INITIAL Q-LIST FOR THE PROSPECTIVE YUCCA MOUNTAIN REPOSITORY BASED ON ITEMS IMPORTANT TO SAFETY AND WASTE ISOLATION[a]

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ABSTRACT

A method for identifying items important to safety based on a probabilistic risk assessment approach was developed and implemented for the conceptual design of the Yucca Mountain repository. No items were classified as important to safety; however, six items were classified as potentially important to safety. These were the shipping cask, the cranes and the truck or rail-car vehicle stops in the cask receiving and preparation area, the hot cell structure of the waste packaging hot cells, the cranes in the waste packaging hot cells, and the waste-handling building fire protection system. In addition, a method for identifying items important to waste isolation was developed and implemented. Two hydrogeologic units of the Yucca Mountain site were classified as important to waste isolation: the Calico Hills nonwelded zeolitic unit and the Calico Hills nonwelded vitric unit. The preliminary Q-list for the Yucca Mountain repository is comprised of the two units of the site classified as important to waste isolation and contains no items important to safety.

INTRODUCTION

Based on the Nevada Nuclear Waste Storage Investigations Project Site Characterization Plan Conceptual Design Report (SCP-CDR), (1) a study was conducted to determine which items (i.e., systems, structures, and components) are important to safety or to waste isolation and should be placed on the Q-list. The Q-list contains those items and activities that are essential to preventing accidents or mitigating accident consequences during the pre-closure operations phase of a repository when waste is being emplaced. The Q-list also contains those items and activities (both preclosure and post-closure) that are important to isolating waste after permanent closure of the repository. Items on the Q-list are subject to Title 10 Code of Federal Regulations (CFR) Part 60 subpart G quality assurance (QA) requirements. These items and activities include both natural (specific geologic units of the site itself) and engineered barriers, and are classified as to whether or not they are relied on to meet the numerical performance criteria in the proposed 10 CFR 60 (that incorporates the 40 CFR 191 requirements for radioactive releases to the environment.

Items considered in this study for the pre-closure period were limited to major structures, systems, and components. For the postclosure period, the site and engineered barriers were assessed. Repository activities such as site characterization, facility operations or procedures, and performance confirmation were excluded from this initial study. Military activities at the adjacent Nevada Test Site (NTS) were also excluded from consideration. As the repository design progresses, site characterization-activities proceed, regulatory positions develop, performance-assessment methods evolve, and more information becomes available, the initial Q-list will be updated in a continuing assessment program.

This paper summarizes methods developed and conclusions reached in identifying items in the SCP-CDR that are important to safety. A brief description of the methods used to determine items important to waste isolation is also presented. To develop the necessary methods, criteria, and definitions commensurate with the available design and site details and consistent with Nuclear Regulatory Commission (NRC) regulations were required. This paper discusses these criteria and definitions.

ITEMS IMPORTANT TO SAFETY

Items important to safety are defined by the NRC as "those engineered structures, systems, and components essential to the prevention or mitigation of an accident that could result in a radiation dose to the whole body, or any organ, of 0.5 rem or greater at or beyond the nearest boundary of the unrestricted area" (2). This definition provides a single criterion, a dose specification, for identifying items important to safety. Another criterion, implicit in the definition, is that the accident must be credible. "Credible accident" necessitates a probability criterion for identifying items important to safety. For purposes of identifying items important to safety, a credible accident has been defined by DOE as a scenario with a probability of occurrence greater than 1×10^{-5} per year (3). Therefore, two criteria have been identified that are used in identifying items important to safety:

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- o the dose criterion--an accident must cause an offsite radiation dose of 0.5 rem or greater to merit consideration in identifying items important to safety; and
- o the probability criterion--an accident must have a probability of occurrence greater than 1×10^{-5} per year to be credible and therefore considered in identifying items important to safety.

If an accident satisfies both of these criteria, then one or more systems, structures, or components essential to the prevention of that accident or essential to the mitigation of the offsite dose consequences of that accident are important to safety. It is important to note that the dose criterion is not a limit for accident consequences; it is a threshold for identifying items important to safety. A dose limit for accidents at a geologic repository has not yet been established in the federal regulations or within the Nevada Nuclear Waste Storage Investigations (NNWSI) project.

Method for Identifying Items Important to Safety

A method for identifying items important to safety was developed in accord with current Department of Energy (DOE) and NRC guidance (4,5). The method is based on a probabilistic risk assessment (PRA) approach as described in Refs. 6 and 7. Figure 1 illustrates this method. A detailed report of the analyses performed with this method and their results is contained in Appendices F and L of Ref. 1. The following is a summary of analyses performed and results obtained in the implementation of the method to identify items important to safety.

The first step (Fig. 1) is the development of facility and systems models. To facilitate the analyses used as the basis for determining items important to safety, the structures, systems, and components of the reference design (1) were grouped into three general areas: Area I, the geologic repository operations area (which includes the underground emplacement area); Area II, the general support area; and Area III, the mining development area (the mining development area is separate from the underground emplacement area). No radioactive waste receipt, handling, preparation, or emplacement operations are performed in Areas II and III; all of these functions are performed in Area I. Therefore, no significant offsite releases can be initiated in Area II or Area III. Area I was further subdivided into nine system areas: (1) the access area (including a bridge), (2) the receiving and inspection area, (3) the cask receiving and preparation area, (4) the unloading hot cell area, (5) the spent fuel and container transfer tunnels in the waste-handling building, (6) the consolidation hot cells, (7) the packaging hot cells, (8) the surface storage vault, and (9) the underground emplacement areas (including the ramp). A detailed discussion of the facility and system modeling effort is contained in Refs. 1 and 8.

Once the reference design for the analysis has been established and the facilities and systems modeled, the next step (Fig. 1), identification and preliminary screening of initiating events, can be performed. Initiating events are divided into two categories: internal initiating events and external initiating events. Different methods were used to identify the two types of initiating events.

A set of internal initiating events that could cause significant radiological releases to the environment was identified and screened using two different methods. In the first method, a standardized accident-scenario survey form was developed and distributed, along with the facility drawings, to the panel of engineers for evaluation of the reference facility design. The members of the panel studied the site plans, building layouts, waste-handling operations, and material block flow diagrams and

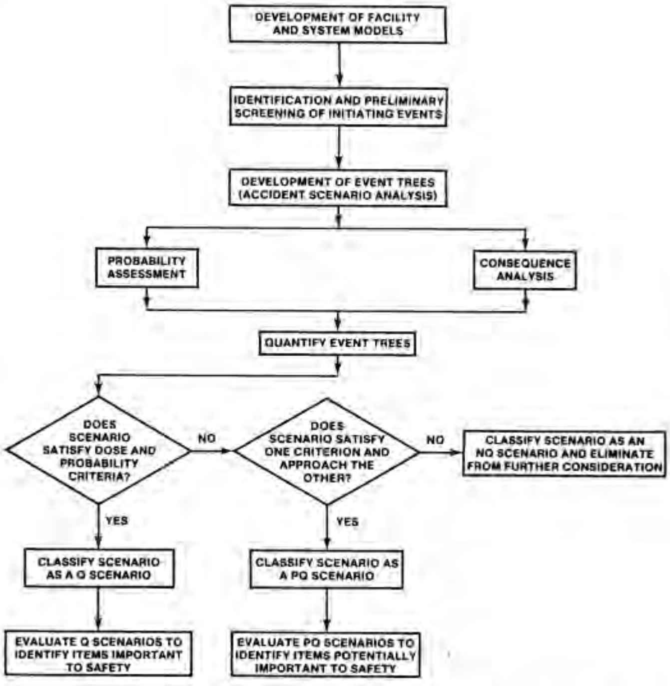

Fig. 1. Method for Identifying Items Important to Safety.

completed a survey form. The panel was instructed to consider, one at a time, each of the nine system areas of the facility and to develop scenarios that could cause significant offsite radiological releases. Review of the completed survey forms revealed that two general categories of accidents were judged to have the potential for significant offsite releases: (1) onsite accidents involving the moving of the waste around the site and (2) accidents involving the handling of the waste in the waste-handling building.

In the second method, interaction matrices were used as a systematic way to develop potential initiating events for each of the nine system areas of the repository facility. These matrices identified major structures, systems, and components in each area and used these items as row and column designators. Each row in the matrix was then analyzed on a column-by-column basis to identify possible interactions between items. For each possible interaction, potential accident scenarios were developed. Interactions that obviously would not result in a radiological release were excluded from further consideration.

The results of the above processes were compiled, compared, and analyzed to develop a set of design-specific internal initiating events for the Yucca Mountain repository. The internal initiating events chosen for further event tree development and quantification are shown in Table I.

External events that could cause significant radiological releases to the environment were identified and screened using the following method. First, a comprehensive checklist of external events, which included both natural and human-induced phenomena, was compiled based on literature surveys of previous safety analyses and probabilistic risk assessments for nuclear facilities (7). This checklist was circulated to the panel of engineers for review to eliminate events either not applicable to the site-specific region near Yucca Mountain or considered to be insignificant with respect to an offsite radiological dose.

A set of screening criteria consistent with NRC's "PRA Procedures Guide" (7) was developed for use by the engineering panel in screening the potential external events for subsequent analysis. These screening criteria include the following:

- applicability to the Yucca Mountain site region (e.g., coastal erosion and hurricanes are not applicable);
- relevancy to preclosure operations (e.g., inadvertent future human intrusion such as archeological exhumation is not relevant);
- likelihood of causing any significant radiological releases to the environment (e.g., drought or frost are not likely to cause significant radiological releases);
- probability or frequency of occurrence (e.g., the probability of a meteorite heavier than 2 lb striking the facility is 10^{-9} per year and thus is not considered); and
- whether the impact of the external event is well within the plant design basis (e.g., subsidence, thermal loading, waste/rock interaction). Certain events such as earthquake, flood, and extreme wind are not excluded since their magnitude may exceed the design basis.

The above screening criteria were applied to each of the external events on the comprehensive list. The following external events were selected for more detailed evaluation before continuing with event tree development:

- flood,
- earthquake,
- extreme wind,
- sandstorm,
- military facility accident/activities, and
- loss of offsite power.

Table I

Internal Initiating Events Chosen For Further Analysis and Event Tree Development

System Area	Internal Initiating Event
o Access area	o Train falls off bridge
o Receiving and inspection area	o Train collision
o Cask receiving and preparation area	o Crane drops cask
o Unloading hot cell	o Crane drops fuel assembly
o Unloading hot cell	o Crane drops DHLW canister
o Unloading hot cell	o Malfunction of container cutting machine
o Consolidation hot cell	o Malfunction of consolidation system
o Transfer tunnel (note that transfer tunnels are in the surface facilities)	o Transfer/storage cart accident
o Packaging hot cell	o Crane drops unsealed container
o Surface storage vault	o Container Transfer Machine (CTM) drops consolidated fuel container
o Surface storage vault	o CTM drops DHLW container
o Underground and emplacement area	o Runaway transporter

Of these six external initiating events, only the earthquake was selected for subsequent event tree development. The screening analyses showed that flood, extreme wind, sandstorm, and loss of offsite power could not lead to offsite releases of radioactivity. Military facility accidents and activities were not addressed in the initial analysis, but it should be noted that underground nuclear tests are considered to be enveloped by the earthquake scenario.

The next step in identify items important to safety (Fig. 1) is the development of event trees. Event trees are used to logically and systematically depict the various accident scenarios developed through the occurrence of intermediate events following a single initiating event. Intermediate events are events that occur after the initiating event and continue an accident scenario. This method has been used extensively in PRAs for nuclear power plants (7). The preliminary state of the repository facility design did not permit development of detailed event and fault trees (as used in as-built or finally designed nuclear facilities); however, event trees were developed for each internal and external initiating event chosen for further analysis. The major considerations used in developing the events trees and intermediate events were

- radioactive particulate release barriers (e.g., cask, cladding, containers, canisters),
- radioactive release mitigators (e.g., heating, ventilating, and air conditioning (HVAC) filter systems, monitoring systems/controls), and
- radioactive release promoters (e.g., volatilization by fires).

The next step (Fig. 1) is a probability assessment of the accident scenarios developed in the previous step. A frequency analysis of an event tree requires evaluating the frequency of occurrence of the initiating event as well as the probability of each intermediate event in the event tree. Frequency assessments of all events were accomplished by two different methods: (1) conducting a literature search for previously published data and (2) surveying a panel of engineers familiar with nuclear facility designs. The panel's assessments were based largely on their insights as to what items will eventually comprise the system being evaluated for its failure frequency and to what specifications the system will be designed.

The literature search involved the review of previous studies and reports to identify and estimate failure rates of systems and components and the frequencies of occurrence of internal events. From these studies and reports, a list of frequencies was developed for events involving systems and components judged to be similar to the Yucca Mountain repository conceptual designs. The categories of internal events searched for in the literature included the following: (1) transportation accidents for trucks and trains, (2) electrical components or systems failures, (3) mechanical components and systems failures, and (4) materials-handling equipment failures.

In surveying the panel, a list of initiating and intermediate events was supplied and the engineers' frequency estimates were compiled. The list included events for which data were also found in the literature. Wherever possible a comparison of published data and the engineering judgment was made to determine the level of conservatism in the engineering judgment of the panel. The range of estimates for each event was assessed and a single representative value established. The engineers were asked to assign an event to one of four categories: (1) anticipated, (2) unlikely, (3) very unlikely, and (4) not credible. "Anticipated" designated something that is going to happen in the range of once per year to once per 10 years. "Unlikely" meant a frequency between once per 10 years and once in the lifetime of the facility (i.e., 10 to 50 years). "Very unlikely" meant the event will probably not occur in the lifetime of the facility but cannot be disregarded; therefore, the event must be treated explicitly in facility design. "Not credible" meant the probability is so low (i.e., about 10^{-5} to 10^{-6} per year or lower) that its occurrence is not considered possible and, therefore, does not need to be explicitly addressed in facility design.

While the probability assessment is being performed, a consequence analysis for each event tree is carried out. To assess the dose received by a member of the public due to a release of radioactivity, the types and quantities of radioactive materials available for release and transport to the environment must be known; these are discussed in detail in Ref. 1 and 8. During an accident that could lead to an offsite release of radioactive materials, certain pathways are more important than others. The airborne pathway is the most important for exposure of the public from the postulated accidents at the Yucca Mountain repository. If radioactive materials in the form of suspended particulates and gases escape from hot cells, containers, canisters, spent fuel cladding, or shipping casks, they could be transported offsite by the wind. Transport of radionuclides is affected by factors such as rise due to buoyancy, changes in wind velocity, the presence or lack of precipitation, building wake effects, dry deposition, etc.

Once the inventory of radioactive material potentially released into the environment is estimated, the dose to an individual at the site boundary can be calculated for the airborne pathway using a dispersion model. Doses were calculated for the maximum exposure to an individual in the passing plume. The doses were calculated as the total of the external exposure from the passing cloud and the internal exposure from inhalation of radionuclides in the cloud. Since each accident scenario or pathway through an event tree will have its own unique set of assumptions and release fractions, it was decided that it would be more efficient to develop a set of generic dose consequence curves for the spent fuel and defense high-level waste (DHLW) than to recalculate dose consequences for each release. To determine the dose consequence for each scenario, adjustments and scaling of the two curves were performed. It should be noted that all dose consequences were of 50-year dose commitments (i.e., the dose that an individual would receive over 50 years from radionuclides that are retained in the body from the initial exposure).

After performing the probability and consequence assessments, this information is used to quantify the event trees. Event trees are quantified by assigning probabilities to the initiating event and to each intermediate event. Release fractions for the radioactive materials present in the scenario also are estimated according to the structure, system, or component and the specific scenario being evaluated. Reference 1 details the development of these release fractions. The probabilities of individual accident scenarios were calculated as the product of the probabilities of the initiating event and the intermediate events. Dose consequences were assessed as described above.

Although not shown in Fig. 1, the dose consequences and probabilities used to quantify the event trees were reevaluated to ensure that adequate conservatism had been applied to accommodate the major uncertainties in the analyses. Although very few changes were made, there were some simple rules applied to the probabilities and to one of the release mechanisms used in calculating dose consequences. The first of these rules was that in cases where probabilities had been assigned based on engineering judgment alone, these probabilities were increased by a factor of 10. An exception to this rule occurred when an event had a probability of 0.1 per year; the factor of 10 was not applied because this would make the event a mathematical certainty. In such a case, the probability was not changed. The probability of the failure of the entire HVAC system was increased by a factor of 10 in all cases. This was done because generic failure-probability data were judged not applicable to the HVAC system envisioned for the repository, and initial estimates to the HVAC total system failure were base solely on engineering judgment. If a probability was obtained from published data, it was considered reliable, and no changes were made. With regard to the modification of one release mechanism, the assumed deposition of particulates in the environment was decreased to account for the uncertainty in particle-deposition mechanisms. A discussion of these changes and the bases for them is contained in Ref. 1. The final results of the reevaluated probability and consequence assessments for all scenarios resulting in nonzero dose consequences are shown in Fig. 2.

Figure 2 is a scatter diagram showing the relationship of the scenario probabilities and consequences to the dose and probability criteria. Of 149 scenarios developed on events trees and quantified, 45 had zero dose consequences; therefore, 104 scenarios are shown in the diagram. Of these, only 3 scenarios exceeded one of the important-to-safety screening criteria and were close enough to the other criteria to be of concern. As will be seen later, these scenarios are classified as potentially important to safety. There were 3 other scenarios which came close to both criteria but were did not exceed either one. These scenarios will be looked at closely in future analyses.

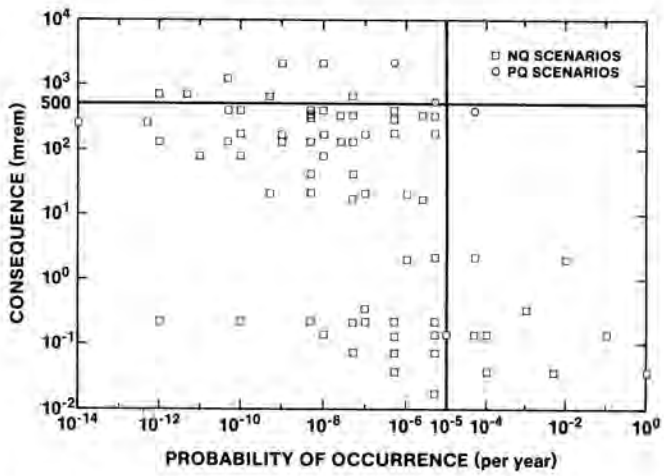

Fig. 2. Scatter Diagram of Scenarios with Nonzero Dose Consequences.

As indicated in Fig. 1, the next step in identifying items important to safety is to screen the accident scenarios to identify those satisfying both the dose and probability criteria. In the screening, the probability and the calculated offsite public dose were compared with the dose and probability criteria. If, and only if, an accident scenario passes both screening criteria, the accident scenario is classified as a "Q scenario." Q scenarios are then further analyzed to determine which of the structures, systems, or components involved in the scenario are important to safety. As mentioned earlier, structures, systems, or components are important to safety if they are essential to either the prevention of the accident or the mitigation of the consequences. Scenarios that do not satisfy the two criteria are classified as either a "Non-Q scenario" (NQ scenario) or a "Potential-Q scenario" (PQ scenario). All NQ scenarios are eliminated from further consideration in identifying items important to safety.

Any scenario not classified as a Q scenario, but, as further study and design take place, judged to have a reasonable potential to be upgraded to a Q scenario, is classified as a PQ scenario. Structures, systems, and components involved in

PQ scenarios require further analysis to determine which items should be placed on a "potentially important to safety list." Such a list is consistent with DOE guidance (4). Specific criteria do not exist to identify PQ scenarios; therefore, engineering judgment was used to identify all PQ scenarios. A scenario was judged to be PQ if, in the opinion of the panel, there were sufficient uncertainties in the present analysis to warrant further investigation and analysis to ensure that the scenario would not exceed the probability and dose criteria.

Determination of Items Important to Safety

As shown in Fig. 1, the next step is to determine which systems, structures, or components are important to safety or potentially important to safety. Once a scenario is classified as a Q scenario or a PQ scenario, that scenario is further analyzed to determine which of the items involved in the scenario should be placed on the list of items important to safety or items potentially important to safety. Further analysis of the scenario requires the evaluation of the systems, structures, and components involved in the scenario to determine what role the items play in the scenario. Items whose failure causes the loss of essential consequence-mitigation processes or whose failure directly causes the release of radioactive materials are classified as important to safety or potentially important to safety, depending on which type of scenario (Q or PQ) is being evaluated. As can be seen from Fig. 2, there were no Q scenarios; therefore, there were no items to be classified as important to safety. These results are consistent with previous studies whose conclusions support the finding that there are no items important to safety for the Yucca Mountain repository (9,10). These results are also supported by statements made by the NRC in Ref. 11. The NRC determined that doses to the public from the most severe accidents at any independent spent fuel storage installation (ISFSI) were far below 0.5 rem at a distance much closer than the boundary of the the Yucca Mountain repository. The surface facilities at the Yucca Mountain repository would be very similar to an ISFSI in purpose and operation. Although these doses were calculated for wet storage, the NRC states that "the staff determined that the release from dry cask storage is of a comparable magnitude to that for wet storage" (11).

Even though there were no Q scenarios, three PQ scenarios were identified. The relationship of these PQ scenarios to the dose and probability criteria may be seen in Fig. 2. Note that these scenarios exceed either the dose or the probability criterion and are very close to the other criterion. Further evaluation of the PQ scenarios resulted in the classification of certain systems, structures, and components involved in those scenarios as potentially important to safety. These items were the shipping cask, the cranes in the cask receiving and preparation area, the packaging hot cell structures, and the cranes in the packaging hot cells. In addition to those items identified through the three PQ scenarios, there were two items that the NNWSI project felt should be on the list of items potentially important to safety that were not involved in any scenario of any event tree developed during this study. The first of these items is the truck or rail-car vehicle stop in the cask-receiving area of the waste-handling building. The failure of this piece of equipment if impacted by a vehicle carrying a shipping cask was judged to be similar in consequence to the other PQ scenario in the cask receiving area. The second item is the fire-protection system in the waste-handling building. No detailed analyses have been performed on fire scenarios in the waste-handling building. However, it was felt that because a fire involving radioactive material is an important dispersion promoter, the fire-protection system in the waste-handling building should be added to the list of items potentially important to safety until future analyses resolve this issue. These two items, along with the rest of the items classified as potentially important to safety, will be the object of future analyses to verify that these items are not important to safety. Table II is a list of items classified as potentially important to safety along with the responsible initiating event, the location of the item, and the probability and consequence of the PQ scenario from the which the item was identified.

ITEMS IMPORTANT TO WASTE ISOLATION

Unlike the term "important to safety," the term "important to waste isolation" is not defined in current NRC regulations or DOE guidance. Therefore, the first step in identifying items (barriers) important to waste isolation is to define "important to waste isolation." The next two steps are to develop and then apply systematic methods for identifying the repository systems, structures, and components that meet the definition. The remainder of this paper discusses the NNWSI project definition of "important to waste isolation." The methods developed for identifying items important to waste isolation and the results of the application of those methods are briefly presented, and reference is made to a more detailed discussion. Barriers determined to be important to waste isolation will be placed on the Q-list along with items important to safety.

Definition of Barriers (Items) Important to Waste Isolation

The basis for defining "important to waste isolation" is the regulations that govern waste isolation. In 10 CFR 60.112, the NRC specifies the overall system performance objective for the repository, requiring assurance that "releases of radioactive material to the accessible environment following permanent closure conform to such generally applicable environmental standards for radioactivity as may have been established by the Environmental Protection Agency with respect to both anticipated processes and events and unanticipated processes and events." The Environmental Protection Agency (EPA) has issued those standards in Title 40 CFR Part 191 (12). The long-term requirements for performance of the repository are given in 40 CFR 191.13, Containment Requirements. This regulation requires a reasonable expectation that the disposal system will hold within stated limits the total quantities of radionuclides that reach the accessible environment over a period of 10,000 years. The requirements are not, however, simple statements of release limits; the quantities that may be released depend on the likelihoods of the releases.

In 10 CFR 60.113, the NRC specifies performance objectives for certain particular barriers, including engineered barriers. These objectives do not have to be satisfied for unanticipated processes and events. Furthermore, 10 CFR 60.113(b) allows the NRC to specify, on a case-by-case basis, other numerical values for the objectives for particular barriers provided that the overall system performance objective, as it relates to anticipated processes and events, is satisfied. In other words, 10 CFR 60 provides for flexibility in the numbers that the particular barriers must satisfy, but the overall system objective must be met as it is written. It is evident that the fundamental requirement is to view the repository as a system and not as discrete, independent pieces.

Neither 10 CFR 60 nor 40 CFR 191 provides a definition of "important to waste isolation" or any criteria to define the phrase. The NNWSI Project has therefore adopted a definition of items "important to waste isolation" as

> those natural and engineered structures, systems, and components essential to the prevention or mitigation of releases of radioactive materials following permanent closure whose likelihoods and amounts would

Table II

Items Potentially Important to Safety For The Yucca Mountain Repository

Items	Location	Dose/Frequency mrem	per year	Initiating Event
Crane, Shipping cask	Cask receiving and preparation area	390	5×10^{-5}	Crane drops shipping cask
Hot cell structure	Packaging Hot Cell	530	5×10^{-6}	Earthquake fails hot cell structure
Crane	Packaging Hot Cell	2,100	1×10^{-6}	Earthquake causes crane to fall on fuel assemblies
Vehicle Stop	Cask receiving and preparation area	NA	NA	Vehicle with cask falls in cask-preparation pit, detailed analysis not performed
Fire protection system	Waste-handling building	NA	NA	Chosen because fire involving radioactive material is a dispersion promoter, detailed analysis not yet performed

exceed the standards established in 40 CFR 191.13.[a]

This definition is similar in form to the definition of important to safety given in 10 CFR 60.2. Furthermore, this definition recognizes that the important issue is to comply with the overall system performance objective.

Under this definition, items important to waste isolation do not automatically include the engineered barriers for which 10 CFR 60.113 sets numerical performance objectives--objectives that may be changed. On the other hand, the definition does not preclude engineered barriers from being important to waste isolation. An engineered barrier (or any other barrier) is important to waste isolation only if its failure would produce releases that exceed the limits in 40 CFR 191.13. This definition has been used to identify the barriers, or items, important to waste isolation for the Yucca Mountain repository. The next two sections describe the methods used for this identification and the results of their application to the repository site and conceptual design.

Methods

The methods used to identify items important to waste isolation can be briefly summarized as follows. Identification begins with the development of a set of scenarios for releases of radioactive materials from the repository. The scenarios include both the expected behavior of the repository and unexpected sequences that could possibly disrupt the expected behavior. In practice, the development of scenarios attempts to identify bounding scenarios whose likelihoods and consequences are great enough to be upper limits to those of other scenarios. The large number of possible scenarios produced by a systematic evaluation of the possible future processes and events are grouped into classes, each of which can be represented by a single bounding scenario. Some potentially significant scenarios currently being considered are:

- Movement on a large fault inside the controlled area but outside the repository,
- Movement on a large fault within the repository,
- Movement on a small fault inside the controlled area but outside the repository,
- Movement on a small fault within the repository,
- Movement on a large fault outside the controlled area,
- Extrusive magmatic event that occurs during the first 500 years after closure,
- Extrusive magmatic event that occurs 500 to 10,000 years after closure,
- Intrusive magmatic event,
- Small-scale exploratory drilling,
- Large-scale exploratory drilling, and
- Incomplete sealing of the shafts and the repository.

The bounding scenarios are then evaluated for their likelihoods of occurrence and their consequences. The likelihoods are estimated from existing data. The consequences, evaluated in terms of the amount of radioactive material released to the accessible environment during the 10,000-year postclosure period, are estimated using current performance-assessment models. The likelihoods and consequences are then compared with the standards in 40 CFR 191.13. Barriers whose failure in these scenarios would produce a violation of those standards are then classified as important to waste isolation. The application of these methods is discussed in Refs. 13, 14, and 15.

[a]Subsequent to this work, new guidance has been developed by DOE and the Office of Geologic Repositories (OGR) that defines important to waste isolation as "those barriers, structures, systems, and components which are relied on to meet the postclosure performance objectives in 10 CFR 60 subpart E." This new guidance will be incorporated into future work; this paper documents what has been done to date.

Studies based on current performance-assessment models have revealed that no single barrier is essential to preventing or mitigating radioactive releases to the accessible environment (13). Because the repository system contains numerous barriers, the failure of a single barrier would not cause a release that violates the standards contained in 40 CFR 191.13. For this reason, none of the barriers meet the definition of "important to waste isolation." However, two hydrogeologic units below the repository are especially important to the performance of the repository system; these are the primary barriers against releases from several classes of scenarios. Since these two hydrogeologic units appear so frequently in the postclosure performance allocation of the repository, the NNWSI project has judged it prudent to classify these units as important to waste isolation. They are the Calico Hills nonwelded zeolitic unit and the Calico Hills nonwelded vitric unit at the site.

Summary

There are no items presently important to safety for the Yucca Mountain repository; however, there are six items that are potentially important to safety. The first of these items is a crane in the cask receiving and preparation area of the waste-handling building. In this case, a crane drop of a shipping cask is actually the initiating event of the PQ scenario that produced this potentially important-to-safety item. The second item potentially important to safety is the shipping cask dropped by the crane. The third and fourth items are a crane in a packaging hot cell and the packaging hot cell structure itself. These items were identified in two different scenarios, both initiated by an earthquake. The last two items are the truck and rail-car vehicle stop in the cask receiving and preparation area and the waste-handling building fire-protection system. The assumptions made and the analyses performed to identify these items were extremely conservative, and therefore future refinement of this work is not expected to alter significantly these results.

As mentioned above, there are no barriers that meet the definition of "important to waste isolation" for the Yucca Mountain repository. However, the NNWSI project has classified two important hydrogeologic units of the Yucca Mountain site as important to waste isolation. These units are the Calico Hills nonwelded zeolitic unit and the Calico Hills nonwelded vitric unit at the site.

Since there are no items important to safety and only two items important to waste isolation for the Yucca Mountain repository, the preliminary Q-list contains only two hydrogeologic units of the Yucca Mountain site. Items considered in this study were limited to major structures, systems, and components for the preclosure period. For the postclosure period, the site and engineered barriers were assessed. Activities such as site characterization, facility operations or procedures, and performance confirmation were excluded in this initial study. Military activities at the adjacent NTS were also excluded. As the repository design progresses, site characterization activities proceed, regulatory positions develop, performance-assessment methods evolve, and more information becomes available, the initial Q-list will be updated in a continuing assessment program.

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