STATE OF NEVADA

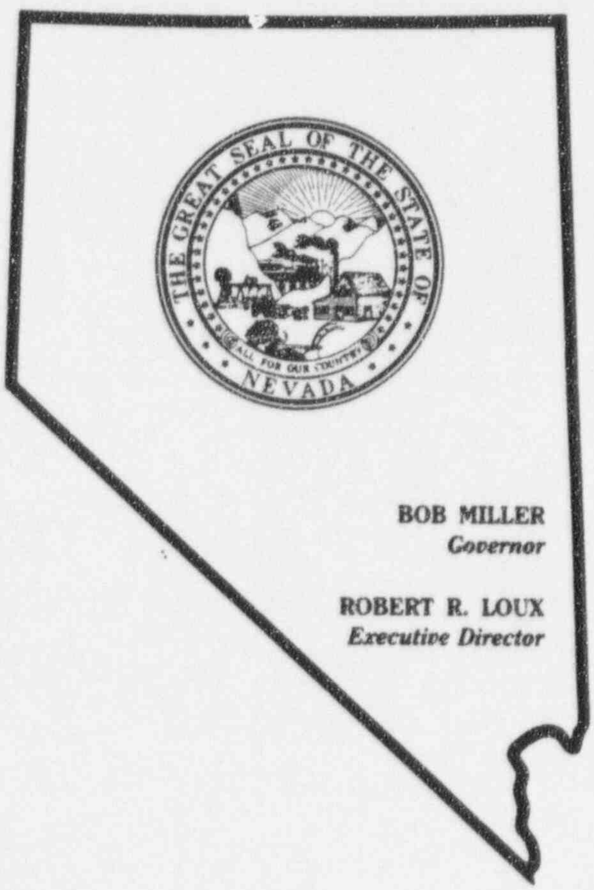
AGENCY FOR NUCLEAR PROJECTS/ NUCLEAR WASTE PROJECT OFFICE

NWPO-TN-007-90

GUIDELINES ON THE SCOPE, CONTENT, AND USE OF COMPREHENSIVE RISK ASSESSMENT IN THE MANAGEMENT OF HIGH-LEVEL NUCLEAR WASTE TRANSPORTATION

by

Dominic Golding and Allen White
Center for Technology, Environment and Development (CENTED)
Clark University
Worcester, Massachusetts



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The Nevada Agency for Nuclear Projects/Nuclear Waste Project Office (NWPO) was created by the Nevada Legislature to oversee federal high-level nuclear waste activities in the State. Since 1985, it has dealt largely with the U.S. Department of Energy's (DOE) siting of a high-level nuclear waste repository at Yucca Mountain in southern Nevada. As part of its oversight role, NWPO has contracted for studies designed to assess the socioeconomic implications of a repository.

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INTRODUCTI

In spite of the shortcomings of probabilistic risk assessment (PRA), the Transportation Needs Assessment (Task 15.2, p. 8) recommended this as the preferred methodology to assess the risks of high level nuclear waste (HLNW) transportation. This recommendation was not without qualifications. While no new basic methodological changes are required, existing methods will need to be adapted and extended to accommodate the particular complexities and data constraints which characterize the HLNW transportation system. A PRA also will need to heed the lessons learned from the development and application of PRA elsewhere, such as in the nuclear power industry. A set of guidelines will aid this endeavor by outlining the appropriate scope, content, and use of a risk assessment which is more responsive to the uncertainties, human-technical interactions, social forces, and iterative relationship with risk management strategies, than traditional PRAs. This more expansive definition, which encompasses but is not totally reliant on rigorous data requirements and quantitative probability estimates, we term Comprehensive Risk Assessment (CRA).*

Guidelines will be developed in three areas:

- the limitations of existing methodologies and suggested modifications;
- CRA as part of a flexible, effective, adaptive risk management system for HLNW transportation; and,
- the use of CRA in risk communication.

The guidelines concerning methodological limitations and modifications will draw on the many critiques and evaluations written to date (e.g., Freudenburg 1988; Lewis et al. 1978; NRC 1984; Gallagher et al. 1984). The guidelines concerning the use of CRA in risk management will draw primarily on the lessons learned in the development and application of PRA in the nuclear power industry. The guidelines on the use of CRA in risk communication will draw of the lessons learned from the disastrous release of the executive summary to the Reactor Safety Study (RSS), and the burgeoning literature in the field of risk communication.

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I. METHODOLOGY OF RISK ASSESSMENT

PRA as traditionally practiced is one of the most rigorous and widely accepted methodologies for assessing risks. Since its origins in the 1970s as a method for estimating the likelihood of accidents and associated damages at nuclear plants, PRA has evolved into the cornerstone of both nuclear plant risk analysis and, more recently, of setting standards for the chemical exposure of human populations. Despite its widespread acceptance, however, PRA as a tool for assessing HLNW transport risks has numerous shortcomings. This section will highlight the major limitations of the methodology, based on several critical reviews, and suggest necessary modifications.

In 1984, the NRC conducted an appraisal of the state-of-the-art in PRA. A summary of the principal findings is presented in Table 1. Focusing on the limitations of PRA, the NRC study concluded that the data bases for low frequency events, external accident initiators, and human failure are poor and underdeveloped, and the data base for low frequency events is unlikely to improve substantially. In terms of human performance, modeling errors of misdiagnosis and potential recovery actions needs improving, and the actual behavior of affected populations during an emergency is not well understood. In addition to the findings indicated in Table 1, the study concluded that completeness is not a principal limitation in regard to the identification of dominant sequences, and uncertainty and sensitivity analyses need to be more widely used and better organized and displayed.

Also in 1984, Gallagher et al. reviewed six out of twenty completed PRAs to assess what level of effort and topic of analysis could have the highest impact in improving PRA. The study found that a moderate effort in the identification of initiating events, the assessment of human errors during normal operations, and the treatment of recovery (especially recovery of human errors) could have a major impact and lead to significant improvements in PRA. A larger effort on human errors during accidents would also yield much improved assessments. They also concluded, in contrast to the above, that the degree of completeness of a PRA does affect the substantive conclusions. Similarly, Hamilton et al. (1986, 7-6) notes that PRA "intrinsically suffers a completeness problem: it is only as good as the imagination of the analyst."

Both studies cited above refer to the use of PRA in regard to nuclear power plants. More recently, Hamilton et al. (1986) evaluated the applicability of PRA to the HLNW transportation system. This study concluded (Hamilton et al. 1986, 5-1 and 7-12) that the philosophy, approach, and organization of PRA are appropriate to HLNW disposal. No new basic methodology is needed, but existing methods may have to be adapted or extended. Additional data may be necessary, but existing methods may have to be adapted or extended. Additional data may be necessary, such as

Table 1

STATE OF THE ART OF PROBABILISTIC RISK ASSESSMENT (PRA) OF NUCLEAR REACTORS

ASPECT OF PROBABILISTIC RISK ASSESSMENT	LEVEL OF DEVELOPMENT	RANGE OF UNCERTAINTY	IMPROVEMENT NEEDED
Qualitative systems analysis (logic modeling) • Internal accident initiators • External accident initiators	High confidence in qualitative insights Medium confidence in qualitative insights	± Factor of 10 ± Factor of (10 to 30)	Modeling common cause failures. Modeling common cause failures.
Modeling human performance		± Factor of 10	Errors of misdiagnosis. Potential recovery actions.
Data base: • high frequency events • low frequency events • internal accident initiators • external accident initiators • equipment and human failure	Fairly good Poor Fair degree of confidence Poor degree of confidence Needs improvement		Not likely to improve substantially.
Source terms due to internal reactor phenomena	Poor confidence	Very large	Extensive research; source terms remain quite large.
Consequences, given source terms and meteorology • mean early fatalities • mean population dose • latent cancer deaths	Reasonably high confidence	~ 0 to 5X ± Factor of (3 to 4) ± Factor of 10	Stochastic uncertainty.
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 1,000 reactor-years. X = nominal estimate.

SOURCE: NRC (1984).

for health effects at low doses of radiation, cask integrity under accident conditions, and the influence of human factors. The authors stress that the methodology is not a constraint; in fact, it generally is more powerful than the available data. Any future risk assessment of the HLNW transportation system needs to address a wide variety of initiating events, including external events such as floods and earthquakes, and the full spectrum of consequences, with greater attention to non-radiological and economic consequences. Special emphasis must be given to the human element in causing, and recovering from, mitigating accidents. Uncertainty is an ever-present problem with all risk assessments but greater use of uncertainty and sensitivity analysis should be made to illustrate the impact of differing assumptions and variable quality of data. Finally, the authors stress the need for clarity of analysis and careful presentation of the results. The latter point stems from criticisms of the Reactor Safety Study, and is a matter of risk communication which will be addressed later in this paper.

We concur with these conclusions and generally support the recommendations. We would, however, suggest that, to improve the methodology and apply it to the case of HLNW transportation, the implications can be viewed more broadly. Given that the methodology is often more powerful than the available data, theories, and models, it is suggested that a CRA rather than a PRA is the appropriate analytical framework. PRA may be especially inappropriate regarding certain low-frequency initiating events such as sabotage and terrorism, catastrophic infrastructure failures, and natural disasters. PRA also appears inappropriate where experimental data on key issues is limited or absent, particularly regarding cask performance under severe accident conditions. A CRA calculates probabilities only where the existing data, theories and models are sufficient to support the use of rigorous quantitative methods. The use of expert judgment should be limited and clearly indicated in a CRA, and the methods of expert judgment need close attention. Judicious use of analogue arguments, e.g., experiences in transporting chemical waste for which certain data types are more abundant, also can serve to enrich the quantitative dimension of a CRA.

The conclusions that a wider variety of initiating events and consequences, including external events and human factors, needs greater attention, should be reiterated and stated more broadly. A CRA should consider the full range of initiating events and the full spectrum of consequences. Historically, PRAs have used extremely narrow definitions of risk that are inconsistent with public perceptions of an adverse consequence and a tolerable risk, and thereby direct managers to inappropriate response strategies. As a result, traditional PRAs may fail entirely to address the risk issues of greatest public concern. A **comprehensive** risk assessment needs to address the complete range of events and consequences, and be able to accommodate a broader view of risk that is more compatible with that of the public.

Freudenburg (1988) suggests that opening the process of risk assessment to input from the social sciences will encourage this kind of comprehensiveness by enlarging the definition of causes and consequences of transportation hazards. In this fashion, CRA will impart greater realism to estimates of probabilities and uncertainties, and yield results which are more coincident with public sensibilities.

We now consider various aspects for achieving this objective.

A. Initiating Events

In assessing the full range of initiating events, the social sciences have significant insights about the issues of human error and human factors, sabotage and terrorism, and organizational deficiencies that hitherto have received only limited attention by risk assessors. Social and psychological research (Fischhoff et al., 1978; Slovic et al. 1985) indicates that low probability/high consequence events are of particular concern to the public. Such events, often ruled out as "incredible" by assessors, should be incorporated in a CRA to maintain public credibility, while serving as the upper bound estimate of plausible damages. Conversely, high probability/low consequence events may be ignored because they appear to be of little significance. Given the nature of the controversy surrounding the shipment of nuclear waste, these events may serve as "signal events" with severe repercussions for the management of the system (Slovic 1987; Kasperson et al. 1988). Every minor accident, with or without radiological consequences, will, at least initially, attract significant media attention and indicate to the public the inherent risks of transporting nuclear waste and the flaws in the risk management system. It is likely that the cumulative and synergistic effects of multiple small events, especially when radioactive materials are involved, will tend to undermine public confidence in risk managers. The CRA, therefore, should consider such events carefully in the hope of identifying vulnerabilities and preventive measures.

1. Human Error

Quality control failures and human errors have been a continuing major problem throughout the nuclear power industry and in hazardous materials transportation. The OTA (1986) estimates that more than 60 percent of hazardous material accidents are the result of human errors. Human error has also been identified as a major contributor to nuclear accidents, such as at Three Mile Island in 1979 and Chernobyl in 1986, and in non-nuclear accidents, such as at Bhopal. Similarly actuarial data indicate that 80 to 90 percent of accidents in the chemical industry and 60 to 80 percent of the accidents in the airline industry involve human error (Joksimovich 1984). In a similar vein, Audin, in a review of human error potential in the transport of spent fuel, stresses the failure of the environmental assessments to properly account for

human error (Audin 1987). The result is a systematic under estimation of risks. It is essential that any CRA not repeat such shortcomings but pay close attention to human errors which might compromise engineered integrity and safety systems.

Following a review of over 20 PRAs, Joksimovich (1984, 264) recommends that "the analysis of human interactions with plant equipment should at least receive the same degree of attention as analysis of systems hardware." Previous PRAs have highlighted the importance of human factors. However, they have also illustrated that large uncertainties in the quantitative estimates of risk result from "the lack of a large data base on human behavior and because evaluation techniques in this area are still in an early stage of development" (Levine and Rasmussen 1984, 253). There have been major improvements over the last 15 years in the way PRAs handle human factors, especially through modeling human errors based on the misperception of plant conditions (Garrick 1984, 276). Unfortunately, many improvements have yet to be made, and the poor data base will continue to be a substantial problem. Consequently, Gallagher et al. (1984, 83-84) concluded that a moderate research effort on human errors during normal operations could significantly improve PRAs, and additional improvements could be accomplished with a more extensive effort on human errors during accidents. This appears to be one area where HLNW transport may benefit from experience with chemical (including chemical waste) transport, for which data bases have been much improved as a result of the 1986 Superfund Amendments. A CRA should incorporate the role of human factors in the transportation of HLNW under both normal and accident conditions.

Human error and poor quality control may be either enabling or initiating factors in a sequence of events leading to hazardous outcomes. They may occur at any stage of the transportation system. This includes design, fabrication, testing, inspection, maintenance, quality control, operation, emergency response, or even in the risk assessment. Many kinds of human error have the same underlying causal mechanisms, consequently, they are treated here as generic factors of concern for both normal and accident conditions.

Some key issues to consider in an evaluation of human error are:

- task, physiological, and cognitive requirements;
- performance shaping factors (e.g., work environment, stress factors, fatigue);
- data availability;
- enabling vs. initiating errors;

- equipment design and operating requirements;
- observation and reversibility of errors;
- organizational structure; and,
- social environment (internal and external to the transport system).

Human errors may cause adverse impacts only if they are not corrected in time. Thus, their effects are dependent to large degree on their observability and reversibility (Levine and Rasmussen 1984). To assess the observability of errors and identify correction methods, the characteristics of the total human/task or human/machine system should be evaluated. The primary interactions are between task requirements (e.g., procedures), equipment and operating characteristics, and human physiological and cognitive capabilities.

In addition, persistent situational features should be evaluated. These include noise and illumination levels, worker fatigue and emotional stress, time pressures, and the organization of the work place. Persistent situational features, often referred to as performance shaping factors, may greatly increase the likelihood and consequences of errors. Emotional stress and time stress are performance shaping factors which have been shown to greatly affect decision behavior.

A complete evaluation of human error in the waste transportation system requires the evaluation of all levels of the socio-technical system. The organization of the workplace, including management-employee interactions, can have a significant effect on performance. Manager personalities, organizational culture, and the regulatory environment can constrain the types of risk management programs that will be acceptable and effective. Such concerns have recently been identified regarding the operation of nuclear power plans and many similar issues appear in the transportation system for nuclear waste (National Research Council 1988).

The types of errors and the rates of human errors are a function of the situation. Therefore, to evaluate fully and effectively the possible sources and impacts of human error, a comprehensive human factors analysis should be completed for the proposed HLNW transportation system. Such analyses should be performed for both transport activities (e.g., driving a truck) and cask handling activities (e.g., loading, packaging). The objective of the analysis is an overall understanding of the work environment, including goals and functions, performance shaping factors, and worker, task, and job requirements. After these issues are evaluated, the sensitivity of the system to human error can be assessed by identifying error types, their probability of

occurrence, potential for recovery, and consequences. Steps in a human/machine systems analysis include:

- describe system goals and functions;
- describe situational and performance shaping factors;
- describe personnel characteristics (including social issues);
- describe task and job requirements (including organizational issues);
- determine situations in which human errors may occur;
- estimate the probability of error occurrence for each identified error;
- estimate the probability that each error will not be corrected;
- develop changes in tasks, equipment, or systems to increase the overall system reliability; and,
- reiterate to evaluate modifications in system.

Whereas the transport of nuclear waste may appear to the lay person as less vulnerable than nuclear plant operations to human errors, such a conclusion is based more on the severity of outcome than the probability of misjudgments. In fact, the ability to standardize effectively and to enforce operating procedures in HLNW transport is confounded by the multiplicity of actors -- supervisors, loader/unloaders, drivers, inspectors -- whose decisions will affect the risk of waste movements from dozens of origins across the nation. In comparison with the insular nature of nuclear power plant operations, such individuals perform in relatively uncontrolled, vulnerable environments subject to the exogenous forces of weather, other roadway users, local highway conditions and a host of other variables beyond their control. Thus, over the course of HLNW preparation, loading, transport, and unloading, the opportunities abound for misjudgments which directly affect ultimate risk levels.

2. Sabotage and Terrorism

Sabotage refers to the deliberate disruption of waste shipments with the intent to steal nuclear materials or to cause harm by the release of such materials. Many of the same reasons that make the transport system vulnerable to human error also make sabotage or terrorism non-trivial hazard initiators. The experience with aircraft hijacking attests to the difficulty of protecting "moving" targets under multiple jurisdictions and

agencies over long distances. By analogy, the pattern of such events -- their timing, spacing, location -- may inform the analysis of hazard initiation in HLNW transportation.

Although it is impossible to quantify the probability of sabotage, the potential consequences can be systemically evaluated. DOE has undertaken such an evaluation by testing radioactive releases caused by sabotaging waste shipments with explosive devices, though such experiments have been questioned on the basis of alleged faulty procedures, documentation and peer review (Audin 1989).

One method of evaluation is vulnerability analysis. The list of parameters to be addressed in such an assessment is almost identical with the list for accidents (see below) since both lead to the potential release of radioactive and toxic materials. In addition, two other parameters are of concern: (1) the probability of sabotage, and (2) its severity. Past experience is too limited to allow reliable estimates of probability. Consequently, either expert judgment must be used, or acts of sabotage may be modeled without reference to probabilities, focusing only on response mechanisms to acts of sabotage and terrorism.

The severity of sabotage attacks runs the entire spectrum. Many acts of sabotage, such as deliberate tampering with vehicles or bridge destruction, result in sequences of events similar to other accidents and may be modeled in the same way. Other acts of sabotage may be deliberately directed towards exacerbating the consequences. Examples are driving damaged casks into urban areas or dumping them into a reservoir system. In the absence of probabilities, modeling these scenarios becomes a worst-case analysis.

B. Consequences

Social science input in a CRA can help to broaden the definition of consequences to bring it more into line with public perceptions. As pointed out previously, the CRA goes beyond PRA by identifying signal events that may have little immediate or obvious consequences in the near term, but severe repercussions later on if thy serve to cast doubts and undermine public confidence in the risk management institutions, or if they interact with high levels of public concern, when they create "signals." It is not expected that a CRA should assess these ultimate consequences in the traditional quantitative sense. Rather a CRA serves to evaluate the significance of these events and to estimate their number and probability, so that the risk management system will be better equipped to respond. A simple but major advantage of such an analysis will be to show risk managers that these events **will** occur relatively frequently and will not necessarily be amenable to "technical fixes."

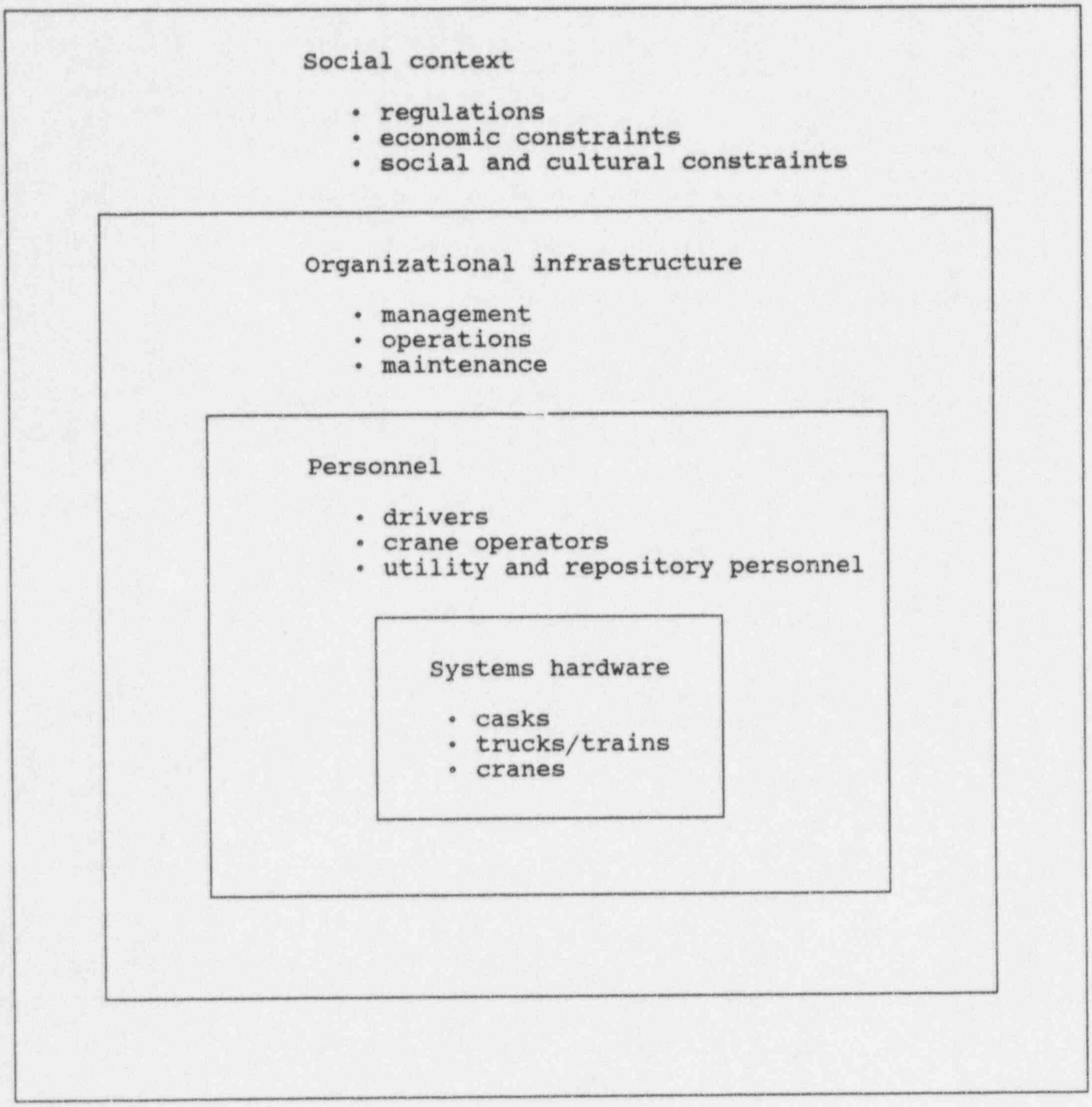
Many such consequences resulting from accidents are not ordinarily incorporated into risk assessments. These include: economic losses; the costs of emergency planning and preparedness; the behavior of public officials and households under emergency conditions; declining property values; psychological stress; and so forth. Some of these broader social and economic consequences might usefully be incorporated in a CRA, but the methodology should not be stretched beyond its limits. Rather, the results of risk assessments need to be integrated with additional socioeconomic and sociopolitical assessments in order to identify additional issues of public concern and how they might be mitigated. Concerns such as economic impacts and questions of institutional capability may have a significant impact on the ability to safely, reliably, and efficiently develop and maintain a transportation system for nuclear waste (NAS 1984). An especially important aspect of such studies is the evaluation of issues related to the social amplification of risk (Kasperson et al. 1988), or those attributes of risk events which tend to trigger adverse public reaction disproportionate to traditional measures of damages.

The need to identify and respond to all types of social concerns argues for a sociotechnical perspective for the risk management of the transportation system. A sociotechnical system refers to interacting components such as: system hardware (e.g., spent fuel casks, trucks, trains, cranes), personnel activities (e.g., drivers, crane operators), organizational infrastructure (e.g., operations, maintenance, management), and social factors (regulations, economics, culture). These are illustrated in Figure 1. If types and degree of interactions of different system components and levels are not taken into account in risk management activities, the result may well be the failure to implement effectively many of the suggested measures (National Research Council 1988).

The preferred approach should make clear that risk estimates are based on specific design and operating criteria, and that these estimates can only approach their true values if the transportation system is operated in accordance with the models and assumptions of the risk assessment. Any modifications to the system components (e.g., operations, maintenance, training) should be fed back into the risk assessment process, thereby creating a dynamic process capable of incorporating hardware and management changes into revised risk estimates. This again supports the need for a comprehensive systems approach to evaluating the transportation system. Concomitantly, new results from a risk assessment then need to be fed forward into developing new control strategies. This approach is particularly important in the transportation system because the technologies, regulatory environment, and institutional structures undergo frequent and often major changes.

Figure 1

A SOCIAL-TECHNICAL VIEW OF THE HIGH LEVEL RADIOACTIVE WASTE TRANSPORTATION SYSTEM



SOURCE: Adapted from National Research Council (1988)

C. Uncertainty

Uncertainty in risk assessments may arise from several sources, including:

- poor or inadequate data;
- the choice of models and assumptions;
- the use of expert judgment as a substitute for poor or missing data;
- the assumptions about human factors, and human and organizational errors; and,
- human error in estimation techniques.

Uncertainty is inherent in any risk assessment. This is especially so in the case of the HLNW transportation system since the final choices on system design have yet to be made, and there is minimal experience with, and inadequate data on, a number of system elements.

Much of the detail concerning uncertainty will be dealt with in a companion paper by Freudenburg. We need to recognize, however, that the same features that motivate our recommendation to recast the PRA into the broader CRA concept at the same time introduce uncertainties into the results. There is little doubt that incorporation of human error and institutional conditions elevate analytical uncertainties because such variables depend heavily on qualitative information and expert judgements. Suffice it to say that recognizing these uncertainties is a partial solution, and a CRA should clearly indicate the models and assumptions used, and the likely range of uncertainty involved. Such clear indications may help to avoid misleading the public into thinking their probability and consequence estimates are etched in stone. Sensitivity analysis should be used to indicate how sensitive the findings are to the assumptions, models, and data used. The results of such sensitivity analysis should be clearly displayed in the reports and executive summary. Continuous, rigorous, independent review of the CRA will aid in the identification of erroneous assumptions, faulty expert judgment, errors of omission and commission, and other unrecognized areas of uncertainty.

D. Some Parameters for Consideration

The CRA should consider the full spectrum of events and consequences under both normal and accident conditions. We, therefore, conclude this section on methodology by suggesting some of the parameters that need to be considered.

1. Normal Conditions

Both radiological and non-radiological risks occur in the normal or incident-free transportation of nuclear wastes. Handlers, crew members, and the public are exposed to the small amounts of radioactivity that penetrate through the walls of the shipping casks and the non-radiological risks such as exhaust emissions from trucks and rail locomotives. As such, there are no identifiable initiating conditions except, of course, the decision to ship nuclear waste and the choice of cask design. Poor quality control and human errors in cask design, fabrication, loading, and handling may exacerbate these normal exposures and can, therefore be seen as enabling events. Moreover, model choices in the transport system lead to measurable changes, and tradeoffs, in routine exposures to the workers and the public (Hoskins 1989). It is possible to identify probabilities for such errors, although the current methodologies are complex, and the data bases for so doing are somewhat rudimentary.

The following illustrative list indicates some of the parameters to be considered in estimating risks under normal operating conditions:

- quality control;
- human error;
- nature of waste material (amount, form, level of radioactivity);
- cask design and fabrication;
- nature of individual transportation procedures (time of day; length, number, and duration of trips; vehicle/train speed and stop-times; shielding factors);
- shipping information (number of casks);
- population at risk (workers, public);
- transport modes (truck, rail);
- transport route (route, length, population density, bridges);
- exposure pathways (inhalation, ingestion, direct);
- dose-response relationships (modeling, extrapolation); and,
- nature of consequences (acute/chronic, morbidity, mortality).

The exposure pathways under normal conditions include direct radiation from the waste package during loading and from the cask during shipping. Given the solid waste form and absence of accidents leading to volatilization or dispersion, inhalation of radioactive gases or particulates is unlikely. The primary route of exposure for non-radiological hazards is inhalation, although skin contact with, or ingestion of, lubricants and fuels is possible for crew members and maintenance staff.

The parameters necessary to estimate doses are also indicated in the above list. To estimate exposure levels at every point in the transportation network, it is necessary to know the nature of the waste material and cask, the nature of the various tasks (e.g., packing, loading, shipping, unloading), and shielding factors. Individual worker doses can be estimated from data on exposure levels during each task, task duration, the number of packages and casks processed, and the number of workers involved.

Estimating doses received by the public requires data on exposure levels at various distances from the cask, the number of casks to be shipped, transport mode, transport route (including data on population proximity and density, and the shielding factors of buildings), and the timing of shipping (including the time of day, the number and length of stops, and the expected population exposed in each case).

Having estimated doses to the various populations at risk, several models are available to predict possible adverse outcomes, including latent cancer fatalities, and genetic effects. There is considerable disagreement and controversy over the extrapolation from known effects at high doses to effects at low doses, and the assessment should provide several alternatives models for calculating the likelihood of adverse health effects, including delayed morbidity and mortality. In this fashion, a range of outcomes may be estimated.

While total, or "bottom line," risk estimates are typically the focus of public debate and policy-making, it is the individual components of risk under normal operating conditions that provide the most useful information for formulating risk reduction strategies. This point cannot be overstated. As HLNW moves through the packaging, loading, shipping, unloading, and emplacement process, each step presents a different set of exposure possibilities for humans and the environment. Furthermore, each step clearly affects all that follow, (e.g., sloppy loading increases shipping risks). Nevertheless, the tendency to look only at final risk figure obscures the relative weight of each step and may lead to suboptimal modal, routing, or other operational decisions. It may be the case, for example, that a transport system estimated to present higher aggregate risks is characterized by (a) less uncertainty, and (b) greater risk reduction opportunities for discrete components. In this instance, a

decision in favor of a lower aggregate risk transport system may be the wrong one. The estimated risk and surrounding uncertainties for each component of the transportation system should, therefore, be presented in addition to the estimated aggregate risk. This concept is equally applicable to accident conditions, the subject of the following section.

2. Accident Conditions

Assessing the probabilities and consequences of accidents is more complicated than assessing the risks of normal conditions, although many of the necessary parameters are the same. Some additional parameters needed are indicated below. Again, this list is intended only to be illustrative:

- frequency of accidents;
- external events (e.g., severe weather, earthquakes);
- common cause and common mode failures;
- nature of accidents (frequency, severity, location);
- release fractions;
- dispersion modeling;
- post accident behavior of officials, emergency workers, and the public; and,
- emergency response.

Human error and quality control take on added significance in accident analysis since they may act as initiating or enabling events in an accident sequence, or they may exacerbate or mitigate the final consequences. Accident frequency, both general transportation accidents and those involving hazardous materials, is an important input for this type of analysis.

Although these data are generally available for both truck and rail transport modes, their accuracy, precision, completeness, and applicability need to be evaluated with respect to CRAs for a transportation system. Special emphasis should be given to external events like plane crashes, bridge failures, earthquakes, floods, as well as combinations of events such as joint impacts of petroleum and pipeline fires with casks released during roadway accidents. Events such as earthquakes and floods may act to both initiate and exacerbate accident conditions, and impede mitigation efforts. An earthquake may initiate an accident by causing a truck to crash and burn, and thus result in a breach of the cask integrity. The earthquake may also exacerbate accident

consequences by causing infrastructural collapse that will hamper emergency response efforts.

The nature of the cask is also particularly important in accident analysis. The design and fabrication of the cask must allow it to withstand a variety of accident conditions. As yet data on cask behavior are limited to a handful of experimental crash and fire tests at the national laboratories (Hamilton et al. 1986, 10-225 to 10-28). However, these data are inadequate for a CRA, and their validity is in question (Resnikoff 1983; Audin 1989).

Intimately related to cask design and integrity is the severity of postulated accidents. This highlights a generic problem in risk assessment: what accidents are considered to be incredible and therefore beyond consideration? It is possible to identify a sequence of events leading to a massive release of radiation and a significant number of adverse effects, but is it reasonable to consider accidents with such low probabilities? These are questions that will have to be addressed at the beginning of the analysis. It is recommended that a broad spectrum of accidents be assessed, including the worst case. To omit such scenarios is to invite skepticism from critics who, in all likelihood, will eventually pose such worst case conditions.

For each accident scenario postulated, the risk assessment must calculate the release fraction or source term (i.e., the amount and type of different radionuclides released into the environment). This release may be in the form of a plume of radioactive particles and gases which will be distributed through the atmosphere and contaminate soil, water, buildings, and equipment in the area. The proximate population may be exposed by direct radiation from the plume (cloudshine) or from contaminated surfaces (groundshine), by inhalation of radioactive gases and particulates, and by ingestion of contaminated food and water. A risk assessment estimates the likely distribution of such materials and subsequent population exposures, using standardized dispersal, exposure, and dose-response models.

Most risk assessments ignore post-accident behavior. A CRA, however, must assess the impacts of the range of public behavior and emergency response activities that follow accidents. These activities may exacerbate or mitigate adverse effects and will vary according to political jurisdiction, population distribution, and other locational factors. Survey research exploring both hypothetical events ("what would you do if ...?") and actual occurrences (e.g., TMI, and acute chemical accidents) reveal a high degree of uncertainty and variation in how affected populations actually behave under different conditions. We do not have reliable predictions of how many households will respond to calls for evacuation, what distance they will travel, and, how long they will remain away. Such behavior directly affects exposure and

ultimately health impacts. A CRA must account for deviations between ideal and actual human behavior during emergency conditions. Because of the paucity of data with which to make such judgements, we believe additional survey research focused specifically on the HLNW transport issue is a necessary ingredient for a future CRA.

II. RISK MANAGEMENT

Risk management involves the identification, estimation, and evaluation of risks, and the selection and implementation of alternative measures to prevent, control, or mitigate adverse consequences. Risk assessment, which we considered in the previous section, generally is considered the initial phase in risk management. As we have discussed, its goal is to identify hazards, estimate the probabilities (where possible) of their occurrence, and forecast the nature and magnitude of their consequences. Risk assessment is, therefore, integral to devising effective risk management. Most of the studies of transportation risks conducted to date are risk assessments, and do not attempt to address the broader issues of risk management. In practice, they should interact iteratively, moving risk management in the direction of risk minimization within the context of meeting necessary social objectives, i.e., the disposal of HLNW.

A. Goals and Objectives

Any risk management system must have clearly articulated, achievable goals. In the present context, its primary goal is to minimize the risks resulting from the transportation of HLNW. A CRA, as we discussed earlier, helps to achieve this by revealing those system components that are most risk prone, thereby directing managers to certain design and operational modifications. These, in turn, allow for a revised CRA. This back and forth process is the essence of effective risk management. Optimally, the assessment-management interplay is a dynamic process that uncovers vulnerable points in the transport system; provides a constant flow of feedback data; and calls attention to previously ignored events and processes that may result in significant risks.

It is **not** the purpose of a CRA to convince the public that the risks are so small that the public should accept them. Previous attempts to use risk assessments in this fashion, such as the Reactor Safety Study (WASH 1400), have demonstrated this fallacy. Instead of focusing on the inordinately low probability estimates, public critics focused on the horrendous consequence of a major accident, and the litany of potential events leading to such an accident. Similarly, risk assessments should not be used as an excuse to conduct somewhat sensationalist "crash tests," such as some of those conducted in Britain by the Central Electricity Generating Board (Snedeker 1989), which are little more than public relations exercises. The British demonstration test "Operation Smash Hit" in which a cask is hit by a speeding locomotive actually subjected the cask to significantly less impact force than the regulatory drop test. Further, there is no empirical evidence that such demonstration testing is effective in changing public attitudes. This proviso, of course, does not preclude appropriate "real world" testing which may provide valuable data for improved risk assessments. Such tests must accurately represent maximum

credible severe accident conditions, and must be conducted under carefully controlled conditions, fully documented, and rigorously peer reviewed. Indeed, destructive testing of casks to determine failure thresholds should be an essential part of a risk assessment program.

B. System Design

Since 1975 when the Reactor Safety Study (WASH 1400) first applied the technique of probabilistic risk assessment to nuclear power plants, over 20 PRAs have been completed, under the sponsorship of the NRC and individual utilities (Joksimovich 1984, Daniels and Canody 1984). PRAs have not been used in developing the basic design of nuclear power plants, however, because the methodology developed after most of the designs were in place and many plants were already constructed, or under construction. PRAs have been most useful, however, in suggesting modifications to existing designs, and operational changes to enhance safety, and indeed performance. (Some of these modifications are discussed in the following section.) Furthermore, the growing realization of endemic problems with existing designs, as uncovered with the use of PRA techniques, has stimulated a reassessment of existing designs and encouraged the notion of developing inherently safe reactors.

In contrast, those developing the nuclear waste transportation system are in a unique position to be able to use a CRA as an aid in designing the optimal system to minimize risk. Conducting such an assessment **prior to** the construction of the system will encourage appropriate design choices, and may help to avoid the costly post-design changes that have plagued the nuclear power utilities.

The major advantage in conducting a CRA is that it forces a rigorous analysis of the complete system in an integrated fashion. Reviews of previously conducted PRAs conclude that conceptual insights about designs are the most important benefits (Levine and Rasmussen 1984; Joksimovich 1984). They have encouraged an "entirely new way of thinking about reactor safety in a logic structure that transcends normal design practices and regulatory processes" (Joksimovich 1984, 265). PRAs have been particularly important in suggesting corrective changes in equipment, maintenance practices, operational procedures, and operator training (Daniels and Canody 1984, 285), as discussed in the following section. At a more general level, PRAs have identified dominant sequences that might otherwise have been given less attention; previously unrecognized accident sequences and unexpected events and consequences; and high probability events that can easily be avoided. PRAs have also underlined the importance of human factors and external events, such as floods, earthquakes, and hurricanes (NRC 1984). A CRA that highlights

these issues prior to construction would be a valuable tool in the design of the transportation system.

Another lesson learned from previous PRAs is that they are extremely plant specific (Garrick 1984, 277), so the findings vary quite significantly from one site to another and among plants of similar but slightly differing designs. This has implications for the design of radioactive waste transportation system. While such a system is more "open" and less complicated than nuclear power plants, it will involve widely varying conditions of climate, terrain, population density. There is, therefore, a diverse array of potentially affected environments (Hamilton et al. 1986, 8-1) and a large number of different system designs. A comprehensive risk assessment will need to accommodate this diversity, and it is likely that the findings will vary significantly among different designs. Indeed assessing the differences between alternative designs will be a major function of the CRA.

Perhaps the overriding problem with PRAs is the calculation of probabilities and degrees of uncertainty. The NRC review of PRAs (Murphy 1984) concluded that: there is still considerable doubt about the statistical techniques of PRA, especially where data are limited; estimates of extremely low probabilities cannot be validated because data for low-frequency events, by their nature, do not exist; and, more work on uncertainties is necessary. In the absence of good data, expert judgment may be used but this is fraught with problems (Freudenburg 1988).

A final problem with attempts to define probabilities is the tendency toward reification of a single number. This is a big problem, and such numbers should not be used as measures of regulatory compliance. This undue attention to specific numbers was a problem in the executive summary of the Reactor Safety Study, where such attention focused on the one in a million estimate of core melt probability to the exclusion of all the qualifying details. Consequently, in response to the Risk Assessment Review Group Report (Lewis et al. 1979) criticizing the Reactor Safety Study, the NRC issued a policy statement to NRC staff. The memorandum advised that "the overall risk assessment results of the RSS . . . shall not be used without an indication of the wide range of uncertainty associate with those estimates," and that quantitative risk assessment techniques "should not be used to estimate absolute values of probabilities of failure of subsystems unless an adequate data base exists" (NRC 1979). Avoiding the presumption that probabilities can and should be estimated for all events and consequences is essential. Rather probabilities should be calculated only where there are sufficient data to replace expert judgment and avoid leaps of faith.

While the calculation of probabilities is problematic, it is also advantageous in that it forces the collection of enormous amounts of data in a relatively rigorous and comprehensive fashion.

This aids in the identification of dominant sequences and hitherto unrecognized events, and is particularly important for its qualitative insights. The NRC review (Murphy 1984, 6) therefore concludes that PRAs are useful provided "more weight is given to the qualitative and relative insights regarding design and operations, rather than the precise absolute magnitude of the numbers generated." A comprehensive transportation risk assessment must internalize these lessons. Particular attention should be paid to the qualitative aspects of system design, with the goal of risk minimization in mind.

While PRAs clearly have a role in evaluating the desirability of alternative HLNW transportation configurations, it is no substitute for a truly comprehensive risk assessment. A CRA should consider the entire sequence of transportation of both defense and commercial nuclear wastes, from initial packaging and loading to unloading of the repository. Moreover, it should extend beyond the borders of the state of Nevada. "Upstream" activities, such as packaging and loading, and generic issues of cask design, human error, and quality control, can have significant impacts "downstream" in Nevada, such as the potential for accidental releases and the exposure of workers unloading casks at the repository. To be truly comprehensive, the risk assessment should also identify the hazard events and consequences arising in the transportation of retrieved waste, since the repository is to be designed with such an eventuality in mind. While transportation to the repository is already controversial, it is likely that the transportation of retrieved waste will be even more controversial, since retrieval would only be necessary because of technological failure.

Because a transportation CRA by definition deals with hazardous materials which originate in and move across multiple political jurisdictions, questions of equity inevitable arise in considerations of system design. It is entirely plausible, for example, that aggregate (i.e., national) risk may be minimized by routing choices which work to the disadvantage of Nevada. By shifting waste shipments of alternative routes, risks to Nevada's population may be reduced but aggregate risks to other populations may be increased. It is entirely appropriate for the risk manager to recognize explicitly these trade-offs and to seek consensus on what reasonable solutions might be developed which balance risk minimization and equity concerns. Ignoring such trade-offs may imperil the entire design process, opening the door to criticism that fairness is being sacrificed to the national interest without due consideration for the host state.

C. System Operation

A CRA is not a static one-time analysis that results in a single document. It should be an ongoing dynamic analysis, integrated with a functioning risk management system, beyond the

design phase and throughout the operation of the HLNW transportation system. It should be continuously updated through a process of review, reevaluation, and data collection. The CRA should be adaptable and adapted to the changing conditions and needs of risk management. Thus, it should become a "living document" and a regular source of reference for those operating the system, so that appropriate engineering and operational changes can be made to ensure the system continues to meet its goal of risk minimization.

Previous PRAs have taught us a great deal about the operation of nuclear power plants. "Virtually every PRA study performed has resulted in changes in procedures and/or hardware which have reduced core damage frequency. This is not surprising, since identifying and implementing such changes is a key benefit of performing as PRA" (MHB 1989, 3-8). It is particularly important to note that many of these changes depend on qualitative rather than quantitative insights. In a recent book-length review of PRA techniques, Fullwood and Hall (1988, 294) concluded that "qualitative knowledge can be used to improve the operation of the facility without a high degree of reliance on the numerical estimates of probability and consequence." This finding further strengthens the emphasis on conducting a comprehensive rather than probabilistic risk assessment.

A transportation CRA can be expected to yield major insights in five generic areas that may enhance systems operation. These insights may come during the design phase and affect future operations, or they may arise from ongoing risk assessment during operation of the system. These five areas include:

- improvements in hardware;
- the identification of external events leading to accidents;
- human factors analysis;
- system management problems and solutions; and,
- the identification of "signal events."

Table 2 gives several examples of the plant modifications that have been indicated as necessary as the basis of PRAs conducted on nuclear power plants. Many of these include modifications to hardware as well as changes in maintenance and operational procedures.

In order to identify such necessary modifications, a CRA will need to consider the complete range of initiating events and likely consequences at each stage in the transportation sequence under both "normal" and "accident" conditions.

Table 2

EXAMPLES OF PLANT MODIFICATIONS MADE OR COMMITTED TO ON THE BASIS OF PRA INSIGHTS

PLANT	PLANT MODIFICATION	PRA	LEVEL
Arkansas Nuclear One	Station battery test scheduling changed to reduce probability of common-mode failures	IREP[a]	1
Arkansas Nuclear One	Ac and dc switchgear room cooler actuation circuitry test procedure established	IREP	1
Millstone	Logic changes made to emergency ac power load sequencer to eliminate single failure	IREP	1
Sequoyah	Procedures changed to ensure that upper compartment drain plugs are removed after refueling	RSSMAP[b]	2
Oconee	Procedure and hardware changes made to reduce the frequency of interfacing system LOCA	RSSMAP	2
Indian Point	Upgrading of charging-pump alternative shutdown power supply to reduce the probability of RCP seal failure	IPPSS[c]	3
Indian Point	Replacement of manual valves with motor-operated valves in fan-cooler service-water lines	IPPSS	3
Big Rock Point	Hardware modification to restrict flow in reject line between condenser hotwell and condensate storage tank	BRP[d]	3

[a] Interim Reliability Evaluation Program.
[b] Reactor Safety Study Methodology Applications Program.
[c] Indian Point Probabilistic Safety Study.
[d] Big Rock Probabilistic Risk Assessment.

SOURCE: Fullwood and Hall 1989

Workers and members of the public will be exposed to risks from the operation of the nuclear waste transportation system even under "normal" or accident-free conditions, in the absence of accidents, inadequate quality control, human error, and sabotage. Under normal operating conditions non-radiological risks include exposure to vehicular emissions, exposure to toxic substances, such as solvents, oils, and petroleum products used in the operation and maintenance of vehicles, and ergonomic and psychological stress. Radiological risks include public exposure to low level radiation during shipment, and worker exposure during packaging, loading, transporting, and unloading waste materials. Current regulations allow low levels of radiation to be emitted from even the best designed casks under the most stringent quality control programs, and in the absence of accidents. A CRA will need to assess the entire range of risks under "normal" operating conditions in order to identify appropriate modifications in hardware, operations, and maintenance.

Accidents include "internal" events or system failure, such as vehicular and loading accidents, or sabotage, external events that originate outside the waste transportation system. Both radiological and non-radiological consequences may be associated with accidents. External events include intentional (e.g., terrorism) and unintentional (e.g., airplane crashes) man-made events, and natural events, such as earthquakes and floods. Again, a CRA will need to assess the entire range of risks under accident conditions in order to identify appropriate system modifications. Power plant PRAs have been particularly useful in identifying significant external events that were previously ignored (Murphy 1984, 6). A transportation CRA should, therefore, pay close attention to external initiating events, especially since the transportation network will cover a wide range of diverse environments.

Unfortunately, many external initiating events occur with such low frequencies that quantification is difficult, and it is unlikely that the data base will improve appreciably in the future (Murphy 1984, 2). So any insights gained will come from qualitative rather than quantitative analysis. Sabotage (internal man-made event) and terrorism (external man-made event) fall into this category of low frequency events. Neither have been adequately treated in previous PRAs, but a CRA will need to pay particular attention to these if only in qualitative terms. The consequences are potentially severe and the transport system is, by nature, particularly vulnerable in comparison with nuclear power plants.

Previous PRAs of nuclear power plants have indicated the significance of operational procedures, maintenance practices, and operator training. As such, they serve to indicate where changes in plant management can improve safety. Similar benefits can be expected from a transportation CRA.

As indicated previously a comprehensive transportation risk assessment should be adapted and adaptable to the overall risk management system, and should be a continually updated source of reference. A similar recommendation was put forward by Garrick (1984, 278) several years ago: "The future of risk management lies in the computerization of the plant specific PRA . . . This concept would truly make risk management a reality and would turn a PRA into a useful living document of the nuclear age." A computerized CRA would amass data on normal and accident conditions throughout the entire transport system. Monitoring, tracking, and record-keeping of activities and events within the waste transportation system would be relatively straightforward given the relatively few shipments, sources, routes, and destinations compared with, for example, the much more complex system of transporting non-nuclear hazardous waste. Such a computerized system would allow the timely collection of a comprehensive data base which could be fed back into the CRA. In this fashion, it could be used interactively to assess the desirability of a wide range of operational adjustments.

In terms of management, a major advantage of a CRA is not so much the outcome as the process. Utilities conducting PRAs found that the process was an invaluable opportunity for in-house engineers and other personnel to become intimately familiar with the plant operations and procedures (Daniels and Canody 1984, 285). Not only does the involvement of in-house personnel improve the quality of the PRA and identify system vulnerabilities, but it also sensitizes in-house personnel to the notion of the limits to technology and that indeed things can go wrong. This is particularly important in an industry that is plagued by feelings of overconfidence and a mindset that, indeed, things can't go wrong. It may also contribute to dispelling the myth that there is always a "tech-fix." It is partly in recognition of these advantages that the NRC now requires all nuclear power plants to perform Independent Plant Examinations (IPEs) to identify plant specific vulnerabilities, and recommends that these limited visions of PRAs be conducted by in-house personnel. Obviously, a CRA would offer similar advantages in the management of the HLNW transportation system.

Finally, in terms of system operation it is important to note that a CRA will identify a large number of high probability/low consequence events. The future designers and operators of the transportation system must recognize that many of these seemingly insignificant events will have major "signal value" to the general public through social amplification and subsequent ripple effects (Kasperson et al. 1988). Many such events cannot be eliminated by design changes or technical fixes. Instead, particular attention will need to be given to operational considerations such as driver training, routing choices to avoid populated areas, emergency planning and response measures. In large part, the magnitude of the consequences of these signal events will be related to their

coverage in the media and the public perceptions of, and trust in, the risk managers' abilities to handle unusual occurrences.

III. RISK COMMUNICATION

A great deal has been written about risk communication in recent years and several excellent guides for risk communicators are available (e.g., Hance et al. 1988; Covello et al. 1988). Nonetheless, there are no guaranteed strategies and looking for them will be a disappointing task. Risk communication about nuclear issues, including the transportation of nuclear waste, will be particularly difficult given the controversial nature of nuclear issues of and the long history of public distrust (Hohenemser et al. 1977; Mitchell 1988). Effective risk communication about high level nuclear waste transportation will require the development of a credible process that encourages public dialogue and trust. Particular attention must be paid to the process as well as the content of risk communications. This section will begin by outlining some general considerations, move on to consider some of the lessons from the release of the executive summary of the Reactor Safety Study, and finish with some more specific pointers about the scope, content, and design of risk communication materials.

A. Goals

A CRA is not intended as a risk communication vehicle **per se**, but it will become such by default. this needs to be borne in mind in both the process and product of the assessment.

It is **not** the goal of a CRA to convince the public that the transport of high level nuclear waste is "safe." As noted earlier, previous attempts to use risk assessments in this way have backfired. The public tends to ignore the estimates of low probability (perhaps rightly, given the considerable uncertainty involved), and focuses instead on the bewildering diversity of initiating events and the potentially dire consequences.

In terms of risk communication, the CRA has two goals. First, by involving designers and operators of the system, the CRA can serve to dispel the myth of invulnerability and the assumption that technical fixes can effectively reduce serious risks to zero. By infusing human behavior into the analysis on both the risk initiation and control side, designers and operators will be sensitized to the role of human error and response in optimizing the transport system.

The second goal is to involve the public in a dialogue about the risks to ensure the comprehensiveness, completeness, and integrity of the assessment and the management system, to demonstrate that no one is trying to mislead the public, and to enhance the credibility of and public trust in the risk management system.

B. Process of Risk Communication

The comprehensive risk assessment is but one part of the risk management system. To be credible and trusted, the risk management system as a whole must be open to public comment and criticism throughout the affected states. Moreover the public should be intimately involved in the establishment and operation of the system through public hearings and committees, such as the local emergency planning committees established under SARA Title III. Similarly, the CRA should be conducted in an open atmosphere with initiatives to secure peer and public review. Unlike the Reactor Safety Study, the CRA must confront comments and criticisms fairly and appropriately. The executive summary in particular, should be widely disseminated for public comment in the affected states.

Risk communication with the operators and designers of the transport system will be achieved in large part through their involvement in the conduct of the risk assessment itself. Other technical experts will refer to the main text. An executive summary will be produced initially during the design phase of the system, and this will need to be revised periodically as new data are gathered and incorporated during the operational phase. The executive summary will serve as the primary vehicle for risk communication with the public directly, and through the media. The summary will therefore need to pay close attention to the lessons learned from the release of the executive summary of the RSS, and the pointers outlined below.

Given the public distrust of the DOE and NRC, any risk assessment conducted by these agencies would have little credibility and would likely be a poor vehicle for risk communication. A credible risk assessment would have to be conducted by an independent body, such as a consortium of the affected states. This body might be given the resources to hire a team of experts to protect the interests of these states. As a conflict avoidance strategy, this team might pursue a joint fact-finding and methodology development mission along with DOE experts, but reserve the right to dissent if it feels appropriate to do so. Since it is intended that the risk assessment become a "living document" with continual updating and application, it would be advantageous to have continued public involvement in oversight and implementation of the analysis.

Drawing from the experience of radioactive waste and hazardous waste facility siting in recent years, a number of options are available to ensure such involvement:

- the creation of a permanent HLNW Transport Oversight Committee, comprising representatives of the populations of the various states encompassed by the transport system, to monitor performance of the system and advise

system operators of public concerns (the Oak Ridge MRS Task Force is illustrative);

- regular publication in the local media and a facility newsletter describing the systems operations, e.g., volume in transit, origin points, volume received at the repository, as well as other operational and performance information;

- regular publication through the same channels of information pertaining to actions taken by management to modify the transport system in response to areas identified for upgrade or improvement;

- the establishment of a system of warnings and penalties wherein certain events, depending upon their severity and frequency of occurrence, a financial penalty, temporary suspension, or longer term stoppage of waste transport.

These are all illustrative of the kind of mechanisms designed to enhance public involvement in ways consistent with the emerging consensus in the hazards management field -- that information access, monitoring and control -- rather than financial benefits -- are the most effective devices to build and sustain public acceptance. There are many others contained in statutes, regulations and negotiated agreements for chemical and low level radioactive waste facilities which can further inform public involvement initiatives in the HLNW transport area (White et al. 1988).

C. Lessons from RSS

The executive summary of the Reactor Safety Study was released in October 1975 along with the main report. The summary was severely criticized for a number of reasons:

- it described only the early health effects arising from nuclear accidents and not the much larger number of delayed effects assessed in the main report;

- it gave no indication of the large uncertainties associated with probability estimates; and,

- it ignored potentially significant initiating events such as sabotage and terrorism.

Consequently, the NRC commissioned the Review Group, chaired by Harold Lewis, to review the report and recommend how risk assessments should be used in the regulatory process. The Lewis Report, as the findings of the Review Group came to be known, found failings in three areas: the executive summary; the peer review; and, the calculation and presentation of probability estimates.

The findings in each of these three areas offer important lessons for the conduct of any future risk assessment for high level nuclear waste transportation.

In regard to the executive summary the Review Group found it "is a poor description of the contents of the report, should not be portrayed as such, and has lent itself to issue in the discussion of reactor risks" (Lewis 1978, viii). Because the summary failed to indicate the full extent of the consequences in the event of accidents, and because it failed to emphasize sufficiently the uncertainties in the calculations of probabilities, "the reader may be left with a misplaced confidence in the validity of the risk estimates and a more favorable impression of reactor risks in comparison with other risks than warranted by the study" (NRC policy statement 1/18/79 p. 2). Lewis et al. (1978) concluded that the summary was not actually a summary of the report but rather a public relations exercise intended to convince the public that reactors were safe compared with the other risks to which the public is exposed (NRC policy statement 1/18/79, note #5). In light of these findings and other criticisms, the NRC withdrew its endorsement of the executive summary in January 1979. The NRC also issued a memorandum to its staff outlining the uses and limitations of risk assessment in general and the RSS in particular.

The Lewis Report also found that in the peer review process, the RSS ignored or evaded cogent criticisms, and where it did respond the response was weaker than it should have been. In terms of accident probabilities, the Lewis Report found that the error bands in the RSS were greatly understated because of: inadequate data; an inability to quantify common cause failures; and questionable methodological and statistical procedure. Hence, the emphasis now put on conceptual and qualitative insights from PRAs rather than an inordinate focus on quantitative findings.

From the NRCs experience with the RSS, any comprehensive transportation risk assessment should:

- estimate the probabilities only where the data are adequate;
- clearly indicate the use of expert judgment, where data are lacking or inadequate, and the range of uncertainty, and the process and experts used
- address the full range of events (including sabotage and terrorism) and consequences (including immediate and delayed effects);
- fully and candidly address all criticisms; and,
- include as an integral part an executive summary which, like the report itself, is subjected to peer review.

The executive summary, furthermore, should:

- **not** be used as a public relations exercise in an attempt to convince the public of the safety of the system;
- accurately portray the contents of the main report, including the major criticisms; and,
- emphasize the limitations of risk assessment and in particular the problems with poor data, expert judgment, and scientific uncertainty.

D. Some Pointers

In addition to the larger issues of the goals and process of risk communication and the lessons to be learned from the RSS, there are some more specific points to be considered concerning scope, content, and design of risk communication materials. These pointers apply mostly to the executive summary, since this will be the principal vehicle for communicating with the public. Many, however, also apply to the main CRA report, and the process of risk communication in general. While this paper considers the role of a CRA in risk communication, it should be remembered that risk communication is much broader than simply conducting a risk assessment and disseminating the findings. All manner of activities during the design and operation of the transportation system will be scrutinized by the media and the pubic and will serve as channels and opportunities for risk communication. A CRA is, therefore, only a small part of the wide spectrum of risk communication activities.

The executive summary should be written clearly and concisely in plain English, using lay terms that are easily understandable to the public. The goals and objective of the risk management system in general and the CRA in particular should be clearly outlined in regard to both the design and operation of the HLNW transport system. The major limitations and assumptions of the CRA should be clearly and fully stated. The goal of the executive summary should be: to enlighten not confuse; to clarify not conceal risk information; and to aid not impede valid inferences about the nature and magnitude of the risks. In short, risk communicators must recognize that while members of the public cannot know all the details, they are quite capable of prudently evaluating conflicting evidence (Freudenburg 1988), playing a role analogous to a Board of Directors who rightfully lean to conservative positions when uncertainties and potentially major harmful consequences may result from misjudgments or leniency.

In terms of scope and content, the executive summary must consider the full range of events from the most common to the most rare, and from the well known to the least understood. This comprehensive coverage is necessary to demonstrate the thoroughness

of the CRA and the risk management system. Particular attention must be paid to sabotage and terrorism even though the data may be slim, because regardless of probabilities the system is perceived to be vulnerable to these events. This, as pointed out earlier, is especially true for a transport system comprising thousands of miles of routes and millions of ton/miles of activity. Similarly, risk communicators must pay particular attention to "signal events" whose immediate consequences may not appear to be large.

The risk communicators must recognize that there are going to be large numbers of these events and that they will cause significant media attention and public concern. Such signal events should not be ignored or trivialized, they should be put in context with other activities of a similar nature. Risk managers must clearly demonstrate how the system is prepared to handle such events, for example, through a comprehensive tracking system and emergency response capability. Constant vigilance and a degree of over-preparedness may be the necessary price for public confidence.

The preference for a CRA rather than a PRA should be explained, noting that for some events and consequences the data are inadequate to calculate probabilities with reasonable certainty. The summary, and the main report, should clearly indicate the assumptions, where data are poor or lacking, and the degree of uncertainty associated with any numerical estimates of probabilities, consequences, and so forth.

The presentation of quantitative risk information is particularly problematic in any risk communication effort. The way in which such information is presented can greatly influence the interpretation by the public, and, therefore, needs particularly careful consideration. The choice of risk measures (such as the number of injuries, fatalities, and accidents) can greatly influence how the information is perceived. For example, saying that there may be as many as 1,000 fatalities over the life of the repository (10,000 years) may elicit a quite different public response than saying there may be only 1 fatality every 10 years. In general, absolute numbers tend to provoke greater concern than do ratios, and the choice of numerator (e.g., injuries, fatalities, accidents) and denominator (e.g., number of shipments, population exposed, miles travelled, years of operation) will frame the information quite differently. Risk communication materials should, therefore, use several measures, some of which may be better understood by different individuals, and which in their totality may give a more rounded perspective as the significance of the risk.

The CRA will include probability estimates where the data are adequate, but these will need careful explanation since most people do not intuitively understand these numbers. This is especially so when the probabilities are extremely small. Graphical presentations including the probabilities of comparable risks can

help (see, for example, Figures 2 and 3 taken from the EPA **Citizen's Guide to Radon**). The graphical presentations in the RSS have been criticized (Hohenemser et al. 1977) because they included only prompt fatalities and failed to indicate the degree of uncertainty. Comparing risks in this fashion is a useful approach but the comparison risks should be carefully chosen. Both immediate and delayed effects should be considered, and the degree of uncertainty around the estimates should be clearly indicated. Like probability uncertainty will need to be carefully explained since it too is not intuitively obvious. In drawing on other risks for comparison, careful consideration should be given to qualitative aspects, such as voluntariness, newness, reversibility, and so on (Figure 4).

Figure 2

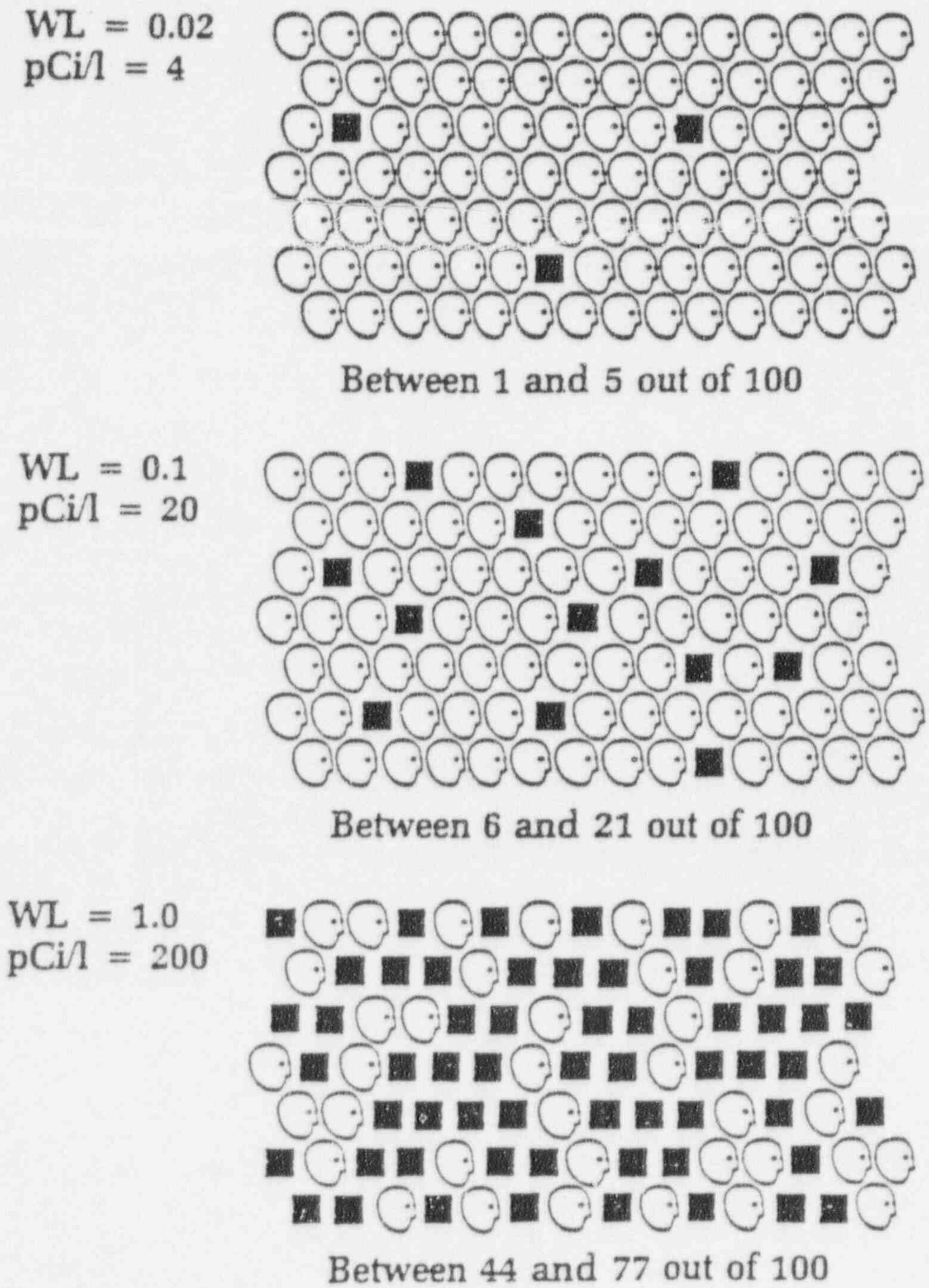
RADON RISK EVALUATION CHART

pCi/l	WL	Estimated number of lung cancer deaths due to radon exposure (out of 1000)	Comparable exposure levels	Comparable risk
200	1	440—770	1000 times average outdoor level	More than 60 times non-smoker risk 4 pack-a-day smoker
100	0.5	270—630	100 times average indoor level	20,000 chest x-rays per year
40	0.2	120—380		
				2 pack-a-day smoker
20	0.1	60—210	100 times average outdoor level	
				1 pack-a-day smoker
10	0.05	30—120	10 times average indoor level	
				5 times non-smoker risk
4	0.02	13—50		
				200 chest x-rays per year
2	0.01	7—30	10 times average outdoor level	
				Non-smoker risk of dying from lung cancer
1	0.005	3—13	Average indoor level	
0.2	0.001	1—3	Average outdoor level	20 chest x-rays per year

SOURCE: EPA (n.d.)

Figure 3

LUNG CANCER DEATHS ASSOCIATED WITH EXPOSURE TO VARIOUS LEVELS OF RADON OVER 70 YEARS



SOURCE: EPA (n.d.)

Figure 4

AN ARRAY OF CONSIDERATIONS INFLUENCING SAFETY JUDGEMENTS

Risk assumed voluntarily	Risk borne involuntarily
Effect immediate	Effect delayed
No alternatives available	Many alternatives available
Risk known with certainty	Risk not known
Exposure is an essential	Exposure is a luxury
Encountered occupationally	Encountered non-occupationally
Common hazard	"Dread" hazard
Affects average people	Affects especially sensitive people
Will be used as intended	Likely to be misused
Consequences reversible	Consequences irreversible

SOURCE: Lowrance (1976, 87)

IV. SUMMARY AND CONCLUSIONS

A CRA is the preferred methodology to assess the risks of HLNW transportation, and would serve as an invaluable tool in the design and operation of the transportation system. By way of concluding the discussion above, we would like to summarize the proposed three sets of guidelines intended to aid this endeavor.

A. Methodological Guidelines

Many methodological improvements in risk assessment techniques have been suggested, and many lessons have been learned in the conduct of previous assessments. In particular, we emphasize that:

- A CRA is preferred to a PRA.
- A CRA should calculate probabilities only where existing data, theories, and models are sufficient to support the use of rigorous quantitative methods.
- The use and limitations of expert judgement should be clearly indicated, and such judgment should be used only where more adequately derived estimates are impossible.
- Sensitivity analysis should be used to illustrate the impact of differing assumptions and variations in the quality of data.
- A CRA should cover all the sequences and phases of the transportation system for both defense and commercial nuclear wastes, and consider the full range of plausible technological configurations, such as new cask designs, model mix, and routing choices.
- A CRA should consider the likely risks involved in waste retrieval.
- The full range of initiating events should be evaluated, with particular attention to human and organizational factors, external initiating events, and sabotage and terrorism.
- The full spectrum of consequences should be carefully evaluated, with particular attention to "signal" events and social amplification.

B. Risk Management Guidelines

The goal of risk management is to minimize risk by selecting and implementing appropriate measure to prevent, control, or mitigate adverse consequences. Risk assessments is an inseparable past of risk management, and an invaluable aid. The guidelines

below will help to ensure that CRA promotes rather than hinders the achievement of these goals and objectives.

- A CRA should **not** be used to attempt to convince the public that the transportation HLNW is "safe."

- A CRA should be used as a risk management tool to achieve risk minimization by indicating optimal design and operational choices.

- A CRA should be developed prior to construction of the HLNW transportation system to encourage appropriate design choices and avoid potentially costly post-design changes.

- A CRA should be used interactively throughout the operatioral phase of the system, to ensure timely operational and engineering modifications and the maintenance of minimum risk levels.

- A computerized monitoring and data collection system should be developed to encourage the interactive use of the CRA during the operational phase.

- A CRA should be fully integrated with the overall risk management system.

- A CRA should be continuously updated through an interactive process of review and evaluation. A CRA should, therefore, become a "living document," and should not be a static one time analysis.

C. Risk Communication Guidelines

A CRA is not intended to be a risk communication vehicle **per se**, but it will become one by default. In light of this, and to encourage credibility the CRA should:

- be conducted by an independent body, acceptable to all stakeholders, particularly the affected states and Indian tribes;

- be conducted in an open atmosphere with considerable room for peer and public review; and,

- include an executive summary as an integral part.

The executive summary of the CRA should be widely distributed, and careful attention should be paid to the scope, content, and design of such a document. The executive summary should:

- not be used as a public relations exercise to convince the public of the safety of the system;
- be written clearly and concisely in plain English for a lay audience;
- clearly outline the major goals, limitations, and assumptions of the CRA;
- address the full range of events and spectrum of consequences; and,
- accurately portray the findings and peer review criticisms of the main report.

Risk communication is complicated and will involve a multitude of activities aside from the distribution of an executive summary. Particular attention should be paid to the process of risk communication and public oversight. A credible process to encourage public involvement should include:

- the creation of a permanent HLNW Transport Oversight Committee, comprising representatives of the populations of the various states and Indian tribes encompassed by the transport system, to monitor performance of the system and advise system operators of public concerns;
- regular publication in the local media and a facility newsletter of data describing the systems operations, e.g., volume in transit, origin points, volume received at the repository, as well as other operational and performance information;
- regular publication through the same channels of information pertaining to actions taken by management to modify the transport system in response to areas identified for upgrade or improvement; and,
- the establishment of a system of warnings and penalties wherein certain events, depending upon their severity and frequency of occurrence, a financial penalty, temporary suspension, or longer term stoppage of waste transport.

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STATE OF NEVADA

AGENCY FOR NUCLEAR PROJECTS/ NUCLEAR WASTE PROJECT OFFICE

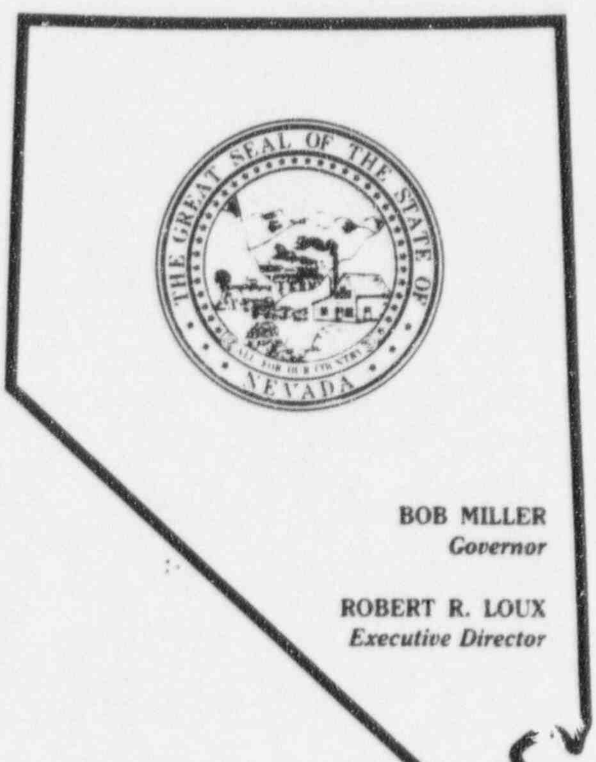
NWPO-TN-005-90

NUCLEAR WASTE SHIPPING CONTAINER RESPONSE TO SEVERE ACCIDENT CONDITIONS: A BRIEF CRITIQUE OF THE MODAL STUDY

by

Lindsay Audin
Consulting Engineer
Ossining, New York

December, 1990



BOB MILLER
Governor

ROBERT R. LOUX
Executive Director

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The Nevada Agency for Nuclear Projects/Nuclear Waste Project Office was created by the Nevada Legislature to oversee federal high-level nuclear waste activities in the state. Since 1985, it has dealt largely with the U.S. Department of Energy's siting of a high-level nuclear waste repository at Yucca Mountain in southern Nevada. As part of its oversight role, NWPO has contracted for studies designed to assess the transportation impacts of a repository.

This study was funded by DOE grant number DE-FG08-85-NV10461.

Contents

Executive Summary and General Conclusions

The Modal Study (NUREG/CR-4829) attempts to upgrade the analysis of spent nuclear fuel transportation accidents, and to verify the validity of the present regulatory scheme of cask performance standards as a means to minimize risk. While an improvement over many prior efforts in this area (such as NUREG-0170), it unfortunately fails to create a realistic simulation either of a shipping cask, the severe conditions to which it could be subjected, or the potential damage to the spent fuel cargo during an accident. There are too many deficiencies in its analysis to allow acceptance of its results for the presumed cask design, and many pending changes in new containers, cargoes and shipping patterns will limit applicability of the Modal Study to future shipments.

In essence, the Modal Study is a good start, but is too simplistic, incomplete, outdated and open to serious question to be used as the basis for any present-day environmental or risk assessment of spent fuel transportation. It needs to be redone, with peer review during its production and experimental verification of its assumptions, before it has any relevance to the shipments planned to Yucca Mountain. Finally, it must be expanded into a full risk assessment by inputing its radiological release fractions and probabilities into a valid dispersal simulation to properly determine the impact of its results.

Procedural Criticisms

The Modal Study was tasked to be an independent verification of the hypothetical accident sections of 10CRF71, the existing framework of cask performance standards. Too often its investigation paralleled or copied aspects of those rules (e.g., sequence and types of accident stresses) for it to be independent of 10CRF71's portrayal of a worst-case reality. The implications of crush and puncture of the outer shell, for example, are almost completely ignored.

The Study itself was peer reviewed after its completion by two research groups (Denver Research Institute and Los Alamos National Laboratory) which were given only a draft of the Study's text to analyze. Many assumptions and calculations were made that were not visible or verifiable by the peer reviewers or the author of this report, nor were the Study's appendices complete (e.g., no coverage whatsoever of spent fuel damage analysis). As a result, the peer reviewers only spot-checked calculations they could readily replicate, and trusted the expertise of the Study's producers for almost all other analyses. This reviewer went into further detail in areas not touched by the peer review (e.g., interactions of stresses, radiological release calculations) and raised questions that can only be resolved by dialogue with the Study's personnel.

So many assumptions and analyses are missing from the text of the Study that it is unclear where engineering judgments end and actual mistakes begin. It is therefore possible that data which appear to be erroneous are simply the results of unacceptable (and hidden) assumptions. When coupled with the Study's often unclear presentation of its methods and resources, a proper review cannot be done. It should be noted that no other in-depth critique of the Study has been performed, and the many questions raised by this report (and those of the peer reviewers) should be seen as the basis for either a more

complete investigation of the procedural details of the Modal Study, or as input to a new study of spent fuel containers and transportation accidents.

Highlights of this Critique

While there are many potential flaws, some are of greater concern than others. Accident rates, for example, are open to question because of deficiencies and applicability of available data, but such errors are less able to influence overall risk than some assumptions concerning the distribution of accident severities, or the various ways in which a cask could leak. Following are some of the problems that may have significant effect on the Study's final results.

- Cask design and accident parameters are significantly oversimplified. The use of strain as the primary variable to define damage does not reflect the results of scale model tests in which failures occurred at seals and welds, not at yield points in the main cask body. The use of mid-lead temperature during a fire conceals the potential for alloying and for cask/seal/fuel damage hours after the fire is out.
- A great deal of information was "created" to fill in missing data on the probabilities of different accident conditions. Some assumptions of random distribution may be invalid, while other judgments regarding severities are outdated or were never verified by experiment or independent expertise (e.g., likelihood and impact of high temperature fires). The absence of benchmarked tests greatly reduces the Study's credibility.
- The interactions of the stresses were not fully analyzed. For example, the presence of an empty neutron shield was found to greatly reduce heat transfer due to fire, but damage to the shield from an impact (resulting in reduction of its insulating capacity) does not appear to be covered in the impact-fire scenario. Similarly, the spent fuel itself was not included in the simulations, so damage to the fuel by a cask collapsing upon it due to sidewise impact was not analyzed. Such lack of interaction could greatly underestimate the heat available to cause damage to the outer shell (possibly leading to loss of the gamma shielding), or the potential release fraction of spent fuel nuclides.
- While admitted in the opening section of the Study, the failure to examine the impact of human error greatly limits the applicability of the analysis to the real world. Actual casks very similar to the representative container in the Study had many problems that should have been examined in the Study's simulations. Most of those deficiencies existed during numerous shipments and some applied to more than one copy of the design.

  They therefore could have been present during many of the scenarios in which the Modal Study assumed "perfect" construction and handling. Human error has proven to be the bane of the nuclear industry, so examination of only mechanical failure is a serious limitation in the Study.

- Computer simulations of cask impacts on a flat surface did not replicate a phenomenon known as "slap down" in which secondary impacts occur. Experimental results with scale models indicate that these secondary contacts (usually at different angles from the first) may experience greater strain than those of the first impact. The Modal Study's software was also inconsistent and limited in its ability to predict the degree of strain beyond a narrow range of severity. There is a strong need for experimental verification of the most vulnerable configuration of the cask at impact and the resulting strains/damage that occur.
- The treatment of spent fuel damage is too simplistic and is based on unrelated tests having little relevance. The sole basis for the release fractions due to impact is test data developed from thermal stress. As a result, the "worst-case" scenario for cladding damage amounts to only a single, 1/16-inch diameter, hole in each 15-foot long fuel rod. Release of re-oxidized fuel pellets is assumed not to occur based on tests where no oxygen was available, contrary to circumstances that would prevail in any cask release. Admitting that experimental data on spent fuel impact is lacking, the Modal Study proceeded with using nearly irrelevant information as the basis for its conclusions that very little radiation would escape the cask.
- The portrayal of the spent fuel itself was also deficient. A major isotope (americium 241) is missing from the truncated list of nuclides available after 5 years of decay, and the gamma output of the structural end parts of the assembly was disregarded. The first item could impact on the particulate hazard and the second on the exposure hazard after lead slump due to an endwise impact.
- Available data on cladding and fuel damage (both experimental and accidental) was not referenced or utilized, thereby limiting the depth of the analysis and the acceptability of the conclusions of accident consequences.

Applicability to Future Shipments

Higher fuel burnup rates, dry storage, rod consolidation and more assemblies per shipment will affect the radiological hazard of future spent fuel shipments. Solid (instead of water) neutron shields, thinner cask shells and use of uranium gamma shielding will greatly affect the response of new casks to impact and fire. The distribution between rail and road shipments may be greatly altered by rail availability at reactors and by erection of a monitored retrievable storage facility. The hazards of other materials shipped with or near spent fuel casks may affect the worst-case fire scenarios.

Taken by themselves, these factors could so alter the accident, cask and fuel characteristics that they alone would call for a new Modal Study. When combined with the deficiencies and uncertainties of the present Study, there can be little question of the need for a new, up-to-date, well-founded and properly reviewed Modal Study. The Nevada Agency for Nuclear Projects is developing a list of improvements for such a future study.

Introduction and Overview

Why a Modal Study?

The purpose of the Modal Study was to examine the validity of existing cask design and certification standards, via engineering analyses of the responses of a representative cask to transportation accidents. To clearly understand the direction (and criticisms) of the Study, some background on the procedures used in cask development is essential.

Cask Design Standards

Shipping casks for irradiated nuclear fuel are considered the primary barrier to a release of radiation in a transportation accident. A great deal of attention has therefore been focused on the design of these containers. While no design can withstand all possible accidents, federal regulations set cask standards that require containers not to leak significantly in the vast majority of accidents, including severe conditions involving fire and impact.

Those rules primarily define performance standards, i.e., the types of conditions that the package must survive. Exactly how those standards will be achieved is left to the designer. While the basic regulations have remained unchanged for over 20 years, several "guides" have been issued to formalize a common approach to meeting the standards. In addition to containing its radioactive contents during an accident, a container must maintain the ability to control criticality (i.e., avoid an accidental nuclear reaction) and limit routine emissions during transport. Finally, a cask design must accommodate the requirements of the transportation and nuclear industries in order to be commercially viable. Limitations on size, weight and internal configuration all come into play.

Cask Certification

Because of the potential for a serious health hazard if the spent nuclear fuel were to escape from the cask, much attention has been paid to proving the ability of the casks to contain radioactive materials and radiation during accidents. Due to the expense involved in destructive testing of actual casks (each costs on the order of a million dollars or more), federal regulations accept scale model or mathematical simulations of tests to verify the safety of a given design.

When a design is finalized, it is described in a "Safety Analysis Report for Packaging" (also known as a SAR or SARP), which follows a format suggested by a regulatory guide. If found acceptable, a license known as a "Certificate of Compliance" (CoC) is issued. Both documents usually include requirements for maintaining and inspecting the container at routine intervals to control its quality. Quality assurance during fabrication is handled by occasional federal inspections of the manufacturing facilities and documentation on materials and staff skills.

Risk Assessment

The basis for accepting the present cask design standards rests on the overall likelihood of fatalities due to cask leakage during spent fuel transit. This conclusion is developed by determining the probabilities of accidents sufficiently severe to release enough radiation such that, upon dispersal, there is a fatal inhalable concentration available to affect the public. The probability of accidents is assumed to be proportional to the number of shipping miles (i.e., the total number of trips x the length of the average trip). Using other statistical and analytical techniques, it is possible to calculate the chances that such accidents will occur during the likely history of spent fuel shipping. If, for example, during 50 years of shipping, only one chance in 40 of a single death is expected, it could be said that only one radiation death in 2000 years (50 x 40 = 2000) is probable. Arriving at such a number involves the multiplication of numerous figures, some very high (e.g., shipping miles) and some very low (e.g., portion of radiation releasable in harmful form). The final result, called "risk" (e.g., one death in 2000 years), is the mathematical product of the probability of an accident and its consequences.

Implicit in all such risk analyses is a grasp of the way probabilities and consequences are calculated. Virtually all studies equate the accuracy of the methodologies involved in quantifying these two factors, as though it were a given fact, regardless of the uncertainties and differences in methods. Any "gray" areas are resolved via "conservative" assumptions, i.e., that the worst case will occur, so that minor methodological errors are avoided in reaching the final conclusion. Such "gray" areas include the validity of accident rate and severity data, and the response of the fuel rods to heat and shock. Defining that credible worst case is, in itself, an uncertain task involving numerous other assumptions.

To appreciate and simplify the difficulties involved, it is often best to look only at the range, in factors of 10 (called "orders of magnitude") that uncertainties could yield in a given area. For example, truck accident rates vary from state to state and even route to route, but the data (from the best to the worst routes) may vary only by a factor of 5 (the worst case is "only" 5 times worse than the best). Thus, one could say there is an uncertainty on the order of .7 orders of magnitude (i.e., 10 to the .7 power is 5). Underreporting of severe accidents has been found (in the DOT accident base) to be as high as 90%[1], so only one out of 10 severe accidents may be listed. That yields another order of magnitude of uncertainty. Since orders of magnitude can be added, a range of 1.7 orders of magnitude is the maximum range across which reasonable people should differ in severe accident rates. By comparison, the portion of fuel released in an accident could vary over several orders of magnitude, depending on the scenario involved. Normalizing all factors into such ranges of uncertainty gives perspective to other variables, as well.

The Modal Study: Purposes and Methods

The lack of applicable full-scale testing has led to criticism of the basic standards as being only theoretical, and meeting them as insufficient to prove safety. To answer these questions, several studies have been performed to better assess the capabilities of containers that meet those criteria. The most recent attempt is NUREG/CR-4829, "Shipping Container Response to Severe Highway and Railway Accident Conditions," also

known as the "Modal Study." Under commission to the NRC, Lawrence Livermore National Laboratory (LLNL), a federally-sponsored facility in California, sought in 1986 to determine two basic characteristics involved in this issue:

- the distribution of accident severities, and
- the response of spent fuel and casks to those conditions.

To do this, LLNL had to "fill in" a great deal of missing information on accidents by using statistical techniques and engineering judgments to "create" a more complete data base. Since no experimental work was to be performed, a simplified cask design was also created to be used as input to various computer simulations of impact and fire. Finally, numerous simplifying assumptions were made to focus the study on the adequacy of existing standards. For example, LLNL found it necessary to assume that its cask was manufactured, maintained and loaded exactly as outlined in its design specification, as is assumed by federal regulations but not always realized in the field.

It was also necessary to restrict the number of critical variables to be examined when characterizing the severity of an accident. While the regulations discuss the height from which a cask was dropped onto a theoretically unmovable surface, for example, LLNL did not find this variable to be useful in determining what accident conditions would yield equivalent damage. It was concluded that strain on the inner shell (an engineering concept that describes the degree of stretching or denting) and temperature at the mid-point of the gamma shielding would be used instead. Cutoff points for these variables were then determined, beyond which it was assumed that the cask would release some of its contents to the environment. The representative cask would then be subjected to the various conditions and analyzed to determine the type and severity of accident necessary to attain or exceed these cutoff points.

By combining the results of its findings on the likely distribution of accident severities with the cask responses to such conditions, it was then possible for LLNL to create a matrix of data that correlated the probability of a set of accident conditions and the radiation releases that would result. These correlations were then compared to similar data developed in a 1977 study, NUREG-0170, also called the "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes." That study was the basis of NRC's conclusion that overall risk (that is, probability multiplied by consequences) of shipping spent fuel was low enough to require no changes to cask design regulations. If the LLNL work yielded comparable results, then the conclusions of NUREG-0170 could be considered reaffirmed.

While not utilizing the same input data or output framework, LLNL translated its findings into a form similar to those of NUREG-0170, and concluded that the overall risk was even less than previously believed; thus, the NRC rules remained acceptable.

Potential Shortcomings

Critics have examined the LLNL work and found deficiencies, some minor and some potentially serious. They range from the validity of the data input to the simulations, to the description of the accidents and the responses of the cask and fuel rods to heat and shock.

It should be understood that the task undertaken by LLNL was, in some ways, herculean: lacking data to analyze, or funds to perform tests, it developed a framework for furthering the analysis of hazardous materials shipments. For that work alone it is to be commended. It is hoped that the deficiencies noted in this report can be ameliorated, and the LLNL assumptions now outdated by changes in cask design can be corrected, so that the debate over cask safety may eventually result in an assessment methodology acceptable to all concerned parties.

Critiques and Questions Concerning Potential Deficiencies

Primary Sources of Criticism

The view of three groups of critics are reflected in this critique:

- the two peer reviewers: Denver Research Institute (DRI) and Los Alamos National Laboratory (LANL)
- The Western Interstate Energy Board reviewers
- the author and reviewers of this critique.

An attempt has been made to incorporate all of the major questions raised by the above, in many cases consolidating them to fit the framework of this report. Since many duplications occurred, no indicator of authorship is given unless important to the credibility of the question.

How this Critique Was Performed

A comprehensive review of the relevant literature was conducted, including a careful reading of the Modal Study, previous risk analyses, the peer reviews and related materials. In addition, a Freedom of Information Act request was filed with the NRC on all correspondence and contracts between the peer reviewers and NRC. It is noteworthy that the overview document to the Modal Study (entitled "Transporting Spent Fuel," NUREG/BR-0111) indicated that all such documentation existed in the NRC's public document room but, even two years after completion of the Study, no effort had been made to allow public access to the peer reviews, nor had anyone sought such access. References cited in the Modal Study and the peer review documents were also obtained and incorporated into the investigation.

The simplifying assumptions and calculations (along with uncertainties due to items not considered by the Modal Study) were then evaluated. Recent changes in cask design, payload and neutron shielding, plus past errors in manufacturing, cask loading and handling were all examined for their potential impact on the LLNL analysis.

Finally, the major and minor questions and perceived deficiencies were sorted into groups to facilitate production of a review document. A draft was produced and reviewed for completeness and clarity prior to offering it to other UNLV consultants for comment. Many minor criticisms were deleted at this last stage to highlight the most important questions. It should be understood that the appendices of the Modal Study do not offer sufficient data in many areas to allow proper evaluation, and actual interview of the Study

staff appears to be the only sure method to ascertain the many hidden assumptions that were apparently made to arrive at some methods and conclusions.

Examination of the Peer Review Process

Two research organizations examined the Modal Study draft report: Denver Research Institute (DRI) and the Los Alamos National Laboratory (LANL). DRI is affiliated with the University of Denver and LANL is a DOE-sponsored facility specializing in nuclear weapons design and development. Neither has any involvement in shipping commercial high level nuclear waste or in the utilization of spent fuel casks. Unlike a member of the nuclear industry, neither had a vested interest in supporting the results of the Study. LANL was a subcontractor to DRI to review the computer simulations used in the structural analyses of the Study (primarily Appendix E). The choice of DRI was quite simple: it had expressed interest in past NRC requests for proposals (RFP), and the only other agency considered was the University of Washington at St. Louis (UW). NRC files show only one response to its RFP, that being from DRI, though another from UW is mentioned in other correspondence. An NRC panel made the choice and there was apparently no requirement for competitive bidding or other rules for selecting a peer review contractor.

While DRI's comments were candid and often critical, it was obvious that it focused primarily on the mechanical aspects of the analysis. Much less attention was given to the sections dealing with probability, accident scenarios and spent fuel responses. This is understandable since the interests and experiences of the DRI personnel (based on the resumes and published papers listed in their proposal) were almost entirely related to mechanics and ballistics, and not transportation or radiation. Any analysis of the peer review must also keep in mind that it examined a somewhat different document than was published. A point-by-point comparison between the final edition and the peer review found, however, that many minor problems cited by it were corrected. Major criticisms, especially by LANL, were either not accepted or else were handled by editing and the addition of text. Most of the fundamental disagreements remain, and this review focuses primarily on them.

Much criticism was leveled at the format and order of presentation, which the principal reviewer (Myron Plooster, a physicist) described as "obscure and difficult to follow."[2] Perhaps most disturbing was the large number of numerical errors, some of which were typographical but many may have been calculational. Plooster states: "it is a certainty we have not found them all. We were still finding numerical data errors in the last week of this review effort." An extensive letter preceded the review report and analyzed an "apparent anomaly in the frequency distribution of thermal damage to truck casks."[3] While he felt that the error would not have a major effect on the overall risk, it did reveal that the calculations may not be entirely reliable. LLNL's response did not specifically acknowledge that anomaly, but instead agreed that there were "input errors" to the thermal analysis simulation.[4]

Several items stand out from the review that demonstrate its lack of depth, which appears to be as much related to the small size of DRI's grant as to the limited relevant experience of the reviewers.

DRI apparently was given only the text of the draft report. No actual calculations, simulation inputs or list of assumptions were examined, beyond what is found in the text. While discussing the analytical (versus experimental) approach taken by LLNL, Plooster states:

> "The analytical approach has the disadvantage that the reader cannot follow the detailed path between input and output, because of the number of complex computations connecting any input datum with the final results. This approach requires an implicit trust, on the part of the reader, in the quality of the programs used. Having said all this, some definitive experiments, or reference to such experiments, would greatly enhance the credibility of the work."[2]
>
> In its comparison review to DRI's paper, LANL echoed this view: "The credibility of the structural response calculations supporting this work can be improved [by]...benchmarking calculations against actual experiments. (Saying that Sandia used a code similar to NIKE-2D to calculate the response of full-scale casks used in crash tests is a rather weak substitute for benchmark calculation!)"[5]

In effect, the reviewers examined nothing more than what LLNL chose to put into its report, and had to make of that what they could. While there are no laws governing peer review, it is not unusual in other professions to examine the full line of researcher's calculations and all of the assumptions, not just those considered worth mentioning.

Many of the more mundane criticisms mentioned by others were also raised by DRI (e.g., American Petroleum Institute (API) accident data, applicability of California highway characteristics), but most of its focus was on the mechanical engineering considerations and how other aspects of the analysis affected them. When examining severe accident scenarios, it felt that a sidewise impact of a truck cask on an abutment or concrete column should have been investigated since "the impact force would be concentrated on only the central portion of the cask, and the ends of the cask could 'wrap around' the structure. In such an impact, bending stresses severe enough to cause tensile failure and rupture of the cask might be achieved." While acknowledging its low probability, DRI felt it was a "plausible accident with the potential for a major radiological hazard." It also expressed concern about the likelihood of such severe accidents:

> "The inhomogeneity and incompleteness of accident data bases makes this the greatest source of uncertainty in this study, in our opinion. The most severe accidents, the only ones with the potential for serious risk to the public, are out in the 'tails' of the probability distributions, where statistical uncertainties are greatest."[2]

And because of its limited examination capacity, "it is not possible to verify any of [the probability analysis] independently; one has only subjective judgment to rely on in evaluating the results." This type of uncertainty permeates the report, leading Plooster to state at one point that "the more closely one reads this report, the harder it is to follow." And after trying to correct the report's many numerical typographical errors, he finally concludes that "we do not have enough information to verify [the Study's] numerical data."

Other problems with the completeness of the analysis were raised (e.g., train sill impacts were always side-on only, always perpendicular to the cask axis), but DRI stopped short of venturing an opinion on the importance of the problems it perceived. Instead, it said "the report as it now stands needs to go through a major quality control process... Every number in this report needs to be checked against its original source... Any scientific journal or publisher receiving a document in this condition would have rejected it out of hand." The public record on the Modal Study indicates that no such detailed followup was ever performed. Nevertheless, with its admittedly limited perspective, DRI felt that "from a technical standpoint, the report is basically sound. No flaws have been found which cast any significant doubt on the major conclusions."

This tune changed later, however, after DRI received the LANL companion report, which also focused mainly on the structural analysis. DRI's lack of knowledge on real cask behavior became obvious in its cover letter to NRC:

> "If the Battelle and Los Alamos scale model experiments [covered in the LANL report] are correct in showing that closure and weldment failure are the most probable structural failure modes, then the foundation of the Livermore analysis, and the use of strain as the response variable, is in question."[6]

The LANL report used two references[7,8] to demonstrate its point that results of experimental tests on cask failure disagreed in some ways from LLNL's theoretical conclusions:

> "In these tests, failure (leakage) was never caused by excessive strain in the parent material but rather at welds or because of excessive deformations at seals."[5]

Both test series in LANL's references used carefully designed and fabricated scale models in 30-foot and 40-inch drop tests, as per 10CFR71. LANL felt that LLNL's use of "conservative" material properties did not address either source of failure.

Perhaps the most telling comment by LANL concerned the pervasiveness of LLNL's choice of strain as the operant structural parameter. LANL points out how many aspects of the Modal Study are touched by that (possibly erroneous) assumption:

> "The difference [between closure failure and maximum plastic strain] can be significant in picking a generic cask since closure failure may be more dependent on peak impact force. Peak impact force would be larger for the 'harder' shielding materials, such as uranium. "The point is that once this choice [of maximum plastic strain] is made in Section 2, the remaining results are totally influenced by it."[5]

LANL also questioned the validity of some of the impact simulations (e.g., IMPASC and NIKE) and, in several instances, reacted negatively to the Modal Study's claims of benchmarking. Aside from the previously mentioned attempt to cite a Sandia analysis that used a different computer code, LANL points out that:

> "IMPASC overpredicted the endwise impact calculation for a truck cask from NIKE by 17%, yet underpredicted the rail cask response by 20%."[5]

Typically, a simulation system will do one or the other, but doing both reveals inconsistencies either in the program or its input data. LANL also felt that the comparison between the strain determined in a physical test and that found from the equivalent damage method was "so poor that some explanation is needed." LANL concluded that "if these results are correct, the equivalent damage technique does not appear to be the best method to use for estimating the effects of impacting real surfaces." That method served as the basis for eliminating some scenarios and reducing the probability of others because their speeds were (from LLNL's perspective) unrealistically high. Replacement with a different technique could yield different results than those portrayed in the Modal Study.

LANL refers to a Battelle Columbus Laboratory study (BMI-2039) of lead-shielded scale models in drop tests, as both a resource for possible benchmarking and as evidence that closure and weldments, not excessive strain, are the most likely source of failure. A second reference discussed uranium-shielded scale models in drop tests. Both studies were obtained and found to strongly support LANL's assertions about weld and closure failures. Each also covered the problems inherent in modeling drops on corners: such tests usually involved secondary impacts, none of which were analyzed in the Modal Study. As discussed in more detail later in this critique, there is a phenomenon (sometimes called "slap down") in which kinetic energy is transferred from the cask end that first strikes a surface over to the opposite end, which is then accelerated as it revolves around the first contact point, or its center of gravity. The references indicate that bending stresses resulted and that the measured strain in the second impact (at the opposite end) exceeded that of the first. This action complicates the modeling of impacts used by LLNL.

While questioning its basic methodology, LANL stopped short of attacking the Modal Study's general conclusions, however, by stating:

> "In general, the reviewers believe that the overall probability conclusions from the study will not be changed significantly by any issues raised in this review, but do believe that the supporting analyses can be stronger."

LLNL'S Responses to the Peer Review Process

Many of LLNL's responses to the points raised by DRI and LANL were either not direct, or else were not substantiated. When it had no good answer, it agreed that more work was essential to settle the problem but blamed "budget and schedule constraints." Many responses began with "we believe," or claimed "conservatism" covered the situation, or else were simply statements that had no more backup than was found in the text. LLNL denied that the tests used in BMI-2039 could be used for benchmarking and provided a reference of its own to point out the difficulty in comparing computer output with test data, which it indicated was often quite inconsistent. This reference (LASL-3306) does support LLNL's view of the problem but likewise discusses the "slap down" phenomenon. It concludes that "none of the [analytical] methods could be substantiated by dynamic measurements made in experiments... most of the methods seemed to be inadequate for the goal of experimental substantiation."[9]

It also pointed out the complexity inherent in determining the most vulnerable drop orientation when secondary impacts were possible. LLNL appears to use this reference to

tell LANL that "your way is no better than mine, so therefore my way must be acceptable." While LASL-3306 does criticize past experimental efforts, it definitely does not support the Modal Study's present theoretical framework.

In essence, these "dueling references" support the notion that much more experimental testing is needed to obtain a realistic perspective on the highly theoretical analyses used in the Modal Study. This discussion indicates that the state of knowledge (at least at the time of the Modal Study) necessary to demonstrate cask safety may be deficient.

LLNL appeared to denigrate the views of LANL's references regarding failures at seals and welds. Without citing any basis for its view, LLNL repeated its assumption that "the seal will not fail at stresses less than yield," though that is apparently what occurred in Battelle's tests. It also stated that "ideally, weld joints should not be present in these areas [near the end closure] where high local strains can occur," though there is no NRC regulation or guideline covering that issue. Focusing again only on the inner containment, it concedes the possibility of local cracking but says "it is not likely that the inner containment will completely rupture." No analysis was made of the potential for lead loss through such failures during a fire. LLNL pointed out that the scale model tests cited in BMI-2039 by LANL were not licensed casks and that a "cask design that results in a 1% lead slump for a 15 foot drop would likely not be licensed or permitted to transport spent fuel." This reveals an interesting aspect of LLNL's response. While LANL had said that its "reference 1" (i.e., BMI-2039) discussed such a lead slump test, it did not say that any of the tested scale models in that study experienced such a result. Examination of BMI-2039 shows no mention of a 15 foot drop test. Apparently LANL's commentary meant to cite a reference in BMI-2039, since lead slump was a topic of that reference. It is unclear how LLNL could have misunderstood LANL, unless it never actually examined BMI-2039. LLNL summarized its responses by re-citing the peer reviewers' comments that their criticisms did not substantially question LLNL's major conclusions.

The reader is left with an uneasiness about the subjective manner in which analytical disagreements were seemingly settled. No calculations were mentioned or shown, and no sensitivity studies were performed to assess the impact of possible deficiencies. Normal peer review processes, such as for a professional journal, are much more rigorous. By contrast, this process involved only an examination of a completed text, without its backup calculations, by agencies with either very limited experience or a narrow focus. It would be difficult to characterize the results as having any significant depth, or to accept them as sufficient confirmation of the Modal Study's credibility.

Discussions of the Deficiencies

Four categories of problems were found:

1. data creation and analysis
2. cask design and response assessments
3. characterization of accident scenarios
4. assumptions regarding spent fuel and its response.

The remainder of this report focuses on these issues.

1. Data Creation and Analysis

Accident Rate Data

The starting point for all accident analyses is an accident rate, usually expressed as a number of accidents per million miles of shipments. The Modal Study used an accident rate from the American Petroleum Institute (API) ostensibly because it covered shipments in containers of size and weight similar to spent fuel casks. The API data was also "judged to be more reliable" than data from the Bureau of Motor Carrier Safety, though no basis was given for such judgment. Rail accident data was taken from the Federal Railway Administration (FRA). Roadway conditions that are related to the types and severity of accidents were developed using California highway characteristics.

Several comments by critics were made on this data, the most cogent of which related to the API and FRA information. While the distribution of physical characteristics along California highways may differ somewhat from the rest of the country, it is unlikely to have a serious effect on the distribution of accident types since, on average, the incidence of grade crossings, etc. was found to be about the same as the average mile of national highways. On the other hand, the typical petroleum shipment (usually gasoline or fuel oil) is quite short (28 miles)[10] and occurs in an urban or suburban area. This makes sense because most petroleum products are moved long distances by pipelines or railroads, not motor vehicles. The accidents involved are therefore likely to be at lower speeds and on local roads. Reportage to API is also voluntary: "inputs are what member companies choose to report," according to DRI. LLNL responded by stating that it believed, because a hazardous material was involved, that reporting was better than other data bases and that travel on non-interstates would yield a conservative accident rate. While this latter fact may be true, the former is not. The United States Department of Transportation (DOT) maintains a system which requires, by law, the reporting of accidents involving vehicles carrying hazardous materials. Careful checking by both critics and federal analysts found that this data base was missing up to 90% of all such accidents, and perhaps 70% of the most serious cases.[11] The truck accident rate could then be low by nearly an order of magnitude (i.e., a factor of 10).

Rail data does not fare much better. A recent study by the U.S. General Accounting Office (GAO) found that "FRA has little assurance that its injury and accident data base is reliable because the railroads GAO visited were not reporting accurately or completely."[12] The degree of error for railroad data was less than that found for highway, but the sampling was limited, so the results could not be used as a correction factor.

By itself, this one source of error does not suffice to cast serious doubt on the results of the Modal Study. It does, however, reveal a naivete about the realities of the shipping world.

Distributions of Accident Severities

To develop the spectrum of accidents involving impact and fire, two data bases were used: impact data came from state and federal agencies' information on actual accidents, while fire data came from a previous analysis that "created" information by statistical techniques and judgments. The two data were mixed together by assuming a random

distribution of fires within the types of impacts that could occur. The lack of real information precludes use of a better data base, but the results should be tested against some other smaller body of data where both impact and fire are known, in order to see if the distribution bears any resemblance to real circumstances. Such benchmarking would add credibility to the Study's assumed distribution of severe accidents. While such data may not be readily available from domestic sources (but may be available from other Western countries), there is no indication that LLNL made any attempt to verify its combination of thermal and impact data.

An example of the potential error that can result from an automatic assumption of randomness is as follows. Since highway routes for spent fuel shipments will, according to a study by the National Academy of Sciences[13], funnel down to a few major corridors, the likelihood of a truck fire involving another vehicle would be affected if that same corridor was commonly used as a prime route for flammable materials. An assumption of a random distribution of highway accidents involving fire would not be sensitive to route funnelling, but benchmarking against actual data on those corridors might reveal if the likelihood of fire was greater than the national average. It should be noted that past spent fuel shipments utilized only a small portion of the national highway network, but independent analysis of DOT data[11] found that nearly half the hazardous material accidents (most involving flammables) occurred on those spent fuel routes, probably because those routes link numerous chemical plants using those materials. It would also not be surprising to learn that truckers who routinely exceed the speed limit do not report that fact, especially if involved in an accident, or that long fires are not randomly distributed among the various collision speeds - but the Modal Study's randomized data appears to ignore those possibilities. Benchmarking could reveal such possible methodological errors.

LLNL did attempt a benchmark of sorts by comparing the results of four recent severe accidents with its own scenario analyses. It is interesting to note that all four occurred between the times NUREG-0170 and the Modal Study were performed and were, in some cases, worse than those previously considered the worst likely to occur. Is it possible that larger vehicles, more hazardous cargo, deregulation, etc., are creating more opportunities for severe accidents? The Modal Study implies, because its work was not contradicted by this small sampling of reality, that the casks are safe. The Study does not, however, consider other real hazards, such as stationary fuel or chemical tanks, that could yield much more serious consequences than those modeled by LLNL. Just such an accident occurred recently in Ohio when a burning butane tanker started a fire in a chemical plant near the railroad tracks[14]. And just as accidents of greater severity occurred after NUREG-0170, worse accidents have occurred since the Modal Study. A train derailment in the United States led to damage to an underground gasoline pipeline adjacent to the railroad. The pipeline later exploded, fortunately not while a train was passing[15]. Such a pipeline could provide an immense supply of thermal energy, leading to a very large fire of long duration. Only a few weeks later, this idea was proven when a leaking natural gas line in the Soviet Union exploded as a train passing nearby ignited the fumes, creating probably the worst rail fire in history[16]. The co-location of rail lines and pipe lines is not random: many use the same rights-of-way. Such real world considerations are absent from the random distributions in the Modal Study's fire analyses.

LLNL's methods of data creation for details of accidents, such as the distribution of impact angles, distances from a fire, etc., cannot be compared to real accidents, however. Actual data is clearly lacking and benchmarking would not be possible. On the other hand, an assumption of random distribution of impact angles does not reflect any effort to model the effects of tiedowns or other factors that could, in reality, skew the data. Proper modeling of this distribution could be important since experimental data indicates that impact at some angles (other than at a right angle) may be more likely to yield puncture of the outer shell of a container.

It is essential that a sensitivity analysis be performed to assess the order of magnitude implications of such assumptions. At the very least, a comparison with other studies that utilized different methods when examining spent fuel accidents (in some detail, unlike NUREG-0170) could shed light on the possible limits of the Modal Study's data creation procedures. Unfortunately, LLNL did not do so.

Quantification of Consequences

The data covering the choice and quantity of isotopes is also open to some question. Table 8-1 lists "only the specific isotopes that are important in performing a radioactive release evaluation." Comparison to other studies of spent fuel accident consequences[17] indicates that LLNL truncated a much longer list, but no criteria are given for its choices. While most of the missing isotopes are present in only small quantities or are not as dangerous as those in LLNL's list, americium-241, a daughter product of plutonium-241, exists in significant quantities and is as lethal as any plutonium isotope. Americium-241 would be of particular concern in shipments of older, high-burnup PWR fuel. Also missing is the cobalt-60 residing in the metal frames that hold the fuel rods. While not involved in a release of material, it would provide direct exposure in areas of lead slump. The absence of these isotopes could seriously underestimate hazard. These deficiencies are discussed further in the section on spent fuel and its responses.

The use of curies in various figures in chapters 8 and 9 conceals the hazard involved because it does not reflect the danger of a curie, which will vary from one isotope to another. Much of the danger comes from particles of plutonium, yet in the Modal Study's case, no more than 7.22×10^{-2} curies will be released, a number that may seem very small to the lay reader. Use of other units instead of curies is essential to yield data comparable to other studies, and to give a clearer picture of the possible hazards involved. Use of curies in a radiation spill is about as clear to the lay public (and many emergency response personnel) as would be the use of moles (i.e., gram-molecular weights) to describe the release of a poisonous quantity of chlorine gas.

2. Cask Design and Response Assessments

The Modal Study assumed a lead-lined cask with steel inner and outer shells, surrounded by a water neutron shield. In a preliminary analysis, LLNL concluded that this configuration was the most vulnerable to impact and fire. Further study would then be automatically conservative. A second level of analysis developed a more detailed version of the cask, using materials and dimensions nearly identical to the NAC-1 container, a truck cask designed for shipping one PWR or two BWR fuel assemblies This similarity is ironic in light of the history of the actual NAC-1 casks. The reader is directed to "A Review of

the Effects of Human Error on the Risks Involved in Spent Fuel Transportation," prepared for the Nebraska Energy Office in 1987, for background in this area.

Description of the Representative Cask and its Materials

Numerous simplifying assumptions accompanied the cask analysis, several of which are best understood by a brief description of the cask materials and their responses to heat and force. Readers with a working knowledge of cask design may skip this discussion and proceed to "Problems with the Cask Simulation."

The Neutron Shield

The water neutron shield assumed by LLNL has been the most common type used on casks in the past. The most important aspect of it in the Modal Study, however, is its absence. The purpose of the water is to absorb neutrons during routine handling and use of the cask. Loss of the water may increase the level of escaping neutron radiation by a factor of 20 or more, but the final result is not considered by the NRC to yield a significant health impact. 10CFR71 allows such increases in the case of an accident. To their credit, the engineers at LLNL created a reasonably accurate thermal simulation of a water neutron shield after its water was lost, a likely event when heat from a fire causes the water to expand and/or boil, opening a pressure relief valve. The result is a dead air space between the outside skin of the cask and the cask's outer shell. That space would act like an insulator, much like the evacuated region in a thermos bottle. LLNL appears to have assumed that this insulating property remains intact, even after impact with other objects.

The Steel Shells

LLNL adopted two variables to describe the response of a container: strain on the inner steel shell, and the temperature at the mid-thickness of the lead shielding. While temperature is a commonly understood measurement, strain is not. It its simplest form, strain is the degree of elongation of a material prior to its failure. Many metals (steel included) will stretch when subjected to sufficient force, and generally do so in three steps. To grasp this phenomenon, consider a coil spring. Pulling on it yields an increase in length, and releasing it results in restoration of its original shape. This type of behavior is called elastic strain. Various types of steel can be stretched about .2% and still remain elastic. Now imagine pulling so hard on the spring that it began to lose its coiling, remaining stretched out of shape. That is analogous to plastic strain in steel: from .2% to 2%, it takes a gradually increasing amount of force to yield a permanent deformation. In the final stage, pulling a little harder will yield much greater elongation, between 20% and 30% beyond the original length, followed by cracking and breaking of the steel[18]. This second stage of plastic strain is one of the most valuable characteristics of steel: even under large forces, it does not break, but rather dents or stretches significantly, maintaining much of its strength until it has become quite distorted. Personal experience in a metals laboratory gave this writer the impression that some steels act like a "super taffy," when pulled sufficiently to cause plastic strain. Machining, cooling, alloying or repeatedly stressing steel can reduce this plasticity, however, so assuming its presence requires

detailed knowledge of the mechanical history of the metal and the conditions existing when the force is applied.

Lead Shielding

Lead also has unique properties when subjected to severe conditions. Due to its high density and softness, it can act like baking dough when it is dropped: it will slump in shape under its own weight, spreading out in all directions. It also melts at a relatively low temperature (about 620°F), soaking up a great deal of heat at high temperatures, instead of merely conducting it. The lead then acts like a temporary insulator due to these (and other) properties. Lead also expands when it melts, and it was not unusual for lead-shielded casks to have empty space available to allow expansion, thereby avoiding pressurizing the cavity between the inner and outer cask shells. Other shielding materials (such as depleted uranium) are harder and do not readily melt, and they react quite differently to heat and shock.

Penetration Sub-Systems

Several penetrations through the ends and/or shells of a cask are common in a spent fuel cask. Most are essential to allow draining of spent fuel pool water when the cask has been loaded. It is typical for drain and vent valves to be installed by drilling into the shells and welding tubing and/or valves into place. In the past, a pressure relief valve was also included to relieve a water-filled cask pressurized by a fire. Such valves will probably be unnecessary for casks filled with inert gases and only residual amounts of water. The bottom end of a cask may be attached by welding it after machining its mating surfaces and drilling holes for alignment pins (attached to the inner and/or outer shells). Finally, lifting trunnions (stubs near the ends of a cask) may also be attached by cutting into the outer shell and welding into place. None of these penetrations were modeled by the Modal Study analysis.

Impact Limiters

At both ends of the cask, relatively soft shock absorbers, called impact limiters, are attached (usually by bolts into the cask lid and its bottom). Made of crushable wood, honeycombed aluminum or similar materials, they are designed to reduce the deceleration of a cask prior to impact. They protect the ends of the cask but offer no protection from sidewise impacts.

The Cask Seal

Finally, there are seals and bolts at the mating surface of the cask lid and its body. The seals are often a flexible elastomeric material that assumes the shape of the channels cut into the lid and body, much like the rubber seal at the top of a thermos bottle. While capable of maintaining their seal up to about 500°F, these materials break down at higher temperatures. Some metal seals can withstand temperatures in the range of 1000°F, but may require replacement with each use and are not favored due to this increase in maintenance. The Modal Study assumed seal failure in the 500°F to 600°F range.

Problems with the Cask Simulation

There are at least three basic problems with the Modal Study cask analysis:

1. its portrayal is seriously outdated by changes in cask design and payload
2. it does not reflect details of construction that create areas and points of vulnerability
3. it fails to account for human errors in cask fabrication, loading and maintenance that could easily compromise cask integrity.

This section will examine the impact of these deficiencies as they affect each of the previously discussed structural parts of the cask. The reader should note how a deficiency in one area creates conditions not examined in other areas. There are major synergistic effects inherent in the problems, and the Modal Study failed to model them, thereby greatly oversimplifying many aspects of its accident simulations.

The Neutron Shield

Almost all new cask designs (and all those proposed so far by the DOE) utilize solid neutron shields on the exterior of the cask outer shell, as versus older designs that used circumferential water tanks. Some are composed of organic materials high in hydrogen content (the key ingredient to shield neutron radiation) that, while not flammable, may vaporize or break down at high temperatures. They are not designed to resist either heat or impact, but instead exist as a means to keep routine emissions at regulatory levels. It therefore cannot be assumed that their presence will have any mitigating effect on heat transfer. This is important because the dead air space left by the empty water neutron shield modeled by LLNL cuts the heat transfer rate into the cask by over 70% (see pp. 6-33 to 6-39 of the Study). Since the time to reach lead melt is roughly proportional to this rate, it is possible that lead melt for a truck cask could be reached in about 20 minutes instead of 1.08 hours (calculated by LLNL), if the dead air space was lost. Seal failure and fuel rod damage may then occur earlier, as well. In realistic terms, this means that a smaller amount of flammable material is needed in an accident to melt the lead and yield high cask temperatures. This could increase the probability of an accident with severe consequences.

Another aspect of LLNL's treatment of the neutron shield regards its capacity to retain its shape when the cask is struck by an object or the cask strikes a flat surface. The outer layer of the shield may be punctured, torn or flattened, contacting the outer shell of the structural part of the cask. Any remaining dead air space on that side will be further reduced if the cask rolls or contacts other obstructions since there is only minor structural support for the neutron shield. The NAC-1 cask, for example, utilized heat transfer fins connecting the outer shell of the cask with the outer layer of the neutron shield. Such fins were designed to conduct heat away from the fuel if it was hot (typical of fuel only recently removed from a reactor), but would also act to conduct heat from a fire into the cask, thereby negating some of the insulating effect of the dead air space.

It should also be noted that, while only a thin dead air space will provide insulation, any puncture of the neutron shield will allow entry of hot gases into the empty shield, thereby nearly eliminating the shield's insulating capacity. While only a portion of the insulating space may be lost when the shield collapses, that would cause very uneven

heating and expansion of the lead, not necessarily near the volume allowed for such expansion. Similarly, impact by a train sill or other hard object could rip away part of the outer layer of the neutron shield, once again creating a pathway for increased (and very uneven) heat transfer. When LLNL examined lead expansion in general, it found that it would yield slight warpage of the inner and outer shells but not enough to create major strain. This may be acceptable when it is assumed that the lead melts evenly throughout the cask, but not necessarily in cases when the expansion is local. Any localized weakness in the shells due to welds, penetrations, etc. could be affected by this local expansion.

An example of this phenomenon occurred during the 1978 Sandia fire test. An outer shell cracked in two places, creating a path for loss of molten lead. The lead had expanded and pressurized the gamma shielding cavity because the manufacturer had failed to drill holes into an adjacent empty space designed to allow for expansion. The shell was locally weakened by a series of welds that used welding rods contaminated with minute amounts of copper[9]. This actual response is a good example of the potential clashes between reality and the Modal Study.

The Outer Cask Shell

The regulatory stresses outlined in 10CFR71 include a drop onto a flat, unyielding surface followed by a second drop onto a steel stump. The Modal Study examined the strain that would result from such theoretical encounters and attempted to find "real" structures that would yield equivalent damage. Impact onto a real flat surface (earth, rock, etc.) could provide similar strain, depending on the impact velocity and the hardness of the surface. The Model Study concluded that most surfaces were too soft to cause such damage, unless the cask were moving at an unrealistically high speed. Rock and reinforced concrete surfaces would, at a realistic speed, provide similar strain.

LLNL also briefly considered a sidewise impact with a bridge abutment or similar structure. It concluded that the chance of such a contact was remote (compared to more likely collision scenarios) and, in its response to a criticism on this issue from LANL, stated that "this type of impact would be similar to that calculated for an impact with a train sill"[4] [i.e., the front of a locomotive chassis]. Absent a confirming calculated analysis or simulation, this opinion is not acceptable. A train sill impact involves contact over a very small area with an object having limited kinetic energy. A bridge abutment is essentially an unyielding column. Impact with it would yield a great deal of lead movement and bending stresses (as the cask ends continued to move while the center of the cask rapidly slowed down) not encountered in the case of the train sill. LLNL should have developed an analysis to show what speed was necessary to yield unacceptable strains due to bending and/or lead movement. The movement of lead also raises the question (even if the cask remained intact) of the gamma output at the point of contact with the column when the cask came to rest.

The Modal Study's analysis of the outer cask shell assumes that, at all points on its surface, it maintains its ability to yield to strain without breaking. As previously mentioned, real casks have numerous welds that may be weaker than pure steel. In addition, poor manufacturing techniques have provided other sources of cask vulnerability. One of the NAC-1 casks had a problem with uneven shielding, so copper plating was

welded to the outside of the outer shell to increase shielding[20]. Not only would such surface welding affect the properties of the shell, but the presence of copper in contact with steel could create, at high temperatures, a melting eutectic point where copper would alloy with the steel, seriously weakening it. This modification only came to light after years of cask usage.

The strain from the drop onto a steel stump could not be duplicated (in LLNL's opinion) by a "real" situation. Only two "real" puncture scenarios were examined: a high speed perpendicular sidewise impact by the end of an I-beam, and collision with a train sill. In both cases, the Modal Study concluded that the shell and shielding would be severely dented but not penetrated. Based on these results, no further investigation of outer shell penetration was indicated. Thus, there was never any analysis of lead shielding loss due to a puncture followed by a fire.

While the basic simulation of striking a flat surface may be acceptable, the puncture study is not. The I-beam impact, for example, was limited to a beam whose height equaled the diameter of the cask. The original concept of the short fall onto the stump was to replicate a cask falling from its trailer (or flatcar) onto a railroad track (a form of I-beam)[21]. The impact would then involve contact with the top side of a much smaller I-beam, not its end, and over a considerably different surface area. The train sill simulation was also open to question. While several impacts were simulated, all were perpendicular to the length axis of the cask: one was in the same plane as that axis, but the others were above that plane to varying degrees, providing glancing blows that would tend to rotate the cask around its length axis. The fact that neither simulation examined impact angles other than those perpendicular to the length axis is important: drop tests on steel stumps have found that the angle of greatest damage is not necessarily 90°. At that angle, strain around the circumference of the stump involves stretching an amount that is nearly the same at all points. At lesser angles, the strain is somewhat compressive on one side of the stump, and involves more stretching on the other.

In recent full-scale drop tests of a prototype Type B container (known as TRUPACT II), the outer shell (designed to a thickness that a computer simulation indicated was sufficient to avoid puncture) ripped on the side where stretching occurred[22]. This thickness of the shell was increased by about 25% as a result. The lack of sufficient analysis by the Modal Study leaves the potential for puncture an open question.

Lead Shielding

Failure to fully investigate puncture and cracking of the outer shell creates the rationale for avoiding consideration of the loss of shielding, a very serious potential problem. The only mechanism for major lead movement covered by the Modal Study is slumping due to an endwise impact of the cask onto a hard surface. Since opening of the outer shell may be a realistic possibility (due to puncture impact angle and/or poor fabrication), examination of the slumping effect alone is insufficient analysis upon which to base the rest of the study.

While other responses of the shielding have already been discussed in the context of the outer shell and insulating effects of the neutron shield, it is noteworthy to consider two more of its characteristics: alloying with steel at 1050°F, and its heat capacity.

As indicated above, absence of a neutron shield greatly accelerates heat transfer and the likelihood of rapidly reaching very high temperatures at the inner surface of the outer shell (as versus the delay inherent in melting most of the lead, which must occur before the lead mid-point temperature exceeds 620°F). Even if the exterior steel layer of the neutron shield was still intact, LLNL apparently did not examine the temperature at the point of surface contact between the lead and the steel outer shell and thus did not consider the potential for alloying at that point. Only when the lead mid-point temperature also reached 1050°F (see page 4-12) did LLNL consider alloying (at which point it indicated that damage was not quantifiable). Alloying of lead with nickel in the steel considerably weakens the shell and can affect its ability to expand under heat, possibly leading to cracking and creation of an avenue for lead loss. Only a very thin layer of lead needs to reach 1050°F for this phenomenon to occur, and LLNL should have determined when that point would be reached in order to properly assess its likelihood.

The heat capacity of lead creates another condition not covered in the Modal Study. All fire simulations in the Study examined the temperatures of the lead mid-point during the fire. Since the temperature of the spent fuel will always be lower at this time due to the buffering effect of the lead, a short fire that doesn't yield a mid-point temperature of 650°F during its duration is not considered to cause fuel rod bursting or oxidation. Examination of the temperature after the fire is out could be crucial, however, to assessing the fuel rod condition. A DOE-sponsored study by Pacific Northwest Laboratories (PNL-2588)[23] examined the fuel temperature during and after a fire and found the highest point was reached hours after the fire was out, due to the delayed heat transfer into the inner shell and the insulating effect of the lead. Very high temperatures resulted, sufficient to burst the rods. LLNL should have examined short fires that cause lead mid-point temperatures of less than 650°F to assess their delayed temperature at the spent fuel. If short fires eventually yield high internal temperatures, the likelihood for a significant release is heightened, since short fires are much more common than long duration fires (at least according to the distribution used by LLNL). This delayed heating effect will be discussed again when the spent fuel's response is also analyzed in a later section of this report.

In closing this section, it should be noted that loss of the gamma shielding would seriously hamper any efforts by emergency personnel, while greatly increasing their risk to exposure. At present, it is doubtful that most firefighters would conduct a careful, 360° radiological survey around a truck or train fire prior to approaching it, unless they knew the hazards of failing to do so when spent fuel is involved. Most firefighters do not carry the necessary equipment, and the DOT Emergency Handbook[24] does not suggest a circumferential radiation check prior to approaching a cask.

The Inner Shell

The inner shell creates the cavity to hold the spent fuel. It is also a cylinder that fills with water when spent fuel is loaded underwater (a requirement to contain its radiation). The previously mentioned drain and vent lines end at the inside surface of the inner shell. Many of the same comments concerning welding and ability to maintain strength while stretching apply also to this steel cylinder. The integrity of the inner shell has another important requirement, however. It must hold the fuel basket in place, remaining rigid and

straight when compressed (as in an endwise impact of the cask against a surface). Impact simulations assume the inner shell remains rigid and straight, and thus does not provide any bending stress on the fuel rods during an endwise impact. It may simulate such stress for a corner impact, but any previously existing bending would exaggerate such stress. While this type of assumption may be acceptable for an ideal case, it should be noted that the actual cask simulated by LLNL, the NAC-1 cask, suffered from a bowing of the inner shell. This problem was not confined to one copy of the container, but rather showed up in several of them[20]. Four out of seven NAC-1 style casks were taken out of service due to this problem, and it was not noticed until several hundred shipments had been made[25]. Had such a container been involved in a severe endwise impact, the bowing could have created a vulnerability to bending or buckling of the shell, which could damage the fuel rods, leading to leakage into the shell. Analysis of such potential weaknesses could give some idea if slight bowing would significantly compromise the shell's integrity. The lack of examination of such real cask problems only adds to the uncertainty of the Modal Study's results.

Penetration Sub-Systems

As previously discussed, the Modal Study did not consider in its damage analyses the various valves and tubing built into a cask. It was felt that valves were protected by design features (e.g., recessing below the surface) and that any damage due to a highly localized load would "limit the escape of any spent fuel material to that which can migrate or be driven out through the small diameter, tortuous passageways presented by the damaged penetration systems" (p. 3-16). While it may be true that chunks of spent fuel would be blocked by narrow cracks or by bends in the tubing, it is not chunks that are the problem. Rather, it is the vapors, gases and fine particles that may be inhaled which create a radiological hazard. Relative to them, any visible crack or tubing is hundreds or thousands of times larger, offering little resistance to dispersion.

Once again, however, reality and the actual NAC-1 cask provide a perspective on LLNL's avoidance of the penetration sub-system as an issue. Prior to their removal from service, at least two of the NAC-1 containers were found to have a chronic problem with valve closure. After several instances of casks arriving with valves open to the inner shell, it was found that the valves were installed backwards, due to confusing instructions[26]. Vibration of the vehicle while in motion apparently opened them. There was no need for a "highly localized load" to open them, nor would there have been a "tortuous path" for the particles, vapors and gases to negotiate.

Finally, the welds involved in installing the tubing also create vulnerabilities even without "highly localized loads." In drop tests without impact limiters, a cask suffered cracks in its welding along its drain lines that extended from the inner shell out to the surface of the container[27], even though the steel around the welds remained intact. Once again, assuming that welds will act just like unworked steel is simply not realistic.

Impact Limiters

While it is valid to model cask impacts with impact limiters, it would have been very useful to examine the situation if a limiter was not attached properly. The Modal Study

assumed limiters and a truck cab (where appropriate) were both present to absorb much of the cask deceleration on impact. LLNL idealized the situation by assuming a perfect cask. The impact limiter is bolted to the end of the cask and is made of crushable wood. An impact at an angle could tear off the limiter, leaving the cask end vulnerable to a second impact (which present regulations do not consider). Such multiple collisions are not unusual in train derailments. The limiter is most effective for its first impact, after which it may be compressed and will not necessarily absorb as much impact. The NAC-1 was once found upon arrival missing bolts that attach its limiter, creating an opportunity for it to come off during an accident[28]. The Modal Study failed to consider such an eventuality, or to model the limiter bolts where they insert into the cask lid.

The Cask Seal

The Modal Study may have been conservative when it assumed that seal failure would occur if strain exceeded .2%, but there is no experimental data cited to support this number. A lower number may be possible. Furthermore, LLNL's thermal data appear to indicate that temperature at the seal would not reach the point of breakdown (about 500°F) but, as previously mentioned, the simulations appear only to cover the period while the fire is in progress (and while the neutron shield provides insulation), and not thereafter when the delayed thermal transfer could be significant. The cask seal does, however, possess a particular vulnerability not evidenced by the other materials. Unless it is a metal seal, it can be dissolved. It is not hard to imagine a rail cask as part of a typical commercial freight train (assuming that dedicated trains are not used*) that also carries a variety of chemicals, some of which may be solvents to the seal. A derailment involving leakage of such a substance could threaten the seal, and no major impact would be needed. A small fire could then provide heat to drive gases out of the cask, perhaps carrying with them particles of fuel surface crud. The Modal Study considered only impact and fire as means to damage the seal.

Final Comments on the Realism of the Modal Study's Cask

LLNL made several simplifying assumptions that, unless closely examined, could be the sources of unseen problems. For example, when simulating the type of steel used in the cask, LLNL used a slightly different variety than that actually in service, apparently due to limitations on its available data. Insufficient information is provided to assess possible

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* Northern States Power Company and the Nebraska Public Power District recently used dedicated trains in two of the largest shipping campaigns in commercial nuclear power history. In the 1988 OCRWM Draft Mission Plan Amendment, DOE assumes that shipments from a monitored retrievable storage facility (MRS), if constructed, will be made by dedicated train (i.e., trains containing only spent fuel as cargo). There are, however, no regulatory requirements for mandating dedicated trains. There is also considerable sentiment within nuclear utilities and DOE defense programs that dedicated trains are unnecessary. Moreover, OCRWM has carefully avoided any commitment to use dedicated trains for shipments between reactors and the MRS, or between reactors and a geological repository, if the MRS is not built. Therefore, it cannot be assumed that current (or next) generation casks will not be shipped in general freight service.

impacts of this alternative choice. Another simplification involved one of the drop simulations. During the sidewise impact of a rail cask on a flat surface, the sheer weight of the shielding nearly flattens the container (see p. 7-9 and Appendix E). It is extremely hard to imagine the welds to the end of the cask not yielding completely in such a case, creating a large avenue for release of fuel chunks and for direct exposure. Unfortunately, the simulation is only two-dimensional, and does not include the mating surface between the cask body and its ends. LLNL should have simulated that surface to determine the likelihood of lid separation. Instead, it simply assumed that releases from severe unsimulated scenarios would be ten times greater than for those it had analyzed.

In several other places, LLNL refers to full scale tests used to verify or benchmark simulations of accidents. In two important cases, it chose to avoid mention of the problems these tests revealed about proper cask fabrication. As previously covered, the failure of an outer shell during a fire test is not discussed. Even worse, however, was a reference to British rail crash tests. Citing the lack of damage involved, LLNL (on p. 6-32) leaves the impression that this test confirmed its analysis. Once again, reality is ignored: the cask in question was a solid forged design, not the welded steel and lead sandwich simulated in the Modal Study. Unmentioned is the fact that the British subjected some of their older welded casks to drop test and found that they cracked along their welds, contrary to the results of their simulations[29]. To their credit, the British retired those containers and now use only forged steel casks. If the British tests demonstrate anything, it is that cask welds are a source of vulnerability, therefore disproving the Modal Study's use of strain as its primary mechanical variable, and supporting LANL's criticisms.

3. Accident Scenarios

10CFR71: Starting at the Destination

LLNL examined a number of accident scenarios, using the 10CFR71 performance tests (i.e., drop, puncture, fire) as a starting point. LLNL discounted the need to examine criticality after a collision and immersion in water (the final 10CFR71 test), because its probability calculations indicated that such a scenario would occur only once in ten million years. In some ways, paralleling the present regulatory scheme made the Modal Study's goal of verifying it almost a self-fulfilling prophecy. It is important that the reader avoid also "signing on" to the 10CFR71 perspective while thought is given to the potential accident conditions that could realistically prevail. The degree to which LLNL did so will become obvious and, to that degree, the Modal Study loses some of its credibility.

But focusing on the order and types of those tests was not the only problem with LLNL's accident scenario analysis. The Modal Study's simulations failed to realistically simulate some characteristics of drops, collisions and fires, and other possible scenarios were deleted from the analysis without sufficient examination. LLNL also failed to sufficiently interact the effect of one accident condition with those that followed it.

The 10CFR71 tests were designed as highly simplified simulations, not of actual accidents, but of the worst conditions that could prevail in almost any accident. LLNL "translated" them into its own parameters of strain and temperature which, it believed, could be used to categorize an accident's potential for causing a radiological release. As

previously covered in the "cask response" section of this report, the strain and temperature considerations are themselves highly simplified replications of reality, also somewhat open to question. For the moment, however, the concept of the conversions to strain and temperature will be accepted.

Collision Simulated by Dropping

The 10CFR71 tests first call for dropping a container, in its most vulnerable orientation, onto a flat unyielding surface from a 30-foot height. Since there are no totally unyielding surfaces (i.e., all real objects will absorb some impact energy), it was essential for LLNL to model a number of "real" conditions and determine the collision speed necessary to equal the kinetic energy that the cask body would absorb in the idealized 10CFR71 drop. This process is referred to as the "equivalent damage" technique. LLNL found that an impact with soft soil would require an impact speed in excess of 150 mph, an unrealistic velocity for a truck under any condition. Hard soil and rock required lower speeds, as did some concrete structures. This approach is acceptable from the standpoint of screening out some types of accidents (e.g., hitting a mound of earth) but only looks at total energy transfer. The most vulnerable angle and point of impact are more difficult to determine.

LLNL considered side drops (i.e., impact at 0°) and end drops (90° impact) and then interpolated linearly between those angles to assess the conditions that would prevail for drops on a corner of the cask. This simplification could lead to a significant error. In the review of cited references and others known to this writer, it became obvious that the determination of the most vulnerable angle and point can only be done with surety through experimentation. One reason for this is the "slap down" phenomenon previously mentioned, in which a drop on one corner results in acceleration of the other end of the cask as it revolves prior to its own contact with the impacting surface. The increase in velocity for the secondary impact (which may also occur at a different angle) may be considerable, and could depend on such items as the flexibility of the impact limiters and location of the cask's center of gravity. The software used by LLNL does not model this phenomenon.

The Modal Study analysis concluded (p. 4-7) that only a .2% strain level would occur at the inner shell during the 30 mph impact (i.e., the 30 foot drop) onto an unyielding surface, so no seal damage would result. Impact speeds of 35 to 55 mph would yield the same result on hard rock, depending on the impact angle (i.e., orientation of the cask to the surface) (p. 6-30). At higher velocities, the strain would no longer be elastic and seal failure is assumed. Note that the impact velocity assumed is that of the first corner to land, not the second corner, which may be moving at a higher speed in a corner drop. Furthermore, the cask lid and body are two separate objects connected by bolts that can flex and bend, so the assumption that distortion of the inner shell is the only criterion for seal failure may be insufficient. In light of these uncertainties, LLNL's conclusion that no seal failure will result at or below the first 10CFR71 drop cannot be accepted without a more dynamic analysis at points along the seal of the cask lid.

Interacting the Effects

Other aspects of the impact with a flat surface also appear to have been simplified, or else were not taken fully into account in later tests. For example, imagine a rail cask involved in a fall onto its tail end, followed by a fire near its lid end. In such a case, lead will have slumped to the rear, removing that heat sink from near the cask seal. One accident condition then creates a worsened (and unexamined) situation for a later aspect of the scenario. Similarly, a side drop onto a hard surface can so distort a cask's shape that the lead will become thinned on the sides of the cask and continued connection with the end plate is doubtful. While the integrity of the lead may remain immediately after the drop (though now thinner at some points), it is now vulnerable to loss (through the damaged end plate connection) when the lead becomes molten in a fire. While LLNL may say that such a fire must exceed the 10CFR71 limit of 30 minutes and is statistically unlikely, recall that this same phenomenon (of cask flattening) will also flatten the empty neutron shield, eliminating much of its insulating capability. Heat transfer rate is increased, possibly to the point that a 30 minute fire is no longer essential to begin lead melt. Many other such cases can be posited, the likelihood of which are not known with any accuracy due to limitations on accident data and/or the simulations. Nevertheless, they have potential for occurrence and these interactions were ignored by the Modal Study even when (as in the case of cask flattening) it postulated the initial step itself.

It may be argued that such combinations are covered by "conservatively" assuming that they fall into the region beyond 2% strain and 650°F fires, where the radiation release is assumed to be 10 times greater than in the next least severe range. Once again, however, the statistical juggling done to marry the impact strain to lead temperature leaves one uncertain as to its validity. The distributions of the two characteristics were combined with very little linkage between them, and their origins were from two different data bases. Even if this data is accepted on faith, however, the multiple of 10 for a release quantity has no basis and, in the postulated flattening case, could easily be off by several orders of magnitude due to increased exposure alone, if the lead shielding were reduced by slumping, or by melting and subsequent lead loss. Failure to follow through on these interactions is a major shortcoming in the Study and again demonstrates its underlying lack of reality.

Potential for Puncture

The possibility of puncture of the outer shell was, in effect, ignored by LLNL's analysis. As previously discussed, simulations involving an I-beam and a train sill were apparently sufficient to convince LLNL that puncture of the inner shell was not within the realm of possibility. While it may be likely for the inner shell to remain intact (though the cask seal has been assumed to leak at 2% strain that results from a 27 mph impact by a train sill), there is also a need to examine the condition of the outer shell. Puncture of the outer shell would open a pathway for molten lead leakage, increasing direct exposure and removing a thermal barrier from the inner shell. As previously discussed, this affects the size and duration of a fire needed to further damage the inner shell, seal and fuel.

LLNL used NIKE-2D, a finite element computer code, to simulate the I-beam and train sill impacts. While a major improvement over the empirical equation used to analyze

puncture in most shipping cask safety analyses, NIKE-2D leaves a great deal to be desired, especially when strains in excess of .2% (i.e., inelastic) are involved. In a 1980 professional paper, NRC structural engineer R.C. Shieh criticized it, saying:

> "The NIKE-2D model also does not possess capability of modeling strain rate sensitive material on inelastic behavior. Therefore, additional efforts are required to improve the computational efficiency and dynamic modeling capability of rate sensitive materials (such as steel and lead) before the NIKE-2D model can become a useful tool in accurately predicting puncture behavior..."[30]

There is thus reason to doubt the validity of the Modal Study's quick dismissal of puncture.

It should be noted that experimental analyses (one of which was performed at LLNL in 1980) indicate that puncture of the outer shell of a lead-shielded cask is indeed a possibility and has, in the past, been underestimated by the empirical equation used in the design of most casks. An interesting finding of one study was that puncture required 50% less energy (i.e., could occur due to a drop from a lower height) after the shell had been heated to about 400°F than when it was cool[31]. A question then arises regarding the likelihood of a fire before puncture, instead of after, as outlined in 10CFR71. One need look no further than the 1978 Sandia fire test for evidence. During the fire, the cask was supported by a rail carriage which collapsed when its steel softened from the heat. The cask fell several feet into the steel rubble, showing how easily contact could be made with a rail track (or other protrusion). "Signing on" to the 10CFR71 order of things limited the Modal Study's examinations of real conditions that could have significant effect on cask integrity.

Concerns about Crush

As with puncture, the Modal Study casts aside any need to closely examine the potential for crushing a cask. While very few scenarios for crush are likely for truck casks (with the possible exceptions of a tunnel collapse or landslide), the rail environment provides several such opportunities. LLNL considered the 200-ton weight of a locomotive resting on-end against a cask, as the worst case and found it did not yield the same damage (p. E-11) as other scenarios. Major derailments can result, however, in greater weights being piled upon one railcar. An NRC study concluded the bounding value in such a case was 550 tons, nearly 3 times the case considered by LLNL[32]. Once again, it is hard to understand LLNL's failure to utilize (or at least comment on) relevant available data. This deficiency simply adds to the uncertainties surrounding its overall analysis.

Fire and "Smoke"

LLNL's attention to fire showed some effort to be conservative, and some of its analyses added valuable insight to this aspect of the problem. Once again, however, there are difficulties with its acceptability. Several problems associated with the interactions of fire and other accident conditions have already been covered, and the simulation of heating the spent fuel will be covered in the next section. For the moment, it is necessary to focus on the fire simulation's underlying assumptions.

While the typical flame temperature assumed may be realistic, the sheer number of assumptions inherent in describing the fire duration, location of the cask and types of flammable materials places the result firmly into the probability "ether." The Sandia analysis used as basis for some of this input (SAND74-0001) is itself based on many (a total of 26) "best guesses," plus some rather old references[33]. It is noteworthy that the data used in that study is significantly dated with regard to types of flammable materials now shipped, the traffic of such materials and the accident rates involving them. It is unclear why LLNL did not try to update the input by using more recent data sources (e.g., the DOT Hazardous Material Information System), but its lack of effort in this area (or any effort to benchmark the result against such data) leaves its fire severity distribution shaky, at best. For example, while much of the flammable material shipped by road and rail is heating or vehicle fuel (the "worst" case included in the Sandia analysis), an increasing amount consists of very high flame temperature materials used in industrial processes (e.g., benzene, propane, acrylonitrile)[11]. Stationary sources of fuel (e.g., storage tanks and pipelines) are also ignored despite the fact that such sources could yield extreme conditions near road and rail lines, as previously discussed under "Data Creation and Analysis." As a result, LLNL's analysis lacks conservatism in its probability assumptions regarding temperature and duration.

Another unsettling aspect of the fire simulation is the disregard for a torch fire. LLNL sets aside any such concern by focusing only on the total thermal input to the cask. LLNL reasoned that, because a torch fire only strikes a small area of the container, it cannot do nearly as much damage as an engulfing fire that transfers a massive amount of thermal energy to the cask. Again the "blinders" inherent in the 10CFR71 approach appear to have blocked awareness of the interactions of accident conditions. While total thermal energy may be a fair way to dismiss the immediate effect of a torch fire on the inner shell it ignores the effect on the outer shell. A 1980 Sandia study of torch fires noted the following:

> "Non-uniform heat input in real fire exposure environments could lead to a number of package design problems unless care is taken by the designer. Local stresses could result in package or seal failure. Also, lead gamma shield material could melt locally away from expansion volumes and the outer shell could rupture, allowing the gamma shield to be totally or at least partially lost."[34]

Loss of a portion of the lead shielding could then expose part of the inner shell to severe local heating, all without raising the average mid-lead temperature to 500°F. Torch fires on railroads are common enough to require that railroad propane tankers be able to withstand them, under DOT regulations[35].

The Modal Study is therefore unfortunately deficient in its examination of several important fire scenarios that could affect the integrity of the inner and outer shells. Coupled with the difficulties previously outlined on its assumptions of a thermal barrier in the empty neutron shield, it is not hard to conclude that there is serious potential for the Study to be a source of erroneous conclusions on the fire resistance of its representative cask.

Taken individually or in toto, these problems show that the Study's portrayal of accident scenarios leaves a great deal to be desired. Its failure to interact the results of

different, but consecutive, accident conditions makes some of its results unreal. Combined with its idealistic and highly simplified view of the cask response, one is left with little confidence in some of its conclusions in this area.

4. Spent Fuel Responses

Examination of the Modal Study's radiological release assumptions also finds numerous reasons for concern. Some of them relate to apparent ignorance of past studies and incidents, while others reflect the potential impacts of problems previously outlined in the "cask response" and "accident scenario" sections of this paper. Unfortunately, the appendix gives no supporting discussion in this area, so the reader is forced to perform research to add perspective to LLNL's conclusions.

Missing Sources of Radiation

The Study lists the isotopes that it considers significant to a possible release (p. 8-6), without benefit of reference. Examination and comparison to other studies of spent fuel confirm the list with two major exceptions:

- americium 241 (Am-241) is missing
- there is no attention to gamma emitters in the structural part of the fuel assembly.

The absence of Am-241 may have resulted from examination of isotopes that exist when fuel is first removed from the reactor. Am-241 does not represent a major nuclide at that time, or even 150 days later, when most past spent fuel transport studies examine the fuel's inventory. Over several years (and especially between 5 and 10 years), however, plutonium 241 decays to Am-241, increasing the americium curie strength nearly 100 times[36]. As a potentially hazardous aerosol, it is as dangerous as any of the isotopes of plutonium, and at 5 years provides a significant portion of the total hazard. If it is actually absent from LLNL's analysis (and not merely a major typographical error missed during all reviews and edits), there is some question about the care taken elsewhere in the Study's radiological analysis.

The failure to include gamma emitters in the structural parts of the fuel is not related to an actual release of materials, but rather is of concern when analyzing direct exposure after lead slump. The end piece and foot piece (see figure 8-1 in the Study) are composed of steel containing cobalt, some of which has been converted to Co_{60} after years in the reactor. Other components of the steel have been similarly converted but do not represent the same hazard as Co_{60} due to their amount, or rapid decay. When an endwise impact of 46 mph was examined, a lead slump of about 3 inches occurred for a truck cask, and about 6 inches for a rail cask. These speeds occurred at the the 2% strain level. Figure 8-7 indicates that (absent any thermal effects) exposure of only .36 curies would result for the truck cask and 27.5 curies for the rail cask. The only way this could conceivably occur is if no curie content was attributed to the head piece of the assembly, a good portion of which would be exposed during such a lead slump. Figure 8-1 indicates LLNL considers the "active length" of the assembly to begin somewhat below the head piece, which would support this conjecture. The head piece, however, contains more than half of the

approximately 2000 curies of Co_{60} in one assembly and gives off (even after 5 years out of the reactor) several hundred rems per hour of gamma radiation, when unshielded[37]. The Modal Study's calculated hazard due to lead slump is inconsistent with this data, which is based on actual measurements of aged spent fuel assemblies. It should be noted that the neutron output of radiation at the head end of a fuel assembly is so intense that, over time, it caused conversion of cobalt in the steel in the lid of an IF-300 cask, causing it to give off an unacceptably high level of radiation on its own[38]. In light of these facts, the direct exposure aspect of the Modal Study requires major revisions.

While it may be just an unintentional omission, fuel crud (i.e., the radioactive surface dirt on the outside of the cladding) and its characteristics are never discussed in the Study; it is only mentioned as a footnote on p. 8-2. This raises an additional question: was crud treated the same as the other isotopes, only to be released when the cladding was breached and fuel pellets damaged? There is evidence of this when one tracks the calculations leading to figure 8-7, in which the released curies are delineated for each of the response regions. Table 8-3, which lists the release fractions due to rod burst or oxidation, appears to be the sole basis for development of figure 8-7, but a check of the reference cited (NUREG/CR-0722, hereinafter referred to as the ORNL study) shows that it was concerned only with releases from the fuel, not the crud layer[39]. If the crud was not "lost" during the analysis, LLNL needs to show why it does not appear in figure 8-7, since the data in that figure then forms the basis for the rest of its conclusions.

The amount of crud on an assembly has been found to vary with the reactor type, age, water treatment and other conditions. In some cases, it has exceeded 300 curies on a single assembly[40]. LLNL used only 21.1 curies. The crud analysis is important for three reasons:

- crud resides on the outside of the fuel, so no cladding damage is needed to release it to the cask environment
- it is shock and heat sensitive, so it can fall off the fuel during an impact, and starts to flake off the rods at only 212°F
- its particles are very small and can form an inhalable aerosol[40].

While the curie quantity of the crud is much less than that of the fuel, it is available for dispersal in the less severe (but much more likely) accident scenarios and requires no other chemical or other mechanism to form an aerosol. If crud release was not treated separately from fuel damage in the Study's analysis, then a large portion of the risk calculations are wrong and the Study's overall calculations and conclusions may be seriously in doubt.

How Much Cladding Damage and Fuel Leakage?

To estimate the fraction of fuel released to the cask environment, LLNL developed percentages of the rods damaged in each response region. It saw this as a two-stage process: the fraction damaged due to impact, followed by damage to the remaining rods due to thermal creep, a phenomenon related to heating of the cladding. To its credit, LLNL made a reasonably conservative assumption of the percent of rods breached in the .2% strain region due to impact, assuming 3% until the thermal creep temperature was reached. Its assumption of 10% damaged in the 2% strain region is, however, a guess not based on

any tests. Some experimental verification is needed. In the 30% strain region, all rods are assumed breached and this guess is acceptable. LLNL also assumed that any nuclides released to the cask cavity would escape to the atmosphere, again a conservative assumption. LLNL's method of determining the extent of cladding damage and the fraction of nuclides released through broken cladding, however, leaves a great deal to be desired.

To estimate the damage to shock, LLNL used data from a 1979 ORNL study aimed at analyzing spent fuel's response to a loss of coolant accident (LOCA) while still in a nuclear reactor. Only thermal (not impact) conditions were involved when very high temperatures (900 to 2200°F) were imposed on spent fuel out of the reactor for about 2 1/2 years. Damage resulted from pressure buildup in the rods, causing them to perforate (i.e., burst through a small hole). The gases, vapors and particles that escaped were measured, and release fractions developed. LLNL states (p. 8-12) that it used the results of those experiments to estimate its own material release fractions. Close comparison of the Study's fraction (Table 8.3) and the ORNL data show significant discrepancies, however, the worst of which involves the fraction of particulate material released. Since most of the dangerous curies are in particulate form, this difference could have a major impact on the degree of hazard. Specifically, the ORNL study found that an average of .02% (i.e., 2×10^{-4}) of the fuel escaped in particulate form, while the Modal Study used 2×10^{-6}, only one-hundredth as much. Unfortunately, the Study does not provide any formulae or calculations to explain the difference. Efforts by this writer to duplicate possible qualifying assumptions were unable to arrive at this factor. For example, the ORNL tests found that each one-foot-long test segment depressurized through a small hole, about 1/16 inch in diameter. If LLNL assumed that a 15-foot rod would also perforate through only one such hole, then the release fraction should be 2×10^{-4} divided by 15, or 1.33×10^{-5}, but this is still about 7 times too high. Correcting for the age of the ORNL fuel (2 1/2 years instead of 5) made a slight difference, but arriving at the Study's fraction was only possible when erroneous assumptions were made.

But this "mystery" is compounded by another: the radiological hazard figures for particles in figure 8-7 do not agree with the basic calculation involving even the 2×10^{-6} release fraction. For example, region R(1,3) (where 100% of the rods are damaged) shows 7.22×10^{-3} curies of particles. Since more than 100,000 curies from the isotope inventory in Table 8-1 could be in particulate form, one would expect at least 100,000 x (2×10^{-6}) curies (i.e., .2 curies) to exist as R(1,3) particles. Again, no basis for this factor of 28 was indicated, nor could one be developed. The Study numbers therefore differ by a factor of at least 28 x 7 = 196 from any straightforward method to adjust the ORNL data. Similar discrepancies were found in some of the calculation of releases of gases and vapors. Until its exact methodologies and calculations are checked by independent reviewers, these results are, at best, suspect. It should be noted that neither peer reviewer was given this information, and the primary reviewer commented several times on the large number of assumptions hidden in the calculations.

Let us assume (for the moment) that all calculations are correct, however. There still remains the validity of using the ORNL thermal test data as a substitute for impact data. As indicated above, the damage occurring in the ORNL tests consisted solely of a single 1/16 inch diameter hole in each rod. There is nc theoretical or experimental analysis to confirm

such a uniform and low level of damage due to all degrees of impact[41]. To the contrary, rods have become brittle in reactors, have broken while being moved, have come loose from their frames and have leaked in casks - all without involvement in a major impact[42]. Furthermore, all the ORNL tests yielding usable particle data involved a steam-helium atmosphere, containing only trace amounts of air. An accident in which a pathway to the environment exists would involve a major influx of air as the inert cask atmosphere diffuses rapidly out through the crack. Unlike steam and helium, air attacks uranium oxide at relatively low temperatures (above 400°F) converting it to U3O8 and breaking ceramic fuel pellets down into an aerosol powder[43]. Such action greatly accelerates release of gases and vapors locked into the pellet structure, while creating a form for the airborne release of all isotopes. The Modal Study implies that it examined oxidation by listing release fractions for gases and vapors related to oxidation in Table 8.3. It does not, however, mention that its basis for showing zero curies for particles resulting from oxidation originates from ORNL test procedures that involved no available oxygen to drive such a reaction.

Other problems exist with the use of ORNL data. At this point, the only conclusion one can come to is that, after a great deal of analysis in other areas, LLNL was confronted by an informational void and, instead of acknowledging that it lacked any valid release data, grasped at whatever it could find to fill the vacuum. LLNL understated the absurdity of its position on p. 9-23: "radiological hazards could be better estimated with pertinent tests performed at high impact conditions for the spent fuel rods."

Other Possibilities for Fuel Damage

Lost in the shuffle of suspicious data is, however, the actual thermal impact that could occur. As mentioned, the fuel pellets will decompose to powder when heated and contacted by the oxygen in air. Once again, LLNL's use of the mid-lead temperature diverts attention from the temperature of the inner shell and fuel. A prior study (PNL-2588) found that the temperature of the fuel will rise significantly hours after a fire is extinguished due to the delayed heat transfer of the gamma shielding (this is true whether it is lead, uranium or steel). The Modal Study gives no indication if it examined such post-fire conditions. Acceptance of the Study's exclusion of fuel oxidation from its release fractions is impossible without discussion of the phenomenon and the provision of post-fire simulation data.

It should be noted that fuel re-oxidation as a phenomenon has not been confined to the laboratory. In 1980, a fuel assembly with several damaged rods (one with cracked cladding) self-heated while in transit in an air-filled cask, breaking down a much larger portion of its fuel into powder than was seen in the ORNL steam-helium tests[44]. The cask (another NAC-1) was heavily contaminated and, when opened under water, released its powder via air bubbles that upon popping at the surface caused the powder to become airborne, contaminating the pool area[45]. Even after several decontamination efforts, so much powder remained in the cask that the mere draining of residual water weeks later cause major problems at a commercial power plant[44]. The area of fuel re-oxidation was so foreign to NRC regulators that they did not react to this incident until petitioned to do so by the Sierra Club. At that point (in 1984), NRC concluded that all casks carrying uranium dioxide fuel must contain an inert atmosphere, even when no cladding defects are detected,

because of the difficulty in detecting such a weakness prior to the actual breech of the cladding[46]. There is thus good reason to doubt LLNL's conjecture that the worst shock damage a fuel rod will see is a single 1/16 inch hole over its entire length.

Several other characteristics of spent fuel may strongly influence its response to shock and heat. Both the ORNL test results and later work done at the Idaho National Engineering Laboratory (INEL) on fuel rod damage due to a major shock (i.e., from an explosion) found that the fuel pellets may shatter back to their grain size (i.e., the size of the particles of uranium before they are pressed and sintered into pellets). The particle size in question is in the aerosol range, making it very fine and able to pass through narrow cracks when airborne. If shock alone reduces some pellets to powder, then it is likely that LLNL's particle release fraction may be low by several orders of magnitude. For example, if only one pellet (out of nearly 100,000 per assembly) shattered back to grain, it could yield several curies of nuclides in dispersible form (as versus LLNL's 7.22×10^{-2} curies in the worst case).

While the cladding would still serve as a barrier to release of the powder, the cladding's own ability to withstand shock is also open to question. Zirconium alloy is designed to operate in water, not air, and will chemically combine with both oxygen and hydrogen, depending on the temperature. The metal may become brittle as a result, leading to cracks along its length, not just pinholes. It should be kept in mind that much of the fuel shipped in the future will have a history of dry storage as spent fuel pool capacity is exceeded and dry storage casks (using inert atmospheres) are increasingly presses into service. Cladding vulnerability appears to be closely tied to storage temperature and surface conditions, neither of which will be known with certainty until a much longer history of dry storage has been obtained.

If shock does yield cladding damage, there is a synergistic effect on re-oxidation, leading to further opening of the cracks or holes. The conversion of UO_2 to U_3O_8 due to heating in air is accompanied by a change in crystal structure and major pellet expansion as it decomposes. This action will spread cracks further apart, exposing more fuel to air, and so on. In the two ORNL air tests (neither of which were cited by LLNL), this began to occur but the expanding fuel eventually blocked the small hole resulting from overpressurization due to heating (but no impact). In the 1980 incident, a larger opening yielded a much greater release. As high temperature accelerates the process, it exposes more fuel surface and also accelerates gas, vapor and particulate release. While quantitative data on the overall impact of these simultaneous processes does not exist, such multiplying effects could quite easily increase released curies by much more than the factor of 10 assumed by the Modal Study for mid-lead temperatures above 1050°F.

LLNL attempts to diminish concern over gas and vapor release by repeating a comment on the potential for them to "plate out" (i.e., condense) as they contact cooler interior cask surfaces on their way to the atmosphere. This notion made sense when relatively "young" fuel (less than 1 year out of the reactor) was considered the norm for shipment, since it was always self-heating in the cask. Older cooler fuel, however, will be heated during a fire by radiation and conduction from the inside of the inner shell, thereby guaranteeing that the shell is hotter than the fuel. Since the cask is being heated from the outside due to fire, the vapors will experience a rise in temperature as they pass through cracks or tubing on their

way to the cask exterior. Condensation prior to release only becomes likely during leakage occurring hours after the fire is out, as the cask surface cools off. Once again, the Modal Study's claims of conservatism don't withstand closer examination. As indicated in another study listed among the Modal Study's references (i.e., NUREG/CR-0811), much more experimentation and research on spent fuel responses is needed before assumptions regarding releases can be made with any credibility.

A Closer Look at Criticality

The last item regarding spent fuel's reaction to impact relates to the maintenance of sub-criticality during and after an accident. Changes to fuel configuration can cause criticality if accompanied by intrusion (or presence) of a moderator such as water. The Modal Study dismissed the chance of such an occurrence as once in ten million years (p. 9-25), using the probabilities in section 5.0. These numbers assume a major impact followed by submergence in an existing body of water. Other combinations of events could, however, mimic aspects of that seemingly incredible scenario. As discussed on p. E-89, "...the rail cask is like a thin-walled cylinder. Under the severe impact conditions, it is unable to support itself." Sidewise impacts with a surface or rigid abutment at high speed could yield collapse of the rail cask inner shell onto the fuel, reconfiguring it into a number of different densities. The same impact could cause the weld to the end cap to crack, thereby creating a path to the fuel. While it is agreed that simultaneous submergence in a body of water would be very unlikely, such an eventuality is not essential to create the criticality scenario. Any fire, even a small gasoline blaze, may prompt firefighters to apply water, thereby providing the necessary moderator. The cask may also contain residual water after loading: such is allowable with rail casks (the IF-300 may hold a number of gallons) and a NLI 1/2 cask - designed to be shipped dry - was once (due to a human error) shipped full of water while containing spent fuel)[48]. It is therefore not essential for a fire or even a cask breach to exist in order for the proper combination of factors to occur. The apparent ignorance of LLNL personnel with respect to actual cask operations and history also apparently blinded them to cases where sub-criticality was not guaranteed by design. After a number of years of use, several spent fuel casks and one plutonium container (all certified by DOE) were found to be vulnerable to uncontrolled criticality in an accident, and were taken out of service[49]. Nor has NRC certification been perfect in this regard. A mathematical error in the design of the IF-300 BWR fuel basket could have caused buckling in a crash, thereby limiting its ability to control the fuel rod configurations[50]. The error was not found until many shipments had occurred, fortunately without a crash. Luckily, the original design analysis was so crude that the math error was later found to be smaller than needed to create a serious hazard. The same mistake in a more sophisticated analysis could have yielded a very different result. Criticality loss cannot be simply wished away by considering only a highly unlikely accident scenario.

The same appendix E discussion also provides a glimpse into another potential limitation on the Modal Study analysis. Unlike truck casks, "the mass of the rail cask contents is very large compared to the mass of the cask...contents are very important to the rail cask calculations and should be modelled to provide more accurate impact forces and g loads and to support the cask as it collapses." (p. E-89) The pressure of a collapsing lead

wall could significantly rearrange the fuel rods and perhaps move or damage any moderating poison rods in the fuel assembly. None of the Modal Study cask models included contents. The forces of the collapsing lead wall as it struck the fuel would provide useful input to analyzing possible cladding and pellet damage, and questions the assumption of a 1/16 hole as the maximum extent of a cladding breach.

In conclusion, the lack of effort by the Modal Study participants to study the response of fuel to shock, temperature and physical/chemical interactions is disappointing, at best. There is a wealth of data on the potential for cladding and pellet damage in air, done to analyze dry storage[51]. These sources are not listed in the Study's references, and one assumes they were not consulted. Yet, in the end, it is the fuel that is the hazard. Spending almost all of its attention on the container and the accident, the Modal Study lost sight of the cask contents, creating a large residue of uncertainty and the possibility that its consequence analysis lacks credibility. Since consequence is half of the risk analysis, the Modal Study has failed to complete its task - and thus has failed to confirm the 10CFR71 standard it sought so desperately to support.

Relevance to Shipments to the Yucca Mountain Repository

Since the Modal Study was published, numerous changes in cask payload and design, as well as the number of shipments, have occurred. While LLNL attempted to be conservative by choosing the design it felt was most vulnerable, some of the changes are not covered by its choices.

LANL pointed out, for example, that strain is even less useful as an indicator when harder gamma shields are involved. Uranium shielding will not yield easily, thereby passing on almost all force directly to the cask's seal and welds. It will also alloy with steel, but will not easily melt, therefore not acting as a heat sink to the same degree as lead.

As mentioned, water neutron shields have been replaced by solid materials that will not have the same thermal characteristics as the dead air space assumed by LLNL. Thinner gamma shields will be used because of the decreased gamma output of the older fuel, more of which will be carried in each cask. Such fuel will probably have a higher burnup rate and will therefore have a higher isotope concentration. The cask-to-payload weight ratio will decrease significantly, making the Modal Study's "no payload" simulations even less relevant.

The mix of shipments will be different since nearly half the reactors lack rail spurs, despite early assumptions that most would utilize rail transit. The total number of shipments may be reduced by the larger capacity of the casks, but that factor also aggravates the direct exposure problems after lead slump (if lead continues to be used).

Likewise, the state of information on fuel conditions has improved, due to the advent of dry storage and rod consolidation research. The high cost of computer power has dropped precipitously, so better and more detailed simulations are possible within a realistic budget. Finally, the Nuclear Waste Fund provides a ready source of capital to perform an improved, updated and more realistic Modal Study.

Conclusion

All of the above considerations point toward the need - and opportunity - for a clearer look at cask safety. The Modal Study was a necessary step in that direction, but not a sufficient one. It needs to be redone and its methodology and results closely critiqued by a competent body of reviewers while it is in progress, if credibility is the desired final result. The present document is unable to satisfy even a brief critical examination. Its flaws provide a breeding ground for bad decisions, from which in the future all parties may suffer.

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