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TOWARD A RISK ASSESSMENT OF THE SPENT FUEL AND HIGH-LEVEL NUCLEAR WASTE DISPOSAL SYSTEM

RISK ASSESSMENT REQUIREMENTS, LITERATURE REVIEW, METHODS EVALUATION - AN INTERIM REPORT

MASTER

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1 INTRODUCTION AND SUMMARY

This report provides background information for a risk assessment of the disposal system for spent nuclear fuel and high-level radioactive waste (HLW). It contains a literature review, a survey of the statutory requirements for risk assessment, and a preliminary evaluation of methods.

The literature review outlines the state of knowledge of risk assessment and accident consequence analysis in the nuclear fuel cycle and its applicability to spent fuel and HLW disposal.

The survey of statutory requirements determines the extent to which risk assessment may be needed in development of the waste-disposal system.

The evaluation of methods reviews and evaluates merits and applicabilities of alternative methods for assessing risks and relates them to the problems of spent fuel and HLW disposal.

1.1 SUMMARY

There are methods, models, and data from which to make a risk assessment of the disposal of spent nuclear fuel and HLW, including transportation and preclosure storage. Although risk assessment is required by present regulations only for the postclosure period, a probabilistic approach to preclosure risk assessment may also be warranted, because it provides the most thorough analysis. Probabilistic risk assessment (PRA) is useful early for guiding design and identifying problems. Presentation of the radioactive waste disposal project to the public demands a thorough analysis of risks.

Previous risk assessments of HLW waste disposal have dealt more with transportation than with preclosure storage. These unanimously indicate that the radiological risks of waste disposal are extremely small. Nevertheless, these assessments have tended to err on the side of accidents that are over-severe. Unless a threshold level for the probability of an accident is agreed on publicly, an issue that will persist is the choice of a "maximum credible accident" because probability-consequence curves typically do not have a sharp cutoff point.

Available assessment procedures derive primarily from safety assessment of nuclear reactors. Although risks of waste disposal are expected to be orders of magnitude lower, reactor experience suggests that particular attention should be given to certain aspects of the PRA, such as human factors and recovery and repair.

Where nuclear reactor safety assessment work applies to waste disposal—as in consequence modeling—assumptions, default values in programs have to be reviewed and revised as appropriate. Among existing computer codes, RADTRAN is a better starting point than CRAC II which was developed for reactors. Besides, some needed data are unique to waste disposal. In particular, source terms describing radioactive releases from damaged waste-disposal packages need some updating and more testing to better the data base.

As for risk, nonradiological effects of accidents are estimated to be larger than radiation effects. There are good historical data bases for analyzing these. The economic consequences of possible waste disposal accidents have not been studied as thoroughly.

Risk assessment can be based on present regulations, or it can aim at a way for better communication with the States and Indian tribes. The risk assessment must also be adequate for the scientific community and the courts and, ultimately, it must satisfy a skeptical public.

2 SCOPE

This report provides material which supports the U.S. Department of Energy Office of Civilian Radioactive Waste Management's efforts to prepare environmental-impact information required for developing a system for permanent disposal of spent nuclear fuel and high-level wastes. An Environmental Impact Statement should treat likely health and safety, environmental, economic, and social impacts, any of which may be either positive or negative for the community concerned. The negative impacts may be due to accidents, or natural phenomena such as tornadoes and earthquakes, or sabotage. This study focuses on the literature of a subset of the total requirements — health and economic risks of accidents — for the waste disposal system from pickup of spent fuel at a nuclear reactor or of solidified high-level-defense wastes at a reprocessing plant until closure of the repository.

The study outlines statutory regulations and requirements of risk assessment, requirements for licensing information, and state of knowledge of accident consequence analysis and risk-assessment methods. We hope that this review describes current capabilities — their strengths and weaknesses — and evaluates the suitability of available methods for waste-disposal systems.

3 STATUTORY AND REGULATORY REQUIREMENTS FOR RISK ASSESSMENT

This section outlines the general statutory and regulatory requirements for risk assessment, with special attention to requirements for assessments of accidents. The statutes as written are often vague about the specific risk assessments needed, but they do define minimum requirements as summarized in Table 3.1. This table does not include requirements related to monitored retrievable storage.

3.1 NUCLEAR WASTE POLICY ACT (NWPA) OF 1982, 42 USC 10101-10226[1]

10132 (b)(1)(E). Each nomination of a site for characterization shall be accompanied by an environmental assessment which shall include:

(iii) An evaluation of effects of site characterization activities on public health and safety and the environment.

(vi) An assessment of the regional and local impacts of locating a repository at that site.

10134 (f). The NWPA specifies that any recommendation for a repository site shall be considered a major Federal action significantly affecting the quality of the human environment. It must, therefore, be accompanied by a final environmental-impact statement, with all that implies for risk assessment requirements, except that there is no need to consider the null alternative of having no repository.

10136 (c)(1)(B). Grants shall be made to each State in which a candidate site for a repository is approved to enable the State to:

(i) Review repository-related activities to determine any potential economic, social, public health and safety, and environmental impacts;

TABLE 3.1. Summary of Statutory Requirements For Health and Environmental Consequence Analysis.

SITE NOMINATION FOR CHARACTERIZATION

Environmental Assessment:

For each site, evaluate effects of site characterization activities on public health and safety and the environment. Include evaluation of risk of transporting wastes to the site.

SITE SELECTION

Environmental Impact Statement:

Environmental Impact Statements currently must include consequence analysis of a worst case with an indication of its probability. They must also include a spectrum of events of higher probability but less drastic impacts. The worst case requirement is under review, and may be changed.*

Comparison of sites in the Environmental Impact Statement must include for each a Performance Assessment that predicts the effects of a repository as a system, preclosure and postclosure. It must evaluate the responses of the repository to conditions that might affect its performance, including natural events, human actions, and interactions between the wastes and the repository. The assessment must include estimates of the effects of uncertainties in data or modeling. Demonstration of compliance with Environmental Protection Agency postclosure standards expressed in probabilistic terms implies some form of probabilistic risk assessment.

LICENSING

Safety Analysis Report:

Determine critical pathways for radionuclide migration to the accessible environment. Evaluate postclosure rates and quantities of radionuclide release to the accessible environment under anticipated and unanticipated conditions.

CONSTRUCTION PERMIT

Environmental Report:

Describe adverse impacts that cannot be avoided.

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*The worst case requirement has been rescinded effective May 27, 1986 (51 FR 15618, April 25, 1986).

(ii) Develop requests for impact assistance;

(iii) Engage in any monitoring, testing, or evaluation required during site characterization;

(iv) Provide information to its residents; and

(v) Request from DOE information and make comments on any site-related activities.

A federal appeals court has ruled that states can conduct independent studies of proposed nuclear waste repositories and DOE must pay for them. The studies may overlap or duplicate DOE efforts. (State of Nevada vs. John Herrington, Ninth U.S.Circuit Court of Appeals, CA NO. 84-7846, December 2, 1985)

10137 (b). In performing any study of an area within a state, DOE shall consult with the State and any affected tribe in an effort to resolve their concerns about public health and safety, environmental, and economic impacts of a repository. DOE shall take such concerns into account to the maximum extent feasible and as specified in written agreements.

10155 (c). Provision of 300 metric tons or more of interim storage capacity at any one federal site shall require an environmental-impact statement. Provision of less than 300 metric tons of capacity at one site shall require an environmental assessment of the probable impacts including:

(iv) An evaluation of the effects on public health and safety, and the environment;

(vi) An assessment of the regional and local impacts, including the impacts on transportation.

The State and Tribal Council shall have the right to participate in a process of consultation and cooperation, based on public health and safety and environmental concerns, in all stages of the development.

10161 (c). Submission of a proposal for a monitored retrievable storage facility requires an environmental assessment, not an environmental impact statement.

3.2 DEPARTMENT OF ENERGY, GENERAL GUIDELINES FOR THE RECOMMENDATION OF SITES FOR NUCLEAR WASTE REPOSITORIES, 10 CFR 960[2]

Nomination for site characterization requires an environmental assessment, including evaluation of the effects of site-characterization activities on public health and safety and the environment, and an assessment of the regional and local impacts of locating a repository at that site. The environmental assessment must specifically include evaluation of the risk of transporting waste to the site and an analysis of emergency response requirements and capabilities related to transportation.

The site-selection phase must include a detailed performance assessment for each site. DOE will predict the effects of a repository as an entire system, during the time it is open for emplacement of waste and after it has been closed. The assessment will evaluate the responses of the repository to the conditions that might affect its performance, including natural events and processes, human actions, and interactions between the waste and the repository. This assessment will be the first based on the more detailed information generated by site characterization activities.

"Performance assessment" means any analysis that predicts the behavior of a system or system component under a given set of constant and/or transient conditions. It must include estimates of effects of uncertainties in data and modeling.

Recommendation of a site requires a final environmental impact statement in accordance with NEPA as modified by Section 114(f). DOE 10 CFR 960 specifically defers to 10 CFR 60, 40 CFR 191, and 10 CFR 20 with respect to health and environmental impact assessment requirements. Guidelines will be revised as necessary to remain consistent.

3.3 NATIONAL ENVIRONMENTAL POLICY ACT OF 1970, 40 CFR 1500-1508[3]

1500. Establishes requirements for environmental impact statements.

1502.22. When there are gaps in relevant information or scientific uncertainty which are impossible or unreasonably expensive to close, and the information is essential to a reasoned choice, the environmental-impact statement must include a worst-case analysis with an indication of the probability or improbability of its occurrence.

46 FR 18026, Questions and Answers[4] - Worst-case analysis must use reasonable projections of the worst-possible consequences of a proposed action. In addition to low-probability/catastrophic impact events, analyses should include a spectrum of events of higher probability but less drastic impact. "One of the federal government's most important obligations is to present the full spectrum of consequences."

50 FR 154 (1985), Proposed Changes[5*] - Eliminate the worst-case analysis requirement in the face of information gaps or scientific uncertainty. Instead, disclose that the information is missing, explain the relevance of the missing information to an evaluation of significant adverse impacts on the human environment, summarize existing scientific evidence relevant to the analysis, and present the agency's evaluation of that scientific evidence.

"Scientific credibility" is the threshold to trigger the requirements of the proposed change. In identifying potentially significant adverse impacts, an agency must forecast those consequences that have a low probability of occurrence but have potentially catastrophic consequences when there is

*The proposed changes become effective May 27, 1986 (51 FR 15618, April 25, 1986.

credible scientific support to suggest that the impact could occur as a result of the proposed action. Analysis should be focused on reasonably foreseeable accidents and generate information and discussion on those consequences of greatest concern to the public and of greatest relevance to the agency's decision. The requirement to disclose all credible scientific evidence extends to what are generally considered "minority views" within the scientific community or those views that are opposite those subscribed to by the agency. Probable remoteness of an impact does not excuse an agency from an evaluation of those impacts when there is a body of data with which an evaluation can be made that is not unreasonably speculative.

The meanings of important terms in this proposal will be illustrated with examples if is is adopted (CEQ 9/19/85).

3.4 NUCLEAR REGULATORY COMMISSION, STANDARDS FOR PROTECTION AGAINST RADIATION, 10 CFR 20[6]

Specifies permissible doses, levels, and concentrations in restricted and unrestricted areas,for public and occupational exposures, with special limits for minors. No risk assessments are required.

3.5 NUCLEAR REGULATORY COMMISSION, ENVIRONMENTAL PROTECTION REGULATIONS, 10 CFR 51[7]

Although these regulations apply generally to a nuclear waste repository, they are superseded by the Nuclear Waste Policy Act and the more specific requirements in 10 CFR 60 except for requirements for an environmental report. Accident risks are mentioned specifically only for transportation of fuel to and from reactors.

3.6 NUCLEAR REGULATORY COMMISSION, DISPOSAL OF HIGH-LEVEL RADIOLOGICAL WASTES IN GEOLOGIC REPOSITORIES, 10 CFR 60[8]

60.21. License applications must include a Safety Analysis Report and an environmental report as required by 10 CFR 51. The Safety Analysis Report must

(i) Determine critical pathways for radionuclide migration from the underground facility to the accessible environment;

(ii) Evaluate postclosure rates and quantities of radionuclide releases to the accessible environment under anticipated and unanticipated conditions, including effectiveness of engineered barriers and performance of major design structures, systems, and components;

(iii) Describe and analyze design and performance requirements for structures, systems, and components important for safety, including margins of safety, mitigations of consequences of accidents of natural and human origin;

(iv) Describe program for control and monitoring of occupational radiation exposures in accordance with the requirements of 10 CFR 20; and

(v) Describe plans for coping with preclosure radiological emergencies.

The Safety Analysis Report is not required to include assessment of health or economic risks of accidents.

60.31. Construction authorization must be based on a determination that the repository design proposed can be operated "without unreasonable risk to the health and safety of the public."

60.101. Satisfying performance objectives and site and design criteria is sufficient to support a finding of no unreasonable risk.

3.7 NUCLEAR REGULATORY COMMISSION, PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIAL, 10 CFR 71[9]

71.5. Licensees must comply with DOE requirements in 10 CFR 20, 21, 30, 40, 70, and 73, and DOT requirements in 49 CFR 170-189.

Subpart E. External radiation standards provide:

(i) Limits at the surface of the package depending on transport variables;

(ii) Limits for vehicle occupants and at two meters from vehicles;

(iii) Limits for escape of radioactive material under specified accident conditions including drop, puncture, fire, and water immersion.

3.8 NUCLEAR REGULATORY COMMISSION, LICENSING REQUIREMENTS FOR THE STORAGE OF SPENT FUEL IN AN INDEPENDENT SPENT FUEL STORAGE INSTALLATION, 10 CFR 72[10]

72.15. A Safety Analysis Report is required for licensing independent spent fuel storage installations. The Report must:

(i) Describe design bases for external events;

(ii) Provide other specified descriptions sufficient to support a finding of an adequate margin of safety;

(iii) Analyze and evaluate the design and performance of structures, systems, and components important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the facility, including margins of safety during normal operations, adequacy of structures, systems, and components provided for prevention of accidents and mitigation of consequences of accidents of natural and human origin;

(iv) Describe means for controlling and limiting occupational radiation exposures within the limits specified in 10 CFR 20, and for keeping exposures as low as is reasonably achievable;

(v) Describe plans for coping with emergencies;

(vi) Estimate quantities of normal annual releases of radioactive material;

(vii) Demonstrate compliance with exposure criteria inside and outside controlled areas (72.67-68); and

(viii) Analyze the potential dose or dose commitment to an individual outside the controlled area from accidents or natural events (during normal operations and decommissioning) causing release of radionuclides or direct radiation to the external environment from a design basis event as specified in Subpart E of 10 CFR 72. It must include (72.61) examination of the frequency and severity of external natural and man-induced events affecting safe operation of the facility and evaluation of potential for radiological and other environmental impacts on the region, with due consideration of the characteristics of the population, its distribution, and the regional environs, including historical and esthetic values. Transportation of spent fuel to the facility must be included with the site evaluation (72.70).

72.20. An Environmental Report is required for licensing as specified in Subpart A of 10 CFR 51.

3.9 ENVIRONMENTAL PROTECTION AGENCY, ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR MANAGEMENT AND DISPOSAL OF SPENT FUEL, HIGH-LEVEL AND TRANSURANIC RADIOACTIVE WASTES, 40 CFR 191[11]

Subpart A. Management and storage (excluding transportation) must be conducted so as to provide reasonable assurance that the combined annual dose equivalent to any member of the public in the general environment, from all NRC-regulated facilities, will not exceed a specified limit, and those from all non-NRC-regulated facilities (defense wastes) will not exceed another limit. Alternative limits for defense wastes may be set under special conditions.

Subpart B. Disposal systems must be designed to provide reasonable expectation that:

(i) Cumulative releases of radionuclides to the accessible environment from all significant processes and events for 10,000 years after disposal will not exceed standards expressed as probabilities of exceeding specified limits. Because of the uncertainties involved, performance assessments need not provide complete assurance that these requirements will be met.

(ii) Undisturbed performance for 1000 years after disposal will not cause an annual dose-equivalent to any member of the public, from all potential pathways in the accessible environment, exceeding specified limits. It will not cause annual average contamination of any ground water supply to exceed specified limits, with special limits for naturally contaminated water.

3.10 DEPARTMENT OF TRANSPORTATION, SHIPPERS – GENERAL REQUIREMENTS FOR SHIPMENTS AND PACKAGINGS, 49 CFR 173[12]

Subpart I – Radioactive Materials. The general requirements include criteria for design basis accidents and emission limits from casks. There are no requirements for accident risk assessments.

3.11 CONCLUSIONS

Accident consequence analyses or risk assessments are required at some stage in the licensing processes for all parts of the high-level waste disposal system (transport including loading and offloading, monitored retrievable storage, preclosure repository, and postclosure repository), but the nature of the analyses required is ambiguous.

All environmental-impact statements must include an accident-risk assessment, currently based on a spectrum of accidents including a worst-case scenario, with an indication of the probabilities of the scenarios in the spectrum. The requirement for worst-case analysis is under review, and the proposed rewording of this requirement would substitute "scientific credibility" and "reasonable foreseeability" as the basis for defining scenarios to be analyzed.* Although the worst-case requirement clearly applies to a repository after closure, it is not clear whether it also applies before closure. A performance assessment is required for the preclosure period.

The repository safety analysis report must determine critical pathways to the accessible environment and evaluate postclosure rates and quantities of radioactive releases for anticipated and unanticipated events, but it need not include assessment of health or economic risks of accidents.

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*The proposed changes become effective May 27, 1986 (51 FR 15618, April 25, 1986).

The performance assessment required at the site-selection stage must address EPA postclosure radionuclide release standards expressed in terms of probabilities, which implies some kind of probabilistic risk assessment, but health risks are not included.

Various requirements from EPA and NRC include the words, "without unreasonable risk to the health and safety of the public." In some cases, meeting standards or design criteria is sufficient proof of compliance. In others, unreasonable risk and requirements for assessment thereof are unspecified.

Transport of high-level waste, per se, does not require a risk assessment, but repository site environmental assessments and the monitored retrievable storage environmental report must include risk assessments of transport accidents.

The monitored retrievable storage environmental report must include an assessment of exposures to the general public from a design basis accident. No risk assessment is specified.

4 ASPECTS OF RISK

Risk is commonly considered to include the probability of a hazard and consequences of exposure to it. If we are more likely to be hurt, we consider a thing riskier. Risk analysis follows this usage. The risk of an event is commonly measured by the product of the probability of occurrence and a measure of the consequences. In the language of probability theory, this is the "expected risk;" if the same event were repeated many times, on average the consequences per event would approach this number. The risk of an activity with several possible adverse outcomes is the sum of the risks of the individual events. This is clearly a simplified measure, and such quantification should be recognized as a rough index of risk rather than a completely adequate description. For example, low-probability, high-consequence risks may be numerically equal to but are clearly different from high-probability, low-consequence risks. This may be accounted for mathematically by "weighting" more heavily those events with more serious consequences.

Statistical risks are different from known risks. That is to say, the concern for a dozen known trapped miners may be disproportionate to that for the unknown 50,000 people who will die next year in automobile accidents. Risks may be concentrated in single accidents or dispersed in populations or over time. Consequences to people may be estimated by various measures of excess mortality or morbidity; consequences may also be estimated by the effect on property and the environment. Involuntary risks do not necessarily equate to those that are voluntary.

Definition of risk is therefore inherently controversial; no one definition is suitable for all problems.[1] How risks compare, therefore, is determined in part by how risks are defined.

Risk assessment (or **risk analysis**; there is no consensus on the precise distinction in these two terms)[2] has been a rapidly developing field in the past decade, particularly in reference to nuclear-reactor safety. **Probabilistic risk assessment** (PRA), as noted earlier, is a technique for integrating different aspects of design and operation to assess the risks and to develop an information base for analyzing plant-specific and generic issues;[3] a summary of two critiques of PRA is given in Section 7.

5 APPLICABILITY OF RISK ASSESSMENT TO PRECLOSURE WASTE DISPOSAL

Risk assessment of HLW waste disposal is unlike that of a nuclear power plant although the analytical tools may be the same. The major activities in waste disposal are transportation and storage, both of which include handling. (Note that the term "storage" is used in this report to include preclosure disposal at the repository.) None of these involves interaction of a large complex of working parts like a reactor. PRA of a reactor typically involves a critical sequence of component failures in a closed system through fault/event-tree analysis; waste disposal requires less complex mechanisms, but on the other hand it is an open system where the eventualities may be harder to define. A reactor is located in one place. Risk assessment of waste transportation has a greater need for general methods of analysis that may be adaptable to a variety of locations.

Appraisal of applicability of current risk-assessment methods to HLW disposal leads to general conclusions:

- No new basic methodology is needed
- Existing methods may need to be adapted or extended
- Analytical methodology that has been developed for evaluating risks from radioactivity is generally more powerful than available data can support; it is not a constraint
- Additional data may need to be collected, possibly by additional testing, e.g., to improve estimates of source terms
- The human element—both in causing and mitigating accidents—has not been thoroughly incorporated
- Models of health effects (commonly entering into policy decision) are not realistic at very low radiation doses, but there are no data to improve them. They provide estimates of upper bounds.

- o Accidents with no radiological consequences should not be neglected; fortunately, good data bases exist for these.
- o Little work has been done on economic consequences
- o Uncertainty analysis for PRA is still in developing stages, but it is applicable to waste-cycle facilities
- o Clarity of analysis and in presentation of results should be emphasized, particularly in view of public interest in this work
- o External accident-causing events, such as floods and earthquakes, have generally not received as much emphasis in nonreactor applications, but the methods used for nuclear reactors are reasonably generic in their application.

These general conclusions are developed in the remainder of this report.

6 OVERVIEW OF ALTERNATIVE RISK ASSESSMENT METHODS

A range of methods is available for assessing risks of accidents, differing in approach and detail of analysis. They include methods for determining the possibility or probability of events, methods for evaluating the consequences of these events, or both. Selection from among them depends mostly on the purpose or objectives of an assessment and the magnitude of expected accidents or consequences relative to acceptable levels. If magnitudes of potential consequences are constrained by physical limits or expected probabilities to acceptable levels, then this can usually be demonstrated with scoping calculations, and nothing is gained by more refined estimates. But if uncertainty is high and/or the range of reasonably foreseeable consequences extends into the unacceptable, then much detail about the full spectrum of possible consequences and their probabilities is needed for informed judgments.

This section gives the risk-assessment methods most applicable to the high-level waste disposal system and its expected range of risks. These include vulnerability analysis, bounding analysis, generic analysis, best and/or conservative values, worst-case analysis, selected scenarios analysis, and probabilistic assessment. They are ordered from the simplest to the most complex, and their relative strengths and weaknesses are discussed. Under appropriate circumstances, each can serve a useful function in risk assessment.

6.1 VULNERABILITY ANALYSIS

Vulnerability aims at assessing the weakness in a system that can lead to injury or damage from failures. It evaluates the ability of a system to recover from partial failures. High probability of failure to recover reveals weak points where engineered safety systems can be improved. Emphasis is on small failures and common events rather than extreme events, and on modest or negligible external consequences rather than catastrophes.

Vulnerability analysis should be part of any scenario analysis or probabilistic assessment. "Engineering judgment" should specifically include system weaknesses or susceptibilities in identifying the most likely outcomes for analysis. Event-tree and fault-tree analysis are highly formalized methods of assessing vulnerability by tracing outcomes of failures forward or backward and examining chains of events that can produce undesirable results. This aids in development of engineered safety systems to prevent high risk events or to mitigate consequences. These methods are integral parts of probabilistic risk assessment, and are discussed in more detail in Section 7.

Although vulnerability analysis could under some circumstances provide useful information by itself, it is not normally used other than as part of a larger effort.

6.2 BOUNDING ANALYSIS

Bounding analysis is a special case of worst-case analysis, and the distinction between them can be fine. A bounding analysis assesses consequences of some extreme event that does not necessarily have any credibility (or even probability measurably greater than zero), but involves absolute physical limits that cannot be exceeded. An example is a "fence post analysis" for a nuclear power plant, in which routine exposures are estimated for a person living continuously at the edge of a reactor exclusion zone, eating only food grown there, drinking water from a well there, etc. No such person exists, but we can be sure that no other person will be exposed to greater risk.

The primary value of a bounding analysis is its simplicity and economy. Under appropriate conditions, scoping calculations suffice to demonstrate that consequences are insignificant. Such an analysis is useful only when the consequences of the accident assessed are so low that there are no conditions under which potential harmful effects can exceed acceptable levels. It dramatizes the safety of a system by showing that it is safe even under the most extreme assumptions.

6.3 GENERIC ANALYSIS

In the absence of a specific location and conditions under which to analyze consequences of an accident, some sort of more general, or generic, analysis must be done. All risk assessments necessarily contain some generic parts; no system is completely specified. This is usually more common in consequence modeling than in engineering systems modeling because of the complexity and natural variability of environmental systems.

A completely generic analysis must be either for a single, somehow representative, environment or for an array of environments showing the sensitivity of results to different variables across their normal ranges. The first approach is common in generic impact statements, where a representative, but nonexistent, environment is used to evaluate different alternatives without need for establishing their (unknown) actual location.[1] Similarly, an array of "representative" but actual lakes and streams might be used to evaluate water quality impacts in a generic way.[2,3]

The sensitivity-analysis approach to generic analysis has been used only to a limited extent, with a few important variables shown over their full ranges and some sensitivity analyses of other variables about their best estimate values.[4] We are not aware of a generic consequence analysis of radiation accidents which includes a large number of variables and parameters over their full ranges.

6.4 BEST ESTIMATES AND/OR CONSERVATIVE VALUES

Best estimates or conservative estimates are used throughout risk assessments whenever information or resources are inadequate to expand the analyses. They are always used when data are lacking on the distribution of a variable or parameter and, since it is generally not possible to include distributions of all variables and parameters, they are used wherever else "analysts' judgments" suggest the variability of results to be relatively insensitive.

Conservative values differ from best estimates by being deliberately biased in the direction producing undesirable outcomes. Often this is necessary because insufficient data are available to determine a best estimate or its uncertainties. At other times known extreme values are used to ensure that results have conservative bias. The intent is to deliberately over-estimate risks because the consequences of being over-pessimistic are preferred to those that are over-optimistic.

Scenarios can be constructed of all best or all conservative estimates. A selected scenarios analysis (Section 6.6) is likely to include both. A scenario based on all conservative values might even exceed a worst case scenario (Section 6.5), because the probability that all parameters and variables would be at their worst levels at the same time is usually so low as not to be worth considering. This is a common fault of worst-case analyses that is not often recognized.

6.5 WORST-CASE ANALYSIS

Worst-case analysis can be:

- o A design-basis worst-case analysis, in which the severity of the conditions to be analyzed is specified by a regulatory agency;
- o Analysis of a "maximum credible accident," the severity of which is based on engineering judgment of the credibility or incredibility of physically possible alternatives; or
- o Some other extreme case, the probability or credibility of which is not addressed ("what if" analysis).

Often this involves selecting the upper extremes of many variables in their worst combinations without consideration for the exceedingly low probability that all of these extreme conditions might occur at the same time. This approach is widely used in compliance assessments and by some regulatory agencies. Council on Environmental Quality regulations under the National Environmental Policy Act currently require a worst-case analysis in environmental-impact statements,[5] but some indication must also be given of the probability of occurrence of the conditions analyzed. This requirement is under revision, and the proposed change requires analysis of something closer to a maximum (scientifically) credible accident than a worst case accident.[6]

In worst-case analysis, a clear distinction must be made between the worst accident sequence and the worst consequences. A worst-case accident sequence generally leads to the largest potential release of radionuclides to the accessible environment; worst-case consequences arise from extremes of conditions leading to the greatest effects per unit radionuclide release. Worst-case analysis does not necessarily include both. Consequences of a worst case accident sequence are often analyzed using best-estimate conditions of weather, population density, etc.

Worst-case analysis does not require collection of much data. Calculations and models are often based on conservative assumptions, so they need not be detailed. It is assumed that if conservative, "back of the envelope" estimates of consequences of extreme events are well within acceptable limits, then there is no need for further analyses. Design-basis accidents are useful regulatory tools because they can be specified to any necessary level of detail with little ambiguity.

The primary disadvantage of worst-case or maximum-credible accident analysis is that the subjectivity of scenario selection leaves the analysis completely open-ended. Because neither "worst" nor "maximum credible" is defined, someone can always think of a more extreme scenario, which leads to accusations of incompleteness and attempts to force risk assessors to examine ever more extreme events without regard for likelihood.

In addition, risk assessors define credibility in terms of probability of occurrence. At some arbitrarily low probability, events are considered possible but so unlikely that they are not worth examining. In contrast, the general public defines credibility in terms of possibility. To them maximum credible means the most severe event physically possible, regardless of its probability of occurrence. Occasionally, through ignorance of the physical processes involved, even the impossible can be considered credible. An attempt on the part of the Council on Environmental Quality to clarify the meaning of "maximum credible" by substituting a specific probability of occurrence failed.[7]

Because of this difference in interpretation of "credibility," worst-case analysis can easily fail to be convincing. Risk assessors may be criticized for selecting a less-than-maximum-credible scenario. Included in this criticism will be accusations of attempting to avoid presenting the "real" risks. However severe the accident analyzed, opponents of the activity under assessment will always be able to suggest something worse that they hope might yield results "bad" enough to stop the activity.

Another disadvantage of worst case analysis is that it makes risks seem greater than they really are. Many will equate maximum credible consequences with expected consequences. The news media are especially prone to this type of exaggeration: it brightens news. If extremely low probability events are included, then the knowledge that such things are possible can make them seem more probable, for no other reason than being included. "Why would they analyze it unless it is something I should worry about?" These misinterpretations create unnecessary fears and biased judgments about relative risks of the activity analyzed.

Yet another disadvantage of worst-case analysis is that it can detract from efforts to identify other less severe events or consequences that can be prevented or mitigated by improved engineering safety systems.

6.6 SELECTED SCENARIOS ANALYSES

Selected scenarios analysis is an expansion of worst-case analysis to include an array of scenarios of different severities. The most severe scenario is also a worst case, but this is not necessarily so, since the definition of worst case is open to interpretation. Selected scenarios analysis is intended to present an array of possible events that provides an overview of the range of realistic outcomes. A worst case having exceedingly low probability might be excluded from the selected scenarios as being unrealistic and therefore providing a biased view of the expected range of outcomes.

In most cases, no attempt is made to estimate the probability of occurrence of the scenarios selected, primarily because of the difficulty, or impossibility, of estimating them. Without probabilities, there is no possibility of estimating risks, as measured by the product of probability and severity of accidents. If probabilities of the scenarios selected are estimated, then selected scenarios analysis can be considered to be a partial or simplified probabilistic risk assessment without a full continuum of consequences.

Selection of scenarios for analysis relies heavily on "engineering judgment" which, like worst-case analysis, leaves results open to criticism for being subjective and vulnerable to selection of favorable scenarios. These criticisms can be avoided somewhat by including estimates of scenario probabilities; these demonstrate the unlikelihood of the most severe case and quantify the relative importance of the scenarios.

Selected scenarios analysis is usually limited to only a few scenarios. A large array of scenarios without their associated probabilities would yield an uninterpretable mass of information having less value for decision making than a few carefully selected scenarios.

Like worst-case analysis, selected scenarios analysis provides little information on events or consequences that can be prevented or mitigated by additional engineering safety systems.

6.7 PROBABILISTIC ASSESSMENT

Probabilistic assessment aims to quantify variability and uncertainty in a rigorously formal way through use of probabilities or probability distributions for as many parameters and variables as possible. There is considerable ambiguity in the literature in the meaning of the terms "probabilistic assessment" and "probabilistic risk assessment." These terms have been applied to any analyses that included probabilities. They range from simple multiplication of the probability per year that an average atom of a radionuclide in the ground will be transfered to a river times the probability that the water will reach human stomachs,[8] to full-scale fault-tree analyses of nuclear power plants, in which as many probabilities and probability distributions are used as is reasonably expected to improve quality of results under budget limitations.[9] Because of the importance of this method to current assessment activities, it is discussed in greater detail in Section 7.

7 CRITIQUES OF PROBABILISTIC RISK ASSESSMENT

In 1975, probabilistic risk assessment, a new approach to evaluating reliability and risks of a nuclear power plant, was introduced in the Reactor Safety Study (RSS), WASH-1400. This approach is based upon the concept of identifying reactor system functions required for mitigating specific challenges (event trees) and estimating the probability of failure of system and functional requirements (fault trees). Since completion of the RSS, reliability- and risk-assessment methods have evolved so that they are generally accepted by the scientific community as providing a reasonable analysis of the safety of a nuclear power plant. During the mid to late 1970's, the Reactor Safety Study Methodology Applications Program (RSSMAP) developed the concept of dominant accident sequences to simplify the construction of detailed event and fault trees. These and subsequent studies by the NRC and the nuclear power industry have made significant advances in the art of probabilistic analysis. Two recent state-of-the-art reviews of probabilistic risk assessment are summarized here.[1,2] While these reviews apply specifically to reactors, this summary draws lessons applicable to radioactive waste disposal.

Probabilistic risk assessment (PRA) is an analysis that: (1) identifies and delineates the combinations of events that, if they occur, will lead to a severe accident or any other undesired event; (2) estimates the frequency of occurrence for each combination; and (3) estimates the consequences.

The emphasis in PRA for reactors has been on identifying internal failure modes leading to serious consequences, mainly core melt but recently also low-probability but quickly developing accidents that bypass the containment structure. The hallmark techniques for this are fault-tree analysis and event-tree development.

Fault-tree analysis is a deductive process starting with an undesired state and identifying the combinations of failures that might lead to it, particularly those "cut sets" with the highest probability of occurrence. Event-tree development is an inductive process in which the subsequent set of possible consequences of a given failure are traced out as a tree of ramifications. The two techniques are typically used together to reduce the number of combinations to be examined.

A major advantage of PRA is that it integrates in a uniform method all the relevant information, including system design, operating practices, operating history, component reliability, emergency actions, and finally, potential environmental and health effects. Thus it is good for identifying weak points in the system. Its limitation lies in that not every element has been developed to the same level.

Typically, PRAs focus on a series of unplanned events that might take place in a complex mechanism of interrelated parts, such as a nuclear power plant, leading to a major failure. The disposal of HLW is unlikely to depend upon the proper functioning of such complex mechanisms, but the analytic techniques used in PRA will probably be applicable to some extent.

The Nuclear Regulatory Commission has recently sponsored an appraisal of the state of the art of PRA of nuclear reactors.[2] Its principal findings are summarized in Table 7-1. Greatest confidence (a range of uncertainty in the results by a factor of ± 10) can be placed in the analysis of internally caused accidents, although the data base for these ranges from fairly good for high-frequency events to poor for low-frequency events. "Medium confidence" (a factor of ± 10 to 30) can be placed in analysis of externally caused accidents, although the data base for these is considered poor. The result of these accidents in releases of radionuclides to the environment ("source

Aspect of Probabilistic Risk Assessment	Level of Development	Range of Uncertainty	Improvement Needed
Qualitative systems analysis (logic modeling)			
o internal accident initiators	High confidence in qualitative insights	± Factor of 10	Modeling common cause failures.
o external accident initiators	Medium confidence in qualitative insights	± Factor of (10 to 30)	Modeling common cause failures.
Modeling human performance		± Factor of 10	Errors of misdiagnosis. Potential recovery actions.
Data base:			
o high frequency events	Fairly good		
o low frequency events	Poor		Not likely to improve substantially.
o internal accident initiators	Fair degree of confidence		
o external accident initiators	Poor degree of confidence		
o equipment and human failure	Needs improvement		
Source terms due to internal reactor phenomena	Poor confidence	Very large	Extensive research; source terms to remain quite large.
Consequences, given source terms and meteorology	Reasonably high confidence		Stochastic uncertainty.
o mean early fatalities		~ 0 to 5X	
o mean population dose		± Factor of (3 to 4)	
o Latent cancer deaths		± Factor of 10	
Consequences, actual		As a function of location cannot be predicted with much precision	
Actual behavior of affected population in an emergency	Not well understood		
Difference between analysts		Factor of 3	

NOTES:

High frequency = often observed in plant operations.

Low frequency = less than once in 1000 reactor-years.

X = nominal estimate

Source: Probabilistic Risk Assessment (PRA) Reference Document, Final Report, NUREG-1050, Sept. 1984.

terms") has a large range of uncertainty at best, and much effort is being made to improving source-term evaluations. (Consequence analysis will be discussed in the next section.)

Human performance in plant operation can be modeled to order-of-magnitude precision, although improvement is needed in errors of misdiagnosis and potential recovery actions. The data base on human failures needs improvement. And the actual behavior of an affected population in an emergency is poorly understood.

In sum, PRA results on nuclear reactors are appraised as useful, provided that more weight is given to insights regarding design and operations rather than the precise absolute size of the numbers generated. Most of the uncertainties are inherent in the problem itself rather than the artifacts of PRA; PRA tends to identify and highlight these uncertainties.

Beyond these generalities, some detailed observations on PRAs that may be relevant to HLW disposal are:

- There is some question as to whether the statistical techniques employed in PRAs have been implemented properly, particularly in assigning probability distributions to parameters based on limited data.
- Completeness does not seem to be the principal limitation when examining the general insights gained from a PRA on dominant sequences, since the data base is large enough so that a rare and unusual type of failure would not be likely to affect the conclusions regarding dominant sequences. (Note: This conclusion is disputed in another study described below.)
- More work needs to be done on propagating knowledge uncertainties (e.g., phenomena), and uncertainty and sensitivity analyses need

to be more widely used and better organized and displayed to assure that users of PRA information know more about the important uncertainties.

- One cannot validate estimates of events that are extremely unlikely. Therefore it is necessary to validate the estimates of many as possible of the elements in the sequence that leads to these events.
- There has been validation of computer codes, mainly through benchmark comparisons. Much remains to be accomplished in this area.
- The validation level of a PRA is not thorough or detailed; however, this level of validation is usually not much worse than the degree of validation achieved by alternative analytical tools.
- Human interactions, including test and maintenance considerations, are extremely important contributors to the safety of plants.
- The reliability of systems, components, and human actions important to safety must be maintained during operation. Degradation in their reliability can sharply increase risk or the likelihood of core melt.
- PRAs are not useful from a quantitative standpoint for some issues. However, PRAs can still provide useful insights even for these issues. For example, sabotage is difficult to quantify due to uncertainty in the frequency of attempted acts and the nature of and likelihood of success for sabotage attempts; however, PRA can still provide good qualitative insights with regard to important (vital) plant areas and weaknesses.

- Criteria for evaluating PRAs include:
  - The scope and depth of the PRA (i.e., does the nature of the PRA reasonably match the needs of the decision?)
  - The degree of realism embodied in the PRA
  - The results of peer reviews, which should be extensive and could add to or subtract from the credibility of PRA results
  - The credibility of qualitative insights obtained from the study
  - The quantitative results of the PRA compared to desired safety levels, and the uncertainty bounds surrounding the PRA analyses
  - The results of sensitivity studies that show the risk of major uncertainties

7.1 HOW ADDITIONAL EFFORT MAY CHANGE PRA RESULTS

Although assessments will continue to improve, PRA—like many other forms of analysis—intrinsically suffers a completeness problem: it is only as good as the imagination of the analysts in recognizing potential failure modes. A measure of this limitation is given in a recent NRC-sponsored study in which additional effort was given to completed PRAs to determine how it might affect the results. The biggest changes resulted from further consideration of human actions, initiating events, and the treatment of recovery.

As of August 1984, about 20 probabilistic safety analyses on specific nuclear power plants had been completed. Six of these were analyzed by Science Applications, Inc., to gain insight into how the choice of analytical methods can affect the results. The primary measure was how additional effort beyond the baseline level changed the importance and ordering of dominant accident sequences.

These findings contradict those reported above in two respects:

- o Common cause analysis of hardware failures, a topic of frequent concern from a methodological point of view, was found generally not to require additional attention in practice
- o The degree of completeness affects the substantive conclusions in some other areas of analysis to an important extent.

The results were presented in the form of effort-impact matrices summarized by a suggested effort-impact profile for future PRAs shown as Table 7-2. (The results have been somewhat simplified by eliminating some topic areas having to do with transients, loss-of-cooling accidents, and AC power systems, which are specific for nuclear power plants and cannot be generalized reasonably to nuclear waste disposal.)

In this table, the effect on the ordering of dominant accident sequences of an increase in the absolute level of effort is shown. The results can be described by the three clusters labeled I, II, and III. Cluster I, large effort, high impact topics, have great importance as high priority topics in future PRAs, and they require a large effort. Cluster II, moderate effort, high impact topics, are those where additional effort is most cost-effective. Cluster III, smaller effort, low impact topics, can be adequately treated with nominal levels of effort. In planning a PRA, the selection of the level of effort to be expended is not trivial; it can have significant impact on the resources required to perform the study and on the acceptability of the results. The conclusions drawn by Science Applications, Inc., from this evaluation are as follows:

Increasing Impact on Ordering of Dominant Sequences →

↓ Increase in Absolute Level of Effort

		No change	Slight re-arrangement; no inter-change be-tween domi-nant and non-dominant	Major re-arrangement of dominant or minor interchange of dominant and non-dominant	Slight in-terchange of dominant and non-dominant sequences	Major re-ordering of dominant versus non-dominant sequences
		1	2	3	4	5
None	A	Data base System inter-action analy-sis. Modeling of test and maintenance	III			II
1-2 man-weeks	B		Common mode human error analysis		Determin-ation of frequency of init-iating events	
Up to 1 man-month	C		Aggregation of initiat-ing events	Common cause analysis	Modeling of logic sys-tems. Human errors dur-ing normal operation. Determin-ation of system success criteria	Treatment of recovery. Identifi-cation of initiating events. Environmen-tal quali-fication
1-2 man-months	D					I
2-6 man-months	E				System hard wired de-pendency analysis	Human errors dur-ing acci-dents

Table 7-2. Suggested Effort-Impact Profile for Future PRAs.

Cluster I. Large effort, high impact topics.

- Human errors during accidents is the topic for which the highest level of effort is most clearly indicated. The "usual screening approach" to treating human errors generally leads to over-conservatism in the estimate. Recommended level: detailed human-error analysis.

- System hard-wired dependency analysis, not as clearly relevant to the HLW waste disposal issue, is to identify and quantify the impact of hard-wired system dependencies, including shared components. Recommended level: use Boolean reduction code.

Cluster II. Moderate effort, high impact topics.

- Treatment of recovery is especially important because, again, failure to consider it leads to over-conservatism, sometimes by large margins. Recommended level: recovery of human errors and action faults, but not individual component faults, to be considered.

- Identification of initiating events refers here specifically to transients and loss-of-cooling-accidents, but perhaps can be considered more generally. Recommended level: generic plus plant-specific data.

- Environmental qualification of equipment was modeled on four levels. Recommended level: estimate environmental conditions at the time of the accident and use manufacturer's specifications of equipment.

- Determination of frequency of initiating events refers to transients and the more general topic of accident-initiating events. Recommended level: generic (e.g., EPRI NP-801) plus "classical" use of plant-specific data, as opposed to the more rigorous two-stage Bayesian analysis.

- <u>Modeling of logic systems</u> refers to the quantification of logic actuation systems. Recommended level: use simple non-detailed models.
- <u>Human errors during normal operation</u>, miscalibration of sensors, leaving a valve aligned in an unsafe position after test or maintenance. Recommended level: "non-detailed" human error analysis.
- <u>Determination of system success criteria</u>. Recommended level: perform realistic, plant-specific analysis, as opposed to only those specified in the FSAR.
- <u>Common cause analysis of hardware failures</u>. Recommended level: analysis performed on components determined by engineering judgment, as opposed to a detailed comprehensive analysis.

<u>Cluster III. Small effort, low impact topics</u>.

- <u>Common-mode human-error analysis</u> identifies common-mode failures and evaluates common-mode errors. A topic of frequent concern from a methodological point of view, this appears generally not to be a high-priority topic. Recommended level: use "engineering judgment" as opposed to a detailed analysis.
- <u>Aggregation of initiating events</u> determines the number of accident sequences to be examined. Recommended level: aggregation of initiators along functional (or phenomenological) lines.
- <u>Data base</u> refers to component reliability. Recommended level: use only generic data, as opposed to using plant-specific data and sophisticated Bayesian analysis techniques.

- System interaction analysis covers the treatment of systems interactions other than hard-wired systems interactions. Recommended level: no analysis.
- Modeling of test and maintenance outage contributions. Recommended level: use only generic data for maintenance frequencies and test and maintenance outage times.

An example of the role that PRA plays in regulatory thinking is revealed by NRC's proposed policy on severe accidents. This will require a PRA "and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health, safety, and property". However, in deciding upon safety acceptability, the NRC staff will use "an approach that stresses deterministic engineering analysis and judgment complemented by PRA". (Emphasis added)[3] Thus, PRA is seen as an adjunct to traditional methods of analysis, but its use is being institutionalized by NRC.[4,5,6,7]

One of the insights resulting from PRA is that individual nuclear reactors are different. The original Reactor Safety Study addressed two prototypical reactor types: a boiling-water and a pressurized-water reactor. The twenty or so PRAs that have been completed since indicate that the vulnerabilities of each plant are unique. This would seem to suggest caution in the expectations of generic analysis, but remember that reactors are mechanically far more complicated than disposal activities.

PRA is a method for collecting and treating the existing body of knowledge. Any set of PRA results, therefore, will reflect the incompleteness and inherent variability of the data base, as well as the limitations and simplifications of the modeling procedure that results from our state of knowledge. However, an important attribute of the PRA method is that it can measure the effects of the limitations in data. This is done by propagating uncertainties through the analysis or by performing sensitivity analyses. Uncertainty and sensitivity analyses, however, need to be better organized and displayed.

The applicability of experience with reactor PRAs to risk assessment of spent fuel disposal must be judged in the light of the greatly different scale of risks between the two. With this in mind, it seems reasonable to conclude that:

- Following NRC's proposed policy on severe accidents, PRA should be viewed as complementing to deterministic analysis
- The philosophy, approach, and organization of PRA are appropriate to analysis of HLW disposal.
- A wide variety of initiating events should be addressed
- Recovery actions should be included
- Special emphasis should be given to the human element, both as a possible cause of accidents and for its part in recovering from them
- Uncertainty and sensitivity analysis should be performed on the results.

Much of the recent development in reactor risk-assessment methodology has been in improving estimates of (i) the kinds of failures that can bypass reactor-safety systems and (ii) the magnitudes of resulting emissions of radioactive materials. Little new work has been done on consequence analysis methods. For the waste-disposal process, especially for waste transport, the emphasis must be reversed. The kinds of failures (accidents) of concern that can occur are relatively few and easily delineated, but the locations of those failures, and the associated affected environments, are either extremely diverse (as in waste handling at each reactor site) or are essentially infinite (as in transport accidents). Since consequences are location-specific, much more attention must be given to the details of consequence analysis in the absence of specific locations and detailed environmental data. This has significant effects on selection of appropriate risk assessment methods and the representativeness of results. Uncertainties are necessarily high.

This section outlines the methods available for the various stages of consequence analysis in a general way and discusses their uncertainties and applicability to analyses of accidents in the waste disposal system. Many of the models and assumptions used in reactor-accident consequence analysis relate to the magnitudes of accidental radionuclide releases. Such large-scale accidents are not possible in the waste-disposal system, so some care must be exercised in transfering methods, models, and assumptions. The section concludes that the models available are, in general, adequate to the risk-assessment task, and that those used for reactor accidents can, with appropriate adjustments, be transfered to waste-disposal accidents. In many cases, the best of the available models are more detailed and sophisticated than the quality of the data available for the waste-disposal system can support, so little or nothing is gained by using them. Another conclusion is that while the uncertainty of some of the models is large, this has little effect on model selection, since no better alternative models are available.

8.1 INFORMATION

The information needed for consequence analysis in the waste-disposal system is relatively straightforward. Given a source of radionuclides released to the environment, consequence-analysis methods should determine the following:

- o Dispersion and transport of radionuclides through air, water, and food chains;
- o Resulting concentrations and radiation emissions;
- o Persons exposed to emissions;
- o Consequences of exposure, including effects of land and food contamination, costs of mitigation or cleanup, and direct effects on humans.

Methods differ primarily in the level of detail included in the assessment, rather than in the nature of the information provided.

The principal pathways to man that should be modeled are:

- o External radiation directly from casks or released contents;
- o Radionuclide puffs or plumes in air yielding external radiation from cloud immersion and inhalation of radionuclides;
- o Deposition of radionuclides from the air to the ground and surface waters yielding external radiation from the ground, contaminated agricultural products, and contaminated water;
- o Resuspension of deposited radionuclides;

- Dispersion of radionuclides in surface and ground water yielding contaminated irrigation water and drinking water.

The consequences that must be modeled include:

- Acute radiation sickness, cancers, birth defects, and genetic effects from radiation exposure with associated medical costs;
- Costs of emergency response and evacuation;
- Costs of cleanup, interdiction, relocation, etc.;

Pathways and consequence analysis methods are discussed separately below.

Note that a distinction is made here between consequence models and consequence-analysis codes. The term, "models," is used to mean mathematical representations of relatively inseparable physical or biological processes, such as meteorological dispersion, radiation health effects, etc. Consequence-analysis codes are linked combinations of models representing complex systems, and they follow materials and their consequences through several environments. CRAC[1] and its derivatives (CRAC2,[2] CRACIT,[3] NUCRAC,[4] and many developed in other countries[5]), and RADTRAN[6] and its revisions,[7] are examples of risk assessment codes.

8.2 EXTERNAL RADIATION

Exposure to external gamma and beta radiation is treated as a standard inverse square relationship with distance, adjusted for buildup, absorption, and sheltering, from a stationary-point source, from a moving-point source, from immersion in a semi-infinite cloud, and from ground "shine" from an infinite plane. Standard equations are available in the literature for each type of source.[8] Results for infinite extrapolations are biased on the high side, since no contamination is infinite, but this bias is compensated by computational convenience. Methods differ in including some or all of these sources and in the level of detail included in the sheltering assumptions. In many cases, sources are omitted because they are known _a priori_ to be insignificant for the size of the source term assessed. External beta radiation, for example, is normally included only for cloud immersion, because the maximum distance of travel of beta particles in air is only a few meters.

Assumptions about sheltering have high uncertainty, especially if locations and times of day are not specified, and can have a large influence on results. Average sheltering rates are realistic only for the most general of assessments.

8.3 ATMOSPHERIC DISPERSION

Atmospheric dispersion (diffusion plus transport) modeling ranges in complexity from simple calculations under average weather conditions to detailed trajectory analyses that follow pollutants downwind using measurements of short-term wind speed and direction and accounting for effects of complex terrain including buildings and mountains. Gaussian plume modeling of one form or another is nearly universal for nuclear accident consequence analysis. Selection of an appropriate model is based on generality of the analysis, need for detailed information, and availability of meteorological data, not model availability. The models include deposition based on assumed particle sizes or reactivity of gases, so estimation of surface contamination is an integral part of the meteorological modeling.

The uncertainties contained in meteorological model results include reducible error and inherent uncertainty.[9] Reducible error arises from poor quality meteorological and air-quality data (including uncertainty about location as in transportation accidents) and from fundamental inadequacies of the models. More detailed models include more physical processes and are therefore often assumed to provide higher quality results. But these models require correspondingly detailed data, with their associated uncertainties. If the quality of the data is not appropriate to the capabilities of the models, then the results are no better than those of simpler models with less stringent data requirements.

Inherent uncertainty arises from the basic stochastic nature of the meteorological processes modeled, such as turbulent winds and atmospheric instability. The magnitude of this inherent uncertainty is a function of the time scale of model results. The longer the span of the average conditions modeled, the smaller the inherent uncertainty attributable to stochastic processes and the higher the precision of model results.

The total uncertainty of Gaussian plume modeling results attributed to various levels of complexity and detail is illustrated by the relationships shown in Table 8.1.[10] These are representative of other meteorological modeling results as well. The Gaussian plume model is highly accurate under the simplest of conditions, but can have an uncertainty range as high as a factor of 30,000 (i.e., meaningless results) under the most complex conditions. In Table 8.2[11], a similar relationship can be seen for averaging time in a different set of models. Here a broad range of models is shown to agree fairly well (high precision), but not to be especially accurate in predicting annual averages. There is much less agreement for shorter averaging times. The studies on which Table 8.2 is based established that simple wind rose models are adequate for estimating annual average concentrations, and that Gaussian trajectory models are as good as the more complex 3-dimensional models for estimating shorter-term averages. The inability of more complex models to produce more precise estimates is related to lack of resolution in the meteorological data. The data in those studies

TABLE 8.1. Summary of the Estimated Uncertainty Associated with Predictions from Gaussian Plume Atmospheric Dispersion Model

Conditions	Range of the ratio $\frac{\text{Predicted}}{\text{Observed}}$
Highly instrumented site; ground-level centerline concentration within 10 km of a continuous point source	0.8 - 1.2
Specific hour and receptor point, flat terrain, steady meteorological conditions; within 10 km of release point	0.1 - 10
Ensemble average (e.g., monthly, seasonal, or annual averages) for a specific point; flat terrain; within 10 km of release point	0.5 - 2
Monthly and seasonal averages, flat terrain, 10-100 km downwind	0.25 - 4
Complex terrain or meteorology (e.g., sea breeze regimes)	0.01 - 300
Low wind speed, inversion conditions	
Smooth, unforested terrain	$\geq$1.3
Flat, forested terrain	20 - 40
Hilly, forested terrain	50 - 500

Source: Reference 10.

TABLE 8.2. Statistical Precision of Models

	No. Predictions	Pearson's R	Spearman's ρ	Kendall's τ	Average Bias, pCi/m^3	RMSE, pCi/m^3	Slope	R^2
Annual	13	0.85	-	-	-24	31	1.33	0.74
Monthly	273	0.51	0.57	0.41	-16	44	0.78	0.28
Weekly	229	0.45	0.62	0.46	-33	134	1.10	0.21
Weekly*	169	0.39	0.38	0.26	-44	164	1.18	0.15
Twice-Daily	200	0.40	0.52	0.42	1	161	0.48	0.18
Twice-Daily*	72	0.31	0.11	0.08	3	280	0.35	0.15

* Points near origin (zero) excluded.

Source: Reference 11.

were inadequate to support the detailed requirements of the more complex models.

The usefulness of detailed meteorological dispersion modeling is also constrained by availability of correspondingly detailed data on population distributions and distributions of surface waters, sensitive land uses (e.g., dairy farming), and land values. Little is gained from increased detail on spatial distributions of concentrations if it does not significantly increase the precision of contamination and exposure estimates.

8.4 SURFACE-WATER DISPERSION

The state of the art of surface-water quality modeling is well advanced. A wide variety of mathematical models for assessing transport of radionuclides in surface waters ranges from simple ratios or algebraic models to sophisticated multidimensional models based on numerical solutions to the advection-diffusion equation and associated hydrodynamic equations. Because surface waters are confined, there is less opportunity for compounding of errors with distance from the source, so the spatial precision of surface water models is greater than that of air-quality models, but most of the parameters for larger scale models (more than a few hundred meters) are location-specific, so the models have less generality than do air quality models. A flat-plane Gaussian Plume air-quality model can be used anywhere meteorological data are available or can be approximated. Surface water models are equivalent to air-quality models for complex terrain; beyond one-dimensional stream models, however, water-quality models must be specific to the physical characteristics of the water body modelled.

Surface-water transport modeling currently exceeds our abilty to apply the results usefully to long-term consequence analyses, so quality of results is constrained mostly by availability and quality of data, not model availability.

Because health- and environmental-effects estimates are based on cumulative exposure over long periods, the more complicated models with their correspondingly more detailed results in time and space are generally not needed for radiation risk assessment. For most assessments, simple box models can be used to estimate how much radionuclide reaches humans by various pathways without modeling the physical processes involved. Transfer coefficients quantifying proportions of the radionuclides in one environment entering another environment (e.g., proportion of deposited material reaching surface waters, proportion of surface waters used for drinking and irrigation, etc.) can be estimated from national average data and applied in sequence to the total radiation entering surface waters from an accident. Such a set of transfer coefficients is a simplified national average model that is not location specific. A localized model would contain local rather than national transfer coefficients.

8.5 FOOD-CHAIN TRANSPORT

Models of radioisotope transport from soil and water through terrestrial and aquatic food chains to humans are well established and generally based on combinations of kinetic rate constants or proportionality coefficients for transport among environmental components (e.g., between soil and plants) and equilibrium concentrations calculated from them. These models have been developed primarily for studying long-term releases of radionuclides. They can be applied to short-term accidental releases, but at considerable increase in uncertainty associated with values of model parameters.

Of all radionuclide-transport models, food-chain models have the highest uncertainty. This is partly because the underlying processes being modeled are so variable in natural environments that the simplifications necessary for model tractability are extreme. Most of the models available are of the same general form and rely on the same measurements of radionuclide transfer rates. Differences are confined mostly to assumptions used for poorly quantified (or unquantified) parameters.

In a recent comparison of model results, models having similar forms and using common sources of data yielded differences in results of two orders of magnitude.[12] In addition to this long-term modeling uncertainty, the variability in the natural environment must necessarily be an order of magnitude or more, and another layer of uncertainty of unknown magnitude would be added by using long-term average transfer rates to estimate short-term exposures. At best, results of such models are just reasonable guesses of general outcomes.

8.6 HEALTH EFFECTS

Nonradiological health effects of waste-disposal accidents can be estimated from general statistics on industrial and transportation accidents, so are simple and straightforward. But there is considerable controversy over estimation of low-level radiation health effects. A two-step calculation must first convert estimates of exposure to estimates of doses to affected tissues, the doses must be converted to estimates of effects on those tissues, and then the results must be extrapolated to effects on the general population. These steps are discussed separately below.

8.6.1 Dose

Internal dose to affected tissues is not normally measured directly except in radiological experiments designed to do so. Instead, these doses are inferred from estimates of intake of radionuclides by inhalation or ingestion, and application of radiation physics and mathematical models of metabolism and transfer of materials in the body.[13] The objective of the modeling is to estimate for each affected tissue the amount of energy absorbed per unit radionuclide exposure. Although the biological processes modeled are complex, the individual models used to represent them are relatively simple. There are many models, however, so collectively they are complex and their results are subject to uncertainties from many sources. A lung model, for example, is used to estimate retention of radionuclides or radionuclide-bearing particles of different sizes in different parts of the lung. Some of the particles are absorbed directly, and some are cleared by ciliary action and ingested. There is uncertainty about the basic form of the lung model, about the average values of the parameters in the model, and about the distribution of values of the parameters in the population. Other models are used for ingested materials, for transport of absorbed radionuclides to critical organs, and for external exposure from immersion in contaminated air and standing on contaminated surfaces. Each of these has similar kinds of uncertainties.

The results of modeling efforts are collected in tables listing estimated radiation doses to critical organs per unit radionuclide exposure for each radionuclide of concern. No further exposure-dose modeling need be done by risk assessors.

To date only two complete sets of model results are available for general use. The International Commission on Radiological Protection (ICRP) has developed a complete set of models for internal exposure based on a standard "reference man" having average physical and chemical characteristics.[14,15] Tables have been generated for inhalation and ingestion of each radionuclide of concern. Kocher has developed similar tables for exposure to external radiation from immersion in contaminated air and from standing on contaminated surfaces.[16]

Considerable controversy about these models arises because the results are for a "reference" (not average or median) adult male. There are significant differences in the values of model parameters among individuals, particularly as related to age and sex. There is normal variability in rates of intake, rates of transfer among tissues, and organ masses ranging from a factor of two to about two orders of magnitude. Differences related to growth of tissues are particularly large. Figures 8.1 and 8.2, for example, show estimated age-specific dose rates to bones from chronic ingestion of strontium-90 and to the thyroid from chronic ingestion of iodine-131.[17] Differences are strongly related to the rates of growth of the exposed tissues, the timing of which differs among tissues.

Another level of uncertainty arises from the fact that many of the estimates of physiological and metabolic parameters used in these models are based on only a few measurements, often from atypical humans or other species. The tables are currently being expanded to incorporate age- and sex-specific differences. Tables have been developed for age-specific risks of lung cancer from inhalation of radon[18] and work is under way at Oak Ridge National Laboratory to develop age-specific tables for a broad array of radionuclides, including strontium, plutonium, americium, curium, and cesium.[19]

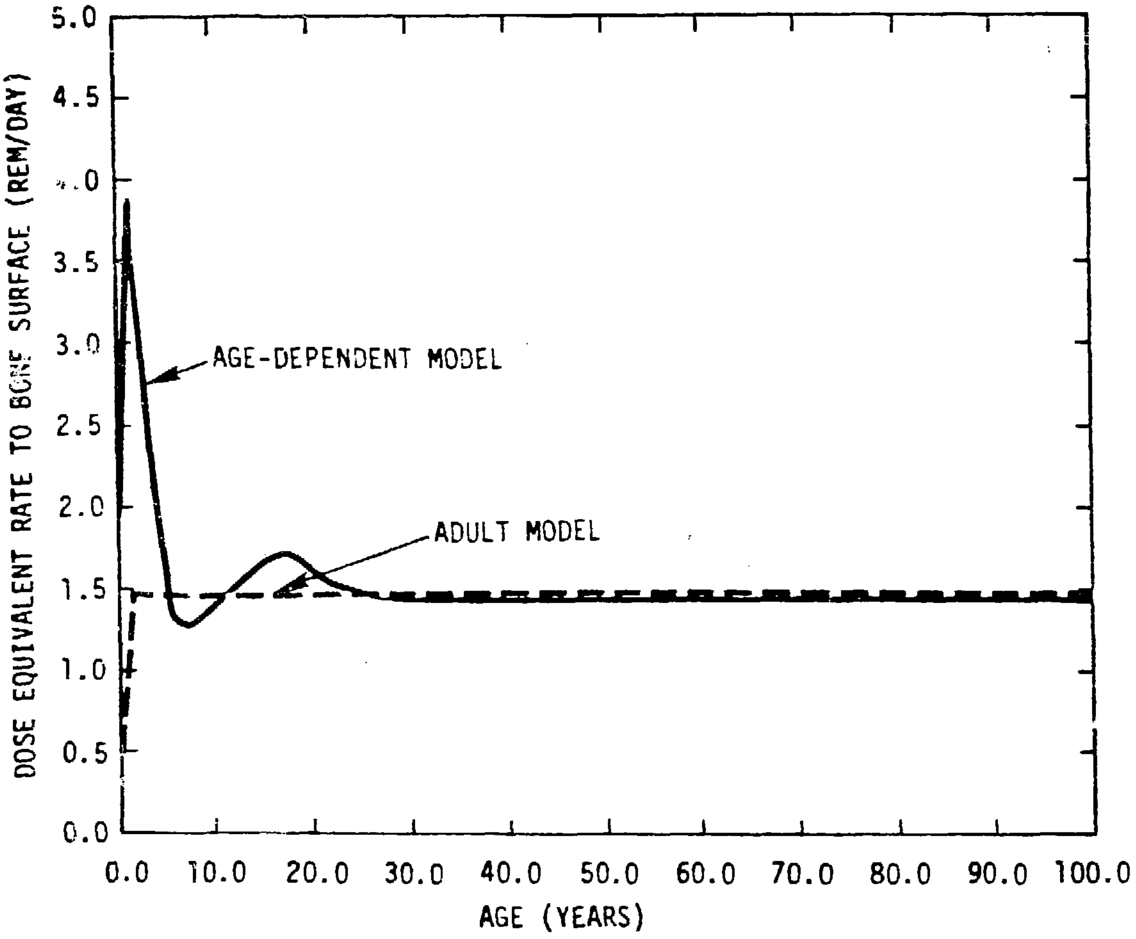

Figure 8.1. **Dose rate from chronic ingestion of strontium-90 in water at a concentration of 1 µCi/ℓ.**
Source: Reference 17.

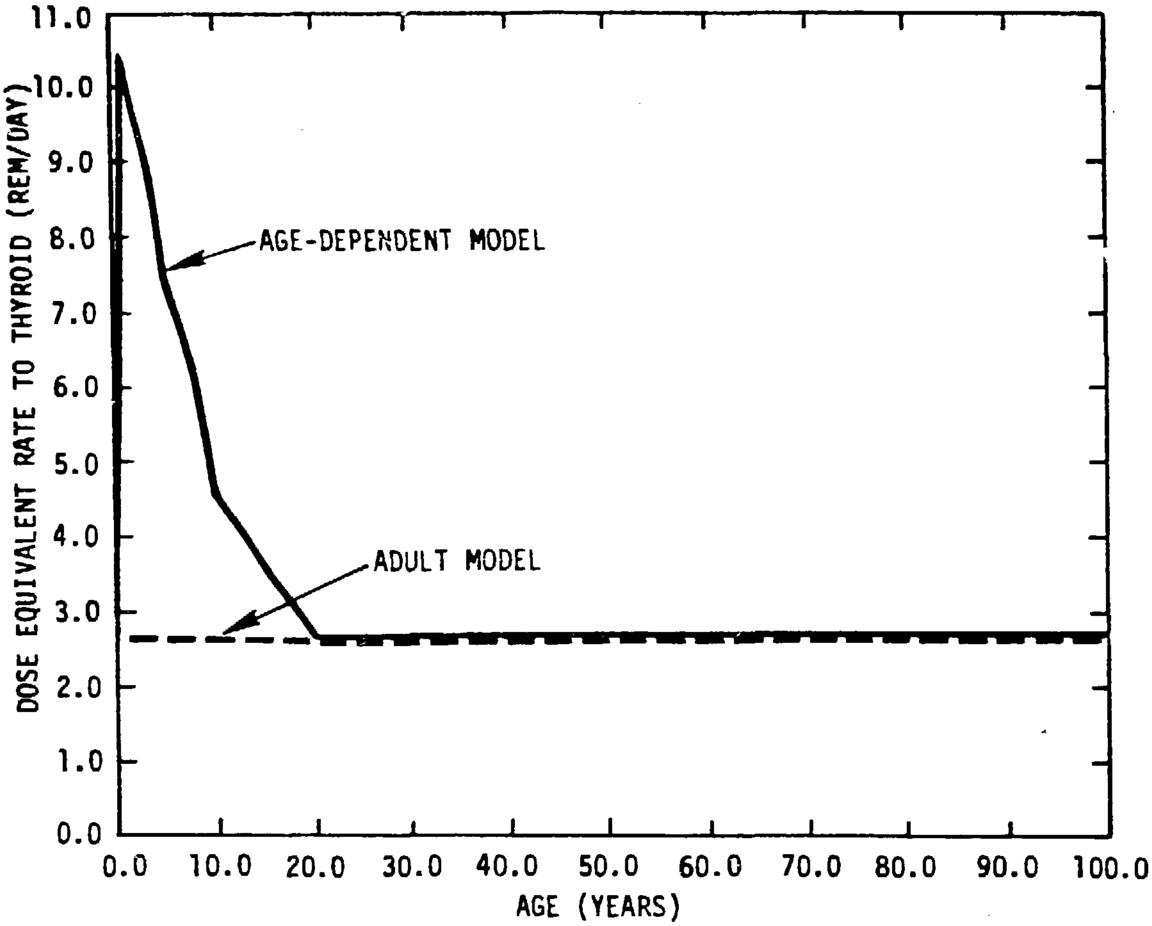

Figure 8.2. **Dose rate from chronic ingestion of iodine-131 in water at a concentration of 1 μCi/ℓ.**

Source: Reference 17.

8.6.2 Effects

Like dose from radiation exposure, health effects of low-level radiation doses are not measured directly and cannot be predicted precisely. In general, dose-response estimates draw on human experience, but most of the data providing insights into mechanisms and model structures are from a large body of animal studies. The only useful data available on humans are from survivors of the atomic bombing of Hiroshima and Nagasaki, from some cases of medical treatments with X-rays or radioisotopes, and from a few radiation accidents and occupational exposures (e.g., uranium miners).[20] Data from the more moderate levels of human exposure are exceedingly difficult to interpret because effects tend to be statistically indistinguishible from normal. There are large uncertainties attributable to random variation in measurements, uncertainty about functional relationships, variability in individual sensitivities, and extrapolations from one species (or population) to another and from high doses and dose rates (in time) to low doses and dose rates. In addition, few of the exposed human populations have been observed long enough to determine the full, lifetime effects of their exposures.

A broad range of assumptions must be made to extrapolate radiation health effects from the high doses at which they were observed to the low doses normally received, and to separate them from cancers from other causes. These are incorporated into mathematical dose-response functions expressing relationships between radiation doses to susceptible tissues and probabilities that radiogenic cancers are induced. The shape of the mathematical relationships is related to assumptions about mechanisms of radiation carcinogenesis, the ability of human tissues to repair radiation damage, and cell-killing effects of higher exposures that eliminate potentially cancerous cells.

A number of functional forms have been postulated and tried, but only two have currency, partly because they are simple with few parameters to be estimated (an important consideration in estimating relationships from a small body of low-quality data), and partly because of analyses showing that other functions are inconsistent with available data.

- The linear dose-response function assumes that effects are directly proportional to dose at all dose levels (no repair, no threshold below which there are no effects).
- The linear-quadratic dose-response function assumes effects are nearly proportional to dose at very low doses and proportional to the square of dose at high doses.

The first function yields higher estimates at low doses than the second.

Much scientific opinion is to the effect that cellular repair mechanisms make dose rate important to effects. Small amounts of damage done over a sufficient length of time can be repaired (or eliminated), while the same amount of damage done in a short time can overwhelm cellular repair mechanisms.[21] All of the dose-response relationships, which are based on high dose and high dose rate exposures, may, therefore, be excessively conservative for normal exposure levels and rates.

There is also uncertainty about the transferability of results among populations (or ages or sexes) having different underlying cancer rates that might be caused by different susceptibilities. Relative models distribute risk of cancer as a function of exposure in proportion to underlying rates in the normal population. Absolute models distribute the excess risk uniformly without consideration for possible differences in susceptibilites. Overall, evidence favors the absolute model, but there are several circumstances under which this model would clearly be incorrect.[21] Resolution awaits better understanding of the fundamental mechanisms of radiation carcinogenesis.

The Reactor Safety Study[22] provided an early attempt to deal comprehensively and quantitatively with these problems, and the health effects models developed there have provided a basis for most of the health consequence analyses of recent years. The Reactor Safety Study, however, did not attempt to quantify uncertainty in these models. In 1985, the NRC released improved radiation dose-effects models that explicitly account for uncertainty from many sources, incorporating the latest epidemiological data.[23]

The resulting radiation dose-effects model is actually a collection of tissue- and effects-specific models, current capabilities of which are given in Table 8.3. The models are based, where possible, on epidemiological analyses of human data. Otherwise, they are based on extrapolations of animal data. All models for genetic effects and some models for early effects of acute doses are based on animal data.

8.7 COSTS

Except for a few studies of transportation accidents,[24] most of the studies of costs of accidents involving radiation releases have been for postulated nuclear power plant accidents,[25] the sizes of which far exceed anything expected from the high-level waste and spent fuel-disposal process.[26] Nevertheless, both contain similar elements and general characteristics.

Initial costs of accidents include all emergency-response requirements and monitoring requirements to establish the extent of contamination, if any. These costs are incurred to establish needs for further action and are relatively insensitive to severity of radiation release. They are more likely to be related to any physical damage done (e.g., number of cars involved in a

Table 8.3. Effects for which quantitative risk estimation models have been developed.

Index (i)	Effect	Model developed for: Mortality	Morbidity	Organ-specific dose
	Early and Continuing Effects			
1	hematopoietic syndrome	x	–[a]	bone marrow
2	pulmonary syndrome	x	–	lung
3	gastrointestinal syndrome	x	–	small intestine/colon[b]
4	prenatal/neonatal deaths	x	–	fetus[c]
5	prodromal symptoms	–	x[a]	abdomen[d]
6	lung function impairment	–	x	lung
7	hypothyroidism[e]	–	x	thyroid
8	acute radiation thyroiditis	–	x	thyroid
9	skin damage	–	x	basal cells of epidermis[f]
10	cataracts	–	x	lens of the eye
11	sterility	–	x	ovaries/testes
12	microcephaly	–	x	fetus[c]
13	mental retardation	–	x	fetus[c]
	Late Somatic Effects			
14	leukemia	x	–	red bone marrow
15	bone cancer	x	–	bone
16	breast cancer	x	x	breast
17	lung cancer	x	x	lung
18	gastrointestinal cancer	x	x	lower large intestine
19	thyroid cancer	x	x	thyroid
20	skin cancer	–	x	face
21	other cancers	x	x	
22	leukemia -<u>in utero</u>	x	–	fetus[c]
23	other cancers -<u>in utero</u>	x	–	fetus[c]
24	benign thyroid nodules	–	x	thyroid
	Genetic Effects			
25	single gene - dominant	–	x[g]	ovaries/testes
26	single gene - X-lined	–	x[g]	ovaries/testes
27	chromosome - numerical aberration (aneuploidy)	–	x[g]	ovaries/testes
28	chromosome - structural aberration (unbalanced translocations)	–	x[g]	ovaries/testes
29	multifactorial diseases	–	x[g]	ovaries/testes

[a]There is no clear differentiation between the hematopoietic and prodromal syndromes. Lushbaugh defines all symptoms between anorexia and death as "acute hematologic syndrome." The symptoms considered by Lushbaugh include anorexia, nausea, vomiting, fatigue, and diarrhea. These models permit prediction of each of these symptoms. However, in this taxonomy, they are identified as prodromal symptoms.

[b]Dose to small intestine is important for brief exposure. Dose to lower large intestine is important for protracted exposure.

[c]Technically, it is the dose to the embryo or fetus depending upon the stage of development. For the first seven weeks, or fifty days, the term "embryo" is appropriate.

[d]Midline, midplane upper abdominal dose.

[e]Although not originally identified as a separate health effect of interest, a model was developed for thyroid ablation.

[f]For a depth of 0.1 mm and an area of 35 to 100 cm^2.

[g]Incidence of each type of genetic effect is modelled. Fractions of each class of defect that is fatal are also given.

Source: Reference 23.

truck accident) and needs for traffic and population control. These costs are simple functions of the numbers of persons involved, which are related to the severity of the nonradiological impacts of an accident, with additional expenditures for radiation monitoring.

Subsequent costs are related to the amount of radiation released and its dispersion characteristics as they affect land and surface water contamination and human exposure. They can be primary, in that they are directly related to actual contamination levels, or secondary, in that they are related to concerns about contamination rather than actual contamination. Primary costs include relocation (temporary and permanent), decontamination and disposal of contaminated materials, agricultural product disposal, land interdiction, loss of income, and medical expenses.[25] Secondary costs include litigation, loss of economic value of land, structures, crops, tourist trade, etc. because of public concerns, and associated economic ripple effects from general loss of value or income. Government expenditures on cleanup, etc., also generate some economic benefits.

For accidents of the size possible in the waste-disposal process, the more extreme costs sometimes associated with reactor accidents are unlikely to be incurred. Cost-estimation models are mostly multipliers based on average values from the literature (corrected to current dollars), and their complexity and detail depend on availability of suitable information. They can be as simple or as complex as the modelers are willing to make them. Uncertainties are high, however, so the quality of the data cannot support excessively complex cost models. Some work is needed on estimating costs of cleanup indoors, which is not included in current models.

8.7.1 Litigation

Studies of litigation costs generally assume that such payments are transfers of costs from affected parties to the parties causing the damage, and do not represent an increase in total costs except for the legal fees involved.[25] For the immediate effects of an accident (assuming no punitive damages are awarded) this is probably true, since causal relationships are relatively easy to establish. But this is clearly not the case for long-term effects such as birth defects and cancers. The problem arises from our inability to assign specific causality to health effects that can be caused by like-kind radiation exposure from any source as well as by chemical carcinogens (known or unknown). Persons who are exposed to radiation in an accident will assume, or their lawyers will assert, that any ensuing cancers or genetic effects of appropriate types have been caused by the exposure from the accident. Some proportion of these cases inevitably will succeed, producing awards for damages possibly in excess of those properly attributed to the accidental exposures.

8.7.2 Costs of Public Concerns

Some economic costs are not closely related to the amount of radiation released in an accident, but instead are functions of public concerns about radiation contamination, some of which is irreducible and some of which is heavily influenced by the nature and extent of news media coverage. Emotional stress and associated health effects and costs are one example. Another is refusal to buy land, structures, crops, etc., somehow identified with an accident, but not actually contaminated, or, in the case of land and structures, formerly contaminated areas that have been decontaminated to safe levels. Such costs are difficult to quantify with more precision than general bounding analyses.

9 STATE OF THE ART: PRECLOSURE STORAGE RISK ASSESSMENTS

There have been several risk assessments of the waste-disposal stages of the nuclear fuel cycle,[1,2] culminating recently in two preliminary probabilistic risk assessments of preclosure storage.[3,4] Although a wide range of analytical methods have been used, there have been heretofore few quantitative assessments making use of fault/event trees. Data bases exist on component failure rates, but seldom on actual fuel cycle operating experience.

Causes of accidents ("accident initiating events") are not well defined in this area. Among these, external events and operator errors may be important to risk, but they have seldom been considered in detail.

Alternative approaches to accident consequence analysis and their relative merits are summarized in Table 9-1.[5]

There is a need for extraction and consolidation of consistent conclusions from previous analyses. Other improvements needed in risk assessment of storage and processing are as follows:[5]

- Sensitivity and error analysis to rank plant and process factors by their contribution to or reduction in risk
- Systematic compilation of improved failure rate data bases for components
- Better data for reliability and efficiency of the high-efficiency particulate air filters used in canisters
- Analyses using more sophisticated aerosol production and transport models
- Methods for analyzing highly degraded conditions, such as those that might result from earthquakes or floods.

TABLE 9-1. Alternative Approaches to Preclosure Storage Consequence Analysis

Alternative Approaches to Accident Consequence Analysis	Relative Merits of Methods for Application to Radioactive Waste Disposal
Preliminary hazards analysis	Convenient qualitative first step but usually focused on one component or hazard at a time.
Postulate accidents directly	Simple but difficult to reproduce and defend.
Fault trees	Widely accepted. Identifies dominant risk contributors and gives quantitative determinations of absolute system risks.
GO methodology	Combines advantages of fault trees with partial failures and time dependency, but complex

Source: Reference 5.

The two preliminary risk assessments reported in 1985 are Preliminary Repository Underground Design Safety Assessment Report, prepared by Roy F. Weston, Inc., for the Office of Civilian Radioactive Waste Management, DOE,[3] and High-Level Waste Preclosure Systems Safety Analysis: Phase 1, Final Report, prepared by GA Technologies, Inc., for NRC.[4] A brief comparison of the contents is given here.

Both reports have the same objectives: (1) to establish a methodology for preclosure repository-risk assessment, and (2) to identify which systems, components, and structures are important to safety and waste isolation. Both take a probabilistic approach to the analysis. Both are an initial phase rather than a complete analysis, but they differ in the scope of the first cut:

- The DOE report treats the public radiological health and safety aspects of only underground activities in basalt, salt and tuff repositories, extending the analysis to the quantification of event trees
- The NRC report treats both radiological and nonradiological effects on workers as well as the public, considering surface and support activities as well as those underground, but only in the basalt repository; its development of event trees does not extend to their quantification.

A comparison of the contents of the two reports is tabulated in Table 9-2.

9.1 HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS PHASE 1 REPORT, NUREG/CR-4303

GA Technologies, Inc., has completed the first phase of the High-Level Waste Preclosure Systems Safety Analysis project of the NRC, a report leading to the specification of a sample problem to demonstrate a methodology for

Table 9-2. Comparison of Two Preliminary Risk Assessments at Preclosure Stage

	Roy F. Weston, Inc. Preliminary Repository Underground Design Safety Report, Nov. 1985		GA Technologies, Inc. High-Level Preclosure Systems Safety Analysis, Phase 1, Final Report, July 1985
Types of repository	Basalt, salt, and tuff		Basalt
Activities analyzed	Underground activities: development, waste emplacement, waste retrieval and permanent closure		Surface, subsurface, and support activities
Waste form	6 MTU canisters of 5-year old spent fuel		Spent fuel and high-level waste
Criteria	Public radiological health and safety per 10CFR 60.2 Regulatory impacts	Public radiological health and safety Importance to waste isolation	Radiological risk to the public Radiological and nonradiological risk to workers Repository availability Long-term waste isolation Financial impact
Basic information	15 target systems preliminarily identified as important to safety and waste isolation 6 flow charts of principal steps in developing and operating a repository		J. M. Davis, Conceptual System Design Description, Nuclear Waste Repository in Basalt, BWI-SD-005, April 1983
Initiating events considered	10 possible accidents resulting in radiological impacts, reduced to 2 bounding scenarios	18 possible accidents affecting waste isolation, retrieval, or closure, reduced to 4 bounding scenarios	153, considering equipment failures, human actions, local personnel injuries, external events, and challenges of plant barriers to to radionuclide release; each event classified as to likely frequency of occurence and the severity and type of consequence
Event trees		Underground flooding Explosion/fire Seismic events Improper construction techniques or operator errors	29 stemming from those potential initiating events not rated as "low" in either event frequency or consequence severity
Fault trees	Accidents with short-term radiological effects		4 based on intermediate events in the accident scenarios, focused on ventilation/filtration systems
Radiological release fractions	"Base case" estimates of burst, diffusion, and fuel pulverization per WASH-1400 and NUREG/CR-1288		E. L. Wilmot et al., Report on a Workshop on Transportation Accident Scenarios Involving Spent Fuel SAND80-2012, May 1980 E. L. Wilmot, Transportation Accident Scenarios for Commercial Spent Fuel, SAND80-2124, Feb. 1981 E. Walker, A Summary of Parameters Affecting the Release and Transport of Radioactive Material from an Unplanned Incident, BNFO-81-2, Sept. 1978
Types of consequences	Short-term radiation hazards	Impacts on waste retrieval; Impacts on repository accessibility; Short-term radiation hazards; Long-term radiation hazards	Public or occupational radiological exposure Occupational nonradiological consequence Impact on repository availability Long-term effects
Degree of quantification		Probability and severity of consequences of event trees estimated by panel of experts	Data compiled for next phase: Initiating event frequencies Intermediate event probabilities Basic event failure data Human error rates Occupational injury data

identifying structures, components, systems, and operations that are important to safety.[4]

Based on a conceptual design of a repository for spent fuel and HLW in basalt, the analysis consisted of the following steps:

1. Analysis of material flows to identify 153 initiating events, considering equipment failures, human actions, local personnel injuries, external events (e.g., earthquakes, tornados), and challenges of plant barriers to radionuclide release. Each such event was classified as to likely frequency of occurrence and the severity and type of consequence: public or occupational radiological exposure, occupational nonradiological consequence, impact on repository availability, or long-term effects

2. Preparation of 29 event trees stemming from those potential initiating events not rated as "low" in either event frequency or consequence severity

3. Preparation of 500 accident scenarios (i.e., feasible sequences of successful and failed events) classified by type of consequence

4. Developmemt of 4 fault trees based on intermediate events in the accident scenarios focused on ventilation/filtration systems

5. Selection of 11 accident scenarios for the next phase of the study in which the fault trees will be reduced and quantified

6. Preparation of quantitative data for the next study phase including initiating-event frequencies, intermediate-event probabilities, basic-event failure data, human error rates, radiological data, and occupational-injury data

7. Evaluation of methods for ranking the importance of repository elements in the next phase.

Retrieval operations are treated separately from emplacement operations in order to provide a measure of relative risk due solely to retrieval operations in addition to emplacement risk.

The authors emphasize that the present results do not represent a preliminary evaluation or order-of-magnitude estimate of repository safety. From the analysis so far, however, they observe that the preliminary design of the primary and secondary confinement exhaust ventilation/filtration systems are not completely independent, nor are their respective backup units, leading to the possibility of a common-cause failure. Although the source term per disruptive incident is smaller than in a reactor, they expect that more disruptive events could occur in the preclosure phase of the repository because of the many canister-handling operations over a long time interval (p. 3-3).

For further work, the authors recommend emphasis on the radiological consequences to the public and the workers (p. 1-6). The radioactive material source of greatest concern consists of the small particles liberated from the essentially monolithic waste matrix outside the local canister containment and into the interior atmosphere of a repository or into the environment, particularly aerosolized particles of Aerodynamic Equivalent Diameter (AED) of 10 micromillimeters or less because they are regarded as respirable (p. 3-6).

A key issue in consequence calculations is the determination of the amount of radionuclides released from failed canisters or fuel pins as a result of accidents (p. 7-3). Because release fractions for gases and volatiles have been characterized to a large extent in previous studies, no new modeling is required. However, large data uncertainties exist because of the limited data and the accident-specific nature of the release (p. 3-7). For transport of airborne particulates in confined areas, an indoor air quality model is proposed, while the Gaussian plume dispersion model (CRAC2) will be used to determine aerosol release to the environment (p. 8-2).

The water pathway will not be considered in further analysis (p. 3-20) because, except for flooding, radionuclide release via the water pathway is a slow process that is not likely to affect the preclosure safety of the repository (p. 3-19). It is noted, however, that mining experience indicates that the presence of undetected groundwater, abnormal seepage, and sump pump failures have caused subterranean flooding in the past (p. 2-44).

9.2 PRELIMINARY REPOSITORY UNDERGROUND DESIGN SAFETY ASSESSMENT REPORT, Roy F. Weston, Inc.

Roy F. Weston, Inc., has completed a preliminary scoping evaluation of repository subsurface systems, demonstrating a probabilistic approach to risk assessment and aimed at identifying elements of the repository that should be classified as important to safety or waste isolation.[3] The procedure was as follows:

1. A list was prepared of 15 target systems preliminarily identified as important to safety and waste isolation, and their interrelationships were charted.

2. A series of 6 flow charts was prepared showing principal steps in the development and operation of a repository.

3. Possible accidents ("event scenario candidates") were identified: 10 resulting in radiological impacts from waste handling, and 18 affecting waste isolation, retrieval, or closure.

4. A comparison of these indicated that maximum risks could be evaluated using 6 bounding scenarios: 2 of the radiological events and 4 of those events related to waste isolation. Subsequent analysis of the two types of accidents differed.

5. Making use of a "fault-logic network," the two radiological events were evaluated by the expected damage to a 6 MTU canister of 5-year old spent fuel resulting from (i) dropping the hoist cage down the elevator shaft in basalt or salt, or (ii) a crash of the transporter on the underground ramp with a resulting fire in the tuff repository.

6. Three types of radioactive release processes were considered: burst release, diffusion release, and releases due to fuel pulverization. In each case, release fractions were assumed, and diffusion into the environment was calculated.

7. It was concluded that neither the hoist system for salt or basalt nor the waste transporter and ramp system for tuff appears to require classification as "important to safety" inasmuch as the radiation dose at the repository boundary resulting from these accidents does not reach the 0.5 rem value specified in 10 CFR 60.2. (See discussion below.)

8. For the 4 events related to waste isolation, a probabilistic risk assessment was performed. After discussion with experts on potential accident progressions, the first step was to construct event trees of five to eight branches for the following 4 events: (i) underground flooding, (ii) explosion/fire, (iii) seismic event, and (iv) improper construction techniques or operator errors.

9. The event trees were quantified using the judgment of a panel of experts as to the frequency of occurrence of the four initiating events, the conditional probabilities of subsequent branches of the event trees, and the consequences of the accident sequences. Consequences were judged as severe, "some," or negligible in four categories: impacts on waste retrieval, impacts on repository accessibility, short-term radiation hazards, and long-term radiation hazards.

10. The risk evaluation then consisted of a compilation and review of the maximum probability of severe or minor impacts from each of the four types of initiating events by the four consequence categories for each of the three types of repository.

A brief summary of the conclusions of this part of the assessment is as follows:

o The overall probability of severe consequences in any preclosure year is on the order of one in a million. The probability of severe short-term radiation hazards may be an order of magnitude greater, however, primarily because of risks from possible explosions and fires.

o Underground flooding has the potential for severe impacts on repository performance in both salt and basalt. The latter case is contrary to the findings of GA Technologies. However, this result stems from the assumption that there is a relatively high probability of flooding that exceeds the design capacity of the pumping system, a weakness easily remedied by upgrading the pumping system.

o Explosion and fire can result in severe short-term radiation hazards in all three types of repository, but they are less likely in tuff because explosive gases are not expected and have not been detected.

o Seismic events can lead to severe impacts on waste retrieval and repository accessibility at the tuff site primarily because of the possiblity of common-mode shaft/ramp failure, but no significant long-term radiation hazards appear to result.

o Improper construction techniques and operator errors are not likely to be sources of high risks during preclosure.

o Because the risks are so low, none of the waste handling systems, structures, or components evaluated needs to be classified as "important to safety." Repository elements that may be important to safety or waste isolation include:

- Flood preventative and mitigative systems in salt or basalt
- Isolation-level layout and ground support systems relating to waste emplacement and the stability of underground openings in tuff
- Specific components of systems for fire protection, waste retrieval and performance confirmation system in all three types of work
- Systems, structures, or components—such as liners—of the exploratory shaft, assuming that it will be incorporated into the repository system.

9.3 PRELIMINARY EVALUATION OF PRECLOSURE RISK REPORTS

Both the Weston and the GA Technologies reports are professionally competent initial evaluations of the preclosure risks in a repository. In both cases the preliminary incomplete nature of the work is made clear, and the initial findings are reported as tentative. That these initial findings are not entirely consistent—for example, in the evaluation of flood hazard in the basalt repository—is grounds for renewed attention in the studies but not for concern about the work. Indeed, the two studies illustrate the value of parallel studies that can be checked against each other to provide greater reliability in the final product.

Equally revealing but less reassuring is what the studies have in common, what in the terminology of reliability analysis might prove to be the basis for common-cause failure. An example of this is the uncertain—and largely arbitrary—estimates of radioactive release fractions in the event of damage to the waste containers. This is pointed out in the NRC report and illustrated by the DOE report.

In the words of the NRC report,[4] "reliable estimates of release fractions are difficult to obtain largely because of the accident-specific nature of the release and the lack of adequate experimental data to support postulated release assumptions. This large uncertainty in the release fraction has to be recognized and accounted for in future work." (p. 8-2)

An example of the need for such release assumptions is the Weston estimate of the consequences of the accidental drop of a waste fuel canister down the waste-handling shaft.[3] "Cladding failure is postulated to occur upon impact for fifty per cent of the fuel rods in the canister; a fraction of the radionuclides in the fuel gap (100% of the noble gases and tritium; 50% of the other radionuclides, p. 33) will escape as a "burst release" upon cladding failure. Furthermore, one per cent of the fuel is postulated to be pulverized upon impact; twenty-five per cent of the resultant fuel particles are assumed to be initially suspended in air, forming a dust cloud with particle sizes ranging from 1 to 500 microns (of which 2% are postulated to have aerodynamic diameters of 10 microns or less, p. 34). Reductions in releases to the environment due to plateout ('Fifty per cent of the radionuclides other than the noble gases and tritium are assumed to be plated out during the subsequent transport up the shaft.' p. 33.) and gravitational settlement have been considered in modeling radiation transport." (p. 10) "For the case where exhaust flow velocity is assumed, transport of dust particles up the shaft is possible and a plateout factor of 5 (i.e., a reduction by 80%) is assumed." (p. 34) (Emphasis added)

The results of the calculations based on this string of assumptions are as follows: For the "anticipated" operational scenario in which the waste handling shaft will experience insignificant exhaust ventilation, the maximum dose at or beyond the repository boundary will be 130 mrem to the bone. For the "unlikely" scenario in which there is exhaust ventilation flow in the waste handling shaft, the maximum overall dose is 400 mrem to the bone. Inasmuch as 10 CFR 60.2 establishes 500 mrem as the permissible radiation dose at this point, the Weston report comes to the conclusion that the hoist system does not need to be classified as "important to safety." (p. 2)

In fairness, it should be noted that this means that, on the basis of these first-cut assumptions, other elements of the repository system may be more likely to cause off-site radiation hazards, and in a system of quality assurance they should therefore receive greater attention. It does not mean that the hoist system is unimportant to safety. Recognizing that this inference only establishes a relative priority for safety attention, however, there are still grounds to question the merits of this judgment:

1. The numbers are too close for comfort. The 130 mrem of the "anticipated" scenario are 26% of the 500 mrem criterion; the 400 mrem of the "unlikely" scenario are 80%. Considering the number of arbitrary assumptions made in the calculations, results of this magnitude argue against downgrading the importance of any system as central and prominent as the hoist system.

2. Radiation hazard to the public is an insufficient criterion of repository safety, whatever the letter of present NRC regulations. It is notable that in the NRC-sponsored study the consequences considered also include radiological and nonradiological hazards to workers, effect on repository availability and long-term isolation, and financial impacts. Contrary to the GA Technologies recommendation that emphasis by placed on radiological consequences (pp. 1-6, 8-2) and particularly in view of NRC's response to the recent fatal accident at Sequoyah Fuels Corporation, it can be anticipated that nonradiological as well as radiological hazards to workers will have to be considered.

3. Conspicuous failures, such as the accidental drop of a hoist cage, must be safeguarded against even if they result in no casualties. Any such failure would be perceived by the public as evidence that the waste-disposal program is mismanaged and dangerous. The crash of a loaded hoist cage—even without any casualties—could potentially be OCRWM's Three Mile Island.

To gain the full benefit of these two preliminary analyses that have been made of preclosure repository risk, a more exhaustive comparison than this deserves to be made. It is one thing to recognize the large uncertainty in release fractions, as the NRC report comments, and another for it to be "accounted for in future work." In particular, common sources of data, such as estimated release fractions, need to be evaluated, and needed experimental work defined.

10 STATE OF THE ART: TRANSPORTATION RISK ASSESSMENT

This section discusses various aspects of risk assessment of transportation accidents, including an initial critical literature survey.

Although risk assessments of even the most severe transportation accidents have not resulted in high radiation doses, they have drawn public and scientific attention and created controversy. The numerous published risk assessments of transportation accidents reflect this controversy either explicitly or implicitly.

Possible reasons for this are: (1) the unique public attitude to and perception of <u>any</u> radiation risk; (2) the fact that there are severe transportation accidents (not involving radioactive material) apparently makes a severe radioactive transportation accident more plausible; (3) uncertainties and ambiguities in models and parameters that describe the consequences of transportation accidents; and (4) the explicit regulatory requirement that in cases of scientific uncertainty or gaps in relevant information "risk assessment shall include a worst case analysis and an indication of the probability or improbability of its occurrence."[1*] Such a stipulation leaves the definition of "worst case" open to various individual interpretations, leading to debate on the credibility or incredibility of certain accident scenarios.

This section focuses mainly on severe potential transportation accidents, since only analyses of such accidents have resulted in doses to the "most exposed persons" in excess of 100-200 millirem, the average "natural" background dose received annually by the public from cosmic and terrestrial radiation, x-rays, etc.

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*The worst case requirement has been rescinded effective May 27, 1986 (51 FR 15618, April 25, 1986).

The aspects of severe transportation accidents discussed in this section are: the impact environment, the fire environment source terms (i.e., amounts of releases of radionuclides that the damaged cask/fuel system), probabilities of occurrence, and doses. Although the section concentrates mainly on truck accidents, most parameters relate also to rail accidents.

10.1 IMPACT

"Impact" has several definitions. One is "collision between a package and some other body, where the force of the collision is applied over a wide area of the package." The emphasis on "large area" is aimed to distinguish between "impact" and "puncture" which is defined as "being struck by an object having the potential for penetrating the container."[2,3]

The regulatory requirement concerning impact is "free drop",[4] namely a free drop of the specimen through a distance of 9 m (30 ft) onto a flat, essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected.

Severe impact may damage the cask/fuel system in the following ways:[5] (1) In those casks equipped with valves, it may cause a valve failure. (2) It may damage the closure seals designed to prevent leakage between the cask head and body. (3) It may cause a small breach (probably having a cross-sectional area of less than 6.4 cm^2) in the cask body or cask head. Such a breach — not more than a fine crack — is considered to be the "worst case" for cask damage.[5] (4) It may cause the fuel to rupture; i.e., cracks can be produced in the fuel cladding as a result of a severe impact.

The first three mechanisms relate to cask damage, providing possible pathways to radionuclides that for some other reason might have been released from the fuel to the cask cavity. The fourth mechanism relates to damage to the fuel rods themselves. None of the above mechanisms is necessarily accompanied by other mechanisms, but a combination of two or more mechanisms is considered possible under severe impact conditions.

The main factors influencing the severity of an impact leading to damage to the cask and its contents are: E, the energy available to damage the package, and m, the package's mass. Energy is determined by the velocities involved in the impact. The energy "lost" in an impact is given by:[2]

$$E = \frac{1}{2} \frac{m_1 m_2}{m_1 + m_2} Vo^2$$

where m_1, m_2 are the masses of the cask and the object it hits.

Vo is the relative velocity of the colliding bodies.

The "puncturing" phenomenon is determined by the ratio V/R, i.e., the ratio between the relative velocities between the "probe" (the puncturing body) and the cask, and the tip radius R of the puncturing body. For a conically shaped probe, whose tip radius is R, the total work required for penetration, W (= $1/2m\ V^2$) is given by[2]

$$W = \frac{\pi R^2 \quad \tau Y}{2}$$

where τ is the thickness of the "hit" plate

Y is the plate yield stress

and hence: $$\tau = \frac{m}{\pi Y}\left(\frac{V}{R}\right)^2$$

Figure 10.1 gives some historical data on velocity changes due to impact in highway truck accidents as a function of truck weight.[7] The weight representing the typical gross weight of spent fuel highway shipments is 30 to 40 tons. Another way of giving accident data for trucks and trains is shown in Figure 10.2.[8]

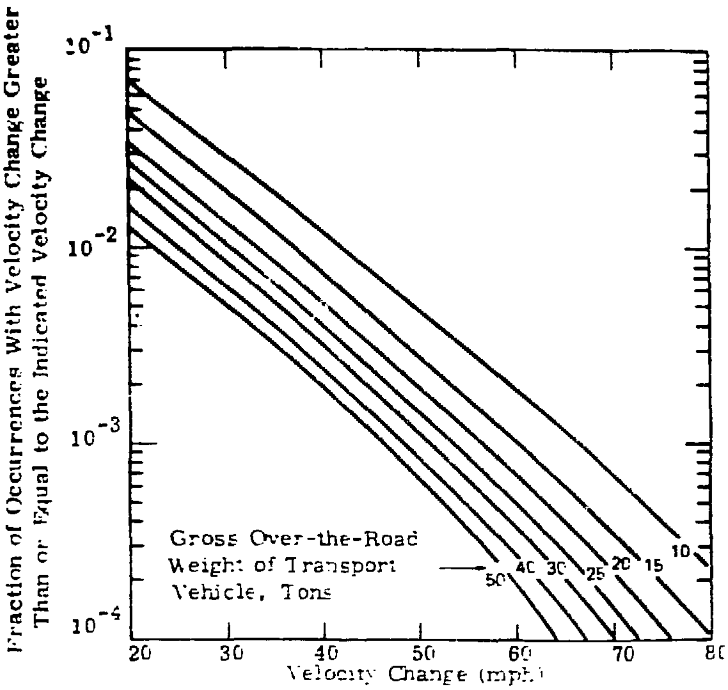

Figure 10.1 Velocity change due to impact in a highway transportation collision accident. Source: Reference 7.

Table 10.1 (a). Frequency distribution for speed differential between impacting road vehicles (package moving)*

Impact speed (mph)	Number
0 - 9	0
10 - 19	0
20 - 29	1
30 - 39	5
40 - 49	2
50 - 59	8
60 - 69	5
70 - 79	0
80 - 89	1
Total	22

Source: References 11 and 12.

Table 10.1 (b). Frequency distribution for speed of trucks hitting fixed objects (package moving)*

Speed (mph)	Number
0 - 9	0
10 - 19	0
20 - 29	1
39 - 39	2
40 - 49	4
50 - 59	12
60 - 69	4
70 - 79	3
80 - 89	0
Total	26

References 11 and 12.

*Note: These frequency distributions are conditional, i.e., they relate to severe accidents only.

An analysis of (limited) statistical accident data was made by Pickard, Lowe and Garrick, Inc.[11,12] Approximately 90 truck accidents were analyzed. Table 10.1 shows frequency distributions for speed changes in cases where moving trucks ("moving packages") hit fixed objects or moving vehicles.

From these figures it can be inferred that velocity changes in the range of 40 to 70 mph should not be precluded in a worst case analysis. While such velocity changes, or decelerations, might be experienced by a vehicle in an accident, they are not likely to be experienced by a cask. The cushioning effect of the vehicle and the cask tiedown system, etc. can reduce the deceleration of the cask to less than half that of the vehicle. For example, in a test performed by Sandia National Laboratories, a truck was stopped dead in a 60 mph head-on collision with a hard surface. The truck experienced a velocity change of 60 mph, but the velocity change of the cask upon striking the truck cab was only 27 mph.[9,10] Because of this effect, the regulatory requirements for a 30 ft. cask drop (equivalent to a 30 mph velocity change for the cask[7]) simulates vehicle accidents of much higher severity as shown on the far right side of the distribution in Figure 10.1. This makes an assumption equalizing vehicle and cask velocity changes extremely conservative.

The regulatory requirement concerning cask puncturing is "a free drop of a specimen through a distance of 1 m (40 inches) in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar, mounted on an essentially unyielding horizontal surface.[4] The bar must be 15 cm in diameter, with the top horizontal and its edge rounded to a radius of no more than 6 mm and of a length to cause maximum damage to the package, but not less than 20 cm long. The long axis of the bar must be vertical."

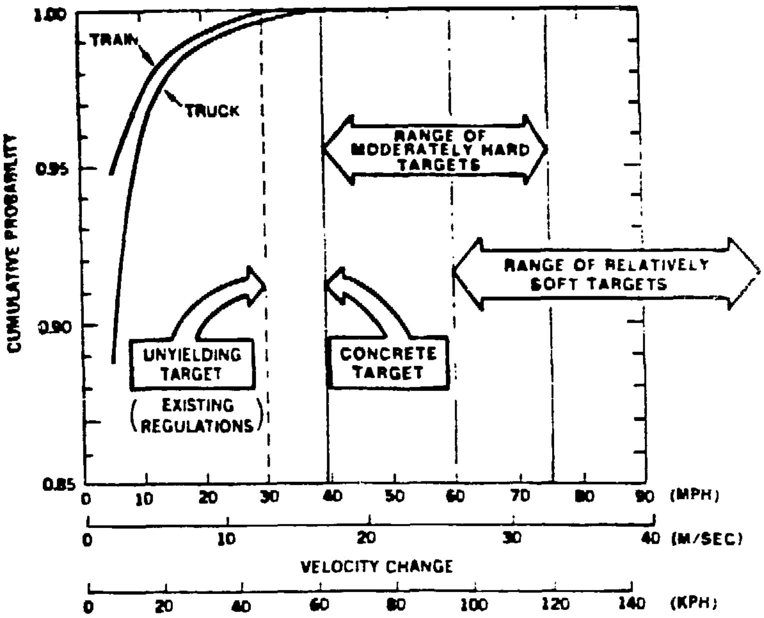

Figure 10.2. Damage equivalence diagram—impact environment. Source: Reference 8.

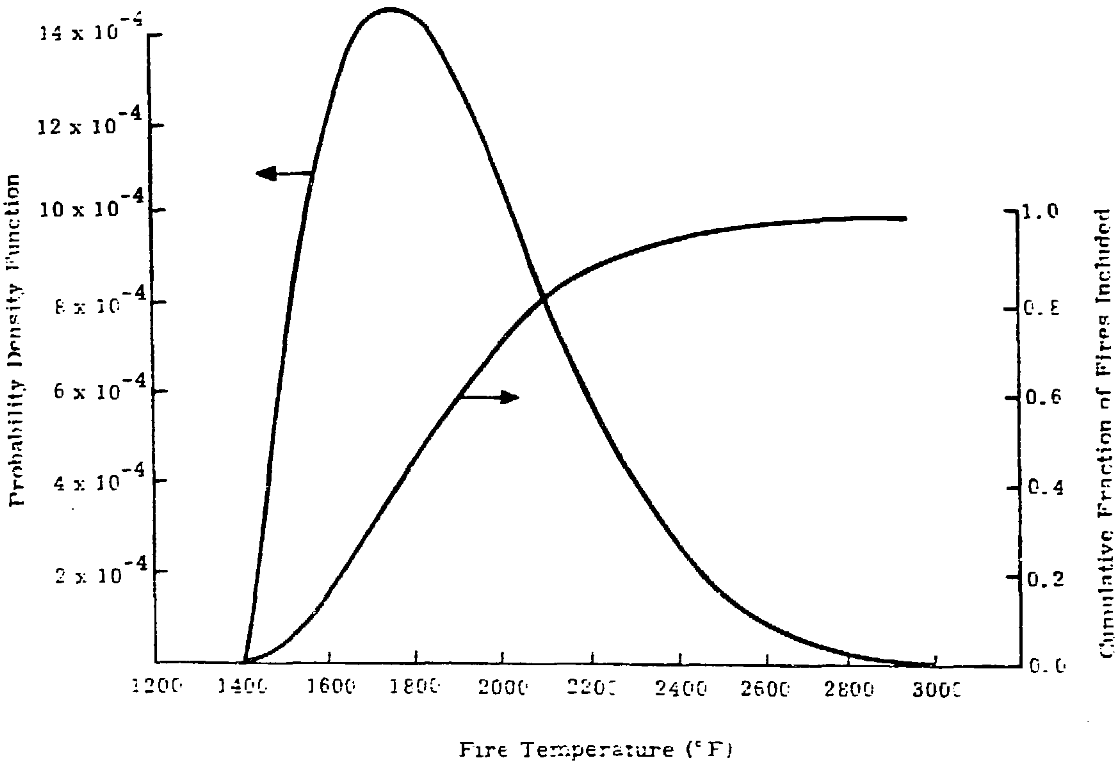

Figure 10.3. Probability distribution of truck-accident fire temperature. Source: Reference 2.

As mentioned above, V/R is the critical measure for a "puncture environment." In the Sandia study,[2] it was shown that, based on a truck puncture accident rate of ~5×10^{-7}/mile and a probability of 0.07 that a given container is involved in such an accident, the probability of getting V/R = 50 or more (making a puncture likely according to the IAEA regulatory standards) is approximately 10^{-9}/mile.

10.2 THE FIRE ENVIRONMENT

As will be discussed in Section 10.3, the dominant factor influencing the potential releases of radioactive fission products from the spent fuel into the cask cavity (thus making them available for releases into the external environment) is the temperature of the spent fuel. In a fire, the maximum temperature that can be reached by the fuel rods depends on two main factors: (1) duration of the accidental fire and (2) temperature of the fire.[2,3,7,13] Of course, the size of the fire and its location relative to the cask also determine the final cask temperature. In the regulatory requirement related to fire test conditions, it is stipulated that the licensee has to expose the cask to a fire at 800°C lasting for one-half hour with an emissivity coefficient of 0.9.

The fire temperature is determined mainly by the flame temperature of the fuel involved in the accident. The duration depends on the amount of fuel available to burn. The temperature and the duration depend also on the types and the quantities of combustible material present in the fire environment such as the cargo, the truck's cab interior, tires, etc.

It appears that most hydrocarbon fuels, such as JP-4, diesel and gasoline, yield similar flame temperatures in open burning, generally falling in the range of 1400°F to 2400°F (750°C - 1300°C).[2,7] The probability density function proposed was

$$f(T) = 1.77\times10^{-5}\ (T - 1400)^{0.83}\ \exp -\left[\frac{(T - 1400)^{1.83}}{550}\right]$$

where T is measured in °F.[2,7]

The temperature distribution function is shown in Figure 10.3.

Fire durations were somewhat more difficult to assess. In the absence of sufficient data, a Monte Carlo analysis of fire duration was performed at Sandia National Laboratories.[2,3,7] The results are shown in Figure 10.4. Some of the inputs to the Monte Carlo program concerning the amounts of fuel available for burning are worth mentioning here: (1) single-vehicle truck accident: 0 to 200 gal (expected number (μ): 120 gal); (2) truck/auto collision: 0 to 250 gal (μ= 150); (3) truck/truck collision: 0 to 500 gal (μ = 300); (4) truck/tanker collision: 0 to 10000 gal (μ = 5000). It was also assumed that truck/tanker collisions were 2% of all truck/truck collisions. In another study[12] analysis of historical statistical data yielded somewhat longer fire durations. These are shown in Table 10.2. The conclusion of this study was that "truck fires occur relatively often, given a severe accident. These fires can burn sufficiently long and intensify enough to damage structural materials. Furthermore, the fires are generally near the truck's cargo if they do not completely envelope it."[12] Table 10.3 gives the areas affected by the fires analyzed.

A third statistical analysis of fire duration is given in Table 10.4.[14,15] In this table, accident severities are categorized by impact velocities and fire durations.

To conclude, both fire duration and temperature can be determined probabilistically relatively easily for input to a probabilistic

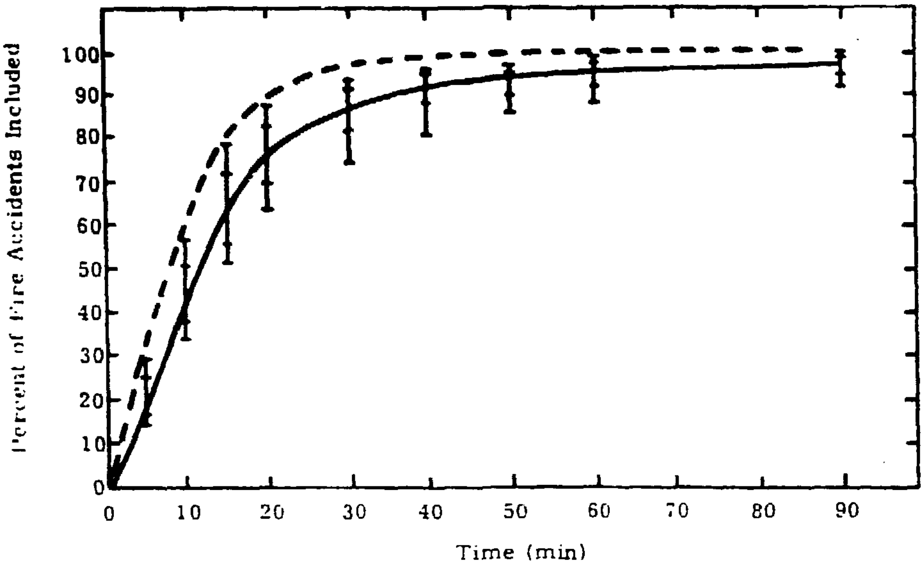

Figure 10.4. Duration of fires in truck accidents involving fire. Source: Reference 2.

Table 10.2. Fire duration histogram (truck accidents)

Duration (hours)	Number
0 - 2	5
3 - 4	1
5 - 6	2
7 - 8	0
9 - 10	0
11 - 12	0
13 - 14	0
15 - 16	0
17 - 18	0
19 - 20	0
21 - 22	1
Total	9

Source: References 11 and 12.

Table 10.3. Frequency histogram for area affected by fire (truck accidents

Area (10^5 ft^2)	Number
0.0 - 0.1	1
0.1 - 1.0	3
1.0 - 10.0	2
10.0 - 100.0	1
Total	7

Source: References 11 and 12.

Table 10.4. Traffic accident probabilities

Severity category	Vehicle speed (mph)	Fire duration (hr)	Probability per vehicle mile		
			Rail	Truck	Barge
Minor	0–30	<1/2	6×10^{-9}	6×10^{-9}	–
	0–30	0	4.7×10^{-7}	4×10^{-7}	1.6×10^{-7}
	30–50	0	7.3×10^{-7}	9.3×10^{-7}	1.4×10^{-6}
Total			7.3×10^{-7}	1.3×10^{-6}	1.7×10^{-6}
Moderate	0–30	1/2–1	9.3×10^{-10}	5×10^{-11}	–
	30–50	<1/2	3.3×10^{-9}	1×10^{-8}	8×10^{-9}
	50–70	<1/2	9.9×10^{-10}	5×10^{-9}	2×10^{-9}
	50–70	0	7.5×10^{-8}	3×10^{-7}	3.4×10^{-8}
Total			7.9×10^{-8}	3×10^{-7}	4.4×10^{-8}
Severe	0–30	>1	7.0×10^{-11}	5×10^{-12}	–
	30–50	>1	3.9×10^{-11}	1×10^{-11}	9.3×10^{-11}
	30–50	1/2–1	5.1×10^{-10}	1×10^{-10}	1.3×10^{-9}
	50–70	1/2–1	1.5×10^{-10}	6×10^{-12}	3.3×10^{-10}
	>70	<1/2	1×10^{-11}	1×10^{-10}	–
	>70	0	8×10^{-10}	8×10^{-9}	–
Total			1.5×10^{-9}	8×10^{-9}	1.6×10^{-9}
Extra severe	50–70	>1	1.1×10^{-11}	6×10^{-13}	2.3×10^{-11}
	>70	1/2–1	1.6×10^{-12}	2×10^{-13}	–
Total			1.3×10^{-11}	8×10^{-13}	2.3×10^{-11}
Extreme	>70	>1	1.2×10^{-13}	2×10^{-14}	–
Total			1.2×10^{-13}	2×10^{-14}	–

Source: Reference 14.

transportation accident risk assessment. When only severe cases are analyzed (for decision making purposes), it seems that fire durations higher than 0.5 hr and fire temperatures higher than 800°C should not be precluded.

10.3 RADIONUCLIDE INVENTORIES AND RADIOACTIVE HEAT GENERATION RATE

Table 10.5 shows the inventories of those radionuclides found to be the most hazardous, as functions of the time elapsing from the day they were discharged from the reactor.[16] The inventories were obtained using the ORIGEN-2 code.[18] The inventories of fission products are functions of fuel burnup and power density in the reactor. Significant variations among different fuel types are not expected (except for activation products such as Co-60), as long as the basic nature of the fuel — say, enriched U — is the same.

Figure 10.5 shows the heat generation rate in the fuel as a function of increasing age. This parameter has a significant influence on the fuel rod temperature and can be crucial when the fire-induced maximum fuel temperature is analyzed. In a worst case analysis,[5] a 120 to 180 day old fuel, having a decay heat generation rate of ~13 kilowatts/assembly, was considered. Five year old fuel will have a heat generation rate of less than 1 kilowatt/assembly, and thus its initial temperature would be much lower. (Note: Even in a transportation accident "worst case" analysis,[5] it was argued that if the fuel was over two years old the burst rupture mechanism (to be explained below) could not occur unless a hotter and longer fire (than the fire assumed in the analysis) occurred.)

10.4 FUEL BEHAVIOR AND RADIONUCLIDE RELEASES

Considering the inventories and the specific radiological hazards of the various isotopes[5,16,17] it turns out that Cs-134 and Cs-137 are responsible

Table 10.5(a). Major contributors to inhalation exposures

Fission products			Actinides and daughters			Activation products		
Radionuclide	Inventory	Rem/μc[1]	Radionuclide	Inventory	Rem/μc[1]	Radionuclide	Inventory	Rem/μc[1]
Sr 90	68570	1.3	Pu 238	2600	303	Mn 54	55	6×10^{-3}
Y 90	68570	8.10^{-3}	Pu 239	314	330	Co 60	7880	1.5×10^{-1}
Ru 106	20000	4.4×10^{-1}	Pu 240	542	330	Ni 59	5.3	2.7×10^{-3}
Rh 106	20000	1.5×10^{-4}	Pu 241	103800	5.8	Ni 63	660	6.3×10^{-3}
Sb 125	4550	1×10^{-2}	Am 241	1036	522	Sn 119M	43	5×10^{-3}
Te 125M	1100	7×10^{-3}	Cm 244	1926	274	Sb 125	500	1×10^{-2}
Cs 134	32060	4.6×10^{-2}				Te 125M	120	$8.7c10^{-3}$
Cs 137	98440	3×10^{-2}						
Ce 144	15540	3.5×10^{-1}						
Pm 147	37860	3.4×10^{-2}						
Eu 154	7600	4×10^{-2}						

1. Whole body equivalent doses per microcurie inhaled.
2. The main contributors were selected for each group separately. Thus, it may happen that one isotope from the actinide group contributes more than the whole activation products group.

Source: Reference 16.

Table 10.5(b). Overall inventories of radionuclides for various decay periods (in curies per 1 MT UO_2 charged to the reactor)

	0	150d	1.0y	5y	10y	30y	50y	100y	200y	300y	1000y
Activation Products	2.9×10^{5}	4.9×10^{4}	3.4×10^{4}	1.4×10^{4}	6.2×10^{3}	8.5×10^{2}	5×10^{2}	3.3×10^{2}	1.6×10^{2}	7.7×10^{1}	5.7×10^{0}
Actinides and Daughters	4.8×10^{7}	1.6×10^{5}	1.4×10^{5}	1.1×10^{5}	8.8×10^{4}	3.8×10^{4}	1.9×10^{4}	7.2×10^{3}	4.8×10^{3}	4×10^{3}	1.8×10^{3}
Fission Products	1.8×10^{8}	5.1×10^{6}	2.5×10^{6}	4.9×10^{5}	3.2×10^{5}	1.9×10^{5}	1.2×10^{5}	3.6×10^{4}	3.5×10^{3}	3.9×10^{2}	2.0×10^{1}

Source: Reference 16.

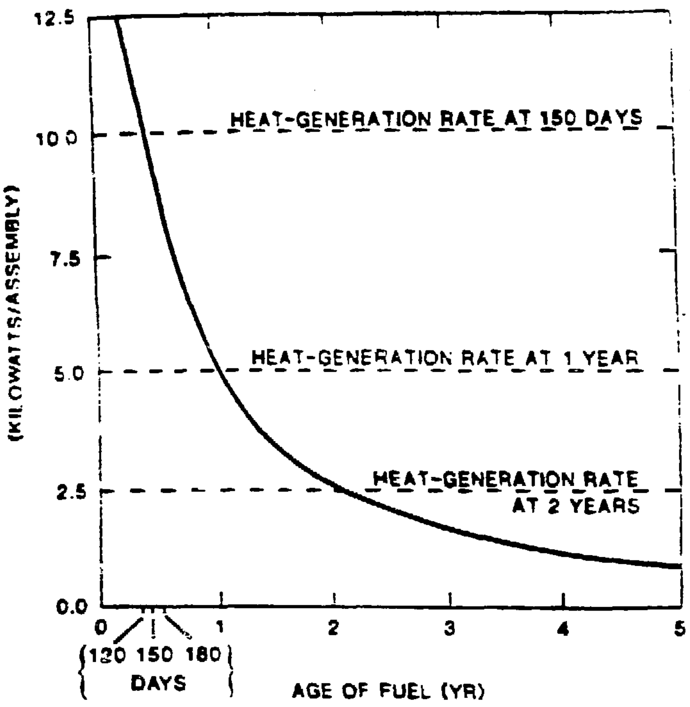

Figure 10.5. Decay-heat generation rate from PWR spent fuel. Source: Reference 5.

for a considerable part of the final collective and individual doses obtained in radiation risk assessments. The following discussion relates mainly to these radionuclides. However, most of the discussion also applies to other radionuclides.

Several mechanisms leading to radionuclide releases during transportation accidents have been identified.[5,6,19,20] It should be emphasized that the radionuclides have to pass two consecutive pathways to be released into the environment: the first from the fuel rod to the cask cavity, and the second from the cask cavity into the external environment. This section deals mainly with the first pathway.

To get any significant releases from a fuel rod, it must rupture (i.e., be "breached" or "cracked"). If the rod fails to maintain its integrity, fission product gases, produced during reactor operation and contained under pressure in the fuel-cladding gap, will be vented through the breach/crack in the fuel cladding and released to the cask cavity. [Note: only a relatively small fraction of the total fuel fission product inventory is present in the fuel-cladding gap].

The cladding can rupture as a result of either a severe impact (see Paragraph 10.1) or a severe thermal environment, often called "burst rupture."[5,6,19,20] Burst rupture can be due to increase of pressure inside the rod caused by an increase in fuel temperature. The pressure build-up may cause mechanical deformations in the rod and finally rupture of the cladding, creating a hole of a few millimeters. The gap inventory, especially that in the vicinity of the hole, will be vented. High enough temperatures may cause other vaporized fission products to diffuse through the gap and finally find their way out of the rod through the hole/crack. [Note: A type of release which is not necessarily caused by a rupture of the fuel rod is also mentioned in the literature.[5,6] This is the release of the crud that plates on fuel surfaces during reactor operations. The dominant radionuclide is cobalt-60 (Co-60). Some of the crud tightly adheres to rod surfaces and is not expected to desorb as a result of even a severe impact. However, a fraction of the crud that adheres relatively loosely to the surfaces — including cask surfaces — may be released to the cask cavity, and if the cask is damaged, a certain fraction of Co-60 can be released into the environment. In any case, analysis of inventories, release fractions, aerosolization processes and radiological parameters for extremely severe accidents with air-cooled casks leads to the conclusion that the doses caused by crud releases of Co-60 are approximately two orders of magnitude lower than those caused by burst ruptures even in cases where burst ruptures are not followed by processes such as oxidation (to be discussed below).]

The threshold fuel rod temperature at which burst rupture begins is extremely difficult to define. The threshold temperature strongly depends on a variety of factors such as heating rate of the fuel ($^{o}C/min$), fuel age, and fuel conditions (degree of embrittlement, etc.). In one example an incipient burst rupture temperature of 565^{o}C was proposed, and a value of 671^{o}C for the expected failure temperature was determined.[22] Other references[19,21] point to somewhat higher temperatures (higher than 700^{o}C) as characteristic of burst ruptures.

Lorenz et al.[19,20] suggested the following equation to calculate burst rupture releases in the temperature range of 700^{o}C to 900^{o}C:

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

where,

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

V_B = volume of plenum gas vented, in cm^3 at 0^oC 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 ^oK

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

Since Mo = M x f

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-cladding gap

Lorenz's equation becomes:

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

In the absence of "direct" experimental data related to burst ruptures of fuel rods contained in a shipping cask exposed to hydrocarbon fuel fires, the equation can be adequate for use in transportation accidents risk assessment provided that the values of the various parameters are selected carefully.

In Lorenz's equation, α, A and C are adjustable constants determined to fit experimental results. V_B and A are characteristic design structural parameters of the particular fuel rod and depend on the operational regime of the nuclear power plant. Therefore, these parameters are not (and cannot) be subject to much controversy, provided that the entire equation is agreed upon. However, f and T may be controversial, and special attention should be given to the selection of their values in various risk assessments. (The term

Table 10.6. Parameters for Burst Release Equation.

Radionuclide	$\alpha \left(\frac{g}{cm^3}\right) \cdot \left(\frac{g}{cm^3}\right)^{-a}$	a	V_{B3} (cm)	M(g) (Ci)	f	A (cm^2)	C (^oK)	T (^oK)
Cs	3.49	0.8	1100	6.4	0.2	1100	7.4×10^3	1123
I	0.163	0.8	1100	0.53	0.2	1100	3.8×10^3	1123

Source: Reference 5.

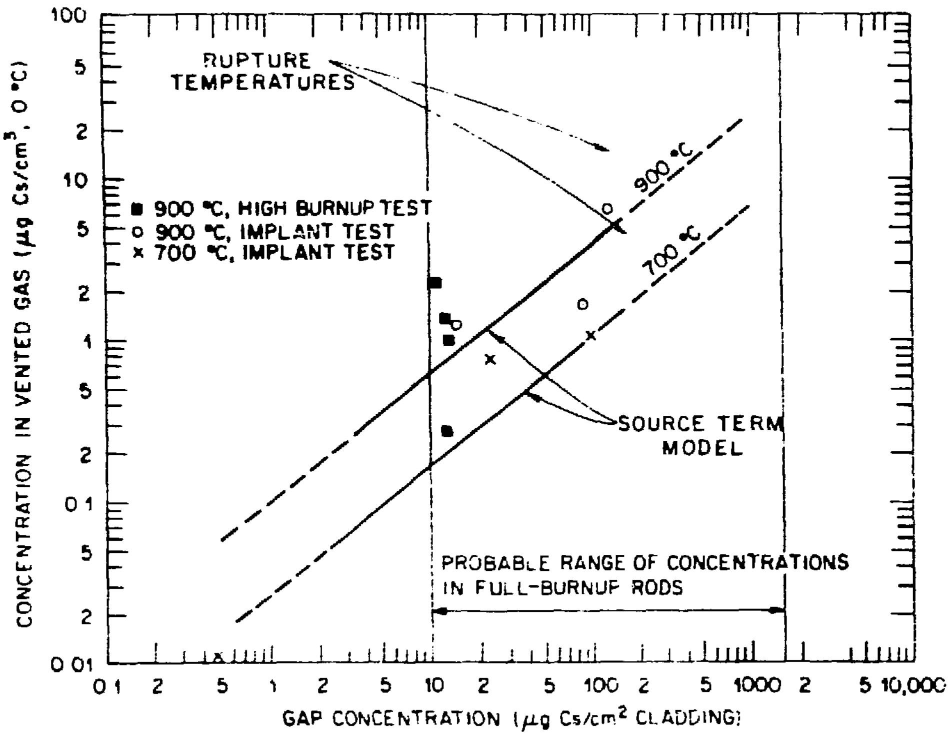

Figure 10.6. Concentration of cesium in gas vented at rupture. Source: Reference 20.

"various" relates to various possible degrees of conservatism in transportation accident risk assessments.)

One set of parameters[5] is given in Table 10.6. Elsewhere,[28] it is assumed that the values of f and T should probably be lower. The possible controversy over the values of f and T is discussed further.

10.4.1 Fraction (f) of Cesium Found in the Fuel-Cladding Gap

This parameter determines the activity of cesium (Cs) available for an instantaneous release in case of a burst rupture. A value of 0.2 was selected for f (leading to $Mf/A = 0.001$ g(ci)/cm^2).[5] This means that 20% of the overall Cs is assumed to be found in the fuel-cladding gap. From Lorenz's experiments, as well as from the theoretical GAPCON-THERMAL-2 code[23] developed by Beyer et al.[24] it can be inferred that f is highly dependent on fuel burnup (MWD/MT) and specific power rating (kW/ft). It should be stressed that:

(1) The H.B. Robinson PWR fuel used in Lorenz's experiments[19,20] had a 30,000 MWD/MT burn-up and a "relatively low" power rating of 6 to 10 kW/ft. This led to only 0.3% of the overall inventory (i.e., f = 0.003) of Cs that was found as "gap inventory."

(2) The range of Mf/A suggested by Lorenz et al. appears in Figure 10.6. It is clear that in the Sandia worst case analysis the extreme right hand side value of this range was chosen.

(3) Checking the calculated f-values for several types of fuels[23] (BandW 15x15, BandW 17x17, C-E 14x14, C-E 16x16, W 14x14, W 15x15, W 17x17, Exxon 15x15, Exxon 8x8, GE 7x7, GE 8x8) (see, for example, Figure 10.7) leads to the conclusion that f-values in the range of 0.2 should be used only for extremely high power ratings, say 15 kW/ft. It is obvious that such power rating values do not represent the average fuel rod, and it is quite questionable whether it can represent any "group" of fuel rods

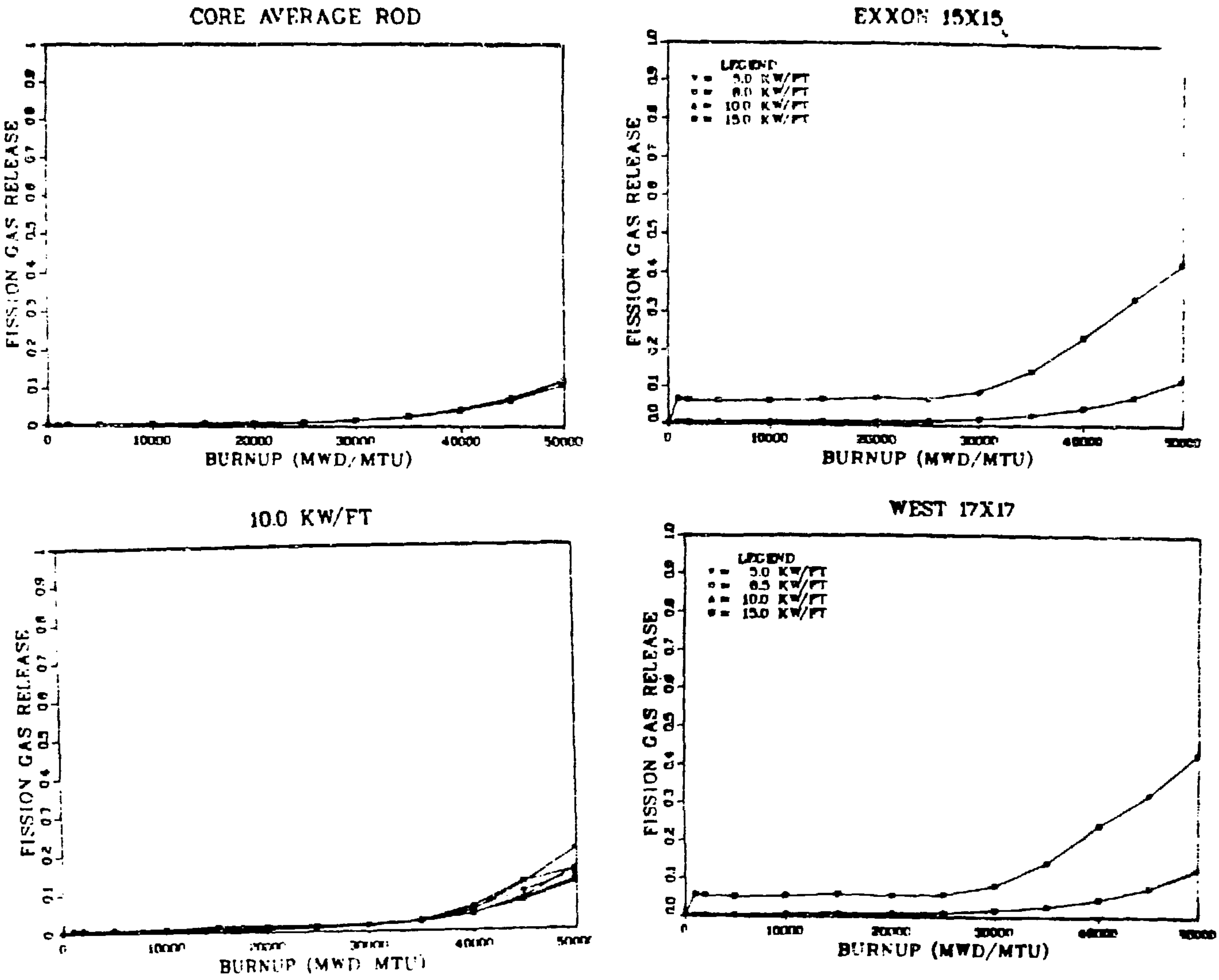

Figure 10.7. Fission gas gap releases as function of burnup and power rating. Source: Reference 23.

present in the shipping cask. In other words, even if we assume, conservatively, that all the fuel rods in a certain cask has been subjected to such high power ratings during power plant operation, the probability of an accident involving the low percentage of fuel rods having such large gap inventories is obviously smaller than 1.0. (This is because the fraction of high power rating fuel in the whole mass of transported spent fuel is low.)

(4) Moreover, today, power plants operate in fuel regimes having low power ratings. Values of f lower than 0.05 or even than 0.01 may be more appropriate for worst-case analyses.

(5) The assumption that f = 0.01 to 0.02 would cause M_B (mass of element released in the burst) to be lower by a factor of 6 to 11 than those obtained in the SANDIA study.[5]

Based on these observations it can be concluded that even for the worst case analysis, values lower than 0.2 should be chosen for f, or at least the probability of an extremely severe accident (severe impact and severe fire) to a cask containing fuel rods having an average f-value of 0.2 must be considered.

10.4.2 Maximum Credible Fuel Temperature

As pointed out in Chapter 10.2, the maximum credible fuel temperature (MCFT) in a fire depends on two main parameters: (1) duration of the accidental fire and (2) the fire temperature. The values of these parameters were also discussed. Now the question is: what is the maximum fuel rod temperature that can be reached under extremely severe fire conditions? Figures 10.8 and 10.9 (curve (1) in both figures) give the maximum fuel temperatures ("hottest fuel pin temperature", meaning that the temperature is not reached in all fuel rods) calculated for one-half hour and two-hour fires

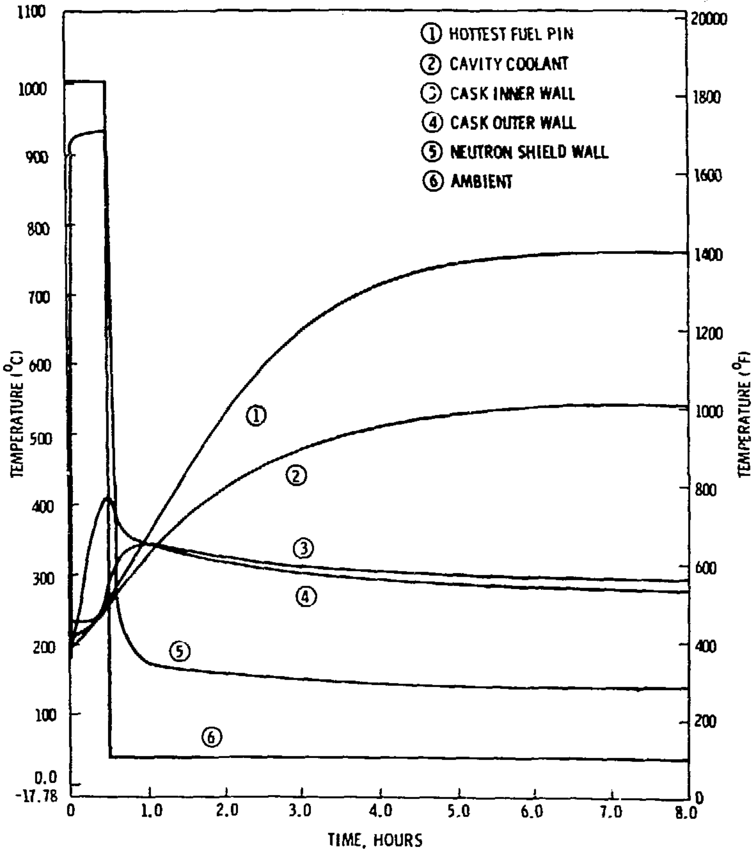

Figure 10.8. Half-hour fire at 1010°C.
Source: Reference 13.

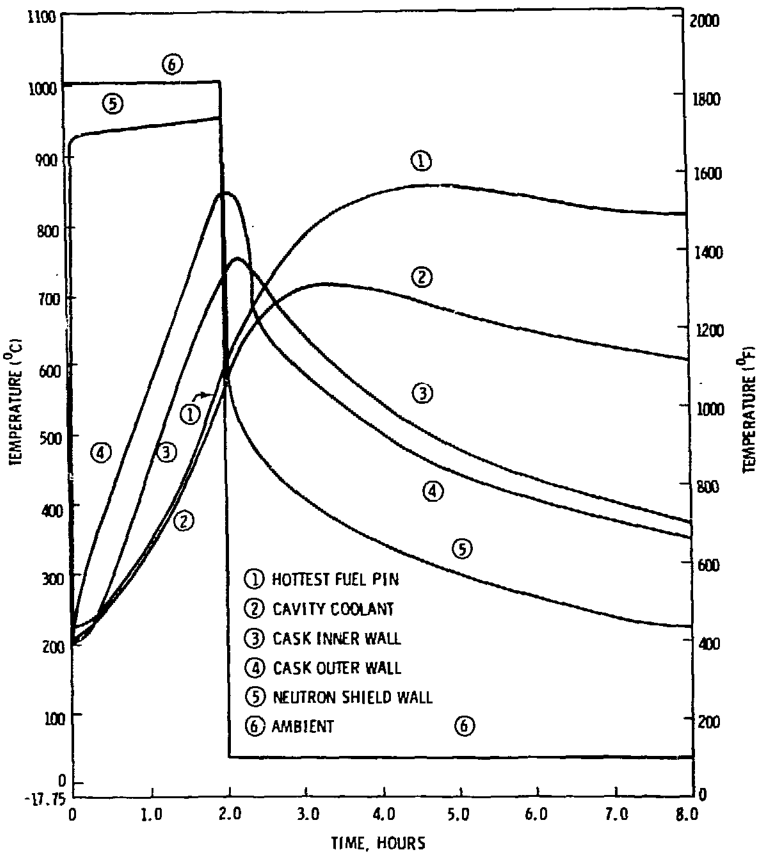

Figure 10.9. Two-hour fire at 1010°C.
Source: Reference 13.

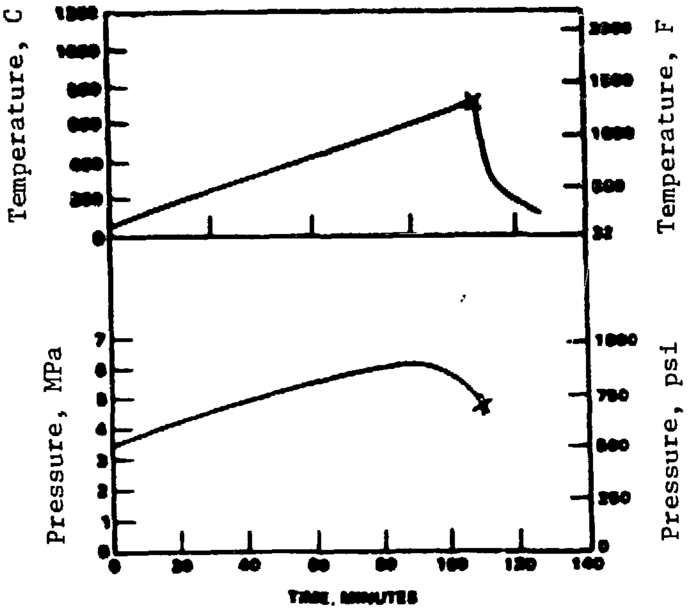

Figure 10.10(a). Typical test history of irradiated specimen. Source: Reference 25.

Test No.[a]	Condition	Pressure MPa (psig) Maximum	Pressure MPa (psig) Failure	Failure Temperature, C
1	Unirrad.	5.93 (860)	3.52 (510)	782
2	Unirrad.	5.96 (865)	4.31 (625)	800
3	Irrad.	6.14 (890)	4.76 (690)	727
4	Unirrad.	4.41 (640)	1.38 (200)	860
5	Irrad.	4.76 (590)	1.72 (250)	810

(a) Heating Rate: 4.4 C/min.

Figure 10.10(b). Summary of test results. Source: Reference 25.

Table 10.7. Cask analysis for hypothetical fire test conditions.

Cask	Coolant	Design Heat Removal Capacity (kw)	Spent Fuel Age (days)	Assemblies	Peak Fuel Temperature After Fire Test
NAC-1 & NFS-4	water	11.5	120	1 PWR/2 BWR	510°C
NLI 1/2	helium	10.6	150 PWR 120 PWR	1 PWR/2 BWR	594°C
TW 8/9	air	35.5/24.5	150	3 PWR/7 BWR	525°C
NLI 10/24	helium	70.0	150	10 PWR/24 BWR	533°C
IF-300	water/air	76.0 62.0	120	7 PWR/18 BWR	858°C 518°C

Source: Reference 5.

at 1010°C.[13] It can be easily seen that the highest temperatures, 750°C and 850°C, will not be reached before 3 to 4 hours have elapsed after the onset of the fire. It also means that some fuel rods will have temperatures for which burst ruptures can be postulated. [Note: Pacific Northwest Laboratories (PNL) curves[13] relate to relatively "young" fuel, i.e., 120 to 180 days old].

Experimental data obtained by Battelle Columbus Laboratory (BCL)[25] are shown in Figure 10.10. The conclusions are quite similar; i.e., it takes approximately two hours before a fuel rod in a cask exposed to 1050°C will reach a temperature of 720°C (still within the "burst rupture" range).

Other important phenomena caused by high fuel rod temperatures are diffusion and oxidation. Of course, these can only take place after a burst rupture. "Burst release" is defined as "release at time of rupture."[19] Diffusion release is a release from the gap space of those fission gas isotopes which were not released during the burst. These are released over longer times, mainly from gap inventories relatively remote from the crack/hole. The diffusion equation is:[19,20]

$$M_D = M_O \{1 - \exp[-(R_O t/M_O)]\}$$

where,

M_D = mass of element released by diffusion, in g

t = time at diffusion temperature, in h

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

R_O is defined by the equation:

$$R_O = \delta (W/P)(M_O/A)^a \exp[-\gamma/T]$$

where,

W = the width of the radial gap, in m

P = system pressure, in MPa

δ and γ are adjustable constants experimentally determined.

This equation expresses a simple diffusion from a depleted source.

In the Sandia study[5] release fractions of 3.6×10^{-3} were obtained from Cs burst rupture releases and only 4.7×10^{-4} for diffusional releases. Since for both releases, the fractional releases through the cask breach and other parameters (such as the respirable fraction of the total releases) were the same, it turned out that diffusion added no more than 10% to the burst release doses.

The oxidation process requires that "bare fuel" is exposed to air, that is, fuel surfaces along the fuel cladding gap be in touch with sufficient quantities of air to cause oxidation of fission products.[5] Although relatively high release fractions from air-cooled casks were expected due to oxidation, the final contribution of this mechanism to the overall doses (i.e., doses from burst rupture and oxidation) was less than 30% because the fraction of respirable aerosols due to oxidation was 20 times less (0.05 vs. 1.00) than the corresponding burst release fractions.[5]

10.5 PROBABILITIES

The appearance of WASH-1400[26] stimulated the use of a probabilistic approach in the field of risk assessments. Some probabilistic work was done to assess the risk of nuclear waste.[27] However, for the transportation of spent fuel casks, only partial probabilistic assessments of accidents were made. Extremely difficult problems still to be resolved have hindered any attempt to perform a full comprehensive PRA for transportation accidents.

The probabilities of severe truck and train accidents have been assessed, using historical data supplied by several federal agencies.[7,8,11] The probabilities covered ranges of accident-related velocities and of fires (including their duration). All possible types of accidents were analyzed, i.e., truck/standing object, truck/auto, truck/truck, truck/tanker, etc.

In spite of some differences in interpretations of data and the results, it can be concluded that these probabilities are available and reliable. In other words, the "initiating event" can quite easily be determined (probabilistically) for transportation accidents. The real problem arises when attempts are made to assess the behavior of the cask and the fuel under severe accident conditions, especially conditions that are "beyond design basis" (Class 9).

Unlike nuclear power plants where many systems (e.g., valves, pipes) have a large historical data base concerning their behavior in extreme conditions, only few data exist for casks and spent fuel. Thus, in several risk assessments the probabilities of initiating events (more severe than those represented by the regulatory tests) were assessed, and it was further implicitly assumed that any beyond-design-basis accident will cause severe release conditions. In other words, conservative values rather than ranges of values (or rather than "best estimate" values) were assigned for the various parameters. This approach was aimed to meet the regulatory requirement of worst case analysis. For example, it was assumed that any beyond-design-basis accident (severe impact and severe fire) would cause a maximum Cs and iodine (I) release, ignoring any possible probability distribution functions or at least ranges for the values of parameters determining the releases.[5] This led to a probability of $\sim 10^{-6}$/y to get a severe beyond-design-basis accident. This number should be interpreted as the very upper value for the worst case rather than a "probabilistic expectation value."

There are several parameters to which, despite the uncertainties, probabilities (either in the form of PDF's or as "expected probability values") can and should be assigned.[28] These parameters include f, T (see

TABLE 10.8. Health Effects Due to Vehicular Transportation Accidents.*

No.	Reference	Health Effects	Comments
1	(NUREG-0170[30]	LCF Annual Societal Risk: 5.4×10^{-3} Latent Cancer Fatalities/year EF Annual Societal Risk: 5.0×10^{-4} Early Fatalities 150 latent cancer fatalities in a very high-density site	for 1975 level fuel shipments
2	WASH-1238[21]	Probability that 100 people or more get 10 millirem: 10^{-5} Probability that 10^4 people or more get 10 millirem: 7×10^{-6} ----- Probability that 100 people or more get 10^4 millirem: 1×10^{-6} Probability that 10^4 people or more get 10^4 millirem: 1×10^{-9}	per 1000, mile train shipment
3	SAND82 -2635[31]	Early latent cancer fatalities (three assembly cask): 1/13 Total latent cancer fatalities (three assembly cask): 4/14 Peak Total bone marrow doses (30 m): 424-900 millirem Peak Total bone marrow doses (1400 m): 12-26 millirem Peak Thyroid dose (30 m): 215-460 millirem Peak Thyroid dose (1400 m): 9-18 millirem	1. The accident is caused by an explosive attached to the cask (simulating sabotage) 2. LCF values are shown by expected/peak values.
4	SAND84 -0062[32]	Expected 2×10^{-7} to 1×10^{-8} person x rem/km truck Expected 3×10^{-6} to 1×10^{-7} person x rem/km train	
5	RAE[17]	Doses to the most exposed person: ~10 rem from impact+burst+oxidation : ~ 6 rem from impact + burst : ~0.2 rem from impact.	Based on accident releases (for a rail cask) from Ref.[5].
6	NUREG/2325[33]	Expected value of: 0.1 person x rem/year	All shipments (1985 level)
7	NUREG/CR-0743[34]	Expected values of: 10^{-5} early morbidity/year 10^{-3} latent cancer fatalities/year 10^{-3} genetic effects/year	1. Limited to New York City area. 2. The values are per shipment year.
8	Eicholz[14]	Maximum "centerline" dose (rem): skin - 1.2, total Bone marrow-0.02, lungs-8, bone-6.	1. Based on USAEC study from 1972. 2. The doses are caused by rail accident.
9	Fullwood et al[35]	Maximum collective whole body dose: 5.2×10^{-7} person rem/trip	"trip" is ~ 1000 miles.
10	PNL2588[13]	Expected 4.5×10^{-5} fatalities/year (Societal Risk) Maximum individual risk: 2×10^{-12} death/person year	for 180 day old fuel.
11	Weston[36]	External dose: ~ 10 millirem Inhalation dose: ~ 97 millirem	Waste transporter accident occurring within a repository (impact + burst).

*NOTE: The results shown in this table cannot be compared with each other without carefully checking the assumptions and models used in each study.

Paragraph 10.4) and the probability of not taking any successful emergency action such as fire fighting.

10.6 PARTIAL SURVEY OF RELEVANT DATA

Several sets of experimental data have been used or should be used in transportation-accident risk assessments. As pointed out in the previous paragraph, these data are not sufficient for a comprehensive PRA, but a lot can be learned from them that can be applied in any kind of transportation-accident risk assessment.

The first set of relevant data relates to the "design-basis fire," that is, a 800°C fire for 0.5 hrs. Table 10.7[5] shows analyses made for several types of licensed casks, for hypothetical fire test conditions. The peak fuel temperatures obtained in these tests were most probably lower than those needed to start a massive burst rupture process. It should be emphasized again that when beyond-design-basis accidents are analyzed, a 1000°C fire lasting, say, two hours, should not be precluded.

The second set of data was obtained in several experiments performed at Sandia National Laboratories in 1977-1978. These experiments were performed to resolve doubts and uncertainties that had arisen concerning the ability of the regulatory tests to really encompass and represent "real" severe accident environments. They included two crashes of a tractor trailer rig carrying a spent fuel cask that was propelled into a 690 ton concrete block at speeds of 60 mph and 84 mph. In a third test, another truck-type spent fuel cask was mounted on a tractor trailer rig and struck by a 120 ton locomotive going 81 mph. In the fourth test, a rail-type spent fuel cask impacted the 690 ton target at 81 mph. In the last experiment, a burn test was performed, exposing a cask to a two hour fire of JP-4 fuel. From the experimental results it was concluded that "in none of the five tests would there have been any significant release into the atmosphere, had the cask been carrying spent fuel."[9] Still, the extent to which the results of these experiments can be applied in a worst case analysis is claimed to be questionable.[5]

In another set of experiments performed at ORNL,[5,19,29] releases of "gap inventories" of Cs and I were determined under various fuel rod pressures and temperatures. Fuel rods of various burnups and power ratings were tested. Amounts of releases due to bursts and slow diffusion were compared to those assumed for gap inventory releases in WASH-1400.[26] As discussed in the previous paragraph, the empirical burst release equations were used in worst case analysis performed at Sandia National Laboratories.[5]

Experiments intended to (1) identify those conditions within a transportation cask that produce fuel cladding failure and (2) determine by test the magnitude of the source terms so produced were performed at Battelle Columbus Laboratory.[25] Failure temperatures of the fuel rods were found to range from 727°C to 860°C. It is interesting that the maximum pressures in the rods were observed <u>before</u> the rods failed due to the swelling of the specimen. Again, it was concluded that cladding failures were caused by bursting and that the burst openings were so small that they were difficult to discern without magnification. Other conclusions are also listed in this study. Curves obtained in these experiments are shown as an example in Figure 10.10.

10.7 RADIATION DOSES AND HEALTH EFFECTS

Regardless of the controversy among scientists concerning probabilities and sizes of doses, collective doses, and maximum/expected health effects caused by severe transportation accidents, the severity of the health effects results is relatively low and needs to be put in proper perspective. It

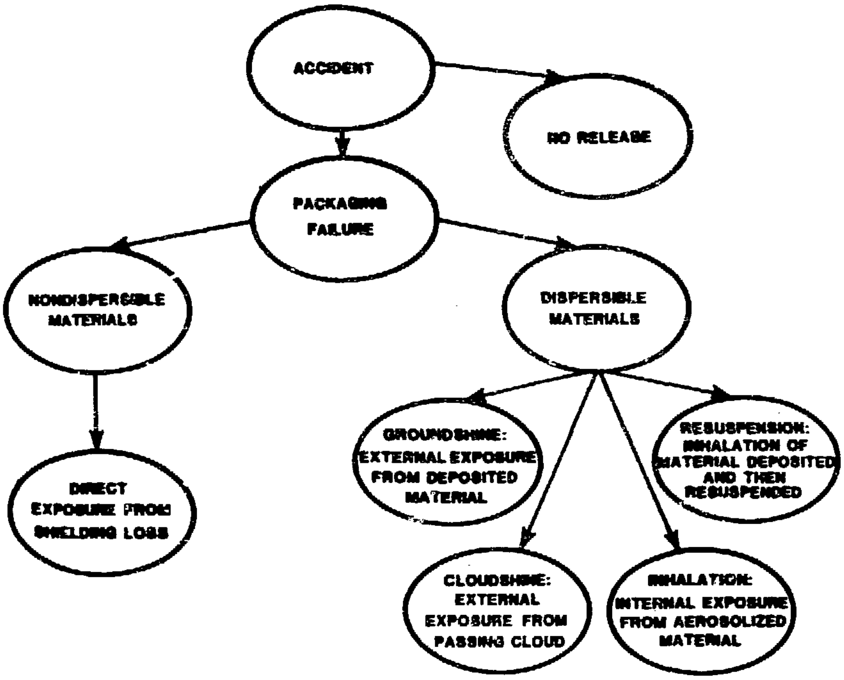

Figure 10.11. Accident dose pathways in RADTRAN2. Source: Reference 37.

should be emphasized that variability and differences in the sizes of health effects are due mainly to differences in postulated accident scenarios and different source terms (i.e., different release fractions were assigned even to identical scenarios). No major differences seem to appear in the consequence modelling of the accidents. The RADTRAN 2 computer code was used in various transportation accident risk assessments to calculate the health and the economic effects from accidents of varying severities. Figure 10.11 shows accident dose pathways modeled in RADTRAN. Similar pathways were assessed in other consequence models as well.

Table 10.8 shows transportation accidents health effects assessed in various studies. Although the numbers in the table cannot be compared to each other before the specific assumptions and models used in each particular study are carefully analyzed, it can be inferred that both expected and maximum health effects from severe transportation accidents are very low compared to other risks to which both individuals and the public at large are exposed. Such comparisons have been extensively discussed in the literature.

Even the extremely severe accident (impact + burst + oxidation) did not yield doses in excess of 10 rem to the "most exposed individual" with a probability of less than 10^{-6}/year.[5,17] If someone were exposed to 10 rem, then following the exposure, his annual increase of cancer death probability due to radiation is approximately $4x10^{-5}$/year (10^{-4} cancers/rem x 10 rem/25 years (the average remaining lifetime)). The combined probability is then $10^{-6}x4x10^{-5}$, that is, a combined individual risk of less than 10^{-10} death/year for the person who would receive the greatest dose. If we further assume that the risk is homogeneously distributed among say, 10^{5} people living close to transportation routes, we get an extremely low average individual risk of less than 10^{-15} deaths per year per person. This risk is less than one tenmillionth of the probability of being struck by lightning. Analysis of this extremely severe, low probability accident occurring in a densely populated area[17], resulted in an overall radiation-induced increase of latent cancer effects in the affected population of not more than 0.003%, given the occurrence of the accident.

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