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Transportation-Accident Scenarios for Commercial Spent Fuel

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Prepared by Sandia National Laboratories, Albuquerque, New Mexico 87185
and Livermore, California 94550 for the United States Department
of Energy under Contract DE-AC04-76DP00789

Printed February 1981

Sandia National Laboratories
energy report

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Printed in the United States of America

Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

Price: Printed Copy \$5.25; Microfiche \$3.00

SAND80-2124
TTC-0156
UC-71

TRANSPORTATION-ACCIDENT SCENARIOS FOR COMMERCIAL SPENT FUEL

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February 1981

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ABSTRACT

A spectrum of high severity, low probability, transportation accident scenarios involving commercial spent fuel is presented together with mechanisms, pathways and quantities of material that might be released from spent fuel to the environment. These scenarios are based on conclusions from a workshop, conducted in May 1980 to discuss transportation accident scenarios, in which a group of experts reviewed and critiqued available literature relating to spent fuel behavior and cask response in accidents.

ACKNOWLEDGMENTS

I am indebted to many people, including all of the spent fuel scenario workshop participants, for their support and technical discussions during the preparation of this report. I would like to specifically mention a few because of their extra interest and concern about the completion of this report: N. C. Finley, R. E. Luna, J. D. McClure and J. M. Taylor, all of Sandia National Laboratories; A. P. Malinauskas, of Oak Ridge National Laboratories and R. L. Schlegel, of NUS Corporation. In addition, I wish to thank F. W. Bingham for his critical reviewing and editing of this report.

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TRANSPORTATION ACCIDENT SCENARIOS FOR COMMERCIAL SPENT FUEL

Summary

The release of radionuclides from spent fuel to the cavity of a shipping cask and then to the environment is a complex process that could involve many plausible release mechanisms and pathways. These pathways were evaluated in a workshop which considered available data and analysis in this area. Six mechanisms for release of radionuclides from the spent fuel assemblies to the cask cavity and three pathways for release from the cask cavity were judged to be credible. In order to simplify the discussion of release scenarios, the relationships of the mechanisms and pathways were described using fault trees in which the basic components become the mechanisms for release from the spent fuel to the cask cavity and release pathways through the cask shell.

Releases during transportation accidents involving commercial spent fuel were described using two fault trees shown in Figures 1 and 2. Figure 1 diagrams the first phase of an accident, the phase in which the cask still contains some coolant. During this phase, release to the environment can only occur in two steps that are represented by the two major branches of the tree. The first step (left branch) is to release radionuclides to the cask cavity and the second (right branch) is to provide a pathway, for material that has been released into the cask cavity, from the cavity through the cask shell to the environment. Figure 2, which represents the second phase of release, shows how pathways and release mechanisms are related when the cask cavity has lost its coolant. Figure 2 is similar to Figure 1 except that it depicts an event that has progressed in time such that the cask no longer contains coolant.

Using Figures 1 and 2, five accident scenarios were described by selecting various events and combinations of events described in the fault trees. The mechanisms and pathways considered in each of the scenarios are presented in Tables I and II. Table I lists all mechanisms and pathways for Phase 1 and Table II lists them for Phase 2. As can be seen in the tables, additional mechanisms and pathways are assumed to become involved as the number of the scenario becomes larger. The first four scenarios are representative of severe accidents involving water-cooled casks; the last is representative of gas-cooled casks.

The fractions of radionuclides released were then estimated for each scenario by totalling the releases from each combination of mechanisms and pathways considered. The fractions (of the total cask inventory) of each radionuclide released are given in Table III. The fractions can be used to determine the absolute quantity released by multiplying them by the total radionuclide inventories contained in the cask. The release fractions in Table III can be separated further into that released (but remaining on-site) and the fraction released which is aerosolized and in the respirable size range. The aerosolized fractions are given in Table IV for each of the scenarios. The radionuclides considered are those calculated to have the greatest consequence if they were inhaled or deposited on the ground.

It is unlikely that any of the scenarios leading to a release will occur. Accident environments severe enough to lead to a release by these scenarios are very unusual. No more than one in a thousand accidents is likely to trigger Scenario 1 and one or two events in one million accidents is an upper limit estimate for occurrence of the others.

*See TABLE IX

Figure 1. Fault tree for Phase 1.

Figure 2. Fault tree for Phase 2.

TABLE I
Mechanisms and Pathways Considered in Each Scenario (Phase 1)

Scenario	Mechanism - Step 1			Pathway - Step 2			
	Crud	Impact Rupture	Leaching	Valve or Penetration	Seal	Small Breach	Severe Breach
1	yes	no	no	no	yes	no	not credible
2	yes	no	yes	no	yes	no	not credible
3	yes	no	yes	no	yes	no	not credible
4	yes	yes	yes	yes	yes	yes	not credible
5	yes	yes	no	yes	yes	yes	not credible

TABLE II
Mechanisms and Pathways Considered in Each Scenario (Phase 2)

Scenario	Mechanism - Step 1			Pathway - Step 2			
	Burst Rupture	Diffusion	Oxidation	Valve or Penetration	Seal	Small Breach	Severe Breach
1	no	no	no	no	yes	no	not credible
2	no	yes	no	no	yes	no	not credible
3	yes	yes	no	no	yes	no	not credible
4	yes	yes	no	yes	yes	yes	not credible
5	yes	no	yes	yes	yes	yes	not credible

TABLE III
Total Fraction of Material Released from the Spent Fuel to the Environment

Radionuclide	Scenario				
	1	2	3	4	5
Co 60	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-2}
Noble Gases	-	3×10^{-5}	1×10^{-1}	1×10^{-1}	1×10^{-1}
Cs 134	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	1×10^{-3}
Cs 137	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	1×10^{-3}
I 129	-	9×10^{-6}	6×10^{-4}	4×10^{-4}	4×10^{-3}
Sr 90	-	5×10^{-7}	2×10^{-6}	4×10^{-6}	9×10^{-7}
Ru 106	-	-	1×10^{-6}	1×10^{-6}	4×10^{-5}
Actinides	-	5×10^{-7}	2×10^{-6}	3×10^{-6}	9×10^{-7}

TABLE IV

Total Fraction of Aerosolized, Respirable Material
Released from the Spent Fuel to the Environment

Radionuclide	Scenario				
	1	2	3	4	5
Co 60	1×10^{-3}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-2}
Noble Gases	-	3×10^{-5}	1×10^{-1}	1×10^{-1}	1×10^{-1}
Cs 134	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	1×10^{-3}
Cs 137	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	1×10^{-3}
I 129	-	9×10^{-6}	6×10^{-4}	5×10^{-4}	4×10^{-3}
Sr 90	-	5×10^{-7}	2×10^{-6}	4×10^{-6}	9×10^{-7}
Ru 106	-	-	1×10^{-6}	1×10^{-6}	4×10^{-5}
Actinides	-	5×10^{-7}	2×10^{-6}	3×10^{-6}	9×10^{-7}

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Introduction

A wealth of transportation accident scenarios for spent fuel have been presented to the public in environmental impact statements and environmental assessments. An unfortunate aspect of the way these scenarios have been presented is the appearance of a lack of consistency, which often results from the myriad of assumptions used to generate the scenarios.

This presentation distills the assumptions used in a large number of scenarios developed over the last eight years. During distillation, assumptions that no longer need to be made are eliminated when experimental or analytical data now exist. However, when data are still not available, this presentation incorporates conservative assumptions that are expected to overpredict consequences. The scenarios that have been reviewed during the preparation of this presentation are included in the supplementary bibliography, and the scenarios of particular importance and relevance are referenced directly in the text.

The result of the distillation is a group of scenarios that are presented here. Their applicability has been made as universal as possible to shipments of commercial spent fuel. In order to ensure their applicability, these scenarios have been reviewed by a group of experts, the same group of experts who participated in a recent DOE-sponsored workshop on spent fuel scenarios.¹ This presentation reflects a consensus of the participants when assumptions and judgments are made.

The scenarios define the fraction of the spent fuel inventory that might be released to the environment as a result of a transportation accident, but they do not define the consequences of material released during the scenarios. This presentation can be used, therefore, as a source document containing release fractions for credible worst-case scenarios involving the transportation of commercial spent fuel.

This presentation should not be considered to be a final representation of transportation accident scenarios for commercial spent fuel. As additional experimental data become available, they will be used to reexamine assumptions made here and to produce a new or modified set of scenarios.

The word "credible," used frequently in this document, is intended to mean "worthy of belief or plausible." As such, it is subject to a wide variety of interpretations, depending on the experience or point of view of each individual using the word. This imprecision is regrettable, but in this presentation, credible refers to what is believable or plausible to those experienced in transportation, handling and research associated with spent fuel.

Organization of Presentation

This presentation comprehensively discusses how a set of transportation accident scenarios is developed for spent fuel, develops the scenarios and calculates what the expected releases would be from the scenarios. The first aspect discussed is the philosophy of worst-case scenarios as well as the regulations that require them. Then, one reads about the release mechanisms and pathways by which radioactive material can be assumed to escape from the spent fuel to the environment. After many possible pathways have been examined, what needs to be accounted for in writing an acceptable generic scenario is detailed. Next, environments that could be obtained in a real transportation accident are discussed. The mechanisms, pathways and environments are then put into perspective using fault tree analysis: the use of fault trees is discussed, and transportation accident scenarios are defined by following branches of the fault trees. By analyzing each branch, releases expected from the scenarios are calculated.

Even though the major purpose of this presentation is to define the releases expected from the accident scenarios, the probability of occurrence for each of the scenarios is also estimated.

Worst-Case Scenario Analysis

As mandated by federal regulation (40 CFR 1502.22), a worst-case analysis of environmental impacts must be performed when data do not exist and cannot be obtained or are too expensive to obtain. This regulation considers no bound, however, to the limit of accident probability that must be considered. Unjustifiably conservative scenarios can be postulated, but they have no practical meaning because of their very low probabilities. This presentation defines and examines a spectrum of credible, though highly unlikely, accident scenarios. The likelihood of these scenarios is then discussed at the end of this paper.

Although this paper limits its conservatism to what is justified by experimental data and the judgments of experts who participated in the recent workshop,¹ it willingly recognizes the paucity of data available to support more realistic assumptions and the need to be conservative when data do not exist. This presentation will, for an interim period, provide a reasonable basis from which environmental consequences of transporting spent fuel can be calculated; data gathered in future studies will refine the analyses presented here.

Transportation accidents involving spent fuel have never resulted in a release of radioactive material to the environment, so it is very difficult to define a credible worst-case in which radioactive material is released. However, a majority opinion of the participants of the workshop was that a large breach (greater than 6.4 cm^2 (1 in^2) cross sectional area) is not credible to consider¹.

Mechanisms and Pathways

A radiological health problem exists when radionuclides are released from the spent fuel, somehow escape from the cask to the environment and eventually reach people. A number of mechanisms and pathways could allow a release to the environment. The way in which they are combined will be discussed in detail in a subsequent section that discusses fault tree descriptions of the spent fuel scenarios. This section specifically discusses only the mechanisms and pathways.

They can be put into two categories that correspond to two necessary steps for release to the environment (Figure 3). The radioactive material must first be released from the spent fuel to the cask cavity, and then this material must be released from the cavity to the environment. A third step that could also be considered separately is the release of radionuclides within the spent fuel rod itself, but since most of the release in this step occurs while the fuel is in reactor, it is combined with the first step. The first step is governed by characteristics of the spent fuel, and the second step is governed by characteristics of the spent fuel shipping cask.

Figure 3. Steps of release in a spent fuel transportation accident.

Before this presentation proceeds, terms used to discuss these events will be defined. The basic elements of a cask and a spent fuel assembly will be defined very generally.

Four frequently used terms are rod, cladding, fuel-clad gap and assembly (see Figures 4 and 5). A fuel rod (oftentimes referred to as a fuel element) is the smallest component of importance to transportation accidents. A fuel rod is simply a tube made of zircaloy (some are stainless steel) that contains reactor fuel pellets. The tubing itself is referred to as the cladding. The void space between the cladding and the fuel is referred to as the fuel-clad gap. The rod contains, under pressure, fission product gases generated in reactor and an inert fill gas added during manufacture. A bundle of rods fixed together by structural members is called a fuel assembly.

Figure 4. Schematic diagram of a fuel rod used in pressurized-water reactors.

Figure 5. Schematic diagram of a fuel assembly used in pressurized-water reactors.

When the fuel in a reactor has been used, it is removed. When this spent fuel is transported, it is carried in heavy casks, each holding one or more spent fuel assemblies. The capacity of a cask is commonly given in terms of the number of assemblies it can carry.

Several terms are used to describe a cask (Figure 6). In very general terms, a cask has two major components: a head and a body. The head is bolted to the body. At the union of the two components is at least one seal (and frequently two) that prevents release of coolant to the environment. The head and body are massive; they provide great strength and shielding from the radiation emitted from the spent fuel. Spent fuel is placed in the hollow core of the cask, called the cask cavity. Small components, referred to as valves and penetrations, are used for filling and draining the cask cavity.

Figure 6. Spent fuel cask schematic.

A special term, "waterlogged rod," is used to describe a rod that has failed in a reactor. The term arises because water is drawn into the fuel-clad gap through a point of failure in a rod as a reactor is cycled through various power levels and shutdowns. From a safety standpoint, the waterlogged rod is different from an unfailed rod: it no longer is highly pressurized with fission gases and its original fill gas, it may have soluble fission products leached from the fuel-clad gap and it may be more brittle and susceptible to impact damage.

The rods can fail from a number of causes, the most important of which to this presentation is hydriding of the internal surfaces of the cladding. Hydriding will be discussed further in the subsequent discussion of the impact rupture mechanism.

The mechanisms and pathways will be discussed using these definitions.

Mechanisms for Release from the Spent Fuel Assembly to the Cask Cavity (Step I)

The release from the spent fuel is dictated by mechanisms that are related to properties of the spent fuel. At least six mechanisms can be postulated: impact rupture, burst rupture, diffusion, leaching, oxidation and crud release. Each mechanism is distinct; however, diffusion, leaching and oxidation are important only after a rupture has occurred.

Impact rupture is a relatively easy mechanism to understand because it is simply the release of radioactive material by the mechanical disruption of the cladding and subsequent depressurization of the fuel rod. The mechanical force of an impact can cause a fuel rod to bend or otherwise deform and subsequently, when coupled with the venting of the fill gas and fission gases, can produce a driving force to actually release materials contained in the fuel-clad gap. Of course, the driving force must be sufficient to actually rupture the cladding.

Rupture of the cladding can occur more easily in a waterlogged rod that has failed due to hydriding. Hydriding occurs when hydrogen gas, which is liberated from impurity moisture in the fuel, can react with the zirconium in the fuel cladding to form zirconium hydrides. The reaction is referred to as hydriding. The hydriding causes a localized embrittlement of the cladding because zirconium alloys form hydride phases that have low ductility and through which cracks may easily propagate. The most common types of hydriding defects are pinholes in cladding blisters caused by hydriding and small cracks which have propagated through hydrided areas of cladding.

As a result of the embrittlement, hydrided cladding may be damaged more easily than unaffected rod cladding. The damage to cladding may be severe if the cladding has been grossly hydrided, but gross hydriding is an unusual occurrence. Waterlogged rods should occur no more often than once in 10,000 rods,² and not all of these rods fail as a result of hydriding. In addition, waterlogged rods normally occur in groups, that is, several failed rods occur in the same assembly. Thus, the number of assemblies containing failed, hydrided rods is smaller than the number of failed rods.

In hydrided, waterlogged rods, the mechanical force of impact is the only driving force available to release radioactive material because the rods are depressurized before the impact. However, should a failed rod self-seal, it may become slightly repressurized.

Burst rupture is an analogy to impact rupture. Burst rupture occurs in a severe thermal environment while impact rupture occurs in a severe impact environment. As a spent fuel rod is heated, pressures will increase inside a rod until it may burst. The effect is exacerbated by rapid rises in temperature and very high temperatures. As the rod is subjected to increased pressures, it mechanically deforms until it bursts, creating a hole that has a diameter approaching a few millimeters. The release of

pressure through the hole will vent material from the spent fuel-clad gap, material which is in the vicinity of the hole.

Once a spent fuel rod has been ruptured, vaporized fission products can diffuse in the fuel-clad gap and out the rupture opening. Very high temperatures increase the likelihood of this mechanism, referred to as the diffusion mechanism.

Fission products and spent fuel may be leached from the surface of the spent fuel pellets or from the fuel-clad gap if water can come into direct contact with the spent fuel. This mechanism, which requires that a rod must have been ruptured, is accelerated when the temperature of the water is elevated.

The fifth mechanism, oxidation, is normally expected to occur in the immediate area of a cladding breach and can be initiated on a large scale only after severe disruption of the rod cladding. If the spent fuel left bare by a rupture is exposed to severe thermal environments and flowing air or steam, the UO_2 may be oxidized to UO_{2+x} , changing density and cracking macroscopically. The increased surface areas release significantly more fission products. Steam atmospheres are much more neutral toward UO_2 .³

The crud-release mechanism is unique among the mechanisms because essentially no fission products are released and because the fuel cladding does not have to rupture for a release to occur. Some fission products may be present in solution in storage pools from rods previously ruptured in service and they may deposit with the crud in small concentrations. The mechanism simply involves the release of the corrosion products (crud) and some traces of fission products by impact, vibration and abrasion or a severe rapid thermal transient. As a result, crud would be expected to be dislodged in most severe transportation accidents. In addition to crud on the fuel rods, there is crud on the cask and structural components of the spent fuel assembly that can also be dislodged in an accident.

Another mechanism, the zirconium-water reaction, has been postulated by Resnikoff⁴ as important. This reaction, if credible for transportation accidents, would have three potentially hazardous results: hydrogen gas would be liberated, heat would be given off as the reaction proceeds and the cladding could become embrittled. However, this exothermic reaction does not even begin to become important in a water-cooled reactor environment until temperatures over 980°C are reached and may become significant above about 1100°C.⁵ A minimum limit of 980°C may be applicable to a transportation accident involving a water-cooled cask, but, since this temperature is considered to be about the maximum temperature the external surface of the cask might be subjected to and since, in a transient thermal environment considered credible for a transportation accident, the spent fuel temperature would not exceed the maximum surface temperature of the cask, the zirconium-water reaction is not a credible mechanism for a transportation accident scenario.

The six credible release mechanisms can result in a release from the spent fuel to the cask cavity, but release to the environment can occur only if a pathway through the cask exists. Potential release pathways through the cask are described in the following text.

Pathways for Release From the Spent Fuel Cask Cavity to the Environment (Step 2)

Spent fuel casks have not been experimentally tested to failure, and consequently, experimental failure threshold values do not exist. As a result, the following subsection relies almost entirely on discussions that took place during the spent fuel scenario workshop in May 1980.¹

A release from the cask cavity to the environment may occur if the cask is compromised in a few ways. Casks that have valves used to fill the cask cavity can be compromised

if these valves are in some way damaged so as to destroy the valve, to cause it to stick open or to sever the valve piping leading to the cask body. The valves can be vulnerable to both heat and impact, but casks are designed to protect valves from such environments. Some casks do not have valves but have penetrations that allow access to the cask cavity. These penetrations may be more vulnerable to compromise than other parts of the cask even though they are protected. In spite of their vulnerability, because valves and penetrations present a very small target requiring a precisely positioned impact or a localized fire in order to fail, the probability of their failure would be expected to be less than the probability for failure of a cask closure seal.

Cask closure seals that prevent leakage between the cask head and body can be damaged so as to create a pathway from the cask cavity to the environment. Such damaged seals would not be expected to provide a pathway with a large cross section because only small sections of the seals would be likely to fail. The pathway could indeed act as a filter for particulate releases. The damage to cask seals may result from either impact or heat. For example, a closure seal could fail if head bolts yield sufficiently to create a release pathway. The head bolts could be deformed mechanically or possibly by differential thermal expansion resulting from uneven heating of the cask head and body. Nonetheless, like valves and penetrations, seals are designed to minimize the likelihood of damage resulting from severe environments. This is the type of failure that resulted in a cupful of coolant being lost after an 134 kph (84 mph) crash test performed by Sandia National Laboratories.⁶

Another release pathway is a small breach (cross sectional area less than 6.4 cm^2 (1 in^2) of the cask body or head. Such a breach would most likely be a fine crack that would allow only limited releases and would protract release times. This sort of a breach is very unlikely and was not evidenced in the severe tests conducted by Sandia.⁶

Because the small breach is very unlikely and a large breach would be even more unlikely, a large breach is not considered credible.¹ No pathway that has a cross sectional area greater than 6.4 cm^2 (1 in^2) is considered credible for the conditions expected in even the most severe transportation accidents.

Other pathways exist but are not considered here. These other pathways might include human error (e.g., incorrectly torquing head bolts) and sabotage, which are thoroughly discussed in Reference 7.

A combination and sequencing of the failure mechanisms for the spent fuel and the release pathways through the cask will produce a release to the environment. Both a release from the spent fuel and a pathway from the cask cavity must be created before radioactive material will reach the environment.

Necessary Parameters for Generic Scenarios

A group of generic scenarios is difficult to formulate for a complex system of casks, spent fuel characteristics and modes of transportation. Variations in cask design as well as spent fuel characteristics must be accounted for in the formulation of a generic scenario. Casks are designed differently for different modes of transportation. Cask designs determine which pathways of release from the cask cavity to the environment are possible and which spent fuel release mechanisms can be initiated. The properties of the spent fuel also are important in determining which spent fuel release mechanisms to the cask cavity are possible as well as in determining the release source terms. The mode of transportation influences how much spent fuel can be carried in a cask due to weight restrictions and influences what the maximum accident environments would be if an accident should occur. This particular section will discuss

the effects that cask design, fuel characteristics and mode of transportation have on generic scenarios.

Cask Design

Many cask designs are presently licensed for the transportation of spent fuel. As can be seen from Table V (from Reference 8, p. III-10), the characteristics of these casks differ markedly. Designs differ in weight, dimensions, shielding, coolant, number of spent fuel assemblies carried, decay heat removal capacity, type of closure seals and use of penetrations and valves. Casks of different design will respond differently to a given transportation accident. Consider several examples.

TABLE V
Licensed and Available Shipping Casks for Current-Generation Spent Fuel⁸

Cask Designation ^a	Number of Assemblies		Approximate Loaded Cask Weight		Usual Transport Mode	Shielding		Cavity Coolant	Maximum Heat Removal, kW
	PWR ^b	BWR ^b	MT	(Ton)		Gamma	Neutron		
NFS-4 or NAC-1	1	2	23	(25)	Truck	Lead & steel	Borated water & antifreeze	Water/air ^g	11.5/ 2.5
NLI 1/2	1	2	22	(24)	Truck	Lead, uranium & steel	Water	Helium	10.6
TN-8	3	-	36	(40)	Truck ^c	Lead & steel	Borated solid resin	Air	35.5
TN-9	-	7	35	(38)	Truck ^c	Lead & steel	Borated solid resin	Air	24.5
IF-300	7	18	63	(70)	Rail ^d	Uranium & steel	Water & antifreeze	Water	76.0 ^e
NLI 10/24	10	24	88	(97)	Rail	Lead & steel	Water	Helium	97.0 ^f
TN-12	12	32	97	(107) ^g	Rail	Steel	Borated solid resin	Air	135.0

^aCask initials: NAC = Nuclear Assurance Corporation
NFS = Nuclear Fuel Services, Inc.
NLI = NL Industries (previously National Lead Company)
TN = Transnuculaire
IF = "Irradiated Fuel," symbol used by General Electric Company

^bPWR (Pressurized Water Reactor) and BWR (Boiling Water Reactor).

^cOverweight permit required. The TN-8 and TN-9 casks are also licensed for rail and water shipments.

^dTruck shipment for short distances with overweight permit.

^eLicensed decay heat load is 62 kW.

^fLicensed decay heat load is 70 kW.

^gWith a cask body extension for abnormally long fuel, the loaded cask weight is 103 MT (113 tons).

If a cask is equipped with external valves, a valve might break open or otherwise be damaged so as to provide a direct path from the cask cavity to the environment. If the valves do not exist, as they do not in some cask designs, then this particular pathway is no longer possible.

As discussed earlier, at least one seal is used between the head and body of a cask to ensure the leak tightness of the cask cavity. The seal may be made from an elastomer or a metal. Both are adequate but have different failure thresholds. Particularly, for the thermal environment, the differences can be very large: 280°C for teflon¹⁰ and probably greater than 500°C for metal.

The type of coolant used in the cask cavity determines whether some potential spent fuel release mechanisms can be initiated. For example, the air in an air-cooled cask allows oxidation to be a potential release mechanism. The coolant in other casks

minimizes this possibility. The water in the water-cooled casks could leach the spent fuel rods should the cladding be breached in some manner. But, if a cask contains a coolant other than water, leaching is not a credible mechanism unless waterlogged rods, which already contain water from the reactor, are present in the cavity.

The design heat loading is critical in establishing the maximum fuel and cladding temperature that can be realized during accident conditions. Spent fuel in some casks may become hotter than the spent fuel in others if the casks are involved in fires that produce identical, external thermal environments.

As these examples show, specific design features are important in determining what can happen in an accident. Spent fuel characteristics can be equally important.

Spent Fuel Characteristics

Characteristics of spent fuel can vary markedly depending on its age (time since it was removed from the reactor), burnup (original amount of radioactivity) and type (what reactor it was used in). Each of these characteristics influences the rate of decay-heat generation, which is critical to a generic scenario because many of the spent fuel release mechanisms are thermally initiated. As spent fuel ages, the rate of decay-heat generation decreases. As a result, older fuel may not generate sufficient heat to initiate such mechanisms as burst rupture. Low-burnup fuel does not generate as much heat as high-burnup fuel. Therefore, for a given fuel age, low-burnup fuel is less likely to be ruptured. The type of reactor that the spent fuel comes from also influences heat generation rate. Pressurized Water Reactor (PWR) spent fuel normally undergoes higher burnup than Boiling Water Reactor (BWR) spent fuel.

The burst-rupture mechanism is very sensitive to the type of fuel. Generally, PWR spent fuel is more likely to fail by this mechanism than BWR spent fuel. Table 10 of Reference 11 shows that PWR spent fuel would be expected to rupture at temperatures about 100°C lower than temperatures at which BWR spent fuel would be expected to rupture.

The physical condition of the spent fuel can also determine whether some release mechanisms are possible. For example, a waterlogged rod that has failed by hydriding may be more likely to fail by the impact-rupture mechanism.

Mode of Transport

The mode by which spent fuel is shipped influences the worst credible environment that can be postulated for a scenario. The mode of transport determines the cask weight and capacity, and in addition, it determines the potential accident environments. For example, the rail mode is generally thought to provide the greatest source for fires because a train may be carrying an enormous supply of flammable materials with the spent fuel shipment. The impact created during a rail accident may be considerably different from that for a truck accident. A spent fuel cask carried on a railcar may be buffered by other railcars so that it will not receive a very large impact. On a truck, the buffering may not be as great.

Summary

Many factors, including cask design, fuel characteristics and mode of travel, must be considered when developing a scenario. A generic scenario necessarily deals with the worst set of factors possible, and in some cases, the combination of factors used in developing a worst-case scenario may result in one that unrealistically predicts results greater than those predicted if specific cask design, fuel characteristics and mode are considered.

Accident Environments

Two questions normally arise about accident environments: (1) What are the extreme environments that can be produced by very severe transportation accidents? (2) Are these environments sufficient to cause failures that would allow a release to the environment?

In general, the damage resulting from these worst-case accident environments has not been thoroughly investigated. However, the effects of hypothetical test conditions (Title 49, Code of Federal Regulations, Part 173.398) have been examined in detail for each specific cask design as part of the licensing process before it can be used in commerce. In performing these examinations, the vast majority of all real accident conditions have been analyzed as the discussion on accident probabilities will explain.

Most Severe Environments Produced in Transportation Accidents

The best estimate of the most severe environments will be obtained by reviewing historical transportation accidents. Such a review is available in at least four references (10, 12, 13 and 14). The most severe thermal environment appears to be easier to quantify than the most severe impact environment. Two conclusions of these references are that a maximum credible fire temperature would be about 1000°C and that a maximum credible fire duration would be one to several hours (2 hours is most often selected in constructing scenarios).

In the licensing process for spent fuel casks, the licensee must analyze the effects of a 1/2-hour fire at 800°C. In addition, the effects on cask components such as valves, penetrations and seals and the maximum temperatures that may result in the spent fuel rods must also be determined. Table VI presents the results from

TABLE VI
Cask Analyses for Hypothetical Fire Test Conditions

Cask	Coolant	Design Heat Removal Capacity (kW)	Spent Fuel Age (day)	Assemblies	Peak Fuel ^{a,b} Temperature After Fire Test
NAC-1 & NFS-4	Water	11.5	120	1 PWR/2 BWR	510°C ^c
NLI 1/2	Helium	10.6	150 PWR 120 BWR	1 PWR/2 BWR	594°C
TN 8/9	Air	35.5/24.5	150	3 PWR/7 BWR	525°C
NLI 10/24	Helium	70.0 ^e	150	10 PWR/24 BWR	533°C
IF-300	Water/Air	76.0 62.0 ^e	120	7 PWR/18 BWR	858°C ^d 518°C ^f

^aSARP references (15, 16, 17, 18, 19 and 20).
^bCompare to Table III, p. 14.
^cAverage temperature for analysis using 12.4 kW in original SARP.
^dDesign changes may reduce this number from its original SARP value.
^eMaximum approved heat load for shipment.
^fFrom NUREG-0069, p. 4.21

analyses for currently licensed casks (taken from their Safety Analysis Reports for Packaging (SARP)). The hypothetical-fire analyses predict a range of temperatures for the various casks; an average value for the maximum fuel temperature is calculated to be about 540°C. A credible worst-case fire (2 hours at 1000°C) would result in considerably higher temperatures.

The test-condition analyses use a maximum heat loading that would correspond to a full load of fuel that is 120 days or 150 days old. If older fuel is shipped, the decay heat generation rate is considerably reduced. Figure 7 shows the effect of increasing age on heat generation rate. If 1-year-old fuel is shipped instead of 120-day or 150-day fuel, the heat rate is halved; 2-year-old fuel results in 1/4 the heat rate of 150-day fuel. Therefore, the maximum fuel temperature would be lower for older spent fuel.

Figure 7. Decay-heat generation rate from PWR spent fuel.²²

The thermal environment in a credible worst-case fire appears to be sufficient to cause a release from the spent fuel to the cavity and to create a failure pathway from the cask cavity to the environment. The release fractions, the mechanisms initiated and the pathways created are discussed in a subsequent section.

The maximum impact environment is not so well quantified; however, analyses of cask designs indicate that a credible impact environment can be postulated in which the spent fuel and the cask will fail. A good discussion of this impact damage is found in References 3, 10 and 23. A good discussion of thermal analyses is found in Reference 24.

The conclusion to be drawn from the discussion above is that extreme conditions can be postulated that are credible (though very unlikely) and that could result in a release of radioactivity to the environment. These conditions are based primarily on analysis and need considerable experimental substantiation.

Failure Thresholds

In this section, the failure thresholds for the spent fuel and for the casks are discussed.

Spent Fuel. As Table VII shows, the failure-mechanism thresholds for spent fuel are reasonably well defined. The thermally activated mechanisms generally require temperatures in excess of 400°C in order to release measurable quantities of radioactive material. As temperatures increase to the range of 600°C to 700°C, each of these mechanisms except the zirconium-water reaction could be activated and could significantly contribute to release.

The impact thresholds have been analytically considered. No experimental data exist from real accidents because no accident has occurred that has involved a severe enough impact to be considered a worst-case.

Data which exist were generated by tests run at Sandia National Laboratories (SNL) and Oak Ridge National Laboratory (ORNL). The SNL tests⁶ simulated severe accident conditions for spent fuel casks. In each test, the casks contained unirradiated fuel which was to have been used for the nuclear ship Savannah. The fuel did not fail in any of the tests, but in some tests, it was mechanically deformed or bowed. The regulatory drop test from 9 meters (30 feet) onto an unyielding target has been performed at ORNL using the same type of fuel. Some assemblies were bowed, but no cladding failed.³ The applicability of these test results to irradiated, commercial fuel is questionable.

Nevertheless, unfailed spent fuel is quite rugged and capable of sustaining severe impact environments.¹⁹ The values listed in Table VII for impact rupture show a range of impact-failure limits for spent fuel. The range is wide because specific

TABLE VII

Failure Thresholds for Release from Spent Fuel to Cask Cavity

	Impact Rupture	Burst Rupture ^a	Leaching	Crud	Diffusion	Oxidation	Zirconium-Water
Thermal	NA	Incipient failure 565°C ¹¹ Expected failure 671°C ¹¹	Occurs at room temperature but usually accelerated by elevated temperature (fuel must have been failed)	Undefined thermal shock (probably not credible)	>400°C (fuel rods must have been failed)	>430°C ^{1,25}	>1000°C ⁵
Impact	Side Impact Predicted thresholds to fail spent fuel: 71 g to rupture ³ 45 kph (28 mph) cask velocity ¹⁰ Experimental thresholds to fail spent fuel: 122 g no failure ⁹ End Impact Predicted thresholds to fail spent fuel: 38 g to bend ³ 60 kph (41.4 mph) cask velocity ¹⁰ Experimental thresholds to fail spent fuel: >234 g to rupture ⁹	NA	NA	Probably less than regulatory test conditions	NA	NA	NA

^aBurst rupture is a complicated mechanism that is dependent on many factors in addition to temperature such as heating rate, fuel age and fuel condition.

cask and impact geometries are used. Because the values are dependent on so many variables, they should be applied only to the exact configurations for which they are calculated in the references.

Cask. The failure thresholds for spent fuel are such that releases from the spent fuel to the cask cavity could occur. The release of this material from the cavity to the environment is determined by the failure thresholds for the cask. As shown in Table VIII, these thresholds are generally not known. They are very dependent on cask design. If the accident environment is severe enough (capable of producing damage much more severe than the regulatory tests), the casks can be analytically shown to fail. The maximum credible failure, however, is considered to be a small breach such as a crack in the cask body or head.¹

Even though these failure thresholds are shown by analysis to be attained under very unlikely accident conditions, no release of radioactive materials from spent fuel shipments has ever occurred. Releases can be postulated in keeping with a worst-case analysis even though they would be very unlikely.

TABLE VIII
Breaching Thresholds of a Cask Allowing Release
from Cask Cavity to the Environment

	Penetration or Valve Failure	Closure Seal Failure	Small Breach	Large Breach
Thermal	30 minutes, 1000°C valve seal failure ¹⁰	30 minutes, 1000°C fire (Teflon seal) ¹⁰	Unknown, but greater than for a seal failure	Not credible ¹
Impact	Unknown, requires direct hit on a penetration or valve	Side Impact 64 kph (40 mph) cask velocity ¹⁰ 131 kph (82 mph) locomotive velocity ²³ End Impact 77 kph (48 mph) cask velocity ¹⁰ 131 kph (82 mph) cask velocity ²³ Side-Center 74 kph (46 mph) cask velocity ³ Puncture (see Reference 3)	Unknown, but greater than for a seal failure	Not credible ¹

Fault Tree Description of Event Sequence

Fault-tree diagrams have been used to sequence and combine events that result in a release of radioactivity to the environment. A fault tree is a graphical representation of the relationship between specified events and an ultimate undesired event, which in this study is the release of radioactive material to the environment. The relationship between events can be very complex and the use of a fault tree is an attempt to reduce the confusion resulting from such complexity.

Before examining the fault trees drawn for the release of radioactive materials, the logic and the symbols used in the fault trees will be explained. The logic proposed for release of radioactive material from the spent fuel is shown in Figure 8. The release to the environment is considered to occur in two phases, which are defined according to the sequence of the events during a transportation accident. The specific thermal and impact environments define the length of time the release remains in a phase. For example, when a transportation accident occurs there may be an immediate release of radioactive material to the cask-cavity coolant and a subsequent rapid

Figure 8. Spent-fuel-release logic.

release of the coolant through an impact-caused breach. Then additional delayed releases of radioactive material may occur from a prolonged fire that may accompany the accident. These delayed releases are the second phase while the immediate releases are considered the first phase. Consequently, two fault trees are drawn in order to present the phases separately.

Seven symbols are used to construct fault trees in this presentation. They are shown and labelled in Figure 9. The rectangle is referred to as an event symbol; within each rectangle, a particular event will be described. The hexagon with an attached rectangle is called an inhibit-gate; its function is to restrict the sequence of events in a fault tree until a particular criterion or limit has been satisfied. The purpose of this symbol and the others will become clearer as the fault trees are described.

Figure 9. Fault-tree symbols.

And-gate and or-gate symbols have functions similar to that of the inhibit-gate except that more than one event controls the gate. The and-gate requires that two or more events must occur before the events in the fault tree can continue. The or-gate requires that at least one of multiple input events must occur before the events continue. The value-gate requires that a specific value or list of comparators is used to decide if the events can progress beyond the gate.

The circle and the triangle describe initiating events. There is no fundamental difference between them except that events with a characteristic or set of characteristics in common may be identified by using a common symbol.

In fault tree construction, each of these symbols is linked to show relationships and time sequence.

Figure 10 is the fault tree for phase 1, the release of radioactive material to the environment along with the coolant. The rectangular box at the top of the fault tree describes the undesirable event, which is the release of radioactive material to the environment. Proceeding down the fault tree, one encounters a hexagon with an attached rectangular box. This inhibit gate requires that, in order for coolant to reach the environment, the cask must be oriented properly (e.g., the breach or crack must be below the level of the coolant). The next symbol encountered is an and-gate requiring that both events below it must occur for the event above it to occur: if radioactive material is to be released to the environment while in the coolant, radioactive material must be in the coolant, and a release pathway to the environment must exist.

The fault tree is said to branch at this point. Following the left hand branch, the inhibit-gate requires that coolant be in the cask cavity when radioactive material is released to the cavity. The next gate encountered is an or-gate requiring that in order to have a release of radioactive material to the cask cavity only one of three mechanisms needs to be initiated: the leaching mechanism, the crud mechanism or the impact-rupture mechanism. As an example, the leaching-mechanism branch will be described. In order for the leaching mechanism to occur, the cask must be water-cooled. Furthermore, in order to initiate the leaching mechanism a spent fuel rod must be ruptured somehow. The rupture can occur as the result of the impact-rupture mechanism that has resulted from an accident that has caused damage greater than that resulting from the regulatory test conditions (Table IX) for unfailed fuel or as the result of impacting a waterlogged rod at less than regulatory conditions (possibly). Some additional leaching of an unimpacted, waterlogged rod may occur as well as long as it has not resealed itself.

TABLE IX

Accident-Environment Categories

Impact	Fire	
1a	1b	Environment that produces conditions less severe than regulatory conditions
2a	2b	Environment that produces conditions somewhat greater than or equal to the regulatory conditions
3a	3b	Environment that produces credible conditions much greater than the regulatory conditions
4a	4b	Not credible

Figure 10. Fault tree for Phase 1.

In an attempt to quantify the impacts necessary to initiate a mechanism, a value-gate is placed above each initiating event. The value-gate gives a value (for example 1a) that is described in Table IX. Because of the uncertainties in quantifying impact environments, the table does not include absolute values. However, for each environment, fire and impact, four relative values, 1-4, are given; as the value increases, the severity of the conditions increase. These values should be used only for comparing branches. The end of the first major branch has been reached. Now the second one must be considered.

The right-hand branch will now be traced. In order to have a pathway, it is necessary that one of four events occurs: a seal must fail, a valve or penetration must fail, a small breach must be created or a large breach must be created. Following the branch for a failed penetration, one sees that a penetration must exist if a penetration is to fail. A penetration may fail by either an impact or a fire, but, in accordance with the value-gates, the conditions generated must be at least as severe as the regulatory-test conditions. Similar branches exist for each of the other cask-failure mechanisms.

Figure 10 represents the first phase of release; Figure 11 represents the second. The second phase, as described previously, is the delayed release of radioactive material after the coolant is released. Reading down the fault tree, the reader discovers that the pathway must be large enough to allow radioactive material to escape. A release occurs only if two branches of the fault tree are satisfied: radioactive material must be released to a coolant-free cask cavity, and the radioactive material must have a pathway from the cavity to the environment. The second branch is essentially identical to the second branch described in Figure 10 and will not be discussed again. Following the first branch, it is seen that there must be a driving force to get the radioactive material out of the spent fuel and into the cavity. A mechanism for release must exist and can be any of four: crud dislodged, burst rupture, oxidation or diffusion. Each of the four can independently or in combination result in a pathway of release to the cask cavity from the spent fuel. Examining the diffusion branch as an example, it is seen that diffusion is a possible mechanism if a rod has failed due to the burst-rupture mechanism or if a rod has ruptured, if a severe thermal environment (sufficient to vaporize cesium) exists, and if the cavity is free of coolant. The rod rupture can occur in either of two ways to satisfy the requirement for the diffusion mechanism. The other branches, heat source

and coolant-free cask, can be followed in an analogous manner. Each branch ends with an initiating event that must have a severity as indicated by the value-gate. This discussion completes the fault tree for the second phase in release to the environment.

By combining the two fault trees in Figures 10 and 11, a complete fault tree is generated. The complete fault tree can be used to define a series of accident scenarios. The scenarios can be created by following different branches of the or-gates, remembering that for an or-gate to be satisfied only one input event is required. However, even though only one is required, several could occur. As a result, the worst scenario might be envisioned as the one in which all events leading into or-gates occur.

A discussion of accident scenarios is given in the following section.

Figure 11. Fault tree for Phase 2.

Scenarios Postulated

As can be seen in the two fault trees, many possibilities exist for potential accident scenarios. In this section five scenarios are selected as being credible: four for water-cooled casks and one for air-cooled casks. No intervention during the scenario is assumed to mitigate the consequences (e.g., no attempts to remove cask from a fire).

Scenario 1

The fault tree path describing this scenario is shown in Figures 12 and 13. The path is indicated by heavy black lines.

In this scenario, only an impact environment has been assumed during the accident. The environment is sufficient to knock crud loose and to cause a cask closure seal to fail. Valves or penetrations are assumed not to fail because the impact zone was not near them. The environment is sufficient to activate the crud release mechanism in phase 1, but since no waterlogged rods are assumed to be in the cask, no rods fail from the impact. None of the mechanisms in phase 2 occurred because the spent fuel was assumed to be more than 180-days old. It must be further noted that, even though the thresholds for the impact-rupture mechanism and for the seal-failure pathway are both assigned a value of 2a, an impact that results in seal failure does not necessarily have to result in impact rupture of the spent fuel. Conditions of impact might be such that the impact zone is in a critical location for seal failure, but in a location that results in little fuel damage.

Figure 12. Fault tree for Phase 1 (Scenario 1).

Figure 13. Fault tree for Phase 2 (Scenario 1).

Scenario 2

The fault tree paths, indicated by heavy black lines and describing this scenario, are shown in Figures 14 and 15.

In this scenario, a severe impact is assumed to occur with an accompanying fire that lasts less than one-half hour. The environments are sufficient to initiate the crud release mechanism and the leaching mechanism in phase 1 and to initiate the diffusion mechanism in phase 2. Rod failure required for leaching and diffusion is assumed to result from waterlogged rods. The cask failure pathway is through a seal. Again, valves or penetrations are assumed not to fail because the impact zone was not near them. The cask must drain its coolant

fairly rapidly so that the diffusion mechanism can be initiated. The decay heat from the spent fuel and the external fire are assumed to be sufficient to allow the diffusion mechanism to proceed.

Figure 14. Fault tree for Phase 1 (Scenario 2).

Figure 15. Fault tree for Phase 2 (Scenario 2).

This scenario, as indicated in Figures 16 and 17, is different from scenario 2 in that the burst-rupture mechanism is initiated because the fire is assumed to last for a much longer time. In addition, the spent fuel is assumed to be much younger so as to have a much higher decay-heat generation rate.

Scenario 4 (A worst-case for water-cooled casks)

This scenario considers all spent fuel release mechanisms that are credible in a water-cooled cask and considers a seal failure and a small breach as the pathways for release from the cask cavity to the environment (see Figures 18 and 19). This scenario can be considered a credible worst-case scenario.

Figure 18. Fault tree for Phase 1 (Scenario 4).

Figure 19. Fault tree for Phase 2 (Scenario 4).

Scenario 5 (A worst-case for air-cooled casks)

This scenario considers all mechanisms of release (see Figures 20 and 21) that are credible for an air-cooled cask. The oxidation mechanism is included even though severe impact must have occurred and a replenished oxygen supply must be available. If fuel that has been grossly failed in the reactor is shipped in an air-cooled cask, the oxidation mechanism could possibly take place during phase 1 (before the air coolant escapes), but for ease in presentation, this scenario considers oxidation to occur during phase 2. In either case, whether it occurs during phase 1 or phase 2 is of little significance, since the release factor is unaffected. This scenario can be considered a credible worst-case scenario for an air-cooled cask.

Figure 20. Fault tree for Phase 1 (Scenario 5).

TABLE XI

Summary of Pathways for Release from the Cask Cavity to the Environment
(Step 2)

Scenario	Valve or Penetration	Seal	Small Breach	Severe Breach
1	no	yes	no	not credible
2	no	yes	no	not credible
3	no	yes	no	not credible
4	yes	yes	yes	not credible
5	yes	yes	yes	not credible

These five scenarios are to be considered only as representative, not as exhaustive. It has been determined, at least on an analytical basis, that they are possible. Little attempt has been made to explain how the environments could be created in a real accident; they are simply assumed to exist. Experimental evidence is needed to confirm their likelihood of occurrence because oftentimes analytical modeling cannot sufficiently define the detail that determines whether any particular threshold for release or creation of a pathway can be achieved. The reader is cautioned to refer to the discussion of accident probabilities in a subsequent section, which shows that the likelihood of scenarios such as these is very low.

Release Fractions

Since events and their sequence have been determined, the next section defines the fractions of materials that are released. The release fractions are unique for each event; this section discusses release fractions for events that release material from the spent fuel to the cask cavity, from the cask cavity to the environment and from the environment to people.

From the Spent Fuel Assembly to the Cask Cavity (Step 1)

If spent fuel somehow is damaged and its cladding fails, an important question to be answered is: What fraction of radioactive material is released from the rod to the cask cavity? In this section, the release fractions are estimated for a single rod. When explicitly needed for calculating release fractions, the inventory in a single rod is assumed to be that for a typical PWR rod. Because the release fractions for spent fuel are a function of release mechanisms, each potential mechanism is considered separately. The first three discussed are phase 1 mechanisms and the last three are phase 2.

The release fractions from the impact-rupture mechanism are given in Table XI for various radionuclides. All of the noble gases that are in the fuel-clad gap are assumed to be released. Since no more than 20 percent of the noble gases generated during reactor operation would be expected in the gap of most current spent fuel,²⁶ the release fraction from the rod is 20 percent of the total inventory of the noble gases generated in the spent fuel. The remaining radionuclides are assumed to be released as particulates that would be swept out as the rod depressurizes after being ruptured. The fraction of particulates released is based on the fraction experimentally determined in Reference 25 for the burst rupture mechanism, which will be discussed

shortly. Impact rupture can be expected to generate more particles than were present in the spent fuel before impact and available for release during burst rupture. But, since the gas pressure in the rods that sweeps out particulate material is less for impact rupture and, since the release pathway would be expected to be more restricted by cladding deformation, the burst rupture release fractions were arbitrarily reduced to 10 percent and used for impact rupture.

TABLE XII
Release Fractions for Impact Rupture
(Phase 1)

Radionuclide	Fraction of Total Nuclide Inventory Released from a Spent Fuel Rod	Reference
Noble Gases	0.2	26
Cs 134	2×10^{-6}	25
Cs 137	2×10^{-6}	25
I 129	2×10^{-6}	25
Sr 90	2×10^{-6}	25
Ru 106	2×10^{-6}	25
Actinides	2×10^{-6}	25

The release fractions used for the leaching mechanism are presented in Table XIII. Two sets of values are given. The first set is to be applied to waterlogged rods and the second is to be applied to all other rods when the leaching occurs over a relatively short time. The values given for waterlogged rods are considerably greater than for the values given for shorter time periods because the waterlogged rod is assumed to have been leaching for a very long time before an accident. The maximum fraction of cesium that can be leached from spent fuel, reported in Reference 21, is used for the waterlogged-rod release fractions. The iodine values are assumed to be the same. The values for strontium and the actinides are taken from the column for one week of leaching time in Table B.3 of Reference 10.

TABLE XIII
Release Fractions for Leaching
(Phase 1)

Radionuclide	Fraction of Total Nuclide Inventory Released from a Spent Fuel Rod		Reference
	Waterlogged Rods	Other Rods	
Cs 134	3×10^{-3}	2×10^{-4}	10,21
Cs 137	3×10^{-3}	2×10^{-4}	10,21
I 129	3×10^{-3}	2×10^{-4}	10,21
Sr 90	2×10^{-4}	3×10^{-5}	10
Actinides	2×10^{-4}	2×10^{-5}	10

The other values are also taken from Table B.3 in Reference 10, but they are extrapolated for one hour of leaching. In most accidents, very little time will

be available for leaching. If a few rods are ruptured by an impact, the impact may have been severe enough to have caused a breach in the cask that would allow the coolant to escape. As the coolant level decreased, fewer rods would be exposed to the coolant. The leak rate from the cask then would determine the duration of leaching, which in this case might only be about an hour.

The values in Reference 10 are very conservative estimates of the leached fractions because they are based on tests performed on fuel fragments from high-burnup fuel that were in free contact with a flowing leachant.

The amount of crud released in an accident has not been determined experimentally; no data exist to support most of the following assumptions. Judgments made at the recent workshop¹ were that about 25 percent of the crud that plates on spent fuel assemblies during reactor operation and contaminates the cask cavity surfaces is loosely adhering (see Table XIV). The remainder adheres tightly and requires abrasion and chemical treatment for removal. Consequently, 25 percent of the crud is conservatively assumed to be knocked loose as a result of impact. This is in contrast to the 100 percent release assumed in the Urban Study.⁷ The Urban Study assumed that only the crud inventory on the rods may be released; however, this presentation assumes that the inventory not only on the rods but also on the assembly structure and the cask interior may be released. As a result, the two assumptions about the total amount of crud released are comparable. The total release to the cask cavity is assumed to be 25 percent of the entire crud inventory on the spent fuel rods, the assembly structure, and the cask cavity walls. As in the Urban Study, cobalt-60 is assumed here to produce the total impact and is the only nuclide considered for the crud mechanism.

TABLE XIV
Release Fraction for Crud Mechanism
(Phase 1)

Radionuclide	Total Available Inventory	Reference
Co 60	0.25	1

The predictions of release fractions for burst rupture (Table XV) and diffusion (Table XVI) rely heavily on recent work at Oak Ridge National Laboratory (ORNL), where a series of experiments was conducted to quantify and characterize fission product releases under conditions postulated for severe transportation accidents involving spent fuel. The release fractions are calculated using ORNL data and empirical equations.

Once a rod bursts, all of the fission gases in the fuel-clad gap are assumed to be released. From Reference 26, 20 percent of the noble gases generated by fissioning in reactor is estimated to be in the fuel-clad gap so that the total release would be 20 percent of the total noble gas inventory. The release fractions for cesium and iodine were calculated using equation (1) from Reference 27. The equations are based on ORNL experiments and depend on a number of adjustable parameters to fit the experimentally determined release curves. Detailed calculations can be found in Appendix A.

The release fraction for particulates (assumed to be strontium and actinide) is based on the particulate release found in Table 40 of Reference 25. The four values shown in that table are simply averaged and the average is adjusted for fuel rod length. Since the mass of particulates released from the 0.3-meter (1-foot) sections tested in Reference 25 would be the same as the mass released from the 3.6-meter (12-foot) fuel region of a typical PWR spent fuel rod, the release fraction, which is the ratio of the mass released to the total mass in the spent fuel, would be

TABLE XV
Release Fractions for Burst Rupture
(Phase 2)

Radionuclide	Fraction of Total Nuclide Inventory Released from a Spent Fuel Rod	Reference
Noble Gases	0.2	26
Cs 134	0.004	27
Cs 137	0.004	27
I 129	0.007	27
Sr 90	2×10^{-5}	25
Ru 106	2×10^{-5}	25
Actinides	2×10^{-5}	25

TABLE XVI
Release Fractions for Diffusion Mechanism
(Phase 2)

Radionuclide	Fraction of Total Nuclide Inventory Released from a Spent Fuel Rod	Reference
Noble Gases	0.02	27
Cs 134	5×10^{-4}	27
Cs 137	5×10^{-4}	27
I 129	4×10^{-3}	27

reduced by about a factor of ten since a typical spent fuel rod would have roughly ten times the mass of fission products in a 0.3-meter (1-foot) section of rod.

Once the rod has burst, the diffusion mechanism may proceed if steam is present, or the oxidation mechanism may proceed if dry air is present. As a result, this presentation assumes that any rod that fails by burst rupture is also affected by either the diffusion or oxidation mechanism.

The release fractions for the diffusion mechanism are calculated using equations (2) and (3) and Table V in Reference 27. The results are shown in Table XVI and the detailed calculations are found in Appendix A. The diffusion mechanism considered here is only for a steam atmosphere. Diffusion in an air atmosphere is considered to be oxidation.

Very little characterization of the oxidation mechanism has been performed. As a result, conservative assumptions are made regarding the release fractions that are listed in Table XVI. The diffusion-mechanism release fractions were adjusted according to estimates of releases in dry air given in Reference 25.

Oxidation is restricted to areas where the bare fuel is exposed. In order for oxidation to occur, rod failure is assumed to expose fuel surfaces along the fuel-clad gap. The release fractions for cesium and iodine are markedly increased when spent fuel is heated in dry air instead of steam, and significant releases of ruthenium

TABLE XVII
Release Fractions for Oxidation
(Phase 2)

Radionuclide	Fraction of Total Nuclide Inventory Released from a Spent Fuel Rod	Reference
Noble Gases	0.02	27
Cs 134	0.03	25
Cs 137	0.03	25
I 129	0.08	25
Ru 106	8×10^{-4}	25

occur. The release fractions for cesium and iodine were calculated for dry air by calculating the ratios of quantities released in air to quantities released in steam given in Reference 25 and then multiplying the diffusion releases in this presentation by the ratios. Values of 50 and 20 were the ratios calculated for cesium and iodine, respectively. The ruthenium release fraction was assumed to be about 0.01 times the release fraction for iodine.

As reported in Reference 24, the oxidation mechanism results in expansion of the spent fuel and can seal ruptures or holes. Unless the clad is stripped from the spent fuel, the oxidation mechanism may be self-limiting.

The release fractions for all of the spent fuel release mechanisms are found in Table XVIII.

TABLE XVIII
Summary of Phase 1 and Phase 2 Release Fractions for Spent Fuel Release Mechanisms
(Step 1)

Radionuclide	Phase 1				Phase 2		
	Impact Rupture	Leaching		Crud	Burst* Rupture	Diffusion (steam)	Oxidation (air)
		Waterlogged	Other				
Noble Gases	0.2	-	-	-	0.2	0.02	0.02
Cs 134	2×10^{-6}	0.003	2×10^{-4}	-	0.004	5×10^{-4}	0.03
Cs 137	2×10^{-6}	0.003	2×10^{-4}	-	0.004	5×10^{-4}	0.03
I 129	2×10^{-6}	0.003	2×10^{-4}	-	0.007	4×10^{-3}	0.08
Sr 90	2×10^{-6}	2×10^{-4}	3×10^{-5}	-	2×10^{-5}	-	-
Ru 106	2×10^{-6}	-	-	-	2×10^{-5}	-	8×10^{-4}
Actinides	2×10^{-6}	2×10^{-4}	2×10^{-5}	-	2×10^{-5}	-	-
Co 60	-	-	-	0.25	-	-	-
*Diffusion or oxidation releases must be added to burst releases when the burst rupture mechanism occurs.							

From the Cask Cavity to the Environment (Step 2)

In general, there are very few data to support assumptions about how much a cask restricts the release of radioactive materials once they have reached the cask cavity. Past conservative assumptions were simply to treat the cask as though it did not restrict release at all. In other words, the cask was assumed not even to exist once an accident had occurred.

This presentation bases its release fractions from the cavity to the environment on the collective judgment of a workshop conducted to analyze transportation accidents¹. The judgment is not founded on a wealth of supporting data, but it is supported by the experience of cask designers, transporters, material scientists and others familiar with cask properties. Tables XIX and XX reflect the judgment of the majority of participants in the workshop.¹ Release fractions are given for particulates, gases and volatiles; for water-cooled and gas-cooled casks and for the periods when the coolant is in the cavity and after it has been released. The rationale for the judgments is given in the following text.

Before they reach the environment, the radionuclides released from the spent fuel must pass through small passages that in most cases will be relatively cool. As a result, there are many places where the radionuclides can condense, settle, plate out or be filtered out before reaching an exit hole to the environment.

All noble gases released while water is in a cask cavity were assumed to be released while about 50% would be released after the water escaped. For gascooled casks that are maintained at greater than atmospheric pressure, Reference 3 predicts that 60 percent of the gases in the cavity would be discharged when the gaseous coolant was initially released. Even though not all casks are pressurized, a value of 60 percent was selected. Once a cask has been initially depressurized, release fractions for the noble gases would be considerably less. However, if a fire heats the gases in a cask, 50 percent release of the gases in the cavity can be predicted.³ So, a value of 50 percent was selected for the noble gas release fraction once a water-cooled or gas-cooled cask has had its coolant discharged.

TABLE XIX

Release Fractions from Cask Cavity to Environment
(Step 2)

Radionuclide	Water-Cooled Cask		Gas-Cooled Cask	
	In Water (Phase 1)	After Water Released (Phase 2)	In Gas Coolant (Phase 1)	After Coolant Released (Phase 2)
Noble Gases	1.0	0.5	0.6	0.5
Volatiles (Cs, I)	1.0	0.05	0.05	0.05
Particulates (Sr, Act, Co)	0.5*	0.05	0.05	0.05
*Value of 1.0 used for leaching.				

TABLE XX

Fractions Important in Pathway to People

Radionuclide	Fraction <10 μ m (aerodynamic diameter)	Fraction Aerosolized			
		Release with Water		Release with Gas	
		Fire	No Fire	Fire	No Fire
Noble Gases	1.0	1.0	1.0	1.0	1.0
Volatiles (Cs,I,Ru)	1.0	1.0	0/0.1*	1.0	1.0
Particulates (Sr,Actinides,Co)	0.05**	1.0	0.01	1.0	1.0
*Iodine **Value of 1.0 used for leaching Sr and actinides.					

The volatiles are released on a molecular scale and would not be expected to occur as particles larger than respirable size. They would be easily suspended in water coolant, where they would be expected to remain suspended much longer than in a gas coolant; volatiles released to a gaseous cask atmosphere would be much more likely to plate out on cask walls. For these reasons, 100 percent of the volatiles released to a water coolant are assumed to be released to the environment. A much smaller value of 5 percent was assumed for the release to a gaseous cask atmosphere.

The release fractions for particulates correspond to those for the volatiles except for the condition where a water-cooled cask still has its coolant in the cavity. Since the volatiles are very small particles, they would be expected to remain suspended longer in water than the particulates, which have sizes that could be considerably greater. The other values for volatiles and particulates were assumed to be the same.

As described in a later section, the fractions in Table XIX are multiplied by those in Table XVIII and others to calculate a total release fraction. As an example of how the fractions in Table XIX are used, consider a scenario where phase 1 and phase 2 release mechanisms are initiated. The release fractions associated with phase 1 mechanisms in Table XVIII would be multiplied by the fractions in the appropriate phase 1 column of Table XIX, and the release fractions associated with the phase 2 mechanisms in Table XVIII would be multiplied by the fractions in the appropriate phase 2 column in Table XIX.

From Environment to People

Once radioactive material has been released to the environment, a number of other factors become important in determining whether the radioactive material will reach people. Two important factors are the fraction of particles smaller than 10 microns aerodynamic diameter (particles less than this size are respirable) and the fraction of the material that becomes suspended in air. Table XX presents the values for these variables: volatiles, particulates and noble gases. Since the noble gases are already gases, the value for their factors is 1.

All gases released can be potentially inhaled. Because volatiles are released as very small particles, they too are potentially available for inhalation as long as they do not plate out. Cesium that is in solution probably will not be aerosolized

unless there is a fire. Iodine is present in the fuel-clad gap as CsI, which may oxidize to release gaseous iodine; an arbitrary 10 percent is assumed for the percentage aerosolized when there is no fire. Ruthenium is assumed to be released according to the same release fractions as cesium.

Particles released via the burst-rupture mechanism have been characterized in Reference 25. Table 42 in this reference indicates that no more than 3 percent of the particles released are smaller than 10 microns. So a value of 5 percent was assigned. All particles released in water coolant are conservatively assumed to be aerosolized in a fire while only 1 percent were assumed to be when no fire was present. All the particles released with the gaseous coolant are assumed to remain aerosolized.

Release Fractions from Each Scenario

In this section, the release fractions from each scenario are defined, and additional assumptions are discussed. These release fractions are presented in terms of the fraction of the total inventory. No attempt is made to define the total spent fuel inventories in this presentation.

Release fractions have been discussed in previous sections for the releases from spent fuel to the cask cavity (Step 1) and for the releases from the cask cavity to the environment (Step 2). In addition, the fractions aerosolized and respirable were discussed as important factors in the pathway to people. Only one additional factor needs to be discussed in order to use the previously defined factors to generate total release fractions that can be used for environmental-consequence analysis. The factor is the fraction of rod failures that occur during a scenario. Since this factor is completely dependent on the conditions described in the scenarios, it has not been discussed until now. Table XXI presents a list of values for this factor that apply to each spent fuel release mechanism for each scenario. For a review on what mechanisms are considered for each scenario, the reader is referred to Table X.

The use of a rod failure fraction for the crud-release mechanism is not strictly correct because the mechanism can occur even if no rods fail. All of the crud inventory is assumed to be available for release. This inventory includes all that is on the rods, the assembly support structures and the cask interior.

The leaching mechanism can occur whenever a rod has failed while water is in the cask cavity. Waterlogged rods can be leached as well as rods that have failed in Phase 1 from impact. The fraction of rods that might be waterlogged was assumed to be 0.0025, a value has been used previously in Reference 28. This value is believed to be a maximum since it is reported² that 1×10^{-4} would be the expected fraction of rod failures. A value of 0.001 is supported by undocumented studies produced by the United Kingdom Atomic Energy Agency. Ten percent of the unfailed rods were arbitrarily assumed to fail during impact in those scenarios that are severe enough. As a result, 0.0025 is the fraction of rods leached in Scenarios 2 and 3, and 0.1 is the fraction of rods leached in Scenario 4, a scenario which is severe enough to cause impact failure of the rods.

The burst-rupture mechanism is sensitive to the type of fuel. Calculations indicate that PWR rods are much more likely to fail than BWR rods,¹¹ and it was shown to be possible that in extreme environments all PWR rods would fail by the burst-rupture mechanism. Even though it is probably just as likely that a BWR assembly would be shipped as it is that a PWR assembly would be, all rods that had not been failed previously by some other mechanism are assumed to fail by the burst rupture mechanism, in keeping with the NEPA worst-case philosophy. As a result, 0.9975 is used in Scenario 3 and 0.9 is used for Scenarios 4 and 5.

TABLE XXI
Fraction of Rods Failed

Scenario	Mechanism	Fraction of Rods Failed
1	Crud	1.0*
2	Crud	1.0*
	Leaching (waterlogged)	0.0025
	Diffusion	0.0025
3	Crud	1.0*
	Leaching (waterlogged)	0.0025
	Burst Rupture	0.9975
	Diffusion	1.0
4	Crud	1.0*
	Leaching	0.1
	Impact Rupture	0.1
	Burst Rupture	0.9
	Diffusion	1.0
5	Crud	1.0*
	Impact Rupture	0.1
	Burst Rupture	0.9
	Oxidation	1.0
*No rods actually need to fail, but all of the crud inventory is assumed to be available for release.		

The diffusion and oxidation mechanisms can only occur if rods have been previously failed by another mechanism. As a result, all rods that have been failed in either Phase 1 or Phase 2 can be subjected to one or the other of these mechanisms. Except for Scenario 2, where the fraction is 0.0025, the fraction for oxidation or diffusion is 1.0.

All component fractions necessary to calculate total release fractions have been estimated. How the components are related is shown in Figure 22, which is a flow diagram for calculating total release fractions. At the top of the diagram are two parallel rows of boxes representing component release fractions. One row represents Phase 1, and the other represents Phase 2. Otherwise, the rows are identical; only one row needs to be examined closely. The first box on the left represents the release fraction from the spent fuel to the cask cavity, and the second the fraction of rods failed. The product of these two boxes represents the release fraction from Step 1. Their product multiplied times the third box, which represents the release in Step 2 from the cask cavity to the environment, is the total release fraction from Phase 1. The sum of the box for Phase 1 and of the corresponding box for Phase 2 is the total release fraction of radioactive material from a particular scenario. The sum is the total fraction of all radioactive material released to the environment. It is the released fraction of the total inventory of radioactive material in the cask, including particulates, gases and volatiles whether or not they are aerosolized or respirable. These release fractions will be assumed to be applicable to all spent fuel regardless

Figure 22. Flow diagram for calculating release fractions.

of time in the reactor, time out of the reactor (age) or reactor type. The one exception might be the age of the fuel. If the spent fuel to be considered is much older than one year, the likelihood of the burst rupture mechanism is greatly diminished. The actual age necessary to preclude this mechanism is quite dependent on the cask in which it is being carried. However, according to Figure 7, it appears reasonable that if the fuel is over two years old the burst-rupture mechanism could not occur unless a hotter and longer fire than the maximum credible fire postulated in this presentation occurs.

In order to determine the amount of radioactivity escaping from a generic scenario like those postulated here, the total release fraction is multiplied by the entire cask inventory of radioactive material. This particular total release would be used to determine such information as the external exposure source that will be encountered by emergency-response teams and the levels of site and equipment contamination.

More useful information can be obtained by multiplying this fraction by the fraction aerosolized. In Figure 22, the product is represented by the third box from the right. The sum of the boxes for Phase 1 and Phase 2 is the total release fraction from the spent fuel to the environment that is aerosolized. If this fraction is multiplied by the total cask inventory, the amount released and aerosolized is calculated. This amount is useful for determining levels of contamination distant from the site, calculating exposure to the public from ground contamination and calculating external exposure to the public from the passing cloud.

A final and very useful number can be calculated by multiplying this last factor by the fraction of material that is respirable ($<10 \mu\text{m}$ aerodynamic diameter). When the sum of these factors for Phase 1 and Phase 2 is calculated and multiplied by the cask inventory, the result is the amount of release that can be inhaled by people (first box on the right in Figure 22). In some instances, this can be a most important pathway to people.

In summary then, the flow diagram in Figure 20 shows how to compute three factors that are useful in calculating the consequences of the scenarios. These factors,

calculated for each scenario, are presented in Tables XXII, XXIII and XXIV. Because the actual calculations are quite repetitive and long, they have been relegated to Appendix B.

TABLE XXII

Total Fraction of Material Released from the Spent Fuel to the Environment

Radionuclide	Scenario				
	1	2	3	4	5
Co 60	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-2}
Noble Gases	-	3×10^{-5}	1×10^{-1}	1×10^{-1}	1×10^{-1}
Cs 134	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	1×10^{-3}
Cs 137	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	1×10^{-3}
I 129	-	9×10^{-6}	6×10^{-4}	4×10^{-4}	4×10^{-3}
Sr 90	-	5×10^{-7}	2×10^{-6}	4×10^{-6}	9×10^{-7}
Ru 106	-	-	1×10^{-6}	1×10^{-6}	4×10^{-5}
Actinides	-	5×10^{-7}	2×10^{-6}	3×10^{-6}	9×10^{-7}

TABLE XXIII

Total Fraction of Material Aerosolized and Released from the Spent Fuel to the Environment

Radionuclide	Scenario				
	1	2	3	4	5
Co 60	1×10^{-3}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-2}
Noble Gases	-	3×10^{-5}	1×10^{-1}	1×10^{-1}	1×10^{-1}
Cs 134	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	1×10^{-3}
Cs 137	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	1×10^{-3}
I 129	-	9×10^{-6}	6×10^{-4}	4×10^{-4}	4×10^{-3}
Sr 90	-	5×10^{-7}	2×10^{-6}	4×10^{-6}	9×10^{-7}
Ru 106	-	-	1×10^{-6}	1×10^{-6}	4×10^{-5}
Actinides	-	5×10^{-7}	2×10^{-6}	3×10^{-6}	9×10^{-7}

TABLE XXIV

Total Fraction of Aerosolized, Respirable Material Released from Spent Fuel to the Environment

Radionuclide	Scenario				
	1	2	3	4	5
Co 60	6×10^{-5}	6×10^{-3}	6×10^{-3}	6×10^{-3}	6×10^{-4}
Noble Gases	-	3×10^{-5}	1×10^{-1}	1×10^{-1}	1×10^{-1}
Cs 134	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	3×10^{-4}
Cs 137	-	8×10^{-6}	2×10^{-4}	3×10^{-4}	3×10^{-4}
I 129	-	9×10^{-6}	6×10^{-4}	4×10^{-4}	5×10^{-4}
Sr 90	-	5×10^{-7}	6×10^{-7}	3×10^{-6}	5×10^{-8}
Ru 106	-	-	5×10^{-8}	6×10^{-8}	2×10^{-6}
Actinides	-	5×10^{-7}	6×10^{-7}	2×10^{-6}	5×10^{-8}

Probability of Scenarios

The scenarios described previously are not likely to ever occur. It is the purpose of this section to try to quantify this statement. In other words, this section answers the question: Considering the expected number of shipments of spent fuel that will occur in the future and an average number of kilometers of shipment, what is the probability of occurrence of a transportation accident that involves spent fuel and is as severe as the postulated scenarios in this presentation?

To answer the question, this section presents a discussion of the average annual distance traveled by spent fuel shipments and describes experience in the United States with accidents that involve spent fuel shipments. Then, accident rates for all accident severities are presented, and estimates are given for the rates of the scenarios described in this presentation.

Kilometers Traveled by Spent Fuel Shipments

A prediction of an accident rate for spent fuel transportation can be made only if the number of annual transport kilometers of spent fuel is known. Such a number is not precisely known, but an estimate can be made.

The number of commercial spent fuel assemblies that have recently been shipped in the United States is known; from this number, the number of shipments can be estimated. The number of assemblies shipped in the United States since 1964 is reproduced from Reference 9 and presented as Table XXV.

Detailed information was not available for the analysis in Reference 9, but it is known that 2220 of the assemblies shown in Table XXV were from PWR's. Reference 9 did not specify whether the spent fuel transport mode was truck or railcar. However, very few of the shipments were rail shipments; most of the shipments were by truck. Current truck cask designs can transport only 1 PWR or 2 BWR assemblies per truck shipment. From Table XXV, if one assumes that all the spent fuel shipments were made by truck and there was one fuel element per shipping cask, the average number of annual shipments would be 300. If the shipments in Table XXV are actually charted between origin and destination, a reasonable estimate of approximately 320,000 kilometers per year can be obtained.

TABLE XXV

Summary of United States Spent Fuel Shipments⁹

Year	Assemblies Shipped
1964	264
1965	39
1966	165
1967	269
1968	67
1969	217
1970	180
1971	561
1972	93
1973	267
1974	137
1975	492
1976	712
1977	675
1978	512
1979	147
TOTAL	4797
Average shipment = 300 assemblies per year	

It should be noted that the number of spent fuel shipments may be different in the future. The shipment numbers in Table XXV represent the transport of some spent fuel for reprocessing and the transshipment of spent fuel between reactor spent fuel discharge basins to provide space for future discharges of spent fuel. At the present time the main inventory of spent fuel is still resident in the reactor basins. Future shipment patterns depend almost entirely on policies to be announced by the United States government.

The actual and estimated annual spent-fuel-shipment kilometers are displayed in Figure 23. The values in Table XXV are plotted in the lower left of the figure. The average of the actual shipping experience is approximately 320,000 kilometers per year. The estimated shipment kilometers are plotted to the right of the Table XXV data. Early estimates of what might be expected appear to be much too high when compared to the most recent projections announced by the Department of Energy (DOE). The DOE projections remain below 320,000 until about 1987 and then they escalate rather rapidly as away-from-reactor storage sites are made available for storing spent fuel that can no longer be stored at reactor sites. An additional line shown in Figure 23, represents the estimate of the Nuclear Regulatory Commission (NRC) in Reference 29.

Figure 23. Spent fuel shipping estimates.

Accident Experience for Spent Fuel Shipments

No transportation accident involving spent fuel has ever released radioactive material to the environment, and according to the Transportation Technology Center's accident data base,³⁰ only one accident involving spent fuel has occurred since 1971. In this accident, a truck hauling a spent fuel cask containing an assembly ran off a road and overturned, killing the driver. The spent fuel cask was undamaged, and no release occurred.

The accident described above was for the truck mode. Spent fuel transportation experience for the rail mode is very limited because very few shipments have been made by rail. No accidents involving spent fuel have occurred in rail transportation.

Accident Rates for Spent Fuel Shipments of All Severities ("Fender Benders" to Worst-Case Scenarios)

If one applies the annual average estimate of 320,000 kilometers for spent fuel shipments from 1971 through 1979 and recognizes the occurrence of one spent fuel accident, the accident rate is

$$1 \text{ Accident} / (2.9 \times 10^6 \text{ total kilometers}) = 3.5 \times 10^{-7} \text{ spent fuel accidents/kilometer}^{31}$$

This calculation assumes that all the spent fuel shipments have been made by truck, and it is subject to a very large uncertainty because it is based on a very limited accident base. The overall truck accident rate (including all truck shipments

of radioactive and nonradioactive materials) has been previously analyzed to be approximately 1.6×10^{-6} truck accident per kilometer.¹²

Since there have been no accidents involving spent fuel in rail transportation, the overall rail accident rate found to be 0.94×10^{-6} railcar accident per car-kilometer¹² is used here. This rate is based on statistics for all rail accidents involving all types of freight.

Probabilities and Accident Rates for Worst-Case Spent Fuel Scenarios

The accident rates given above are for accidents of all severities, which include "fender benders" and more severe accidents. In order to determine the accident rate for the worst-case scenarios, the fraction of all accidents that are at least as severe as the scenarios must be calculated.

Because there have been no accidents as severe as those postulated in the scenarios, the fraction of accidents at least as severe as the conditions in the scenarios must be estimated. McClure³¹ has estimated 0.1 percent for both truck and rail as the fraction of accidents involving impact that are more severe than the regulatory test conditions. His estimate for the fraction of accidents involving fire that are more severe than the regulatory test conditions is 0.2 percent for rail and 0.1 percent for truck. It is important to remember that the scenarios are more severe than the regulatory test conditions: although the regulations allow limited release of radioactive material, the scenarios postulate gross release.

Figures 24 and 25 in Reference 31 are the basis for the discussion above. The cumulative probability that an accident will produce a velocity change less than a given change is plotted against vehicle velocity change in Figure 24. The regulatory test condition, which is to drop a cask from 9 meters (30 feet) onto an unyielding target, is represented by the vertical line drawn from the 48.2 kph (30 mph) velocity

Figure 24. Probability that the velocity change in an accident will be less than a given velocity change.³²

Figure 25. Probability that the duration of a fire in an accident will be less than a given duration.³³

change. The regulations specify velocity change of the package, but the velocity changes in the figure are vehicle velocity changes, not equivalent package velocity. Since insufficient data exist to perform a detailed analysis of the velocity reduction resulting from crushing a vehicle, it is conservatively assumed that the impact velocity of the vehicle is the same as the impact velocity of the cask. To understand why this assumption is conservative, it is interesting to note that in Sandia tests,²³ when a vehicle hit a very hard surface head-on at 98 kph (61 mph), the resulting cask impact was around 43 kph (27 mph). In another test, where the vehicle velocity change was 135 kph (84 mph), the resulting cask impact velocity was about 99 kph (62 mph).

The regulatory tests also require that the impact be made on an essentially unyielding target. Since no truly unyielding targets occur in nature, a derating scheme can be used to estimate equivalent velocity changes for unyielding surfaces,³² which are labeled as moderately hard and relatively soft in Figure 24. It can be shown that a 63 kph (39 mph) velocity change on a hard target would produce damage equivalent to the 48.2 kph (30 mph) velocity change on an unyielding target.

Since in a real accident environment (e.g., collision with a concrete abutment) at least a 63 kph (39 mph) velocity change would be required to exceed the equivalent regulatory test conditions, Figure 24 is interpreted to mean that 99.9 percent of all accidents are less severe than the regulatory test conditions. Since the accident scenarios are more severe than the regulatory tests, the accident scenarios are more severe than at least 99.9 percent of all accidents.

A similar representation for the fire environment is shown in Figure 25. Cumulative probability curves are shown for truck and rail, and the regulatory test condition

is shown as the vertical line drawn from the 30-minute duration mark. The inherent conservatism implicit with drawing the line is that the regulation requires complete immersion in the flame. Because the accident scenarios are more severe than the regulatory test conditions, Figure 25 is interpreted to mean that the accident scenarios are more severe than 99.8 percent of all train accidents and 99.9 percent of all truck accidents. This percentage is even more conservative because the accident scenarios postulate an 1000°C fire.

Because most of the scenarios consider both fire and impact, the probabilities must be combined. Since the probabilities of fire and impact are assumed to be independent parameters, the fractions of accidents less severe than the scenarios involving both fire and impact are 99.9998 for train and 99.9999 for rail. The precision of such numbers can rightly be questioned because of the lack of data for severe accidents. The order of magnitude of the probability is probably more important; it is in the range of 1 in one million. That is, in every one million accidents of all severities, 1 or 2 accidents as severe as the scenarios involving both impact and fire could be expected. Table XXVI is a tabulation of probabilities for each of the five scenarios.

TABLE XXVI
Probabilities and Accident Rates of the Scenarios

		Scenario				
		1	2	3	4	5
Probability (per accident)						
	Truck	1 in 1000	1 in a million	----->		
	Rail	1 in 1000	2 in a million	----->		
Accident Rate (per kilometer)						
	Truck	3.5×10^{-10}	3.5×10^{-13}	----->		
	Rail	9.4×10^{-10}	1.9×10^{-12}	----->		

As can be seen in Table XXVI, the probabilities are very small. To determine the accident rates (accidents per kilometer of travel), the probabilities are multiplied by the overall accident rates for truck and rail given in the previous subsection. The results are also shown in Table XXVI. If the number of kilometers shipped per year is known, the frequency of accidents (accidents per year) can be determined by multiplying the accident rate by the number of kilometers shipped and then taking the reciprocal of the product. As an example, assuming 320,000 kilometers of spent fuel shipment per year and using the accident rate for rail for Scenario 1, one accident similar to Scenario 1 would be expected to occur no more frequently than once in 3300 years.

The accident probabilities calculated above are based on the limited shipping of spent fuel that has occurred to date. It must be emphasized that projections of future spent fuel accident occurrences must be based upon a continuing review of shipping experience, coupled with projected shipping volumes, which depends on government policies, and the actual locations of away-from-reactor storage sites and repository sites.

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APPENDIX A

Release Fraction Calculations

Release Fractions for the Burst Rupture Mechanism

The amount of fission product released is calculated using Equation (A-1).²⁷

$$M_B = \alpha V_B (M_O/A)^a \exp[-(C/T)] \quad (A-1)$$

where,

M_B = mass of element released in the burst, in g

V_B = volume of plenum gas vented, in cm^3 at 0°C and system pressure

M_O = inventory of element in the gap space, in g

A = internal area of the cladding associated with M_O , in cm^2

T = temperature at the rupture location, in $^\circ\text{K}$

a , α and C are adjustable constants that were experimentally determined.

M_O is calculated using the equation:

$$M_O = Mf \quad (A-2)$$

where,

M = the total mass of a particular radionuclide in the spent fuel rod, in g

f = the fraction of the element that is in the fuel-clad gap

As a result, the equation for the amount of the radionuclide released becomes:

$$M_B = \alpha V_B \left(\frac{Mf}{A} \right)^a \exp[-C/T] \quad (A-3)$$

Finally, the equation for the fraction of the total activity released is simply Equation (A-3) divided by M

$$A_{fB} = \frac{\alpha V_B}{M} \left(\frac{Mf}{A} \right)^a \exp[-C/T] \quad (A-4)$$

The values for the parameters used are given in Table A-I.

TABLE A-I

Parameters

Radionuclide	$\alpha \left(\frac{g}{cm^3} \right) \cdot \left(\frac{g}{cm^3} \right)^{-a}$ Ref 34	a Ref 34	V_{B3} (cm ³) Ref 34	M(g) (Ci) Ref 22	f Ref 26	A (cm ²) Ref 34	C (°K) Ref 34	T (°K)
Cs	3.49	0.8	1100	6.4	0.2	1100	7.4x10 ³	1123
I	0.163	0.8	1100	0.53	0.2	1100	3.8x10 ³	1123

In calculating the values for Table A-I, certain assumptions had to be made. The PWR spent fuel described in Reference 22 was assumed to be representative for burnup, and the fuel was assumed to be 180-days old. The temperature at which burst rupture was assumed to occur was 850°C.

Substituting the parameters in Table A-I, the release fractions for cesium and iodine were calculated from Equation A-4. The results are shown in Table A-II.

TABLE A-II

Release Fractions for Burst Rupture

Radionuclide	Release Fraction
Cs	3.6x10 ⁻³
I	7.4x10 ⁻³

Release Fractions for the Diffusion Mechanisms in Steam

The amount of fission product released by the diffusion mechanism is calculated using Equations (2) and (3) from Reference 27,

$$M_D = M_0 \{ 1 - \exp [-(R_0 t / M_0)] \} \quad (A-5)$$

where,

M_D = mass of element released by diffusion, in g

t = time at diffusion temperature, in h

R_0 = initial rate of release by diffusion, in g/h

R_0 is defined by the equation:

$$R_0 = \delta(W/P)(M_0/A)^a \exp[-\gamma/T] \quad (A-6)$$

where,

W = the width of the radial gap, in μm

P = system pressure, in MPa

δ and γ are adjustable constants that were experimentally determined.

R_0 must be evaluated before the release fraction can be evaluated. Using the relationship for M_0 , Equation (A-2), given in the burst release calculations, Equation (A-6) becomes:

$$R_0 = \delta(W/P) \left(\frac{Mf}{A} \right)^a \exp[-\gamma/T] \quad (A-7)$$

The values for the parameters are given in Table A-III and in Table A-I. When these values are substituted in Equation (A-7), the values for R_0 can be obtained. The results are shown in Table A-IV.

TABLE A-III

Parameters

Radionuclide	$\delta \left(\frac{\text{g} \cdot \text{MPa}}{\text{m} \cdot \text{h}} \right) \cdot \left(\frac{\text{g}}{\text{cm}^2} \right)^{-a}$ Ref 34	W (m) Ref 34	P (MPa) Ref 34	γ (°K) Ref 34	T (°K)
Cs	1900	200	0.1	2.0×10^4	1123
I	120	200	0.1	1.5×10^4	1123

TABLE A-IV

R_0 Values

Radionuclide	Release Rate
Cs	3.8×10^{-4} g/hr
I	2.8×10^{-4} g/hr

These results are now substituted into a version of Equation (A-5) modified to produce a release fraction:

$$M_{fD} = f \left\{ 1 - \exp \left[- \frac{R_0 t}{Mf} \right] \right\} \quad (A-8)$$

A value of eight hours is selected for t . The calculated values for M_{fD} are found in Table A-V.

TABLE A-V
Release Fractions for Diffusion Mechanism in Steam

Radionuclide	Release Fraction
Cs	4.7×10^{-4}
I	4.2×10^{-3}

APPENDIX B

Table of Release Fractions

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APPENDIX B

Table of Release Fractions

Scenario	Phase	Mechanism	Nuclide	Fraction SF to Cavity	Rods Failed	Release from Cavity to Environment	Release from SF to Environment	Fraction Aerosolized	Released SF to Environment/Aerosolized	Respirable	Released SF to Environment/Aerosolized and Respirable
1	1	Crud	Co 60	0.25	NA	0.5	0.125	0.01	0.001	0.05	6×10^{-5}
2	1	Crud Leaching	Co 60	0.25	NA	0.5	0.125	1.0	1×10^{-1}	0.05	6×10^{-3}
			Cs 134	0.003	0.0025	1.0	8×10^{-6}	1.0	8×10^{-6}	1.0	8×10^{-6}
			Cs 137	0.003	0.0025	1.0	8×10^{-6}	1.0	8×10^{-6}	1.0	8×10^{-6}
			I 129	0.003	0.0025	1.0	8×10^{-6}	1.0	8×10^{-6}	1.0	8×10^{-6}
			Sr 90	2×10^{-4}	0.0025	1.0	5×10^{-7}	1.0	5×10^{-7}	1.0	5×10^{-7}
	2	Diffusion	Pu 239	2×10^{-4}	0.0025	1.0	5×10^{-7}	1.0	5×10^{-7}	1.0	5×10^{-7}
			NG	0.02	0.0025	0.5	3×10^{-5}	1.0	3×10^{-5}	1.0	3×10^{-5}
			Cs 134	5×10^{-4}	0.0025	0.05	6×10^{-8}	1.0	6×10^{-8}	1.0	6×10^{-8}
			Cs 137	5×10^{-4}	0.0025	0.05	6×10^{-8}	1.0	6×10^{-8}	1.0	6×10^{-8}
			I 129	4×10^{-3}	0.0025	0.05	5×10^{-7}	1.0	5×10^{-7}	1.0	5×10^{-7}
3	1	Crud Leaching	Co 60	0.25	NA	0.5	0.125	1.0	1×10^{-1}	0.05	6×10^{-3}
			Cs 134	0.003	0.0025	1.0	8×10^{-6}	1.0	8×10^{-6}	1.0	8×10^{-6}
			Cs 137	0.003	0.0025	1.0	8×10^{-6}	1.0	8×10^{-6}	1.0	8×10^{-6}
			I 129	0.003	0.0025	1.0	8×10^{-6}	1.0	8×10^{-6}	1.0	8×10^{-6}
			Sr 90	2×10^{-4}	0.0025	1.0	5×10^{-7}	1.0	5×10^{-7}	1.0	5×10^{-7}
	2	Diffusion	Pu 239	2×10^{-4}	0.0025	1.0	5×10^{-7}	1.0	5×10^{-7}	1.0	5×10^{-7}
			NG	0.02	1.0	0.5	1×10^{-2}	1.0	1×10^{-2}	1.0	1×10^{-2}
			Cs 134	5×10^{-4}	1.0	0.05	3×10^{-5}	1.0	3×10^{-5}	1.0	3×10^{-5}
			Cs 137	5×10^{-4}	1.0	0.05	3×10^{-5}	1.0	3×10^{-5}	1.0	3×10^{-5}
			I 129	4×10^{-3}	1.0	0.05	2×10^{-4}	1.0	2×10^{-4}	1.0	2×10^{-4}
	2	Burst	NG	0.2	0.9975	0.5	0.1	1.0	0.1	1.0	0.1
			Cs 134	0.004	0.9975	0.05	2×10^{-4}	1.0	2×10^{-4}	1.0	2×10^{-4}
			Cs 137	0.004	0.9975	0.05	2×10^{-4}	1.0	2×10^{-4}	1.0	2×10^{-4}
			I 129	0.007	0.9975	0.05	4×10^{-4}	1.0	4×10^{-4}	1.0	4×10^{-4}
			Sr 90	2×10^{-5}	0.9975	0.05	1×10^{-6}	1.0	1×10^{-6}	0.05	5×10^{-8}
			Ru 106	2×10^{-5}	0.9975	0.05	1×10^{-6}	1.0	1×10^{-6}	0.05	5×10^{-8}
			Pu 239	2×10^{-5}	0.9975	0.05	1×10^{-6}	1.0	1×10^{-6}	0.05	5×10^{-8}
			Co 60	0.25	NA	0.5	0.1	1.0	0.1	0.05	6×10^{-3}
			Cs 134	2×10^{-4}	0.1	1.0	2×10^{-5}	1.0	2×10^{-5}	1.0	2×10^{-5}
			Cs 137	2×10^{-4}	0.1	1.0	2×10^{-5}	1.0	2×10^{-5}	1.0	2×10^{-5}
			I 129	2×10^{-4}	0.1	1.0	2×10^{-5}	1.0	2×10^{-5}	1.0	2×10^{-5}
4	1	Crud Leaching	Sr 90	3×10^{-5}	0.1	1.0	3×10^{-6}	1.0	3×10^{-6}	1.0	3×10^{-6}
			Pu 239	2×10^{-5}	0.1	1.0	2×10^{-6}	1.0	2×10^{-6}	1.0	2×10^{-6}
			NG	0.2	0.1	1.0	2×10^{-2}	1.0	2×10^{-2}	1.0	2×10^{-2}
			Cs 134	2×10^{-6}	0.1	0.5	1×10^{-7}	1.0	1×10^{-7}	0.05	5×10^{-9}
			Cs 137	2×10^{-6}	0.1	0.5	1×10^{-7}	1.0	1×10^{-7}	0.05	5×10^{-9}
	1	Impact	I 129	2×10^{-6}	0.1	0.5	1×10^{-7}	1.0	1×10^{-7}	0.05	5×10^{-9}
			Sr 90	2×10^{-6}	0.1	0.5	1×10^{-7}	1.0	1×10^{-7}	0.05	5×10^{-9}
			Ru 106	2×10^{-6}	0.1	0.5	1×10^{-7}	1.0	1×10^{-7}	0.05	5×10^{-9}
			Pu 239	2×10^{-6}	0.1	0.5	1×10^{-7}	1.0	1×10^{-7}	0.05	5×10^{-9}
			NG	0.2	0.9	0.5	2×10^{-4}	1.0	9×10^{-2}	1.0	9×10^{-2}
			Cs 134	0.004	0.9	0.05	3×10^{-5}	1.0	2×10^{-4}	1.0	2×10^{-4}
			Cs 137	0.004	0.9	0.05	9×10^{-4}	1.0	2×10^{-4}	1.0	2×10^{-4}
			I 129	0.007	0.9	0.05	2×10^{-4}	1.0	2×10^{-4}	1.0	2×10^{-4}
			Sr 90	2×10^{-5}	0.9	0.05	9×10^{-7}	1.0	9×10^{-7}	0.05	5×10^{-8}
			Ru 106	2×10^{-5}	0.9	0.05	9×10^{-7}	1.0	9×10^{-7}	0.05	5×10^{-8}
			Pu 239	2×10^{-5}	0.9	0.05	9×10^{-7}	1.0	9×10^{-7}	0.05	5×10^{-8}
	2	Burst	NG	0.02	1.0	0.5	1×10^{-2}	1.0	1×10^{-2}	1.0	1×10^{-2}
			Cs 134	5×10^{-4}	1.0	0.05	3×10^{-5}	1.0	3×10^{-5}	1.0	3×10^{-5}
			Cs 137	5×10^{-4}	1.0	0.05	3×10^{-5}	1.0	3×10^{-5}	1.0	3×10^{-5}
			I 129	4×10^{-3}	1.0	0.05	2×10^{-4}	1.0	2×10^{-4}	1.0	2×10^{-4}
			Co 60	0.25	NA	0.05	1×10^{-2}	1.0	1×10^{-2}	0.05	6×10^{-4}
5	1	Crud Impact	NG	0.2	0.1	0.6	1×10^{-2}	1.0	1×10^{-2}	1.0	1×10^{-2}
			Cs 134	2×10^{-6}	0.1	0.05	1×10^{-8}	1.0	1×10^{-8}	0.05	5×10^{-10}
			Cs 137	2×10^{-6}	0.1	0.05	1×10^{-8}	1.0	1×10^{-8}	0.05	5×10^{-10}
			I 129	2×10^{-6}	0.1	0.05	1×10^{-8}	1.0	1×10^{-8}	0.05	5×10^{-10}
			Sr 90	2×10^{-6}	0.1	0.05	1×10^{-8}	1.0	1×10^{-8}	0.05	5×10^{-10}
			Ru 106	2×10^{-6}	0.1	0.05	1×10^{-8}	1.0	1×10^{-8}	0.05	5×10^{-10}
			Pu 239	2×10^{-6}	0.1	0.05	1×10^{-8}	1.0	1×10^{-8}	0.05	5×10^{-10}
			NG	0.02	0.9	0.5	9×10^{-2}	1.0	9×10^{-2}	1.0	9×10^{-2}
			Cs 134	0.004	0.9	0.05	2×10^{-4}	1.0	2×10^{-4}	1.0	2×10^{-4}
			Cs 137	0.004	0.9	0.05	2×10^{-4}	1.0	2×10^{-4}	1.0	2×10^{-4}
			I 129	0.007	0.9	0.05	3×10^{-4}	1.0	3×10^{-4}	1.0	3×10^{-4}
	2	Burst	Sr 90	2×10^{-5}	0.9	0.05	9×10^{-7}	1.0	9×10^{-7}	0.05	5×10^{-8}
			Ru 106	2×10^{-5}	0.9	0.05	9×10^{-7}	1.0	9×10^{-7}	0.05	5×10^{-8}
			Pu 239	2×10^{-5}	0.9	0.05	9×10^{-7}	1.0	9×10^{-7}	0.05	5×10^{-8}
			NG	0.02	1.0	0.5	1×10^{-2}	1.0	1×10^{-2}	1.0	1×10^{-2}
			Cs 134	0.03	1.0	0.05	2×10^{-3}	1.0	1×10^{-3}	0.05	8×10^{-5}
			Cs 137	0.03	1.0	0.05	2×10^{-3}	1.0	1×10^{-3}	0.05	8×10^{-5}
			I 129	0.08	1.0	0.05	4×10^{-3}	1.0	4×10^{-3}	0.05	2×10^{-4}
			Ru 106	8×10^{-4}	1.0	0.05	4×10^{-5}	1.0	4×10^{-5}	0.05	2×10^{-6}
			Co 60	0.25	NA	0.05	1×10^{-2}	1.0	1×10^{-2}	0.05	6×10^{-4}
			NG	0.2	0.1	0.6	1×10^{-2}	1.0	1×10^{-2}	1.0	1×10^{-2}
			Cs 134	2×10^{-6}	0.1	0.05	1×10^{-8}	1.0	1×10^{-8}	0.05	5×10^{-10}

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