PRELIMINARY PRECLOSURE SAFETY ANALYSIS FOR A PROSPECTIVE YUCCA MOUNTAIN REPOSITORY[a]

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ABSTRACT

A quantitative probabilistic safety assessment was performed for the reference conceptual repository design being used as the basis for the development of the Yucca Mountain Site Characterization Plan. A new methodology to quantify radioactive source terms for the air pathways was also developed and applied in the assessment. The assessment identified 21 potential initiating internal and external events applicable to the prospective site for which 149 potential accident scenarios were defined with event tree analyses. Of these accident scenarios, 13 occurred with probabilities ranging between 10^{-6} to 10^{-9} per year and with estimated offsite 50-year dose commitment consequences ranging from 110 to 1,100 mrem, respectively. These 13 reference accident scenarios represent the most severe consequences identified that the operation of a repository could initiate. The remaining scenarios had either probabilities of occurrence of less than 5×10^{-9} per year or offsite dose consequences of less than 50 mrem.

INTRODUCTION

The Nevada Operations Office (NVO) of the Department of Energy (DOE) is studying the possibility of siting a geologic repository in tuff at Yucca Mountain near the southwestern corner of the Nevada Test Site (NTS). As a participant in the Nevada Nuclear Waste Storage Investigations (NNWSI) Project, Sandia National Laboratories (SNL) is responsible for the conceptual design of the surface and underground facilities at the repository. Bechtel National, Inc., and other contractors, under the direction of SNL, have developed a conceptual design for this prospective facility. The conceptual design (1) serves as the basis of the repository design for the Yucca Mountain Site Characterization Plan being developed by NVO and other contractors.

As part of the conceptual design process, a probabilistic safety assessment was conducted for the reference conceptual design being used for the Site Characterization Plan. This paper summarizes the results of this conceptual level assessment, which was carried out by developing accident scenarios, estimating probabilities of occurrence, and calculating offsite dose consequences. The development of accident scenarios and the estimates of the probabilities of occurrences in the safety assessments were both site-specific and design-specific; this is discussed in Ref. 1. Because the methodology is applicable to other geologic repositories and to a Monitored Retrieval Storage Facility, offsite dose consequences, and particularly the novel approach used to estimate the source terms, are emphasized in this report.

METHODOLOGY FOR SAFETY ASSESSMENT

A safety assessment of the material handling operations, the repository design, and the facility equipment concepts consists of six principal steps (1). These steps are described below.

Conduct Facility and Systems Modeling

Based on the conceptual facility designs, building layouts, and waste inventory, the structures, systems, and components are grouped into a set of system areas that represent and cover the total facility design.

Identify and Screen Initiating Events

From a checklist of initiating events, the internal and external events whose occurrences at the Yucca Mountain site either are not credible or cannot result in an offsite radiological release are eliminated from further analysis.

Develop Event Trees and Accident Scenarios

For all credible initiating events that may potentially result in significant offsite radiological releases, the response and interactions of the structures, systems, and components of the repository and of the radioactive materials in each system area are assessed. Event trees are developed and used to summarize the set of potential accident scenarios associated with each initiating event.

Assess Probabilities of Occurrences

For each accident scenario in the event trees, the likelihood or probability of occurrences of the initiating and intermediate events are estimated on the basis of engineering judgment and from the data base for similar material handling operations.

Assess Dose Consequences

For each scenario in the event trees, the 50-year dose commitment received by a maximally exposed individual located offsite is estimated. Before dose consequences with conventional air pathway dispersion, air transport, and dose consequence models are calcu-

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lated, realistic source terms and releases of radioactive materials from the facility designs are developed.

Establish Reference Accident Scenarios

After the the probabilities of occurrence and the offsite dose consequences are assessed, these numerical quantities are transferred to the event trees so as to summarize the quantitative results of the safety assessments. The quantified event trees represent a set of potential accidents that might occur at the Yucca Mountain repository. Based on various screening criteria, subsets of the accident scenarios can be selected from the full set of quantified event trees for additional specific and detailed analyses.

For example, all scenarios with a probability of occurrence of greater than 5×10^{-9} per year and a dose consequence of greater than 50 mrem were selected for more detailed analyses, which were conducted to establish the initial Yucca Mountain Q-list (1,2). The Q-list contains items and activities sub- jected to Quality Level I QA requirements. Other subsets which focus on the enhancement of overall design safety, development of a set of design basis accidents, establishment of detailed design requirements, etc., can be selected for analyses.

RESULTS

The probabilistic safety assessment concentrated on the Yucca Mountain surface and underground facilities and the operations that are directly involved in the receiving, handling, preparation, storage, and emplacement of radioactive commercial spent fuel and defense high level waste (DHLW) during the preclosure period. These initial assessments excluded the development of accident scenarios associated with retrievability, repository decommissioning and closure, performance confirmation, and criticality. The normal retrieval operations are the reverse of emplacement operations which were included. The plans for decommissioning and closure, performance confirmation, and nuclear criticality assessment have not been developed to the extent that a credible initial safety assessment could be performed.

Accident Scenario Development

Nine system areas were defined based on factors such as waste form physical characteristics, types of confinement barriers impeding dispersion of radioactive materials, facility design layouts, etc. To illustrate this approach, six system areas within the reference main waste-handling building are shown in Fig. 1. The degrees of intrinsic dispersion resistance by the waste forms and the confinement barriers can be correlated with each system area in Fig. 1. For example, the resistance was classified "low" for bare fuel assemblies, "medium" for spent fuel and DHLW in containers or canisters, and "high" for spent fuel or DHLW in shipping casks. Once the system areas had been established, teams of experienced facility designers and safety assessment analysts identified internal and external initiating events and accident scenarios that could potentially lead to significant offsite radiological releases.

Next, accident scenarios initiated by internal events were developed in detail using event tree analyses that took into account intermediate events. These scenarios included crane load drop accidents, fuel consolidation or container-cutting machines failures, and collision accidents involving shipping

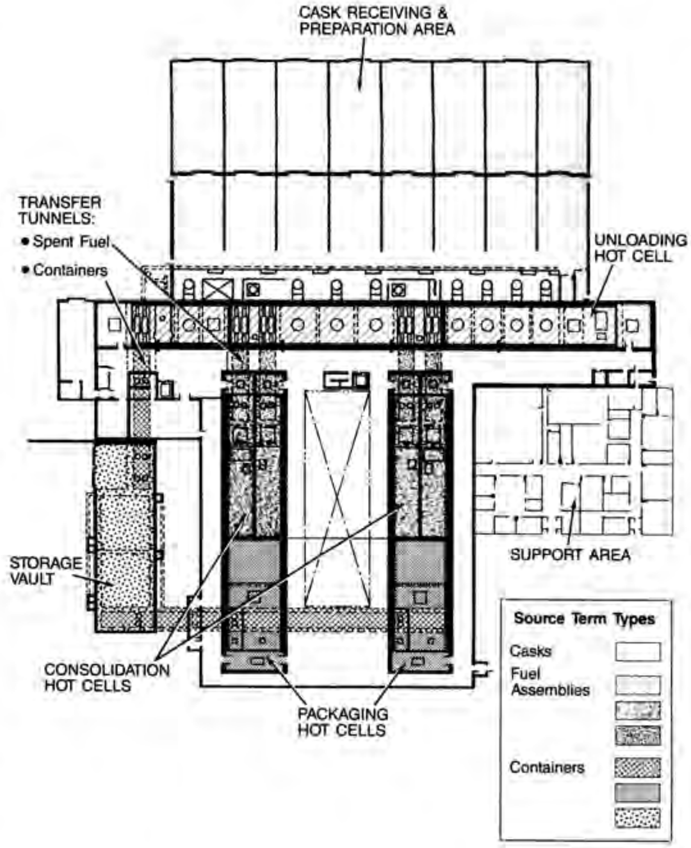

Fig. 1. Six System Areas Within the Reference Main Waste Handling

casks and underground waste transporters. The intermediate events included breach of the primary confinement barriers (e.g., fuel cladding, DHLW canisters, containers and shipping casks), the loss of normal and/or standby high-efficiency particulate air (HEPA) filter systems, and fires subsequent to collision accidents. On the basis of these design-specific accidents, 12 event trees comprising 58 scenarios were developed from 12 internal events. Reference 1 discusses why these 12 initiating events were representative of all credible internal initiating events.

Accident scenarios initiated by earthquakes were developed using similar event tree analyses. From a comprehensive list of external events, a seismic event was selected as the bounding external event that may potentially lead to a significant offsite radiological release. Other external events, such as flood, extreme wind, loss of offsite power, and sandstorms, were estimated to not result in any significant offsite radiological releases, but these initial estimates need to be assessed further. Because of the lack of data and detailed design information, scenarios initiated by military activities or internal fire hazards were not analyzed in this initial study.

Earthquake-initiated accident scenarios included falling structural objects, falling overhead cranes, crane load drops, collisions involving shipping casks and underground waste transporters, and mechanical handling impacts. The intermediate events included breach of the primary waste confinement barriers, the loss of normal and/or standby HEPA filter systems and fires subsequent to collision accidents. Based on these types of scenarios, 9 event trees comprising 91 scenarios were identified for the seismic external initiating event.

These 58 internally initiated and 91 externally initiated accidents are representative of those which could occur in the repository facility and which might be expected to result in upper bounding values for any potential offsite releases of radioactive materials. As the design of the facility continues, the detailed features of the accident scenarios will be refined, but the results are not expected to change significantly. Therefore, the 149 accident scenarios identified constitute a public health and safety design basis for a repository facility during preclosure operation.

Assessments of Probability of Occurrence

The probabilities of occurrence of initiating events and intermediate events were estimated by two different methods: (1) searches for previously published applicable data and (2) engineering judgments provided by experienced facility designers when applicable data were not available. Four ranges were used to categorize the engineering judgment: likely, anticipated, unlikely, and incrediable.

In this study, a likely event is one that occurs once a year to once every 10 years; an anticipated event is one that may occur any time between once every 10 years and once in the facility lifetime (i.e., 10 to 50 years); an unlikely event is one that will probably not occur in the lifetime of the facility, but its occurrence cannot be disregarded, and it must be treated explicitly in the facility design bases; and an incredible event (such as a meteorite falling on the facility) is one whose probability is so low that it is assumed not to occur and therefore does not need to be considered in the design or analyzed in any great detail.

Extremely conservative values were frequently assumed in estimating these probabilities. For example, the probability of a cladding failure in a bare fuel assembly caused by a drop impact was assumed to be 1. The probability of the loss of an HVAC system due to an earthquake was estimated by assuming that the HVAC system was designed for Uniform Building Code (UBC) standards and not Seismic Category I standards. Likewise, the probability of an overhead crane falling in a design basis earthquake was estimated by assuming that the crane was designed without devices to mitigate such seismic failures. However, such design practices would not be followed in the actual design of nuclear facilities.

Because of the lack of data and detailed design information, fault tree analyses were not used to determine the failure frequencies of either initiating or intermediate events. As more data and detailed design information become available, fault tree analyses can be performed.

The probabilities of occurrence for scenarios initiated by the internal events and by the seismic event were summarized in 21 event trees. The results indicated that the probability is very low for any accident scenario that may result in significant offsite dose consequences. This is discussed below.

Assessment of Dose Consequences

Offsite radiation doses to a maximally exposed member of the general public were calculated for each accident scenario. The results depend on the type and quantity of source term release and on the transport mechanisms. This initial study assessed only air pathway mechanisms. Before offsite dose consequences can be calculated with conventional transport models, the quantity of radioactive material available for transport and dispersion must be established for each accident scenario. No regulatory guidance exists for source terms for repository accidents involving spent fuel assemblies or DHLW glass canisters. Therefore, as part of this study, a new approach to the estimation of source terms was developed.

A technical basis for estimating the primary air pathway source terms was derived from (1) a laboratory-scale experimental program (3,4,5) at Argonne National Laboratory (ANL) which characterized the impact fracture responses of brittle glass and ceramic simulated waste forms; (2) NRC Regulatory Guide 1.25 (9), which provided assumptions for evaluating potential radiological releases from spent fuel handling accidents at storage facilities; and (3) an experimental program at Oak Ridge National Laboratory (ORNL) (7,8), which characterized the releases of radioactive materials from the spent fuel pellet-cladding gaps in high-temperature steam and dry environments. Further evaluations of the types of analyses outlined below are required to verify the proposed technical basis.

The ANL work established that the fracture particulates generated in mechanical impacts of unirradiated UO_2 and simulated DHLW bare specimen obeyed a lognormal particulate size distribution (3,4,5). The data indicated that either UO_2 or DHLW glass specimen resulted in approximately the same air- borne particle size distribution for the same impact energy. The mass quantities of particulates less than 10 m in size and the total surface areas of the particulates were found to correlate linearly with the impact energy divided by the total specimen volume. These ANL observations and the theoretical modeling approach provide a sound basis for assuming in this study a mass fraction of 2×10^{-4} for either the UO_2 particulates within a spent fuel assembly or the glass particulates within a DHLW canister with sizes less than 10 m. This fraction corresponds to a free fall drop impact of approximately 30 to 40 feet. Thus, for a PWR fuel assembly with a UO_2 mass of 0.523 MTU, the mass of fracture particulates less than 10 m is assumed to be 100 grams. A DHLW canister containing 1,590 kg of glass would result in 318 grams of particulates of glass less than 10 m size.

These ANL particulate correlations (3,4,5) were based on bare specimens of UO_2 and DHLW glass brittle materials, so the absorption of impact energy in the ductile metallic fuel assembly components and in the DHLW canister was not considered. Such energy losses reduce the mass fraction of particulates and need to be taken into account to develop more realistic source terms in the future. The energy absorption effect should be more important for spent fuel assemblies than for the DHLW canisters. Full-scale tests at Battelle Pacific Northwest Laboratory (PNL) seemed to confirm the ANL correlations for DHLW (6,7). Impact test characterizations (6) of full-scale simulated DHLW canisters in 30-foot free fall drops onto unyielding surfaces indicated that the mass fraction of particulate sizes less than 10 m was 1×10^{-4}. This value is consistent with the ANL work (3,4,5) and the value of 2×10^{-4} assumed in this study. It should be noted that although fracture particulates were generated within the PNL test canisters, the stainless steel DHLW canisters did not breach upon impact and, in fact, did not leak helium in leak tests performed after their impacts (6).

No published data for fuel assembly impact tests are currently known. Because of the large quantity of metal components of a fuel assembly and the varying methods used to hold the individual spent fuel rods in the assembly, a significant reduction in the available impact energy is to be expected. According to the ANL tests (3,5), any reduction in impact energy will result in the production of fewer particulates of airborne sizes. This study assumes that only 20 percent of the fracture particulates predicted by the bare specimen ANL impact model will be generated in impacts of actual spent fuel assemblies. No such reductions were assumed for the DHLW canisters (5); there- fore, the mass fractions for the airborne source terms for spent fuel and DHLW canisters in drop impact scenarios were assumed to be 4×10^{-5} (or 20 grams of UO_2/assembly) and 2×10^{-4}, respectively.

To be both consistent and realistic in the development of the source terms, it must be noted that the UO_2 spent fuel and DHLW glass is generally surrounded by multiple metallic confinement barriers such as cladding, canisters, containers, and (in some scenarios) shielded casks. Gross failure of these confinement barriers resulting in the release of all of the fracture particulates less than 10 m is not realistic for any proposed impact scenario. It is assumed that some cracks (but not gross brittle fracture failures) will be formed in the metallic confinement components, so that most of the respirable particulates are retained. In this study, 10 percent of the particulates less than 10 m was assumed to escape from each metallic confinement barrier. This escape factor of 0.1 was applied as many times as there were redundant metallic confinement barriers. For example, if spent fuel assemblies were in sealed containers in the underground shielded transporter cask, the airborne particulate mass fraction source term of 4×10^{-5} was reduced by a factor of $(0.1 \times 0.1 \times 0.1)$ to 4×10^{-8} of the total mass inventory of UO_2. For a DHLW canister in an overpack, the mass fraction for the source term is 2×10^{-6}.

The above methods were used to quantify the pulverized spent fuel and DHLW glass particulates. Further study is needed to demonstrate conclusively that the approach discussed above is technically sound and to resolve and quantify issues such as irradiation effects on UO_2 spent fuel matrix and the zircaloy cladding and the actual responses of relatively ductile canisters and massive shielding casks. As an example, the ANL data were based on unirradiated UO_2. A reduction in particulate sizes below 5 m was reported in impact tests of spent fuel irradiated to 32,000 MWd/MTU. The reduction was reported as due to intergranular fracture of the irradiated UO_2 specimen, which had an average grain size of 6-7 m (8). In contrast, the ANL data were observed as transgranular fracture of the UO_2 specimen with an average grain size of 9 m but with no reduction in particulate sizes below 5 m (8). This discre- pancy in the limited experimental observations and literature requires resolution.

To quantify the volatile radionuclide and pellet-cladding gap releases, fractions recommended for use in accident analyses were compiled from the literature (9,10,11). These fractions of the total avail- able inventory include 0.3 for ^{85}Kr, 0.1 for ^{3}H, 2.8×10^{-4} for Cs isotopes and 1×10^{-6} for ^{90}Sr (9,10,11). The values for Cs and Sr were deter- mined for simulated high-temperature steam accident environments associated with reactor accidents (10, 11), while the other values were for fuel handling accidents. Because of the lack of known data, no contributions to the source terms from small UO_2 particulates due to previously pulverized UO_2 fuel pellets from reactor operations were considered.

To calculate specific offsite dose consequences, the number of spent fuel assemblies and DHLW canisters, and the source terms of airborne fractions were evaluated for each scenario in the 21 event trees. No credit was taken in the analyses for any retention or plateout of airborne particulates within the hot cells or the ductwork of the ventilation systems. Fuel assemblies were assumed to be 10-year-old PWR assemblies with a burnup of 33,000 MWd/MTU. Wind speeds were assumed to be 1 m/s, and Pasquill's F Stability (moderately stable) conditions were assumed. With the exception of the bridge collision accident scenario, the point of offsite exposure used for the dose calculations was 5 km; this corresponds to a representative distance from the proposed surface facilities to the nearest site boundary. The bridge accident scenarios used a distance of 3.4 km.

Dose calculations were performed using (1) X/Q values obtained from Regulatory Guide 1.25 (Ref. 9) and Regulatory Guide 1.3, (Ref. 12); (2) immersion 50-year dose conversion factors obtained from Regulatory Guide No. 1.109 (Ref. 13); (3) internal 50-year dose equivalent conversion factors obtained from Regulatory Guide 1.109 (Ref. 13), NUREG/ CR-0150, Volume 3, (Ref. 14), and NUREG/CR-0172 (Ref. 15); and (4) the radionuclide inventory (Ci/MTU) of the spent fuel was obtained from ORNL/TM-6008 (Ref. 16). With the methodology of Refs. 17 and 18 and the lognormal particulate size distribution parameters in Refs. 3 and 4, a factor of 20 reduction in the quantity of airborne particulates was applied to account for dry deposition of airborne particulates prior to reaching the site boundary. This deposition factor was applied only for scenarios in which no HEPA filtration was assumed. For scenarios using HEPA filtration systems, credits of a decontamination factor (DF) of 10,000 for a series of two or more HEPA filter systems were assumed.

Based on the above, the doses to a maximally exposed individual located offsite were calculated for the airborne pulverized spent fuel, DHLW particulates, ^{85}Kr, ^{3}H, and the gap releases of Cs and Sr isotopes. Offsite dose consequences were computed for each of the scenarios in the 21 event trees. These results indicated that no dose consequence exceeded 1,100 mrem.

CONCLUSIONS

The calculated doses for accident scenarios initiated by internal events and an external seismic event were estimated for 21 event trees. For the majority of the 149 accident scenarios, the resulting doses from releases of radioactivity to which a member of the general public is potentially exposed was either zero or below 50 mrem. Forty-five of the scenarios indicated no exposure to the offsite public. Only 13 accident scenarios occurred with a dose consequence of more than 50 mrem and an associated probability of occurrence greater than 5×10^{-9}/yr. These 13 scenarios are summarized in Table I.

TABLE 1

Summary of Evaluated Accident Scenarios in Waste-Handling Building with Dose Consequences Greater than 50 mrem and Probabilities of Occurrence Greater than 5×10^{-9} per year

		Scenario Consequences	
Location (System area)	Initiating Event for Scenario	Offsite Dose (mrem)	Probability* (per yr)
Cask receiving & preparation area	o Cask drops	340	5×10^{-6}
	o Earthquake--crane falls on cask	340	5×10^{-8}
	o Earthquake--cask transporter falls onto unsealed cask	290	5×10^{-8}
Unloading hot cell	o Fuel assembly drops	110	1×10^{-8}
	o Earthquake--piece of structure falls on fuel assemblies	110	5×10^{-7}
	o Earthquake--equipment falls on fuel assemblies	110	1×10^{-6}
Consolidation hot cell	o Earthquake--piece of structure falls on fuel assemblies	110	5×10^{-7}
	o Earthquake--equipment falls on fuel assemblies	110	1×10^{-6}
Packaging hot cell	o Unsealed container drops	1,110	1×10^{-9}
	o Earthquake--piece of structure falls on unsealed container	330	5×10^{-7}
	o Earthquake--piece of equipment falls on unsealed container	1,110	1×10^{-6}
Hot cell transfer tunnels	o Earthquake--piece of structure falls on fuel assemblies	200	5×10^{-7}
Storage vault	o Sealed container drops	230	3×10^{-8}

* The probabilities of scenerios in this table includes an assumed annual probability of the seismic initiating event equal to 5×10^{-4}.

As shown in Table I, for the 13 accident scenarios that may potentially lead to an offsite dose larger than 50 mrem, the associated probabilities of occurrence are very low, less than about 5×10^{-6}/yr for scenarios initiated by internal events and less than about 1×10^{-6}/yr for scenarios induced by an earthquake. It is concluded that, because of their low probabilities of occurrence, these accident scenarios are not likely to occur at an operating repository.

The maximum offsite dose consequence to which a member of the general public could be exposed as a result of the accident scenarios identified in this study was about 1,100 mrem. However, the probability associated with these scenarios was about 1×10^{-6}/yr, which makes the accident not credible or at least not likely to occur. Several scenarios were found to occur more frequently during preclosure repository operations. These scenarios included the malfunction of fuel consolidation machines or an overhead crane drop accident in various hot cells, resulting in a release of airborne radioactive materials while the HVAC system remained in operation. The resulting offsite doses, however, were low (about 1 mrem or less). Should the HVAC system also fail at the same time, the doses will be increased as discussed above; however, the probability of both events actually occurring is extremely low.

This initial assessment indicates that the design and operation of a repository at Yucca Mountain will not result in any significant offsite releases of radioactive materials. These results are generally consistent with other earlier assessments for geologic repositories (19,20). This conclusion is based upon the 40-year period of assumed repository opera- tions and the calculated low dose consequences and low probabilities associated with any of the identified repository accident scenarios. However, the effects, if any, on the initial conclusions because of the exclusion of potential scenarios for retrievability, decommissioning, and performance confirmation operations need to be examined. Details of this study are reported in Ref. 1.

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