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

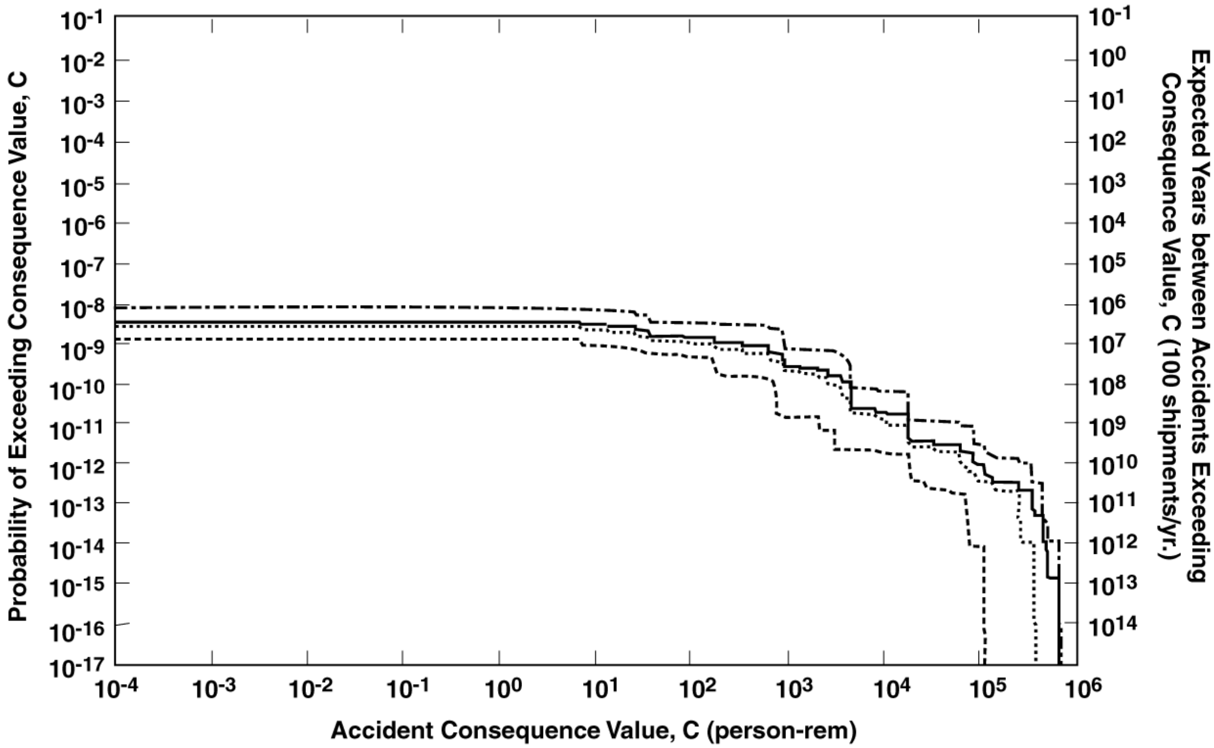

Figure 14.17. Rail Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Monolithic Steel Rail Cask Over the Kewaunee to Savannah River Site Illustrative Rail Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 21 Representative Rail Accident Source Terms.

Mean (———) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

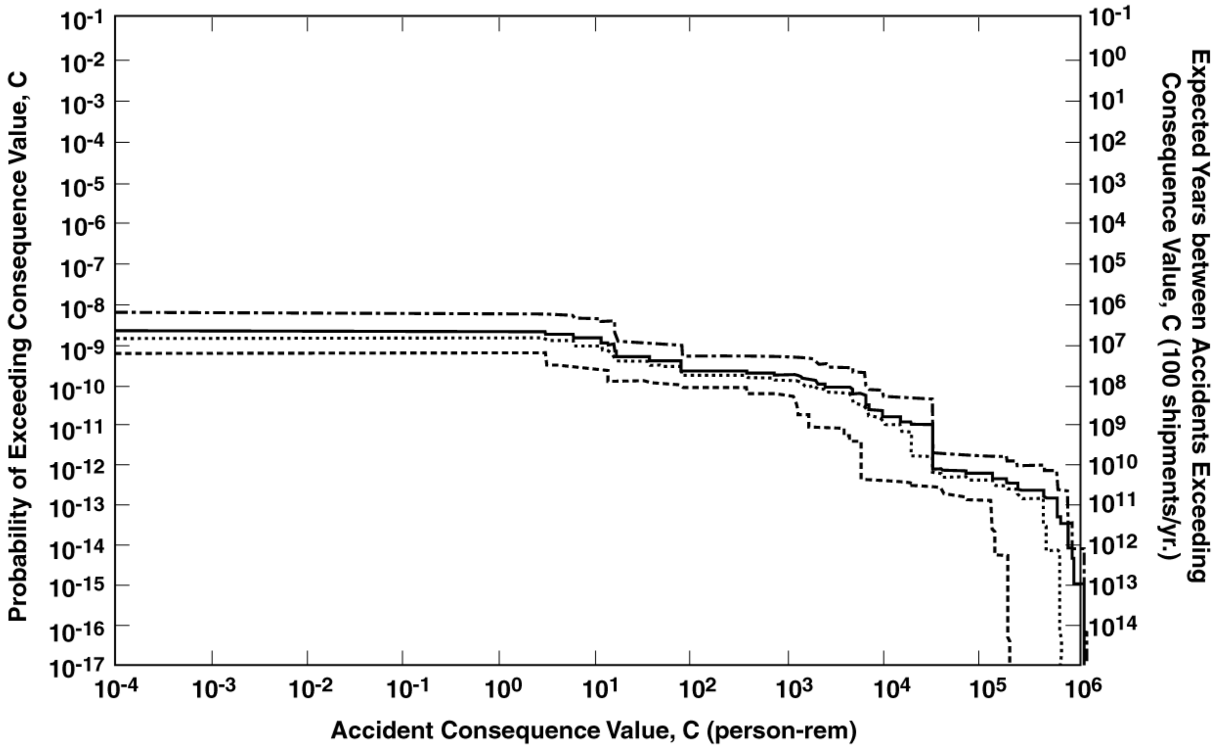

Figure 14.18. Rail Accident Population Dose Risk CCDFs for Transport of PWR Spent Fuel in the Generic Monolithic Steel Rail Cask Over the NUREG-0170 Representative Rail Route. Each Underlying RADTRAN 5 Calculation Generated Results for all of the 21 Representative Rail Accident Source Terms.

Mean (————) CCDF, and 95th (-·-·-·-·-), 50th (··········), and 5th (-------) quantiles

Table 14.6. Incident-Free Population Dose Risks for Rail Transport of PWR Spent Fuel in a Generic Monolithic Steel Rail Cask over Illustrative Routes

Metric	Population Dose Risks (person-rem)			
	Incident-Free			Accident
	Stops[a]	Other[b]	Total	
Crystal River Nuclear Plant to Hanford Site				
Mean	9.70E-03	2.89E-02	3.86E-02	2.44E-06
Standard Deviation	5.71E-03	1.71E-02	1.80E-02	2.08E-06
Maine Yankee Nuclear Plant to Skull Valley				
Mean	1.19E-02	2.75E-02	3.69E-02	3.25E-06
Standard Deviation	7.00E-03	1.62E-02	1.77E-02	2.77E-06
Maine Yankee Nuclear Plant to Savannah River Site				
Mean	1.02E-02	1.66E-02	2.70E-02	3.79E-06
Standard Deviation	6.05E-03	9.84E-03	1.15E-02	3.27E-06
Kewaunee Nuclear Plant to Savannah River Site				
Mean	7.61E-03	1.33E-02	2.09E-02	1.95E-06
Standard Deviation	4.50E-03	7.87E-03	9.06E-03	1.68E-06
NUREG-0170 Truck Route				
Mean	2.05E-03	6.46E-03	8.51E-03	1.11E-06
Standard Deviation	1.21E-03	3.82E-03	4.01E-03	1.03E-06

a. Exposures at rest and refueling stops.
b. Sum of on-link, off-link, and crew doses.

Comparison of these rail transport results to the previous corresponding section, where 200 representative rail routes were constructed by LHS sampling, shows similar results for the accident risk CCDFs as well as for the incident-free doses.

14.9 Rail Routes with Heavy Haul Segments and Intermodal Transfers

In the Reexamination Study, risks were evaluated for the case of spent fuel transport in rail casks by special heavy haul truck transport over short route segments when a nuclear power plant or storage site does not have a rail spur. The short-haul segments that were evaluated were:

- Main Yankee nuclear power plant to railhead at Pejepscot, Maine
- Kewaunee nuclear power plant to railhead at Kewaunee, Wisconsin
- Proposed Skull Valley, Utah, interim storage site to railhead at Timpie, Utah.

RADTRAN 5 calculations from the Reexamination Study give the population dose risks associated with each of these three heavy-haul routes. The route parameters for these calculations included urban, suburban, and rural link distances; population densities; and accident rates. These parameters are shown in Table 14.7 (Reexamination Study, Table 8.10).

Table 14.7. Route Parameters for Heavy-Haul Truck Transport Segments

Aggregate Link	Length (km)	Population Density (persons per km^2)	Accident Rate (accidents per km)
Maine Yankee Nuclear Plant to the Railhead at Pejepscot Mills			
Rural	15	31.6	2.2E-7
Suburban	21	318	4.1E-7
Urban	4.0	2570	5.2E-7
Kewaunee Nuclear Plant to the Railhead at Kewaunee			
Rural	17	38.5	2.2E-7
Suburban	1.0	90.8	4.1E-7
Urban	0.0	NA	NA
Railhead at Timpie to the Proposed Skull Valley Interim Storage Site			
Rural	46	0.21	2.2E-7
Suburban	0.0	NA	NA
Urban	0.0	NA	NA

Other considerations were made, taking into account factors involved with heavy haul transport such as low-speed transport under escort, resulting in the risk values presented in Table 14.8 (Reexamination Study, Table 8.11).

Table 14.8. Heavy-Haul Incident-Free and Accident Population Dose Risks

Metric	Population Dose Risks (person-rem)				
	Incident-Free				
	Stops[a,b]	Other[c]	Total	Accident	Handling[d]
Maine Yankee Nuclear Plant to the Railhead at Pejepscot Mills					
Mean	3.8E-07	5.1E-04	5.1E-04	8.0E-08	1.4E-02
Standard Deviation	2.2E-07	3.0E-04	3.0E-04	4.4E-08	8.5E-03
Kewaunee Nuclear Plant to the Railhead at Kewaunee					
Mean	2.1E-07	1.7E-04	1.7E-04	2.2E-09	1.4E-02
Standard Deviation	1.2E-07	1.1E-04	1.1E-04	1.4E-09	8.5E-0E
Railhead at Timpie to the Proposed Skull Valley Interim Storage Site					
Mean	4.5E-10	4.2E-04	4.2E-04	2.6E-11	1.4E-02
Standard Deviation	2.6E-10	2.7E-04	2.7E-04	1.8E-11	8.5E-03

a. Intermodal transfer stop dose to members of the public.
b. Short segment lengths mean no stops are made for inspections or to refuel, eat, or sleep.
c. Sum of on-link, off-link, and crew doses.
d. Intermodal transfer dose risk to cask handlers.

14.10 Loss of Shielding Accidents

A loss of shielding (LOS) model was developed by the Reexamination Study researchers to evaluate population risk using the RADTRAN STOP model. Three factors were calculated for each accident severity category:

- Severity fraction for each LOS accident case
- Dose rate at 1m from the cask surface after the LOS accident
- Maximum dimension and geometry of the unshielded area.

Severity fractions are determined by accident case scenarios with respect to accident speed ranges with and without fire conditions.

The Reexamination Study authors evaluated the unshielded length as a result of lead slump by finite element structural analysis for end drops. This unshielded length to total cask length, the LOS fraction, is used to calculate a source-strength multiplier. The source-strength multiplier is the number by which the maximum dose rate at 1 m from an unshielded fuel assembly is multiplied to yield the maximum dose rate 1 m from the cask centerline averaged over the entire cask surface for both the shielded and the unshielded portions of that surface.

Table 14.9 (Reexamination Study, Table 8.12) lists the accident conditions and calculation parameters for the LOS cases evaluated for train accidents. Table 14.10 (Reexamination Study, Table 8.13) presents the results of the LOS risk calculations.[14-1]

Table 14.9. Values of Severity Fractions, LOS Fractions, and Source-Strength Multipliers for Ten LOS Accident Cases

LOS Case	Accident Type	Accident Conditions	Train Accident Cases	Sum Cases Probabilities	Severity Fraction	LOS Fraction	Source-Strength Multipliers
1	Collision	end	4,5,6	3.049E-05	1.707E-06	0.052	0.215
2	Collision	end	1,7,8,9	8.273E-06	4.633E-07	0.158	0.637
3	Collision	end	2,10,11,12	5.730E-07	3.209E-08	0.264	1.017
4	Collision	end	3,13,14,15	4.524E-09	2.534E-10	0.368	1.336
5	Collision	corner	4,5,6	3.049E-05	2.201E-05	0.033	0.137
6	Collision	corner	1,7,8,9	8.273E-06	5.973E-06	0.096	0.394
7	Collision	corner	2,10,11,12	5.730E-07	4.137E-07	0.158	0.637
8	Collision	corner	3,13,14,15	4.524E-09	3.266E-09	0.255	0.986
9	Fire Only	T > 350°C	20	4.905E-05	4.905E-05	0.029	0.120
10	Fire	T > 350°C & puncture	16,17,18,19	4.150E-10	1.660E-09	0.500	1.668
11	No LOS				9.999E-01	0.000	

Table 14.10. Results of Loss of Shielding Risk Calculation

Case	Pop. Zone	Length (km)	Acc. Rate (per km)	Sev. Frac.	Probability	Consequence (dose,rem)	Dose Risk
1	Rural	1777	4.40E-08	1.71E-06	1.34E-10	0.0021	2.81E-13
	Suburban	541	4.40E-08	1.71E-06	4.07E-11	0.06	2.44E-12
	Urban	35	4.40E-08	1.71E-06	2.63E-12	0.0051	1.34E-14
2	Rural	1777	4.40E-08	4.63E-07	3.63E-11	0.0071	2.57E-13
	Suburban	541	4.40E-08	4.63E-07	1.10E-11	0.206	2.27E-12
	Urban	35	4.40E-08	4.63E-07	7.13E-13	0.0175	1.25E-14
3	Rural	1777	4.40E-08	3.21E-08	2.51E-12	0.0133	3.34E-14
	Suburban	541	4.40E-08	3.21E-08	7.64E-13	0.385	2.94E-13
	Urban	35	4.40E-08	3.21E-08	4.94E-14	0.0326	1.61E-15
4	Rural	1777	4.40E-08	2.53E-10	1.98E-14	0.0221	4.37E-16
	Suburban	541	4.40E-08	2.53E-10	6.02E-15	0.639	3.85E-15
	Urban	35	4.40E-08	2.53E-10	3.90E-16	0.0541	2.11E-17
5	Rural	1777	4.40E-08	2.20E-05	1.72E-09	0.0013	2.24E-12
	Suburban	541	4.40E-08	2.20E-05	5.24E-10	0.0373	1.95E-11
	Urban	35	4.40E-08	2.20E-05	3.39E-11	0.0032	1.08E-13
6	Rural	1777	4.40E-08	5.97E-06	4.67E-10	0.004	1.87E-12
	Suburban	541	4.40E-08	5.97E-06	1.42E-10	0.115	1.63E-11
	Urban	35	4.40E-08	5.97E-06	9.19E-12	0.0097	8.97E-14
7	Rural	1777	4.40E-08	4.14E-07	3.24E-11	0.0071	2.30E-13
	Suburban	541	4.40E-08	4.14E-07	9.85E-12	0.206	2.03E-12
	Urban	35	4.40E-08	4.14E-07	6.38E-13	0.0175	1.12E-14
8	Rural	1777	4.40E-08	3.27E-09	2.56E-13	0.013	3.32E-15
	Suburban	541	4.40E-08	3.27E-09	7.78E-14	0.377	2.93E-14
	Urban	35	4.40E-08	3.27E-09	5.04E-15	0.032	1.61E-16
9	Rural	1777	4.40E-08	4.91E-05	3.84E-09	0.0011	4.22E-12
	Suburban	541	4.40E-08	4.91E-05	1.17E-09	0.0331	3.86E-11
	Urban	35	4.40E-08	4.91E-05	7.55E-11	0.0028	2.12E-13
10	Rural	1777	4.40E-08	1.66E-09	1.30E-13	0.035	4.54E-15
	Suburban	541	4.40E-08	1.66E-09	3.95E-14	1.01	3.99E-14
	Urban	35	4.40E-08	1.66E-09	2.56E-15	0.0858	2.19E-16
Total							9.12E-11

Note that the total LOS risk (probability times consequence), compared to the earlier PWR steel-lead-steel rail cask results given in Table 14.2, is much smaller than the dispersion accident risk value. Also note that the sum of these two risks, that is, an accident incurring both dispersion and loss of shielding, is within the variability of the dispersion value itself.[14-1]

14.11 Individual Dose Rates

In addition to the population dose rates presented previously in the form of CCDFs, RADTRAN also provides doses downwind of the accident. These doses are dependent on the source term magnitude and assume that an individual is outdoors in the path of the radioactive cloud for the duration of the accident and its subsequent release of radioactive material. Despite these unlikely conditions and the unlikely possibility of an accidental release, there is a potential for persons close to the accident to receive a relatively large dose of radiation.[14-1]

RADTRAN calculations are very conservative; however, while calculated doses may be high, they are not large enough to predict an early fatality from radiation. Note that because of the high degree of conservatism in the RADTRAN calculations, the combination of circumstances required to release material from a modern spent fuel cask are so improbable as to be essentially impossible.[14-1]

14.12 Population Dose Risks for Transport of the Entire 1994 Spent Fuel Inventory

The calculated results from the Reexamination Study, which are discussed in this report, were relative to single shipments of one Type B spent fuel cask by truck or train. The authors of the Reexamination Study also calculated population risks for the shipment of the entire 1994 United States inventory of commercial BWR and PWR spent fuel.[14-1]

Table 14.11 (Reexamination Study, Table 8.14) lists this inventory, including the total numbers of BWR and PWR assemblies, the number of truck or rail shipments required, the incident-free and accident population dose risks, and the sum for transport of the entire 1994 United States inventory.

Table 14.11. Incident-Free and Accident Population Dose Risks for Shipment of the Entire 1994 Spent Fuel Inventory (person-rem)

Spent Fuel Type	Rail Shipments		Truck Shipments	
	Monolithic Steel Cask	Steel-Lead-Steel Cask	Steel-Lead-Steel Cask	Steel-DU-Steel Cask
	Assemblies in Total 1994 Inventory			
BWR	60144			
PWR	44598			
	Assemblies per Cask			
BWR	52	52	2	7
PWR	24	24	1	1
	Required Number of Shipments			
BWR	1157	1157	30072	8592
PWR	1858	1858	44598	14866
Total	**3015**	**3015**	**74670**	**23458**
	Incident-Free Stop Dose Risks[a,b,c]			
BWR	5.1	5.1	460	130
PWR	8.1	8.1	680	230
Total	**13.2**	**13.2**	**1140**	**360**
	Other Incident-Free Population Dose Risks[a,b]			
BWR	18.4	18.4	870	250
PWR	29.5	29.5	1280	430
Total	**47.9**	**47.9**	**2150**	**680**
	Total Incident-Free Population Dose Risks[a,b]			
BWR	24	24	1330	380
PWR	37	37	1960	660
Total	**61**	**61**	**3290**	**1040**
	Accident Population Dose Risks[a]			
BWR	0.0017	0.011	0.010	0.0093
PWR	0.0037	0.018	0.036	0.034
Total	**0.0054**	**0.028**	**0.046**	**0.043**

a. Values have been rounded to two significant figures.
b. Because the probability of occurrence of incident-free doses is 1.0, incident-free doses and incident-free dose risks have the same values.
c. Truck stop dose risks assume shipment without stops to sleep.

Note that Table 14.11 clearly shows accident dose risks to be negligible as compared to incident-free dose risks. Also note the large variation in risk between truck and rail and also between cask types.

14.13 Risk Summary and Conclusions

The Reexamination Study provided a recent and comprehensive set of calculations that are described in this report for their value in illustrating current risk analysis methodology in evaluating population risk as a result of transporting radioactive materials. The Reexamination

Study was concerned with truck and rail transport of spent fuel; however, the methods used are applicable to any radioactive material. MOX spent fuel transport will provide slightly different risk values for population dose rates than were calculated for uranium dioxide PWR and BWR spent fuel by the Reexamination Study, but the methods used will be identical for both types of fuel.

In the Reexamination Study, three representative sets of data were developed:

- Generic design data for four representative casks: steel-lead-steel truck casks, steel-lead-steel rail casks, steel-DU-steel truck cask, and monolithic steel rail cask;

- 200 sets of representative truck and rail route data determined by Monte Carlo Latin Hypercube Sampling from distributions of real route parameters; and

- 19 representative truck and 21 representative train accident source terms calculated by analysis.

Cask response to structural accidents was calculated from finite element analysis for impact of each of the generic cask designs onto an unyielding target in three impact orientations. This impact was related to yielding surface impacts by impact energy considerations.

Cask accidents involving fires were evaluated by one-dimensional thermal analyses of the cask shell and shielding to determine the time to produce cask seal failure and rod failure temperatures.

Leak areas were determined from the structural and heat transfer analysis results and were used to estimate cask leakage rates. Total release fraction values were both calculated from analysis and were also based on experimental results.

Because only impact onto hard rock at high speed appeared to be able to cause a spent fuel cask to leak, the Reexamination Study event trees were updated from the Modal Study using GIS analyses to account for the frequency of occurrence of hard rock.[14-4]

The radiological consequences were then evaluated by RADTRAN calculations based on the derived input data. Two types of consequences were examined: incident-free and accident population dose rate risk. These consequences were calculated for PWR and BWR spent fuel in each generic cask type for the 200 representative routes, for four illustrative United States truck and rail routes, and for the NUREG-0170 truck and rail routes.

The dependence of accident consequences on accident source terms was also determined in the Reexamination Study and compared to the earlier Modal Study results.

The major conclusion of the Reexamination Study was that the NUREG-0170 estimates of spent-fuel transportation incident-free doses are somewhat conservative, and the NUREG-0170 accident population dose risk estimates are very conservative. This conclusion clearly demonstrates that the existing regulations governing spent-fuel transportation are adequate to protect public health and safety.[14-1]

14.14 References

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15. Literature Search

At the beginning of this study, a literature search was conducted to identify articles, papers and reports in the open literature concerning risk in the land transport of MOX fuel. Six databases available through Dialog, a commercial database provider, were searched. The database descriptions that were taken from database summary sheets available on the Dialog web site are described below.

Energy Science & Technology (formerly DOE ENERGY) is a multidisciplinary file containing worldwide references to basic and applied scientific and technical research literature. The information is collected for use by government managers, researchers at the national laboratories, and other research efforts sponsored by the U. S. Department of Energy, and the results of this research are transferred to the public.

NTIS: National Technical Information Service database consists of summaries of U. S. government-sponsored research, development, and engineering, plus analyses prepared by federal agencies, their contractors, or grantees. It is the means through which unclassified, publicly available, unlimited distribution reports are made available for sale from agencies such as the National Aeronautics and Space Administration (NASA), the Department of Energy (DOE), the Department of Defense (DoD), the Department of Housing and Urban Development (HUD), the Department of Transportation (DOT), Department of Commerce, and some 240 other agencies. Additionally, some state and local government agencies now contribute summaries of their reports to the database. NTIS also provides access to the results of government-sponsored research and development from countries outside the United States. Organizations that currently contribute to the NTIS database include: the Japan Ministry of International Trade and Industry (MITI); laboratories administered by the United Kingdom Department of Industry; the German Federal Ministry of Research and Technology (BMFT); the French National Center for Scientific Research (CNRS); and many more.

INSPEC (the database for physics, electronics, and computing) corresponds to the three Science Abstracts print publications: Physics Abstracts, Electrical and Electronics Abstracts, and Computer and Control Abstracts. The Science Abstracts family of abstract journals began publication in 1898. Approximately 16% of the database's source publications are in languages other than English, but all articles are abstracted and indexed in English. Author-prepared abstracts are used when available.

The **Ei Compendex®** database is the machine-readable version of the Engineering Index, which provides abstracted information from the world's significant engineering and technological literature. The Compendex database provides worldwide coverage of approximately 4,500 journals and selected government reports and books. Subjects covered include civil, energy, environmental, geological, and biological engineering; electrical, electronics, and control engineering, chemical, mining, metals, and fuel engineering; mechanical, automotive, nuclear, and aerospace engineering; computers, robotics, and industrial robots. In addition to journal literature, over 480,000 records of significant published proceedings of engineering and technical conferences formerly indexed in Ei Engineering Meetings® are also included.

PASCAL is produced by the Institut de l'Information Scientifique et Technique (INIST) of the French National Research Council (CNRS). It provides access to the world's scientific and technical literature and includes about 450,000 new citations per year. Available in machine-readable form since 1973, PASCAL corresponds to the print publication Bibliographic internationale (previously Bulletin signaletique). Each citation includes the article's original title, and, in most cases, a French translated title; for material since 1973, an English translated title is also provided. Most abstracts are in French. Analyzed documents come from all over the world, in 100 different languages. French journals are particularly well represented. The file's breakdown by language is as follows: English 63%, French 12%, Russian 10%, German 8%, and other languages 7%.

JICST-EPlus (Japanese Science & Technology) is a comprehensive bibliographic database covering literature published in Japan from all fields of science, technology, and medicine. The file contains both the JICST-E and the PreJICST-E files from Japan Science and Technology Corporation, Information Center for Science and Technology (JICST). JICST-E contains bibliographic data, abstracts (when available), and indexing from 1985 to the present. PreJICST-E covers from1994 onward and contains no indexing, but does include bibliographic data, abstracts (when available). Many, but not all, of the articles appearing in PreJICST-E will later be replaced by JICST-E records. JICST-EPlus covers over 6,000 journals and serials, in addition to conference papers, preprints, technical reports and other non-periodicals published by the Japanese government or local governments.

These databases were selected for the relevance of their coverage. In general, the publication dates for the materials indexed go back to the late 1960s or early 1970s, when the electronic forms of the indexes were introduced. Publications in NTIS may go back further.

Once the databases to be included in the search were selected the more challenging task of constructing the search strategy was begun. It is possible to search in titles, subject keywords assigned by the authors or database producers, or abstracts. The first step was to identify any citations which specifically discussed the transport of mixed oxide fuel by the various land modes, and which specifically mentioned risk. Only five publications were identified in this initial search.

Since so few citations were retrieved in the initial attempt, the search was expanded. The selection of search terms is somewhat arbitrary. A publication may discuss risk, land transport, or other concepts without mentioning the specific terms identified by the searcher.

All citations included in the final search discussed transport or transportation of MOX or mixed oxide fuel. There were almost 700 citations linking these two concepts; however, this set would include duplicates, as many of the publications were indexed in two or more of the databases. To this set other concepts were added, including:

- Various modes of land transport, with no further limitation to mention of risk or other concepts;
- Discussion of risk or assessment as a major concept with no reference to mode of transport;
- Any discussion of forms of packaging mentioned risk or assessment; and

- Discussion of transport in European countries including mention of risk or assessment.

The final search included around 100 citations, which were used to compile the database. From this number, some duplicate citations were removed, as were some false hits resulting from the use of the term “transport” to refer to radionuclide migration in the environment as well as transportation.

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